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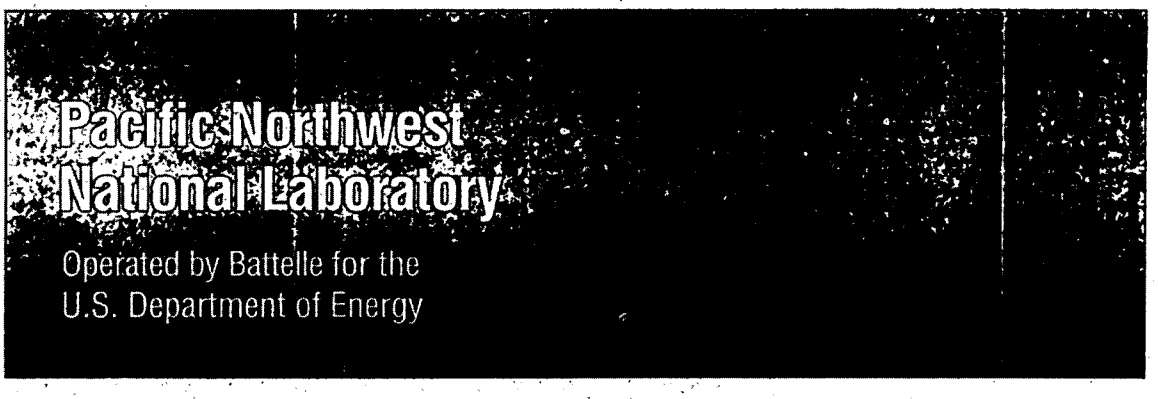

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Abstract

This report presents the first iteration of the *Composite Analysis for Low-Level Waste Disposal in the 200 Area Plateau of the Hanford Site* (Composite Analysis) prepared in response to the U.S. Department of Energy Implementation Plan for the Defense Nuclear Facility Safety Board Recommendation 94-2. The Composite Analysis is a companion document to published analyses of four active or planned low-level waste disposal actions: the solid waste burial grounds in the 200 West Area, the solid waste burial grounds in the 200 East Area, the Environmental Restoration Disposal Facility, and the disposal facilities for immobilized low-activity waste. A single Composite Analysis was prepared for the Hanford Site considering only sources on the 200 Area Plateau. The performance objectives prescribed in U.S. Department of Energy guidance for the Composite Analysis were 100 mrem in a year and examination of a lower dose (30 mrem in a year) to ensure the "as low as reasonably achievable" concept is followed. The 100 mrem in a year limit was the maximum allowable all-pathways dose for 1000 years following Hanford Site closure, which is assumed to occur in 2050. These performance objectives apply to an accessible environment defined as the area between a buffer zone surrounding an exclusive waste management area on the 200 Area Plateau, and the Columbia River.

Estimating doses to hypothetical future members of the public for the Composite Analysis was a multistep process involving the estimation or simulation of inventories; waste release to the environment; migration through the vadose zone, groundwater, and atmospheric pathways; and exposure and dose. Doses were estimated for scenarios based on agriculture, residential, industrial, and recreational land use. The radionuclides included in the vadose zone and groundwater pathway analyses of future releases were carbon-14, chlorine-36, selenium-79, technetium-99, iodine-129, and uranium isotopes. In addition, tritium and strontium-90 were included because they exist in groundwater plumes. Radionuclides considered in the atmospheric pathway included tritium and carbon-14.

Most of the radionuclide inventory in past-practice liquid discharge and solid waste burial sites on the 200 Area Plateau was projected to be released in the first several hundred years following Hanford Site closure and a significant fraction of the inventory was projected to be released prior to closure. The maximum predicted agricultural dose outside the buffer zone was less than 6 mrem in a year in 2050 and declined thereafter. The maximum doses estimated for the residential, industrial, and recreational scenarios, were 2.2, 0.7, and 0.04 mrem in a year, respectively, and also declined after 2050. The radiological doses for all of the exposure scenarios outside the buffer zone were well below the performance objectives.

Significant uncertainties exist in the first iteration Composite Analysis, with the largest uncertainty associated with the inventories of key mobile radionuclides. Other sources of uncertainty in the analysis arose from the conceptual and numerical models of contaminant migration and fate in the vadose zone, and assumptions regarding source-term release models and end states. These uncertainties reflect most on the performance of past releases of liquid wastes and past disposals of solid wastes. The review of existing plumes conducted as part of the first iteration Composite Analysis revealed that the exclusive waste management area and buffer zone should be expanded to include the retired Gable Mountain Pond. The Composite Analysis demonstrated a significant separation in time between past-practice discharges

and disposals, and active and planned disposals of solid waste, environmental restoration waste, and immobilized low-activity waste. The higher integrity disposal facilities and surface covers of these active and planned disposals delay releases, and the delayed releases do not superimpose on the plumes from the near-term past-practice disposals.

Executive Summary

This report summarizes the efforts to complete the first iteration of the *Composite Analysis for Low-Level Waste Disposal in the 200 Area Plateau of the Hanford Site* (Composite Analysis). In this document, the background and performance objectives of the Composite Analysis are described. The methods used, results, and conclusions are summarized. Recommendations are made for work to be undertaken in anticipation of a second analysis.

Introduction

The Composite Analysis was prepared in response to the U.S. Department of Energy Implementation Plan for the Defense Nuclear Facility Safety Board Recommendation 94-2 and in accordance with U.S. Department of Energy guidance[a]. The purpose of the Composite Analysis was to provide an estimate of the cumulative radiological impacts from active and planned low-level radioactive waste disposal actions and other potentially interacting radioactive waste disposal sources that will remain following Hanford Site closure. This Composite Analysis is a companion analysis to published analyses involving four active or planned low-level radioactive waste disposal actions:

- solid waste burial grounds in the 200 West Area[b]
- solid waste burial grounds in the 200 East Area[c]
- Environmental Restoration Disposal Facility[d]
- disposal facilities for immobilized low activity waste.[e]

Because these active and planned low-level radioactive waste disposal actions are located within the 200 Area Plateau of the Hanford Site, the U.S. Department of Energy, Richland Operations Office elected

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(a) U.S. Department of Energy. 1996. *Guidance for a Composite Analysis of the Impact of Interacting Source Terms on the Radiological Protection of the Public from Department of Energy Low-Level Waste Disposal Facilities.* U.S. Department of Energy, Washington, D.C.

(b) Wood, M.I., R. Khaleel, P.D. Rittmann, A.H. Lu, S.H. Finfrock, R.J. Serne, K.J. Cantrell, and T.H. DeLorenzo. 1995. *Performance Assessment for the Disposal of Low-Level Waste in the 200 West Area Burial Grounds.* WHC-ED-0645, Westinghouse Hanford Company, Richland, Washington.

(c) Wood, M.I., R. Khaleel, P.D. Rittmann, S.H. Finfrock, T.H. DeLorenzo, and D.Y. Garbrick. 1996. *Performance Assessment for the Disposal of Low-Level Waste in the 200 East Area Burial Grounds.* WHC-SD-WM-TI-730, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

(d) U.S. Department of Energy. 1994. *Remedial Investigation and Feasibility Study Report for the Environmental Restoration Disposal Facility.* DOE/RL-93-99, Rev. 1, U.S. Department of Energy, Richland, Washington.

(e) Mann, F.M., R.P. Puigh II, C.R. Eiholzer, Y. Chen, N.W. Kline, A.H. Lu, B.P. McGrail, and P.D. Rittmann. Publication planned for March 1998. *Hanford Immobilized Low-Activity Tank Waste Performance Assessment.* DOE/RL-97-69, Rev. 0, U.S. Department of Energy, Richland, Washington.

to complete a single Composite Analysis in support of the four disposal actions. The first iteration of the Composite Analysis only considered sources on the 200 Area Plateau because of their proximity to one another on the Plateau and the distance between the Plateau and other contaminated sites located near the shore of the Columbia River at the Hanford Site.

In addition to the active or planned sources, other radioactive sources exist or are planned for placement in the 200 Area Plateau of the Hanford Site. The sources that are the responsibility of U.S. Department of Energy include

- 149 single-shell tanks arrayed in 12 tank farms
- 28 double-shell tanks arrayed in 6 tank farms
- past-practice (pre-1988) solid waste burial grounds
- past-practice (pre-1988) liquid discharges to cribs, ditches, French drains, trenches, ponds, and reverse wells
- graphite cores from surplus production reactors
- canyon buildings and related structures (e.g. B Plant, Plutonium Uranium Extraction facility and tunnels, T-Plant, U-Plant, Reduction Oxidation Facility, and Z-Plant or Plutonium Finishing Plant).

The commercial low-level radioactive waste disposal facility operated by US Ecology, Inc., located immediately southwest of the 200 East Area was included in the Composite Analysis in accordance with guidance on the content and format of the Composite Analysis, and because of its proximity to U.S. Department of Energy operations.

Performance Objectives

The performance objectives of the Composite Analysis followed U.S. Department of Energy guidance for radiation dose to hypothetical future members of the public.[a] The U.S. Department of Energy Order 5400.5 (and anticipated 10 CFR 834) set the primary dose limit of 100 mrem in a year, but requires that a lower dose be examined (30 mrem in a year) to ensure the "as low as reasonably achievable" (ALARA) concept is followed. The 100 mrem in a year standard is the maximum allowable projected dose from all pathways to the hypothetical future member of the public. In accordance with U.S. Department of Energy guidance, the regulatory period of performance begins at the time of Hanford Site closure, assumed to be in 2050, and continues for 1000 years. In the Composite Analysis, an options analysis and ALARA assessment were to be prepared if the projected dose exceeded the dose constraint

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(a) All doses in the Composite Analysis (except where noted) are in units of mrem effective dose equivalent in a year.

of 30 mrem in a year. The options and ALARA analyses were to consider alternate actions that would reduce the calculated doses and to provide an assessment of cost and benefit.

At the Hanford Site, the approach adopted to achieve comprehensive environmental management involves a complex process of negotiated decisions among the U.S. Department of Energy, the State of Washington Department of Ecology, and the U.S. Environmental Protection Agency. These negotiations govern the U.S. Department of Energy response to Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) and Resource Conservation and Recovery Act (RCRA) requirements for remedial actions. The selection of alternate remedial actions for study needs to be a joint decision of the three parties. At this time, the U.S. Department of Energy is negotiating the cleanup of past-practice sites on the 200 Area Plateau at the Hanford Site with the regulatory agencies. There was insufficient time and information to determine if alternate remedies are necessary from the results of the Composite Analysis and to identify them through the negotiation process. Accordingly, a single remedial action (i.e., leave in place and cover with surface barrier) was analyzed in the Composite Analysis. Consideration of any additional alternate remediations, if necessary, is deferred to the second iteration of the Composite Analysis.

The point of compliance for exposure and radiological dose predictions to a hypothetical future member of the public in the Composite Analysis was a boundary based on anticipated land use at the Hanford Site. In 1992, the Hanford Future Site Uses Working Group,[(a)] comprising representatives from government entities (federal, tribal, state, and local) and constituencies (labor, environment, agriculture, economic development, municipal, and public interest groups), defined the concepts of an "exclusive" waste management area within a surrounding buffer zone on the 200 Area Plateau. This area includes the land within and surrounding the 200 East and 200 West Areas of the Hanford Site, the commercial low-level radioactive waste disposal facility operated by US Ecology, Inc., and the Environmental Restoration Disposal Facility. The first chapter of this report contains figures that show the relationship between the exclusive waste management area and buffer zone, and the waste sites included in the Composite Analysis. The policy of the U.S. Department of Energy is to control and maintain the land within the exclusive waste management area and buffer zone until it can be released to the public. The U.S. Department of Energy has acknowledged that many low-level radioactive waste disposal facilities may never be suitable for unconditional release to the public, and deed restrictions for use of resources such as groundwater may be necessary. The projected doses to hypothetical future members of the public from the low-level radioactive waste disposal actions and all other sources considered in the Composite Analysis were compared with the dose limit of 100-mrem in a year and dose constraint of 30-mrem in a year in the area between the buffer zone and the Columbia River.

Methodology

The process used in the Composite Analysis is necessarily iterative, adaptive, and flexible in order to respond to the constantly changing technical and decision-maker needs. This document discusses the

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(a) Hanford Future Site Uses Working Group. 1992. *The Future for Hanford: Uses and Cleanup, the Final Report of the Hanford Future Site Uses Working Group.* Hanford Future Site Uses Working Group, Westinghouse Hanford Company, Richland, Washington.

initial iteration of the Composite Analysis that has resulted in a deterministic baseline. For the first iteration, estimating doses to hypothetical future members of the public for the Composite Analysis was a multistep process.

- The first step involved estimating the inventories of radionuclides for the various sources present or to be placed on the 200 Area Plateau. A complete and accurate inventory of sources of radioactive materials disposed to ground and stored at the Hanford Site does not exist. Consequently, an inventory had to be estimated based on process knowledge and plans for environmental restoration.

- The second step in the analysis involved calculating the radionuclide release from the various sources, based on knowledge of waste form characteristics and long-term performance calculations (recharge characteristics and geochemical behavior).

- The third step involved predicting transport through the vadose zone to the water table under transient flow conditions. The recharge rate in the vadose zone was allowed to vary with the application of different surface treatments and covers (barriers).

- The fourth step involved predicting transport through the unconfined aquifer. The aquifer was modeled as it responded to the cessation of wastewater discharges from Hanford Site operations and its water table declined during the post-closure period.

- The fifth step in the analysis involved calculating dose based on exposure scenarios for hypothetical future members of the public at locations on the present Hanford Site and comparing those doses with the dose limit and constraint standards.

The Data Quality Objectives process was applied to the Composite Analysis for the 200 Area Plateau although no new data were collected. Existing data on source inventories, waste site characteristics, and the vadose zone and groundwater were compiled and used with release and transport models to predict future radionuclide concentrations in environmental media (air, soil, and groundwater) and resulting radiation doses. However, the U.S. Department of Energy guidance for the Composite Analysis specifically prohibited improvement of data through the gathering and analysis of samples for the first iteration analysis.

Four exposure scenarios defined by the Hanford Site Risk Assessment Methodology were used in the Composite Analysis to estimate radiation doses to hypothetical future members of the public. These four scenarios were used to examine the potential variability in future land use. The four Hanford Site Risk Assessment Methodology exposure scenarios, agricultural, residential, industrial, and recreational, were developed for the Hanford Site to facilitate evaluation of risk related to CERCLA remedial investigations and RCRA facility investigations. Groundwater transport was the primary exposure pathway considered in the Composite Analysis for the 200 Area Plateau. However, a limited analysis of exposure and dose from the atmospheric pathway was included in the all-pathways dose assessment.

The waste source inventories at the Hanford Site were screened to select key radionuclides for study in the Composite Analysis. The effort to screen the list of radionuclides benefited from published

performance assessment and environmental impact analyses and field observations (characterization and monitoring data). Those radionuclides identified as potentially significant contributors to dose in performance assessments for the 200 West and 200 East Area post-1988 burial grounds and the Tank Waste Remediation System immobilized low-activity waste, and the risk assessment for the Environmental Restoration Disposal Facility were assumed to be the key radionuclides in the Composite Analysis. The radionuclides included in the groundwater pathway analysis for future sources included carbon-14, chlorine-36, selenium-79, technetium-99, iodine-129, and uranium isotopes (uranium-233, -234, -235, -236, and -238). In addition, tritium and strontium-90 were included because they currently form groundwater plumes at the Hanford Site. Radionuclides considered in the atmospheric pathway included tritium, carbon-14, and radon-222. However, the surplus graphite cores of production reactors were identified as the only potentially significant source for the atmospheric pathway, and they release tritium and carbon-14, but have no appreciable inventory of radon-222 or its parents.

Results

Prior to conducting the analysis, a review of existing radionuclide contamination in the unconfined aquifer revealed the presence of a strontium-90 plume beneath the decommissioned Gable Mountain Pond. Strontium-90 is relatively highly sorbed in Hanford groundwater and sediments, (e.g., distribution coefficient of 20 mL/g), and will be reduced relatively soon by decay (i.e., half-life of 28.78 years). Because the contamination is relatively immobile and is in the vadose zone sediment column and the groundwater beneath this retired pond, it is assumed it will not be further remediated. In the following discussion of dose outside the buffer zone, the assumption is made that Gable Mountain Pond is included in the exclusive waste management area and buffer zone.

In the Composite Analysis, most of the radionuclide inventory in past-practice liquid discharge and solid waste burial sites on the 200 Area Plateau was projected to be released within the first several hundred years following Hanford Site closure. A significant fraction of that radionuclide inventory was projected to release prior to Hanford Site closure in 2050, and peak concentrations of key radionuclides in groundwater are predicted to occur before closure in 2050. For the agricultural exposure scenario, which results in the highest predicted doses, the maximum dose from the key radionuclides and all sources considered in the Composite Analysis outside the buffer zone at Hanford Site closure was less than 6 mrem in a year. The maximum dose from the agricultural scenario declined thereafter. The maximum doses estimated for the other scenarios, i.e., residential, industrial, and recreational, were 2.2, 0.7, and 0.04 mrem in a year, respectively, and also declined after 2050. The groundwater plumes from existing and future sources considered in the analysis are predicted to migrate away from the 200 Area Plateau in two primary directions, to the east and southeast following the major existing plumes, and to the north. The groundwater flow paths gradually change from an initial radial pattern from the 200 Area Plateau, to an easterly direction as the water table changes in response to cessation of wastewater disposal.

A brief ALARA assessment showed the cost to society associated with population dose during the 1000-year period after Hanford Site closure was between $4 million and $40 million. This estimated range was based on an approximate average dose to an individual of 4 mrem in a year from the agricultural scenario for the 1000-year period, an agricultural community of 1000 people, and a cost to society of between $1000 and $10,000 per person rem. The dose and community population estimates are

conservative; therefore, the cost estimate is biased high. Based on this estimated cost to society, a thorough options analysis and ALARA assessment that would provide a detailed investigation of alternate remediations was not justified at this time.

The radiological doses for all of the exposure scenarios outside the buffer zone were well below the dose limit of 100 mrem in a year and the dose constraint of 30 mrem in a year. However, the predicted radionuclide concentrations in groundwater within the exclusion and buffer zones demonstrate the need for continued land use control and monitoring programs at the Hanford Site to meet the primary objective of the long-term protection of human health and safety. This analysis of future radiological dose to the maximally exposed individual on lands outside the buffer zone supports the concept of retiring the Hanford Site boundary to the buffer zone boundary at the time of Hanford Site closure in 2050. However, the conclusions of the Composite Analysis depend on the inclusion of Gable Mountain Pond in the exclusive waste management area, and continued land use controls by the U.S. Department of Energy to prohibit use of resources (groundwater and land) within the exclusive waste management area and buffer zone for the 1000-year period of analysis.

Significant uncertainties exist in the first iteration of the Composite Analysis, with the largest uncertainty associated with the inventories of key radionuclides discharged and disposed at specific facilities by the time of Hanford Site closure. The inventory for post-closure sources at the Hanford Site was assembled from inventories for specific wastes, waste forms, and waste discharge sites. These prior efforts to define specific inventories occurred at different times, under different U.S. Department of Energy programs, and were not coordinated to produce a single and consistent database for wastes that will reside at the Hanford Site after closure. Inventory characterization is also incomplete because of the costs and limitations of characterization efforts to address specific questions. Consequently, the total inventory of key mobile radionuclides examined in the Composite Analysis includes significant uncertainties. A more thorough examination of uncertainty with respect to inventory must follow development of a more consistent inventory where best estimates of both magnitude and final location of radionuclides are included. It would be advantageous to have an inventory model that could generate alternate realizations based on the range of process parameters assigned to Hanford Site facility operations.

Another source of uncertainty in the analysis arose from the conceptual and numerical models of the contaminant migration and fate in the vadose zone. The conceptual model has considered transient recharge rates and spatially varying retardation factors that govern the leaching of waste and its migration. The recharge rates were designed to represent periods of high recharge and leaching prior to the placement of covers and barriers, and periods of low recharge associated with the vegetation of the site or the placement of covers. The model of geochemical mobility has taken into account the character of the waste and the neutralizing effects of contact with soils and sediments. However, the model has not included preferential pathways such as clastic dikes or unsealed well bores. The vadose zone model employed in this first iteration of the Composite Analysis is one-dimensional, and, therefore, was not able to represent multidimensional effects. The model adopted did not consider the potential influence of non-isothermal or high-density fluids on the migration and fate of radionuclides. Currently, the Tank Waste Remediation System Vadose Zone Program is working to gather field data and establish the conceptual models for contaminant migration and fate in the vadose zone beneath tank farms. Because this program

has just begun and an effort to integrate and coordinate it with other vadose zone studies has just gotten underway, examination of multiple conceptual models of the vadose zone pathway has been deferred until the second iteration Composite Analysis.

Other sources of uncertainty included assumptions regarding source-term release models, and end states for waste sites. These sources of uncertainty are not believed to be as significant as the uncertainties in the inventory estimates or pathway analyses.

Conclusions

Conduct of the first iteration Composite Analysis has revealed the exclusive waste management area and buffer zone should be extended to include Gable Mountain Pond. This first analysis has also shown a significant separation in time between past-practice discharges and disposals, and active and future disposals at the Hanford Site. Liquid discharge sites including cribs and specific retention trenches, past leaks from single-shell tanks, future sluicing losses from single-shell tanks, and pre-1988 solid waste burial grounds all release to the water table in the coming decades. Significant portions of their inventories release within the next century. Active and planned disposals release much later. When modeled with a high recharge rate prior to the placement of final covers, the post-1988 solid waste burial grounds are shown to begin release of their estimated highly mobile inventories of chlorine-36, selenium-79, and technetium-99 within 200 years from the present. However, a scenario that takes credit for the waste isolation afforded by burial containers showed mean travel times of these mobile radionuclides were between 650 and 1150 years. A mean travel time of approximately 1000 years was associated with burial grounds destined to receive the majority of future solid waste disposals. Neither scenario indicated release of radionuclides exhibiting adsorption in the vadose zone, i.e., carbon-14, iodine-129, or uranium. Neither the Environmental Restoration Disposal Facility nor the Tank Waste Remediation System immobilized low-activity waste were shown to release in the 1000-year analysis period. Thus, the higher integrity disposal facilities and surface covers cause releases from active and planned disposals to occur much later in time and not superimpose on near-term releases that have occurred and will occur from the liquid disposal facilities.

This analysis has also shown that concentrations of radionuclides in the aquifer are much higher during the period from now until Hanford Site closure than they will be after closure. Similarly, doses based on the assumed use of groundwater from now until Hanford Site closure are much higher in the period leading up to site closure than in the period after. Contaminants in the aquifer today are a result of the early discharge of large quantities of liquid waste or direct injection at reverse well sites. Consequently, the resulting plumes had relatively high concentrations, and continue to exhibit relatively high peak values today despite years of groundwater transport, radioactive decay, and dispersal. Past rates of liquid discharge, (e.g., ~12,000 curies per year of tritium in ~400 cubic meters per day of liquid discharge between 1984 and 1986), were orders of magnitude higher than any predicted future release rates to groundwater.

Future releases to the aquifer from the liquid discharge sites, tank leaks, tank losses, and burial grounds will occur, but with a greatly diminished driving force as compared to past releases. This is because natural recharge rates and not large liquid releases will drive future leaching and movement.

While more curies of specific radionuclides such as technetium-99 will leach into the aquifer in the future than are present today, they will be introduced at lower rates. Because flow in the unconfined aquifer under the 200 Area Plateau will remain relatively constant, these lower projected release rates from sources in the vadose zone will create plumes with lower peak concentrations. Consequently, the Composite Analysis has shown that future doses through time of Hanford Site closure and beyond will be dominated by the existing plumes of tritium and iodine-129. As tritium concentrations are reduced by migration to the Columbia River, dispersion, and decay, the iodine-129, which is less mobile than tritium in Hanford Site sediments, begins to dominate dose projections.

Dose estimates during the post-closure period are low, less than 6 mrem in a year to the maximally exposed individual in the agricultural exposure scenario. The area of the unconfined aquifer predicted to yield estimates of dose above 4 mrem in a year for the agricultural scenario decreases from approximately 40 km^2 in 2050 to zero by 2085. If inventories of the mobile radionuclides assigned to liquid discharge sites, past tank leaks, future tank sluicing losses, and pre-1988 solid waste burial grounds were increased, higher doses could be tolerated before approaching the dose constraint of 30 mrem in a year. If inventories of key mobile nuclides assigned to active disposals were increased beyond the current inventory limits of the facility, protocols require an analysis to ensure the safety of the disposal action prior to waste acceptance. When high concentrations of key radionuclides appear in waste delivered for disposal, the waste acceptance criteria and protocols employed at active solid waste burial grounds reveal their presence. Because of their potential adverse impact on long-term radiological dose, these radionuclides (when in high concentrations) are contained in waste forms or containers that inhibit leaching and release. Thus, greater inventories of key nuclides could be disposed in active or planned disposals only after a thorough analysis of their potential impact and appropriate actions to ensure their safe disposal.

As a companion analysis for the performance and risk assessments associated with current and planned low-level radioactive waste disposal actions at the Hanford Site, the Composite Analysis has shown that the active and planned dry disposals are safe and will not contribute significantly to radiation dose to hypothetical future members of the public for the 1000-year period following Hanford Site closure.

Future Work

Three key areas where additional data and information will contribute to greater confidence in the second iteration Composite Analysis are

- a fully consistent Hanford Site-wide inventory
- an accepted suite of conceptual models of liquid and dry disposals
- a tested linkage of inventory, release, and vadose zone models sufficient to explain existing plumes.

A Hanford Site-wide inventory model should be created, or an existing model modified, to provide a probabilistic estimate of the magnitude (e.g., mean value and standard deviation) of all key radionuclides

for all significant disposals at the time of Hanford Site closure. The list of key radionuclides should be reevaluated to consider those found to have greater mobility when disposed with organic chelating agents at liquid discharge sites. The concept of mass conservation should be used to ensure the probabilistic distributions are fully consistent with known quantities generated at or imported to the Hanford Site. The inventory model should include credible estimates of radionuclides lost to the atmosphere, discharged as liquid and disposed as solids. The inventory model must include liquids discharged to facilities (cribs, French drains, reverse wells, and specific retention trenches), leaked from tanks, and forecast to be lost from tanks during waste recovery operations. The model should include estimates of radionuclides retained in canyon buildings, permanent filters, and tunnels. It should include a means of estimating the disposal of key mobile radionuclides to all facilities, and it should address the secondary waste streams coming from future facilities and programs including tank waste recovery operations, chemical separations plants, and plants designed to immobilize both low-activity and high-level wastes.

Efforts are now ongoing to provide greater insight into the present location, and past and present mobility of contaminants from past tank leaks at the Hanford Site. This information and that from the study of past-practice liquid discharge and dry disposal sites will lead to greater understanding of contaminant migration in the vadose zone at the Hanford Site. From this knowledge will come peer-reviewed and accepted conceptual models for liquid discharges, tank leaks and losses, and solid waste burials. These conceptual models will identify the applicable recharge rates, geohydrologic formations, dimensionality, and geochemistry of waste-sediment interactions. The potential value of more sophisticated and higher dimensionality vadose zone models to future Composite Analyses should be evaluated. In the second iteration Composite Analysis, the range of conditions defining uncertainty in radionuclide migration should be simulated to capture the associated uncertainty in dose estimates.

Finally, the inventory and conceptual models associated with specific facilities should be tested and evaluated where possible. Each existing vadose zone and groundwater plume is the result of a past waste discharge or disposal. Of particular interest are facilities that received, or are suspected to have received, large inventories of key mobile radionuclides. Of special interest will be those sites with highly uncertain and potentially significant contributions to the composite dose. For such sites, efforts should be made to obtain sound estimates (mean and standard deviation) of the spatial distribution of the mass of contaminants in the vadose zone and unconfined aquifer. Application of inventory, release model, and vadose zone contaminant migration models should yield estimates of mass in the vadose zone and released to the aquifer. These estimates should be in agreement with mass estimates based on field observations. Inventory estimates that can not be supported by reasonable release and vadose zone models when compared to vadose zone and groundwater plume data should be revisited to ensure that overly conservative or bounding estimates of inventory have not been assigned to liquid discharges. If possible, this effort should include facilities that gave rise to plumes in the vadose zone that have not yet reached the water table. Confidence in the present state of contamination in the vadose zone is central to building confidence in projections of future release. Where the results would support the understanding of major contributors to total dose, efforts should be made to sample for and interpret the plume mass of all key mobile radionuclides.

Acronyms

ACGIH	American Conference of Governmental Industrial Hygienists
ALARA	as low as reasonably achievable
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CFEST	Coupled Fluid, Energy, and Solute Transport (code)
Composite Analysis	*Composite Analysis for Low-Level Waste Disposal in the 200 Area Plateau of the Hanford Site*
DBBP	dibutyl butyl phosphate
DNFSB	Defense Nuclear Facility Safety Board
DOE	U.S. Department of Energy
DOE-HQ	U.S. Department of Energy, Headquarters
DOE-RL	U.S. Department of Energy, Richland Operations Office
DOH	State of Washington Department of Health
DQO	data quality objective
DWS	drinking water standards
Ecology	State of Washington Department of Ecology
EDE	effective dose equivalent
EDTA	ethylenediaminetetraacetic acid
EIS	environmental impact statement
EPA	U.S. Environmental Protection Agency
ERC	Environmental Restoration Contractor
ERDA	U.S. Energy Research and Development Administration

ERDF	Environmental Restoration Disposal Facility
FFTF	Fast Flux Test Facility
GIS	geographic information system
HDW	Hanford Defined Waste
HEAST	Health Effects Assessment Summary Tables
HEDTA	N-(2-hydroxyethyl) ethylenediaminetetraacetic acid
HEPA	high-efficiency particulate air filter
HFSUWG	Hanford Future Site Uses Working Group
HGWP	Hanford Groundwater Project
HSRAM	Hanford Site Risk Assessment Methodology
HTI	Hanford Tanks Initiative
ICRP	International Commission on Radiological Protection
ILAW	immobilized low-activity waste
IRIS	Integrated Risk Information System
LLW	low-level waste
MEPAS	Multimedia Environmental Pollutant Analysis System
MMEDE	Multimedia Modeling Environmental Database Editor
NPH	normal paraffinic hydrocarbon (or kerosene)
NPL	National Priorities List
ONWI	Office of Nuclear Waste Isolation
ORIGEN2	Oak Ridge Isotope Generation and Depletion (code)
OU	operable unit

PCB	polychlorinated biphenyl
PFP	Plutonium Finishing Plant
PNNL	Pacific Northwest National Laboratory
PUREX	Plutonium Uranium Extraction (Plant)
RCRA	Resource Conservation and Recovery Act
REDOX	Reduction-Oxidation (S Plant)
RfD	Reference Dose
RFP	Request for Proposal
RI/FS	remedial investigation and feasibility study
ROD	record of decision
SALDS	State-Approved Land Disposal Site
SARA	Superfund Amendments and Reauthorization Act
SEPA	State Environmental Policy Act
SIF	Summary Intake Factors
STOMP	Subsurface Transport Over Multiple Phases (code)
S/V	surface to volume (ratio)
SWITS	Solid Waste Information Tracking System
TBP	tributyl phosphate
TEDF	Treated Effluent Disposal Facility
TLV	threshold limit value
TRAC	Track Radioactive Components (model)

Tri-Party Agreement	*Hanford Federal Facility Agreement and Consent Order Between the U.S. Environmental Protection Agency, the U.S. Department of Energy, and the State of Washington Department of Ecology*
TRU	transuranic
TWRS	Tank Waste Remediation System
UDFs	unit dose factors
UTFs	unit transport factors
VOC	volatile organic compounds
VOI	Value of Information (analysis)
WIPP SEIS	Waste Isolation Pilot Plant Supplementary Environmental Impact Statement

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1.0 Introduction

The *Composite Analysis for Low-Level Waste Disposal in the 200 Area Plateau of the Hanford Site* (Composite Analysis) is a radiological assessment to estimate doses to hypothetical future members of the public from radionuclides from low-level waste (LLW) disposal and all other sources of radioactive contamination at the Hanford Site (Figure 1.1). The first iteration of the Composite Analysis is a companion analysis to the facility-specific risk documentation for the following four active or planned LLW disposal actions:

- post-1988 solid waste burial ground in the 200 West Area (Wood et al. 1995)
- post-1988 solid waste burial ground in the 200 East Area (Wood et al. 1996)
- Environmental Restoration Disposal Facility (ERDF) (DOE 1994b)
- disposal facilities for immobilized low-activity wastes.[a]

The Composite Analysis is part of the documentation required for the continued and planned LLW disposal operations at these four facilities at the Hanford Site.

1.1 Purpose and Scope

The Implementation Plan for the Defense Nuclear Facility Safety Board (DNFSB) Recommendation 94-2 (DOE 1996e) requires that a Composite Analysis be prepared to accompany the performance assessments for the burial grounds and the planned low-activity tank waste disposal, and the risk assessment for ERDF. This Composite Analysis was prepared to provide an estimate of the cumulative radiological impacts from the active and planned LLW disposal actions and other potentially interacting radioactive sources at the Hanford Site. The calculations for this Composite Analysis were performed with a combination of spreadsheet programs, multidimensional numerical models, and geographic information system software. The U.S. Department of Energy Richland Operations Office (DOE-RL) has elected to complete a single Composite Analysis for wastes disposed in the 200 Area Plateau because multiple LLW disposals will occur at Hanford, and many waste sites are present that may interact with the LLW disposals.

A multistep approach was used to estimate doses in the Composite Analysis.

- The first step was to estimate the inventories of radionuclides for the various sources. A complete and accurate inventory of sources of radioactive materials disposed to ground and stored at the Hanford Site does not exist and had to be estimated based on knowledge of the processes that generated the waste.

(a) Mann, F. M., R. P., Puigh II, C. R. Eiholzer, Y. Chen, N. W. Kline, A. H. Lu, B. P. McGrail, and P. D. Rittman. Publication planned for March 1998. *Hanford Immobilized Low-Activity Tank Waste Performance Assessment.* DOE/RL-97-69, Rev. 0. U.S. Department of Energy, Richland, Washington.

- The second step in the analysis was to calculate the radionuclide release from the various sources, based on knowledge of waste form characteristics and long-term performance calculations (e.g., recharge characteristics and geochemical behavior).

- The third step was to predict transport through the vadose zone under transient conditions. The recharge rate in the vadose zone was allowed to vary with different surface conditions and especially surface covers (barriers).

- The fourth step was to predict transport through the unconfined aquifer under transient conditions. Groundwater flow in the unconfined aquifer responded to the cessation of wastewater discharges from Hanford Site operations and declined. Separate analyses of existing contaminant plumes and future releases from the vadose zone were conducted.

- The fifth step in the analysis was to calculate dose based on exposure scenarios for hypothetical future members of the public at locations on the Hanford Site and compare those doses with standards outlined in the Composite Analysis guidance (DOE 1996b). The dose estimates provided represent the effective dose equivalent received over a commitment period of 50 years.

The scope of the first iteration Composite Analysis was to consider all radioactive sources within the 200 Area Plateau of the Hanford Site that could potentially interact with the active and planned LLW disposal actions. The four LLW disposal actions are located on the 200 Area Plateau, located near the center of the Hanford Site.

Chapter 3 describes the Composite Analysis source term. Chapter 4 discusses the release, vadose zone, groundwater, and exposure simulation methods and results. Chapter 5 presents an interpretation of results, a discussion of uncertainties, and suggestions for further study.

An approach for the future reduction of uncertainty and the establishment of greater confidence in subsequent iterations of the Composite Analysis is described in Chapter 2 of this report. The U.S. Department of Energy (DOE) has issued guidance that performance assessments, Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) risk assessments, and the composite analysis are to be maintained. Significant changes in future land use (e.g., the future DOE property boundary), the inventories, the release models, the environment transport pathways, and exposure and dose scenarios would precipitate another iteration of the composite analysis (DOE 1996b).

1.2 Regional Setting

The Hanford Site lies within the semiarid Pasco Basin of the Columbia Plateau in southeastern Washington State (Figure 1.1; Dirkes and Hanf 1997). The site occupies an area of approximately 1,450 km^2 (approximately 560 mi^2) located north of the city of Richland, Washington, and the confluence of the Yakima and Columbia Rivers. This large area has restricted public access and provides a buffer for the smaller areas onsite that were used for research, fuel fabrication, fuel irradiation, the production

of nuclear materials, and the storage and disposal of wastes. Approximately 6% of the land area has been disturbed and is actively used. The Columbia River flows eastward through the northern part of the Hanford Site and then turns south, forming part of the eastern site boundary. The Yakima River flows near a portion of the southern boundary and joins the Columbia River downstream of the city of Richland.

The cities of Richland, Kennewick, and Pasco (known collectively as the Tri-Cities) constitute the nearest population center and are located southeast of the Hanford Site. Richland and Kennewick are in Benton County, and Pasco is in Franklin County. Land surrounding the Hanford Site is used for urban and industrial development, irrigated and dryland farming, and grazing. In 1995, population totals for Benton and Franklin Counties were estimated at 131,000 and 44,000, respectively (Washington State Office of Financial Management 1995). The estimated 1995 populations of the Tri-Cities were: Richland, 36,270; Pasco, 22,500; and Kennewick, 48,130. The combined populations of three smaller outlying communities of the Tri-Cities (i.e., Benton City, Prosser, and West Richland) totaled 13,320 in 1995.

1.3 Site Description

Major operational areas at Hanford are described in the following list.

- The 100 Areas, on the south shore of the Columbia River, are the sites of nine retired plutonium production reactors, including the dual-purpose N Reactor. The 100 Areas occupy approximately 11 km^2 (4 mi^2).

- The 200 West and 200 East Areas are located on a plateau, approximately 8 and 11 km (5 and 7 mi), respectively, south of the Columbia River. Historically, these areas have been dedicated to fuel reprocessing and waste processing management and disposal activities. The 200 Areas cover approximately 16 km^2 (6 mi^2).

- The 300 Area is located just north of the city of Richland. Fuel fabrication facilities were operated in this area, and it is the site of nuclear research and development. This area covers 1.5 km^2 (0.6 mi^2).

- The 400 Area is approximately 8 km (5 mi) northwest of the 300 Area and is the site of the Fast Flux Test Facility (FFTF), used in the testing of breeder reactor systems. Also included in this area is the Fuels and Materials Examination Facility.

- The 600 Area includes all of the Hanford Site not occupied by the 100, 200, 300, and 400 Areas.

- The 1100 and Richland North Areas are located south of the Hanford Site, in the northern portion of the city of Richland. These are support areas that include general stores, transportation maintenance, and the DOE and DOE contractor facilities.

During 1996, the 3000 Area was cleaned up and vacated by DOE and its contractors. All land and facilities within the area were turned over to the Port of Benton, and the 3000 Area designation was retired (Dirkes and Hanf 1997).

Several areas of the Hanford Site (a total land area of 665 km^2 [257 mi^2]) have special designations. These areas include the Fitzner/Eberhardt Arid Lands Ecology Reserve, the U.S. Fish and Wildlife Service Saddle Mountain National Wildlife Refuge, and the Washington State Department of Game Reserve Area (Wahluke Slope Wildlife Recreation Area) (Dirkes and Hanf 1997). Management of the Fitzner/Eberhardt Arid Lands Ecology Reserve was transferred to the U.S. Fish and Wildlife Service in 1997. It is currently part of the National Wildlife Refuge system.

Non-DOE activities on Hanford Site leased land include commercial power production on the land occupied by the Washington Public Power Supply System WNP-2 plant (and partially completed WNP-1 and WNP-4 plants) and operation of a commercial LLW burial site by US Ecology, Inc. Immediately adjacent to the southern boundary of the Hanford Site, Siemens Power Corporation operates a commercial nuclear fuel fabrication facility, and Allied Technology Group Corporation operates an LLW decontamination, supercompaction, and packaging disposal facility. Kaiser Aluminum and Chemical Corporation leases the 313 Building in the 300 Area to use an extrusion press that was formerly owned by DOE. The National Science Foundation is building the Laser Interferometer Gravitational-Wave Observatory facility between the 200 and 400 Areas.

The Hanford Site description and historical site operation information presented here were taken from the introduction section of the *Hanford Site Environmental Report for Calendar Year 1996* (Dirkes and Hanf 1997). More detailed information on the Hanford Site environment is provided by Neitzel (1997).

1.4 Historical Site Operations

The Hanford Site was established in 1943 with the mission to produce plutonium for nuclear weapons (Dirkes and Hanf 1997). Hanford operations have resulted in the production of liquid, solid, and gaseous wastes. Most wastes from these operations have a potential to contain radioactive materials. From an operational standpoint, radioactive wastes were originally categorized as "high level," "intermediate level," and "low level," which referred to the level of radioactivity present.

Some high-level solid waste, such as large pieces of machinery and equipment, were placed onto railroad flatcars and stored in underground tunnels. Both intermediate- and low-level solid wastes (e.g., tools, machinery, paper, and wood) were placed into covered trenches at storage and disposal sites known as burial grounds. Beginning in 1970, solid wastes were segregated according to the makeup of the waste material. Solids containing plutonium and other transuranic materials were packaged in special containers and stored in lined trenches covered with soil for possible later retrieval. High-level liquid wastes were stored in large underground tanks in the 200 Areas.

Intermediate-level liquid waste streams were usually routed to underground structures of various types including cribs, French drains, and specific retention trenches. Occasionally, trenches were filled with the liquid waste and then covered with soil after the waste had soaked into the ground. Low-level liquid waste streams were usually routed to surface impoundments (ditches and ponds). Nonradioactive solid wastes were usually burned in burning grounds. This practice was discontinued in the late 1960s in response to the Clean Air Act, and the materials were buried at sanitary landfill sites instead. These storage and disposal sites, with the exception of high-level waste tanks, are now designated as active or inactive waste sites, depending on whether the site currently receives wastes.

1.4.1 The 300 Area

From the early 1940s to the present, most research and development activities at the Hanford Site were carried out in the 300 Area, located just north of Richland. Until 1987, the 300 Area was also the location of nuclear fuel fabrication. Nuclear fuel in the form of pipe-like cylinders (fuel slugs) was fabricated from metallic uranium shipped in from offsite production facilities. Metallic uranium was extruded into the proper shape and encapsulated in aluminum or zirconium cladding.

Substantial amounts of copper, uranium, and other heavy metals were found in 300 Area liquid waste streams. Until the mid-1970s these streams were routed to the 300 Area waste ponds, which were located near the Columbia River shoreline. In more recent times, the low-level liquid wastes were sent to process trenches or shipped to the solar evaporation facility in the 100-H Area (183-H Solar Evaporation Basins). Discharge to process trenches ceased in December 1994.

1.4.2 The 100 Areas

The 100 Areas are located on the southern shore of the Columbia River in the northern portion of the Hanford Site, where in the past up to nine nuclear reactors were in operation. The graphite cores of the eight production reactors were the host environment for the conversion of uranium atoms to plutonium atoms. Also produced were radionuclides from the fission and activation processes.

When fresh fuel slugs were pushed into the front face of a reactor's graphite pile, the irradiated fuel slugs were forced out the rear into a deep pool of water called a fuel storage basin. After a brief period of storage, the irradiated fuel was shipped in specially constructed railcars to the 200 Areas for processing.

The N Reactor (the ninth reactor) ran from the early 1970s to the early 1980s with the dual missions of electricity and plutonium production. Beginning in 1975, N Reactor irradiated fuel was shipped to the K East and K West Fuel Storage Basins for temporary storage where it remains today. This fuel accounts for the majority of the fuel currently stored underwater in the 100-K Area fuel storage basins. The majority of material produced in N Reactor from the early 1980s until 1987 was processed in the 200 East Area. The remainder is stored in the K Basins.

1.4.3 The 200 Areas

The 200-East and 200-West Areas are located on a plateau at the center of the Hanford Site. Figure 1.2 shows the areas that housed chemical separation plants that received and dissolved irradiated fuel and then separated out the plutonium (Dirkes and Hanf 1997, Figure 1.0.3). At different times and in different plants, three processes were used to perform the separation. Each of the plutonium production processes began with the dissolution of the aluminum or zirconium cladding material in solutions containing ammonium hydroxide, ammonium nitrate, and ammonium fluoride, followed by the dissolution of the irradiated fuel slugs in nitric acid. This chemical processing step produced large quantities of nitric acid solutions containing high levels of radioactive materials. These wastes were neutralized and stored in large underground tanks. Fumes from the dissolution of cladding and fuel, and from other process steps were discharged to the atmosphere.

The first separation process was the bismuth phosphate precipitation that operated from 1945 until 1956 in B and T Plants. This method was supplanted by a second, more efficient method that involved contacting a methyl isobutyl ketone (hexone) organic phase with an aqueous aluminum nitrate solution of plutonium and uranium from dissolved fuel. This process was run from 1952 until 1966 in the Reduction-Oxidation (REDOX) Plant.

Finally, the REDOX process was replaced by a much-improved solvent extraction based on an organic phase that was a mixture of normal-paraffinic hydrocarbon or kerosene (NPH) and tributyl phosphate (TBP) contacting an aqueous nitric acid solution of plutonium and uranium. The Plutonium Uranium Extraction (PUREX) Plant ran this process from 1956 until 1972 then restarted in 1983 and ran until 1988. Wastes from each of these process steps were neutralized and placed into storage tanks. Some tank wastes were directed to cribs and trenches and disposed in the unsaturated soil profile (Waite 1991).

A uranium recovery campaign was undertaken at U Plant from 1952 until 1956. The bismuth phosphate process did not recover uranium from the process stream, and it was decided to recover uranium from the metal wastes stored in the large underground single-shell tanks. These wastes were sluiced from tanks and a process based on the TBP/NPH solvent extraction was applied. The uranium was purified into uranium oxide powder at the Uranium-TriOxide Plant. Ultimately, because of the volume of waste produced in this process, ferrocyanide was used to scavenge or entrap the cesium-137 in a precipitated sludge, and the supernatant was placed into the cribs or trenches.

The REDOX and PUREX Plants produced uranium nitrate for recycle and plutonium nitrate for weapons. Uranium nitrate was further processed in the Uranium TriOxide Plant. Plutonium nitrate was transferred to Z Plant (later called the Plutonium Finishing Plant, or PFP) for conversion to plutonium metal. The conversion processes used nitric acid, hydrofluoric acid, carbon tetrachloride, and various oils and degreasers. Varying amounts of all these materials were in the intermediate-level wastes discharged to cribs. Cooling water from the PFP was discharged to U Pond. Solid wastes containing plutonium were segregated and packaged for storage in special earth-covered trenches.

After separation processing ended at B Plant, the facility was reconfigured. From 1967 until 1976, the reconfigured facility was used to extract strontium from PUREX acid waste and sludges, and cesium from a variety of neutralized supernatants taken from the tanks. The strontium and cesium were concentrated into solid salt materials, melted, and encapsulated. Canisters of encapsulated strontium and cesium are stored today in the Waste Encapsulation Storage Facility.

Evaporators were used to remove excess water and concentrate the tank waste into salt cake and sludge, which remained in the tanks. The evaporated and condensed water contained radioactive tritium and was discharged to cribs.

Large volumes of cooling water and steam condensate discharges to ground have significantly affected the water table by causing the formation of groundwater mounds. Cooling water and steam condensate from B Plant went to B Pond (216-B-3) and those from T Plant went to T Pond (now beneath the 218-W3AE Burial Ground). Cooling water and steam condensates from the U Plant and Uranium Tri-Oxide Plant were routed to U Pond (216-U-10). Cooling water from the REDOX Plant was discharged to the S Ponds (216-S-16 and 216-S-17). Cooling water from the PUREX Plant was discharged to Gable Mountain Pond (216-A-25) and B Pond.

From 1944 to 1988, 526,000,000 gallons (2.0×10^9 L) of highly radioactive chemical processing waste was placed in single-shell and double-shell tanks at Hanford (Agnew et al. 1997). This amount included metal waste that was reprocessed in U Plant from 1952 until 1956, and PUREX sludge and supernatants reprocessed in B Plant from 1967 until 1976. Of this total, 63,200,000 gallons (2.4×10^8 L) were later removed and reprocessed, 129,600,000 gallons (4.9×10^8 L) were discharged to cribs and trenches, and 272,400,000 gallons (1.0×10^9 L) were removed by evaporation. Approximately 61,000,000 gallons (2.3×10^8 L) of waste remain in the tanks. These data reflect conditions on January 1, 1994 as reported by Agnew et al. (1997). The waste volume in tanks will change with time (e.g., waste volume will decline as evaporator campaigns are completed).

1.5 Low–Level Waste Disposal Facilities at Hanford

This section identifies the active or planned LLW disposal facilities and other sources of radioactive contamination under consideration in the first iteration of the Composite Analysis.

1.5.1 Active or Planned Disposal Facilities

The Composite Analysis provides a first estimate of the potential cumulative impacts to a hypothetical future member of the public from the active or planned LLW disposal facilities at Hanford. The Composite Analysis also includes other sources of radioactive material in the ground that may interact with plumes from the LLW disposal facilities. The four active or planned LLW disposal facilities at Hanford are:

- post-1988 solid waste burial ground in the 200 West Area
- post-1988 solid waste burial ground in the 200 East Area
- Environmental Restoration Disposal Facility
- disposal facility for immobilized low-activity wastes.

Each of these disposals is located on the central or 200 Area Plateau of the Hanford Site. Figure 1.3 shows the position of these LLW disposals on the 200 Area Plateau.

In accordance with DOE Order 5820.2A (DOE 1988b), performance assessments have been completed for the solid waste burial grounds located in 200 West Area and in 200 East Area. These burial grounds have received solid waste since DOE Order 5820.2A went into effect (September 26, 1988). Burial grounds in the 200 West and East areas were treated separately in performance assessments by Wood et al. (1995) and Wood et al. (1996). Under the CERCLA program, a Remedial Investigation and Feasibility Study was completed for the ERDF (DOE 1994b). The DOE-RL plans to submit a performance assessment[a] for the immobilized low-activity waste (ILAW) from Hanford tanks to U.S. Department of Energy Headquarters (DOE-HQ) in Spring 1998. Pending review and approval by DOE-HQ, the ILAW will be disposed of in a combination of four existing vaults and new facilities that are now in the conceptual design stage.

1.5.2 Other Sources of Radioactive Contamination

As is apparent from the description of Hanford Site operations, other radioactive sources are present or will be placed on the 200 Area Plateau of the Hanford Site. These sources may create contaminant plumes in the unconfined aquifer at the same time and in the immediate vicinity of plumes generated by the four LLW disposal facility sources described above. These sources are the responsibility of the DOE and include the following list:

- 149 single-shell tanks arrayed in 12 tank farms (i.e., T, TX, TY, U, S, SX, B, BX, BY, C, A, and AX)
- 28 double-shell tanks arrayed in 6 tank farms (i.e., SY, AP, AN, AZ, AY, and AW)
- past-practice (pre-1988) solid waste burial grounds
- past-practice (pre-1988) liquid discharges to cribs, ditches, French drains, trenches, and ponds
- graphite cores from 9 surplus production reactors

(a) Mann, F. M., R. P., Puigh II, C. R. Eiholzer, Y. Chen, N. W. Kline, A. H. Lu, B. P. McGrail, and P. D. Rittman. Publication planned for March 1998. *Hanford Immobilized Low-Activity Tank Waste Performance Assessment*. DOE/RL-97-69, Rev. 0. U.S. Department of Energy, Richland Operations Office, Richland, Washington.

- canyon buildings and related structures (e.g., B-Plant, PUREX, T-Plant, U-Plant, REDOX, Z-Plant [PFP], and the PUREX tunnels).

In addition, a commercial low-level radioactive waste disposal facility operated by US Ecology, Inc. is located immediately southwest of the 200 East Area, and was included in this analysis because of its proximity to DOE operations on the plateau. The treatment of each of these facilities was addressed in the analysis.

1.6 Operation of Low-Level Waste Disposal Facilities

This section provides a brief description of the facilities and their past, present, and expected future operations.

1.6.1 Active or Planned Disposal Facilities

Low-level waste has been disposed in the 200 West and 200 East solid waste burial grounds since nuclear materials production and processing began at Hanford. The initial generators of the majority of disposed waste were the chemical separations plants in each area: T-Plant, U-Plant, REDOX, and PFP, and tank farm operations in the 200 West Area; and PUREX, B-Plant, and tank-farm operations in 200 East Area. Disposals to the 200 West Area LLW facility support both onsite and offsite generators. The U.S. Navy is the only offsite generator contributing to waste disposal in the 200 East Area.

Solid waste disposals have occurred for several decades and as one burial ground filled up, another burial ground was opened. The current method of disposal for LLW is to place waste in an unlined trench about 6 to 7 m deep and of variable length up to about 500 m. Slopes of trenches are angled at about 45 degrees. Waste packages are stacked to within about 2.5 m of the surface, and soil is placed over the packages to grade. Some surfaces have been vegetated with grasses to stabilize the cover. In the future, efforts may be made to stabilize the waste in situ to prevent subsidence and to reduce recharge through the waste deposit.

Active burial grounds are defined as those that have received waste since September 26, 1988. Active disposal trenches are found in burial grounds 218-W-3A, 218-W-3AE, 218-W-4C, and 218-W-5 in 200 West Area; and 218-E-10 and 218-E-12B in 200 East Area. Since September 26, 1988, when DOE Order 5820.2A went into effect, 23 trenches have been open and receiving waste in the 200 West Area burial grounds, and 6 trenches have been open and receiving waste in the 200 East Area burial grounds. One additional trench in 218-E-12B, Trench 94, is dedicated to the disposal of defueled ship reactor compartments generated by the U.S. Navy. The performance assessments for the active 200 West and active 200 East solid waste burial grounds stipulate an expected 30 years of operation from the September 1988 start date.

In the summer of 1996, disposal of wastes generated during excavation and remediation of CERCLA past-practice sites on Hanford began. These wastes are disposed in the ERDF trench. This trench is a

belowgrade excavation that is lined to collect leachate. The excavated material is mounded abovegrade to create a trench of greater disposal volume or capacity. When filled with remediation waste, the trench will be closed with a protective surface barrier. Only remediation wastes originating at Hanford will be disposed in the ERDF. The waste is expected to consist of dangerous and hazardous waste, polychlorinated biphenyl (PCB) and asbestos waste, low-level radioactive waste, and low-level mixed waste containing both dangerous and radioactive waste. The ERDF trench is being developed in stages. Currently it consists of two disposal cells, and approval is now being sought for additional cells. Based on need, it is anticipated the ERDF will be expanded to receive all remediation wastes from Hanford's CERCLA past-practice sites.

Over the last 50 years, radioactive and mixed waste from the production of special nuclear materials has been stored primarily in single- and double-shell tanks in the 200 Areas. Under the Tank Waste Remediation System (TWRS) program, the DOE is proceeding with plans to permanently immobilize and dispose of the low-level portion of this waste onsite in near-surface disposal facilities as outlined in the record of decision (ROD 1997). Wastes will be retrieved from the tanks and pretreated to separate the low-level fraction from other tank waste. The low-level fraction will then be immobilized. Over 200,000 m^3 of LLW will be disposed under this program. An initial or interim performance assessment providing initial insight and guidance to the design of disposal facilities has been prepared for this waste form (Mann et al. 1997). A performance assessment is being submitted to DOE-HQ in the spring of 1998 to seek approval for the construction of disposal facilities and the disposal of waste. It is now anticipated the first of the ILAW will be disposed in four existing disposal vaults with the remaining waste disposed in new disposal facilities. Disposals are forecast to begin in 2002. Authorization to close the disposal facilities is expected around 2030.

1.6.2 Other Sources of Radioactive Contamination

In addition to the disposal of ILAW, releases to the environment originating from the single-shell tanks must also be considered. Sixty-seven single-shell tanks are known or assumed to have leaked. The Hanford Federal Facility Agreement and Consent Order (also known as the Tri-Party Agreement; Ecology, EPA, and DOE 1989) calls for approximately 99% of the waste volume in each of the 149 tanks to be removed. At present, sluicing is the method of choice for the removal of these wastes. It is believed that some contaminated liquid could be lost from each single-shell tank during recovery operations. Finally, each of the single- and double-shell tanks will contain some residual after wastes are recovered, separated, and solidified. These residuals will also release radioactive contamination to the surrounding environment in the future. The end-date milestone (Tri-Party Agreement Milestone M-45) for tank waste retrieval is September 2018.

Shallow-land burial of solid waste has occurred at Hanford since the mid-1940s. Burial grounds closed prior to September 26, 1988 are considered among the other sources of radioactive contamination. Prior to 1970, no distinction was made between transuranic (TRU) waste and LLW. In 1970, the Atomic Energy Commission required that TRU waste be retrievably stored. In the early 1980s, low-level liquid organic waste was segregated from LLW and placed in retrievable storage underground. Low-level

waste was further categorized in 1987 when mixed waste (i.e., waste containing both radioactive and hazardous chemicals) disposal in unlined trenches was discontinued. Contact-handled mixed waste is currently stored in aboveground buildings in the Central Waste Complex. Post-1988 LLW in burial grounds exhibits much lower inventories compared to the inventories of pre-1988 burial grounds. The pre-1988 solid waste burial grounds are designated past-practice units, and their remediation, final closure, and end state will be negotiated through the CERCLA process.

Since initial processing of irradiated fuels began in 1944, liquid wastes containing radionuclides have been discharged to the subsurface. These large liquid discharges have resulted in water table rises of approximately 24.4 m (80 ft) in the 200 West Area and approximately 9.1 m (30 ft) around the ponds near the 200 East Area (Law et al. 1996). In the past decade this practice has nearly ended; liquid waste discharges continue at only a few sites (e.g., the 200 Area Treated Effluent Disposal Facility [TEDF], the State-Approved Land Disposal Site [SALDS], and the 400 Area discharge ponds). This reduction in liquid disposal will result in the Hanford Site groundwater levels eventually reaching pre-Hanford levels. This will have a significant effect on the routing and movement of contaminants in the aquifer, especially at locations on the Hanford Site where the permeability of the Hanford formation currently dominates the total transmissivity of the system. Past discharges occurred to subsurface facilities including cribs, trenches, French drains, and reverse wells. Large volumes of cooling water and steam condensate generated by chemical separations facilities and evaporators were discharged to surface ponds and ditches. Some of the more significant liquid discharges to the subsurface were the intentional discharge of approximately 120 million gallons (4.5×10^8 L) of tank waste in various forms, e.g., first-cycle supernatant, second-cycle supernatant, and scavenged uranium recovery wastes. These sites are designated past-practice units and their remediation, final closure, and end states will be addressed through the CERCLA process.

Nine graphite core production reactors were operated at the Hanford Site between 1944 and 1987. Based on the environmental impact statement (EIS) for the eight surplus reactors (DOE 1989), a record of decision (ROD) was issued to follow a safe storage period with one-piece removal of the reactors to the plateau (ROD 1993). Safe storage at their current location along the Columbia River in the 100 Areas would occur for less than 75 years. Then, each reactor block would be transported intact on a tractor-transporter, from its present location to a 200 West Area burial ground for disposal. Since the EIS and ROD were issued, the B Reactor has been declared a national historic monument. Accordingly, it is possible it will be left at its current location along the Columbia River. This reduces the number of reactors affected by the ROD to seven. The N Reactor was not included in the surplus reactor EIS, and it is probable that it will be removed to the 200 West Area burial ground. Thus, eight reactors are assumed disposed on the 200 Area Plateau in this analysis.

Facilities in which the chemical separations were conducted are long, monolithic, concrete structures. These are known as the canyon buildings and are identified as the 221-B or B Plant, 221-T or T Plant, and other facilities. There are also related nearby structures used in additional process steps, (e.g., the 224-B and 224-T buildings), and storage facilities, (e.g., the two subsurface tunnels at PUREX). Two canyon buildings are in 200 East Area: B-Plant and PUREX. Four canyon buildings are in

200 West Area: T Plant, U Plant, REDOX, and Z Plant (PFP). In general, these structures contain inventories of mixed fission products and mixed activation products; however, they are in fixed or immobile settings inside metal vessels and piping and contained inside monolithic concrete cells. The end state of these structures and associated facilities is being defined through negotiations with regulators; however, the current baseline assumes canyon facilities will be demolished to the cover block grade with the remaining structure covered with a surface barrier.

The commercial LLW disposal facility opened in 1965 on 100 acres located southwest of the 200 East Area. The LLW that is packaged and shipped for disposal at the facility comes from medical practices, scientific research, industrial processes, and nuclear power plants. Prior to 1993, LLW came from throughout the United States to this site; but today LLW comes only from Washington, Alaska, Hawaii, Idaho, Montana, Oregon, Utah, Wyoming, Colorado, Nevada, and New Mexico. Naturally occurring radioactive materials can still come from all 50 states. The US Ecology Site is regulated by the Washington State Departments of Health and Ecology and is expected to close by 2063.

1.7 Waste Management Area Boundary

With regard to offsite exposure to a hypothetical member of the public, the current Hanford Site boundary of greatest interest is the Columbia River. However, the boundary of interest for the Composite Analysis is the future boundary. In 1992 a working group comprising representatives from governmental entities (federal, tribal, state, and local) and constituencies (labor, environment, agricultural, economic development, cities, and public interest groups) vitally interested in possible future uses of Hanford lands and cleanup efforts was formed. Included in the published report of the Hanford Future Site Uses Working Group (HFSUWG 1992) is the concept of an "exclusive" waste management area (Figure 1.4). This area is defined by the squared-off boundaries of the current 200 Areas expanded to include: 1) the land to the east of the 200 East Area (where TWRS privatization facilities are planned), and 2) the land to the south including the commercial LLW disposal site.

Surrounding the exclusive waste management area is a temporary buffer zone composed of the rest of the Central Plateau including the 200 North Area extending north to the base of Gable Butte. The cleanup target for the exclusive waste management area is to reduce risk outside that area and to minimize the size of the buffer zone. The ultimate cleanup target for the buffer zone is to prepare the land for unrestricted use. Thus, in future analyses, the size of the buffer zone may shrink.

However, it is the policy of DOE (1996h) that it "will control and maintain LLW disposal facilities until the disposal facilities can be released." The requirements for release of DOE property are provided in DOE Order 5400.5 (DOE 1993b). The DOE has acknowledged that many LLW disposal facilities may never be suitable for unconditional release to the public. For example, deed restrictions on the future use of the groundwater resource may be necessary.

The collective locations of waste sites within the 200 Area Plateau are illustrated in Figure 1.3a, b, and c. The position of the various waste disposals and other sources are shown. Some liberty was taken

to locate the graphite core disposal in the western portion of the 200 West Area. A specific location has not been chosen yet, so a logical but not precise location has been selected for the purpose of this analysis. Because the land area associated with many of the liquid discharge sites is quite small, they are represented by uniform red dots in the figure. The disposal sites and other contamination sources cover a significant portion of the exclusive waste management area. While both the ERDF trench and commercial low-level radioactive waste disposal facility lie outside the 200 West and 200 East Areas, they lie within the exclusive waste management area.

1.8 Performance Objectives of the Composite Analysis

This analysis will estimate the potential cumulative impact to a hypothetical member of the future public from the active and planned LLW disposals and the other sources of radioactive material to remain at Hanford after Site closure. DOE Order 5400.5 (and anticipated 10 CFR 834) sets the DOE primary dose limit[a] of 100 mrem to members of the public in a year and as low as reasonably achievable (ALARA). This is the maximum allowable projected total dose from all pathways to the future member of the public. An options analysis and ALARA assessment are to be prepared if the projected dose exceeds the 100 mrem in a year limit or a significant fraction of the limit (defined to be 30 mrem in a year). The options analysis and ALARA assessment are to consider those actions that could be taken to reduce the calculated dose and their costs. They are to focus on those sources making a significant contribution to dose. If the projected dose is below the significant fraction of the limit, a brief ALARA assessment should still be performed to determine whether or not a quantitative or semi-quantitative options analysis and ALARA assessment are warranted.

If the projected dose is above 100 mrem in a year, DOE uses the options analysis to identify alternatives that reduce projected future doses to tolerable levels, and selects one for implementation. Performing these calculations provides the DOE with information supporting a comprehensive approach to environmental management that will ensure that the 100 mrem in a year primary annual dose limit is not exceeded in the future and that potential doses are maintained at ALARA levels.

At Hanford, the approach adopted to achieve comprehensive environmental management involves a complex process of negotiated decisions among the DOE, the State of Washington Department of Ecology, and the U.S. Environmental Protection Agency. Even the selection of each alternate remedial action for further study needs to be a joint decision of the three parties. At this time, DOE is beginning to negotiate the cleanup of past-practice sites in the exclusive waste management area. Thus, there has been insufficient time to determine whether alternative remedies are necessary and to identify them through a negotiation process. Accordingly, the options analysis (if necessary) and ALARA assessment will be deferred to the second iteration of the Composite Analysis.

(a) All doses in the Composite Analysis (except where noted) are in units of mrem effective dose equivalent (EDE) in a year.

DOE guidance for the composite analysis (DOE 1996b) requires the analysis present results for a time period of at least 1000 years. Subsequent guidance for performance assessments (DOE 1996f) that are related to composite analyses, requires the analysis cover a period of 1000 years following closure of a disposal facility. For this analysis, the time period is assumed to begin at the time of Hanford Site closure (assumed to occur in 2050). The latter guidance also notes that analyses beyond 1000 years may be appropriate in the sensitivity/uncertainty analyses. This analysis has considered a time period of 1500 years beginning in 1944 and including the 1000 years following site closure.

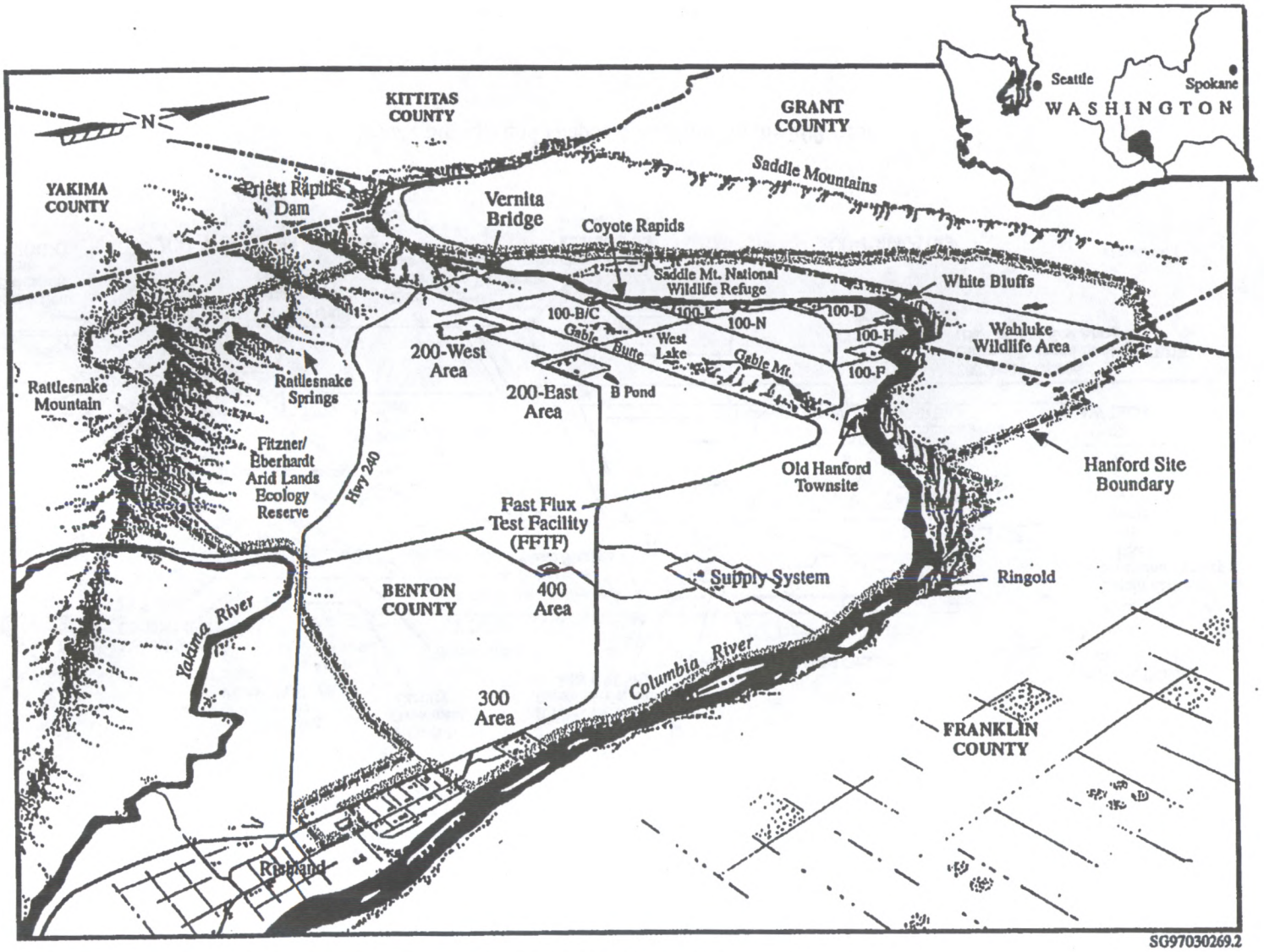


Figure 1.1. The Hanford Site and Surrounding Area

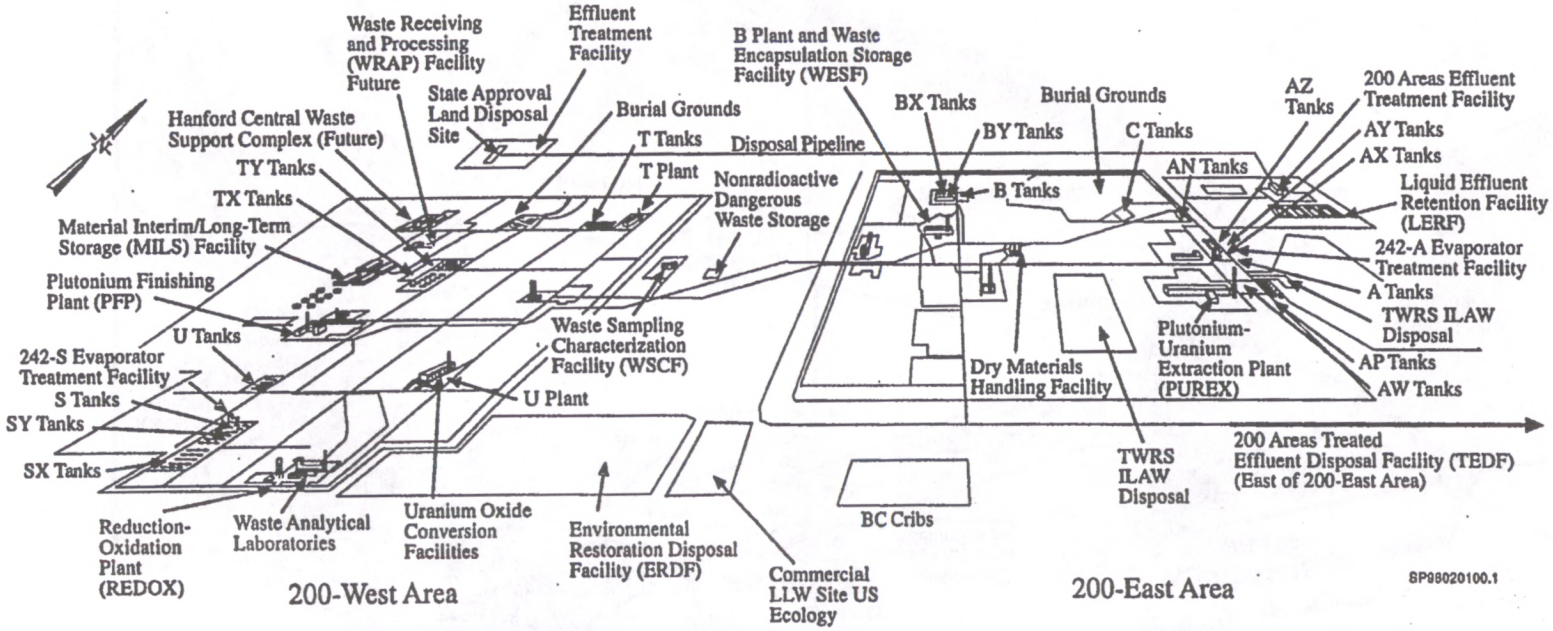


Figure 1.2. Waste Storage and Disposal Facilities in the 200 Area

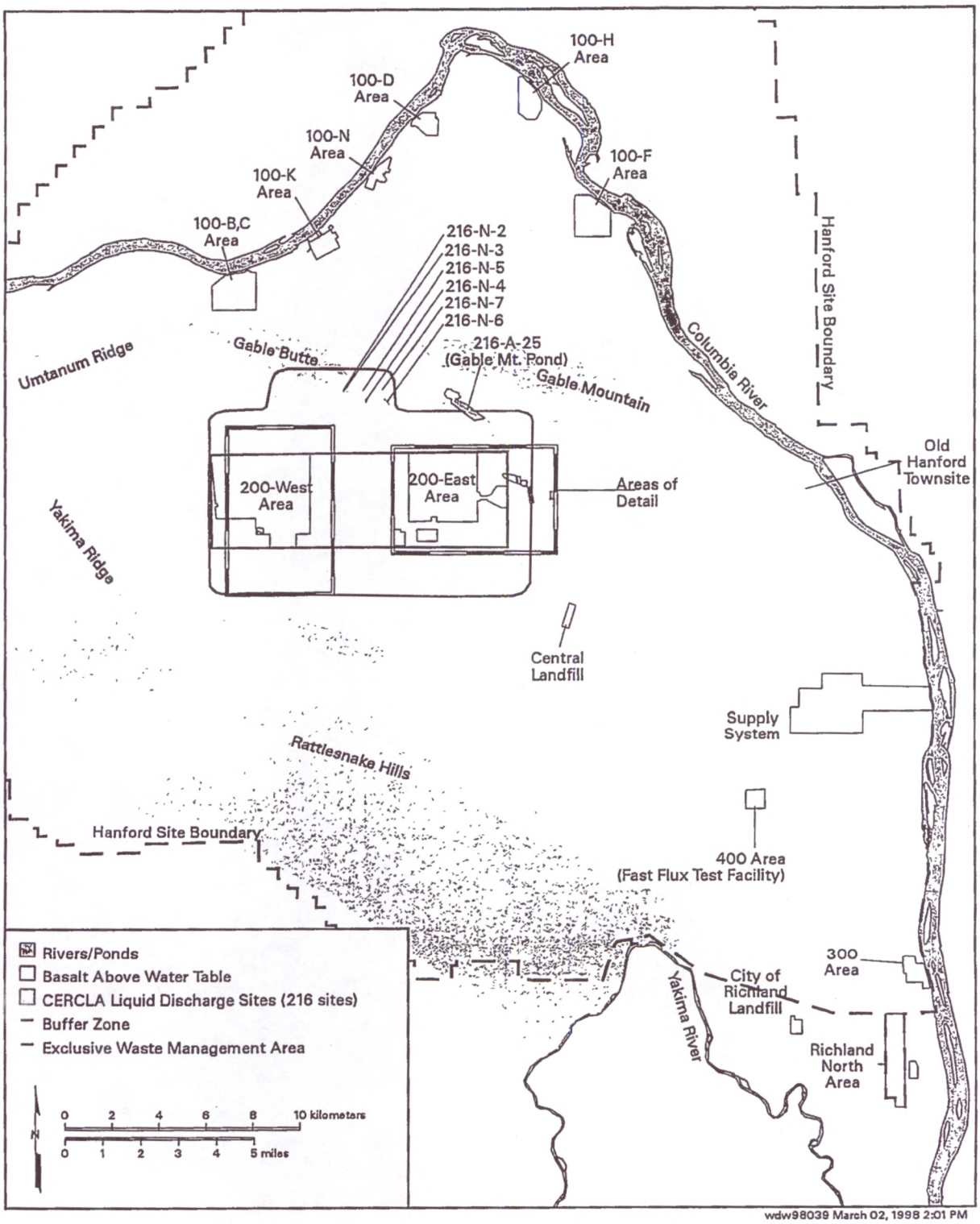


Figure 1.3a. The Exclusive Waste Management Area and Buffer Zone of the 200 Area Plateau at the Hanford Site.

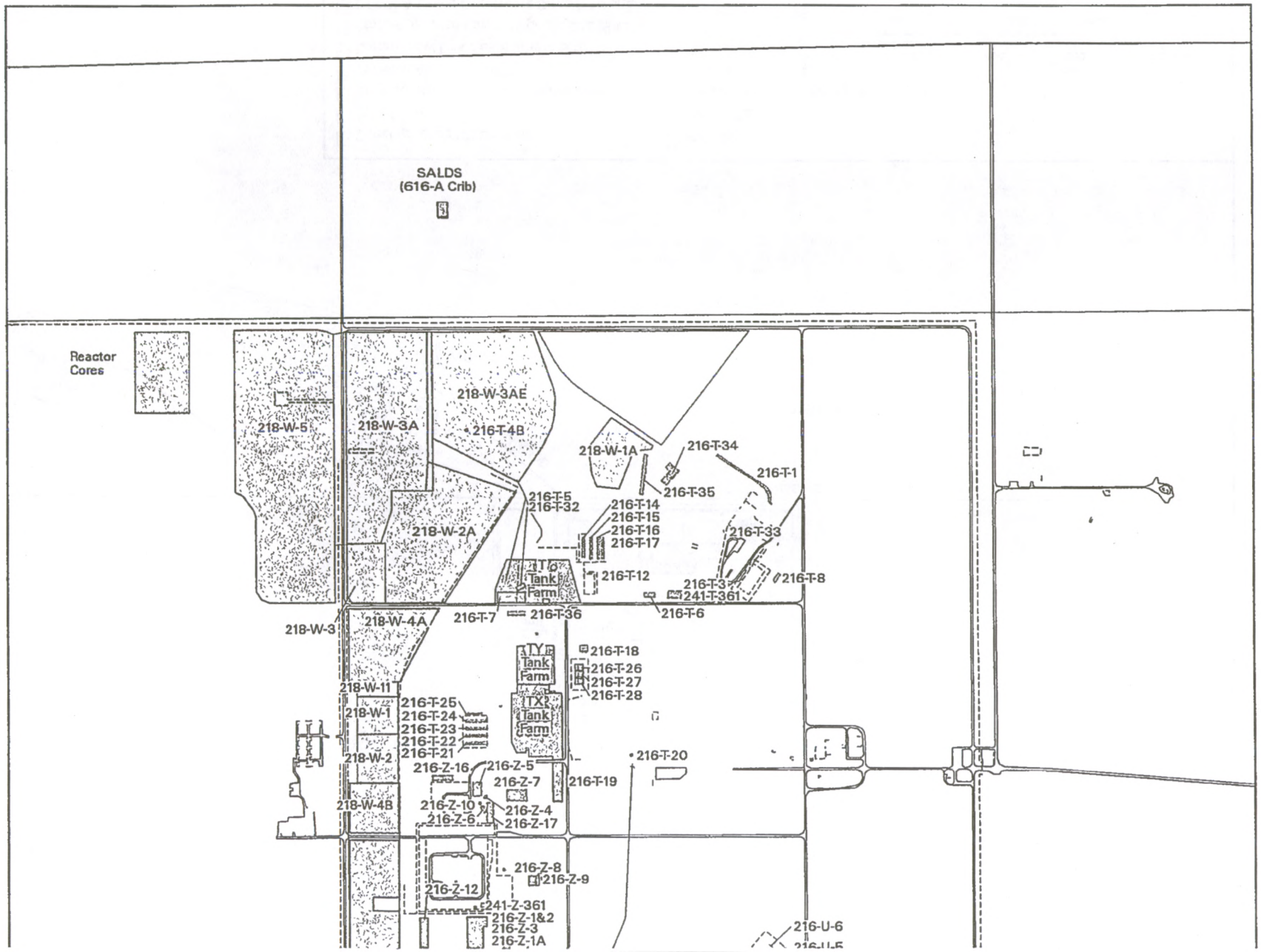
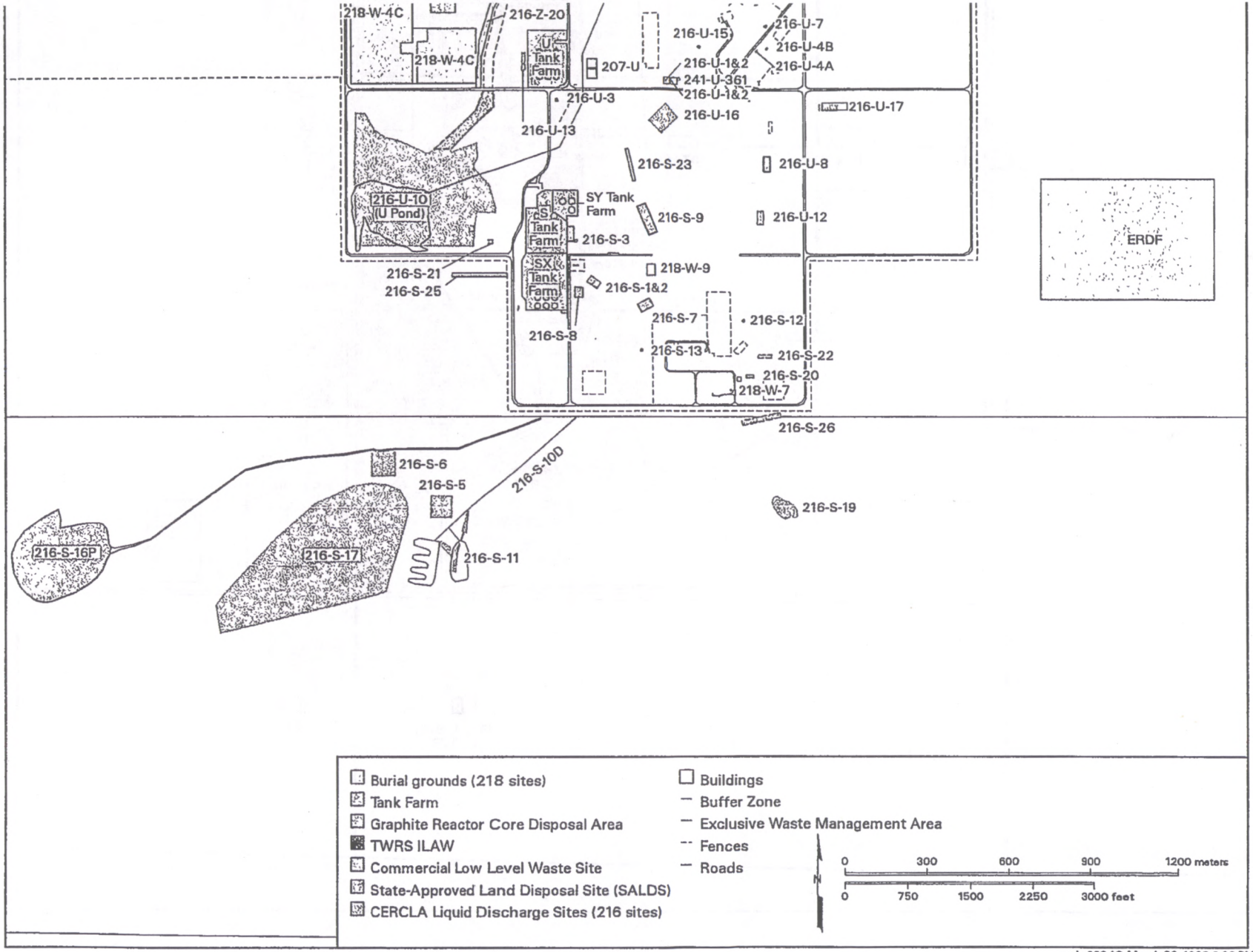


Figure 1.3b. Waste Sites of the 200 West Area and Imm



ate Vicinity Considered in the Composite Analysis.

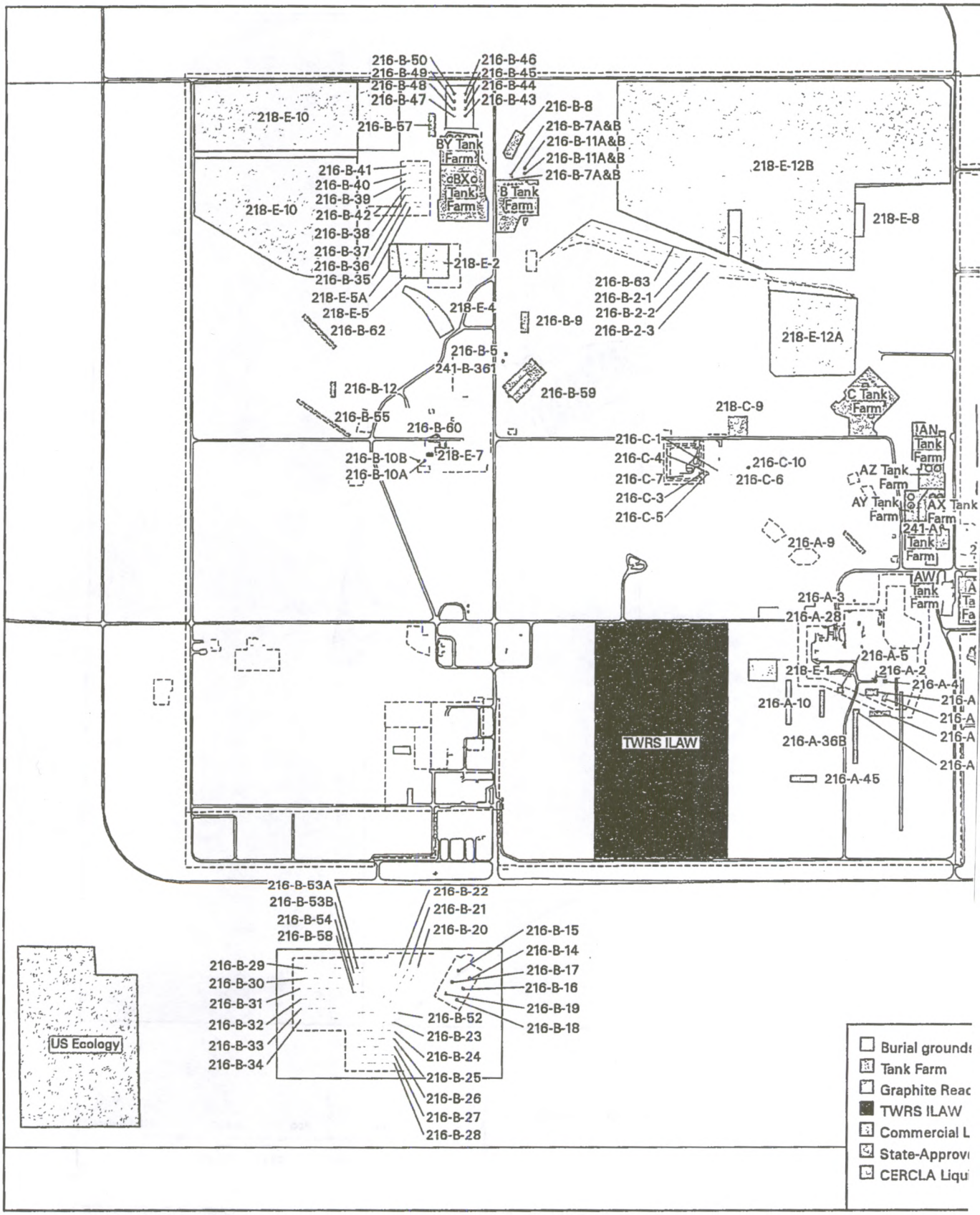
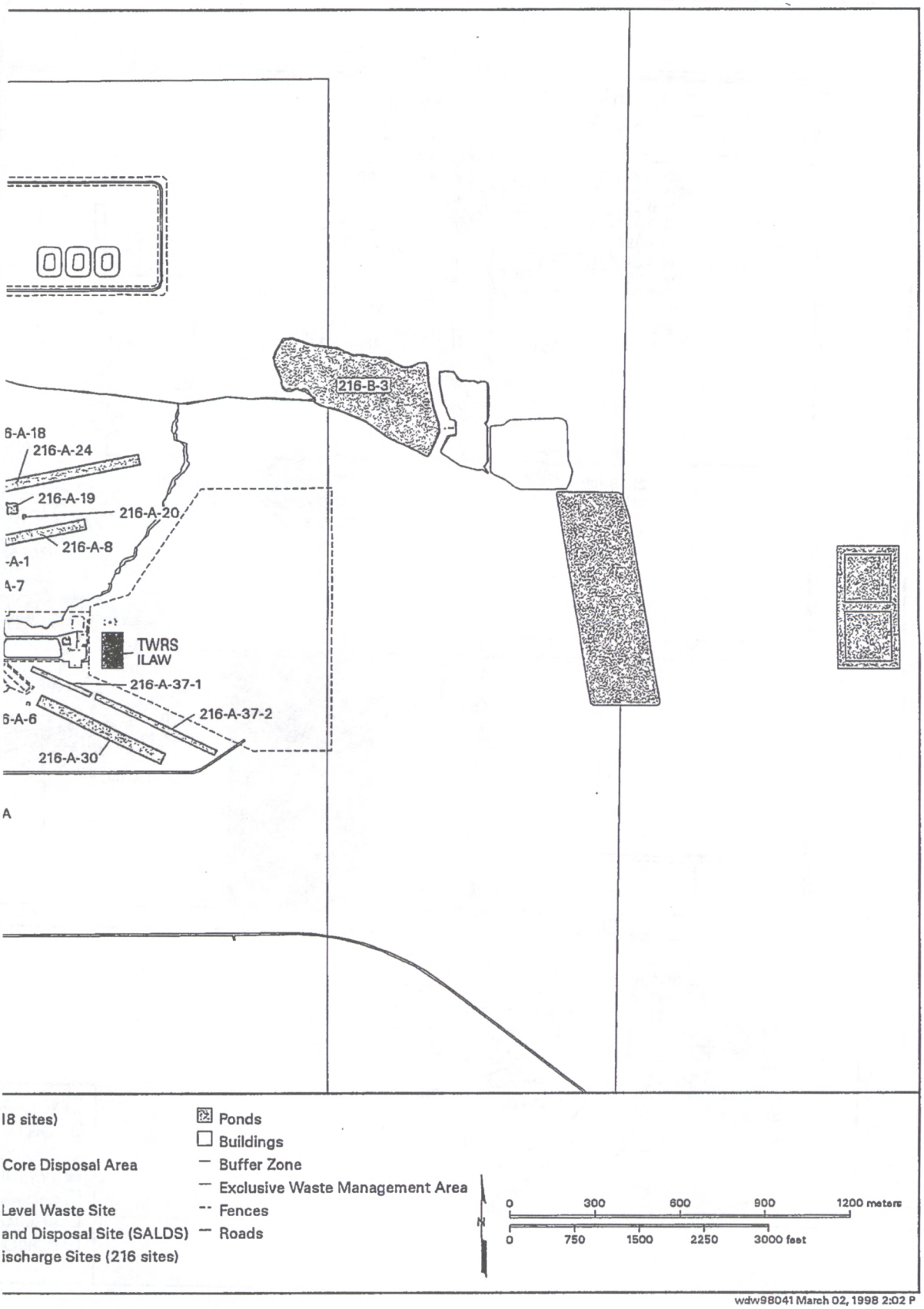


Figure 1.3c. Waste Sites of the 200 East Area and Im



liate Vicinity Considered in the Composite Analysis.

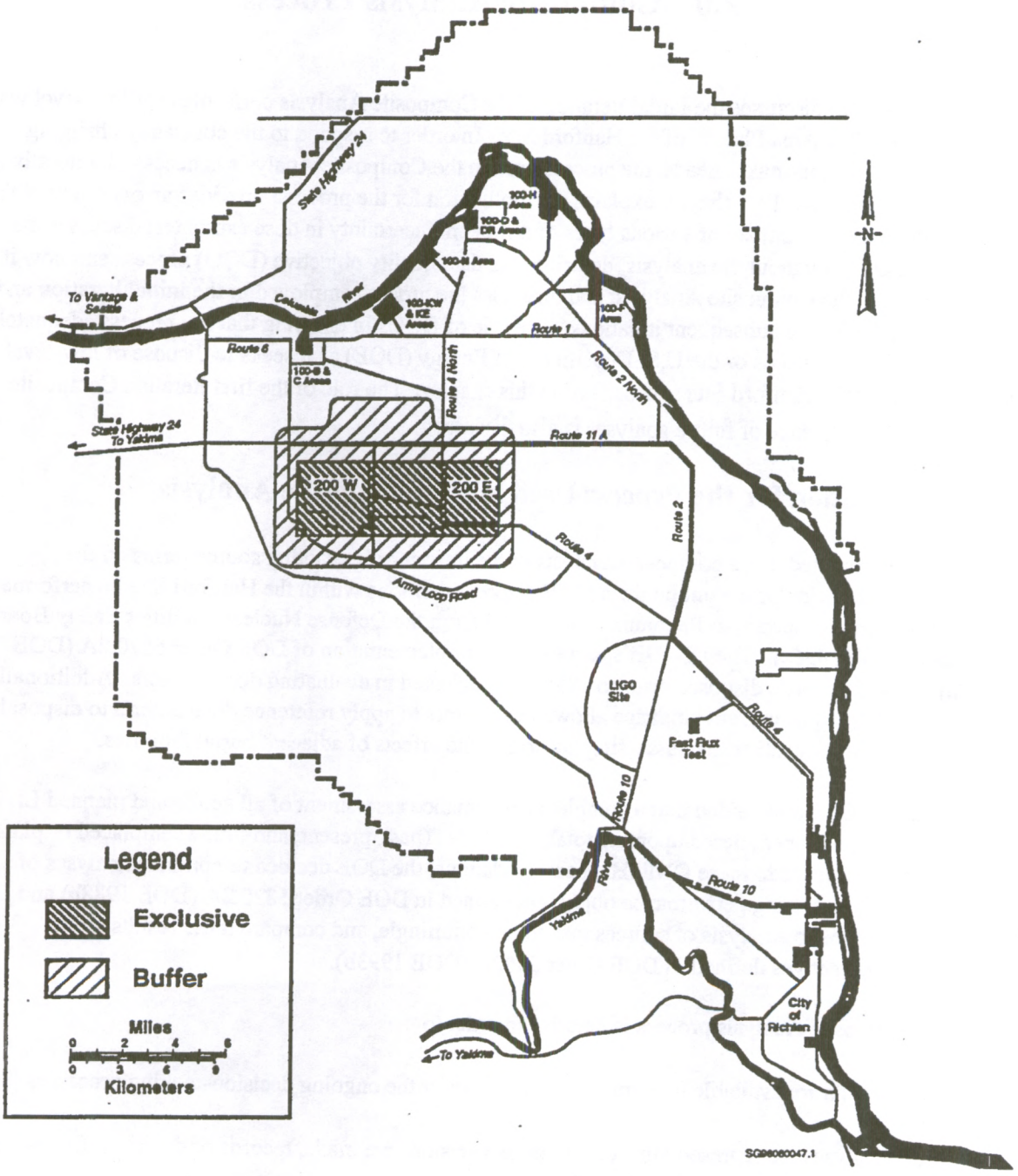


Figure 1.4. 200 Area Plateau with Exclusive Waste Management Area and Buffer Zone

2.0 Composite Analysis Process

This document discusses the initial iteration of the Composite Analysis performed for low-level waste disposed in the 200 Area Plateau of the Hanford Site. In order to respond to the constantly changing technical and decision-maker needs, the process used in the Composite Analysis is necessarily iterative, adaptive, and flexible. This chapter explains the motivation for the process; provides an overview of the process; describes the impact of various types of errors on uncertainty in dose estimates; discusses the decisions made throughout the analysis; describes the data quality objective (DQO) process and how it was adapted for the Composite Analysis; and discusses the process employed in the initial iteration and the process planned for subsequent iterations. The role of DQOs in ensuring that the process adequately reflects the decision needs of the U.S. Department of Energy (DOE) as it seeks to dispose of low-level waste (LLW) at the Hanford Site is described in this chapter. The role of the first iteration Composite Analysis in the sequence of future analyses is also discussed.

2.1 Motivation for the Process Used in the Composite Analysis

The DOE directed that a composite analysis of the impact of interacting source terms on the radiological protection of the public from LLW disposal facilities within the Hanford Site be performed. This action was in response to Recommendation 94-2 from the Defense Nuclear Facilities Safety Board (DNFSB) (DOE 1996e). The DNFSB's review of the implementation of DOE Order 5820.2A (DOE 1988b) found that waste disposed prior to 1988 was neglected in evaluating dose impacts. Additionally, the DNFSB found that current guidance allowed evaluators to apply reference dose criteria to disposal facilities individually rather than assessing the composite effects of adjacent burial facilities.

The DNFSB recommended that a complete performance assessment of all active and planned LLW burial sites be performed, based upon the total inventories (past, present, and future) emplaced or planned for burial. In response to these DNFSB recommendations, the DOE decided to continue analyses of individual facilities using performance objectives defined in DOE Order 5820.2A (DOE 1988b) and conduct a Composite Analysis of sources that could commingle, and compare these results to the performance objectives defined in DOE Order 5400.5 (DOE 1993b).

The Composite Analysis process is iterative in order to

- ensure results are available in a timely fashion to assist the ongoing decision-making processes
- ensure analyses are representative of the site as decisions are made, records of decision (RODs) are issued, and new data are gathered and interpreted
- optimize the scope and scheduling of analysis activities.

While the primary objective and purpose of the Composite Analysis are to provide a holistic view of waste disposal and composite dose impacts to the public, each iteration of the Composite Analysis will be

used to prioritize analysis activities in subsequent iterations. This adaptive analysis process is required, given the significant uncertainty involved in facility end states, inventories of critical contaminants, conceptual models (of contaminant release and transport), computational models (of related release, transport, and exposure), and the assumed future land uses.

A first iteration of the Composite Analysis has been completed. Consistent with the DOE directive for this initial iteration, it established a base case for comparison with the primary dose limit. The DOE directive, acknowledging the iterative process that would be required, stated that no new data should be collected for the first iteration.

The second and subsequent iterations of the Composite Analysis may affect the allocation and scheduling of resources for sitewide LLW management actions (e.g., site remediation) and waste disposal practices (e.g., barriers and waste packaging). Resources could be prioritized toward sites that will result in the greatest reduction in estimated composite dose. Schedules could be altered to favor those sites where delays might result in the greatest adverse impact.

In order for the Composite Analysis to support scheduling and prioritization of sitewide waste management or waste remediation actions, it must clearly articulate the tradeoffs between various objectives. These objectives include: minimizing the risk of underestimating the dose impacts; maximizing the time available before actions must be taken; maximizing the decision maker's confidence that the dose impact assessments are reliable; and minimizing costs. Clearly, these objectives involve tradeoffs. For instance, increasing confidence in dose predictions will result in greater analysis costs. Decision makers formulate the decisions to be made and incorporate the multiple objectives into the decision-making process by working through the seven steps of the DQO process.

The goal of the Composite Analysis is to reduce uncertainty only inasmuch as the reduction in uncertainty will directly affect the actions to be taken. It is not feasible to eliminate uncertainty. Reducing the uncertainty in such complex performance analyses can involve costly laboratory experiments, field experiments, and modeling analyses. The cost of reducing uncertainty is justified, if the reduction in uncertainty would likely alter a waste management or waste remediation decision. The cost of the analysis itself must also be compared to the expected loss from making an incorrect decision. If the analysis itself would cost more than the expected benefit of the analysis (i.e., the benefit to be gained from having improved information on which to base decisions), the analysis should not be performed.

Worst-case analyses are one way to limit the cost of one type of decision error: failure to take action when action is required. However, in a composite analysis, worst-case assumptions can only be applied in a limited manner. Attempting to use worst-case analyses independently for each site is problematic in a composite analysis because this approach neglects the impact of superposition of releases, generally is not able to prioritize actions, and violates sitewide mass conservation. The worst-case dose impact for a specific site is generally defined by the earliest and largest feasible release. However, the worst case for the composite dose from numerous sites is a function of the superposition of each site's plume. Defining the worst-case scenario for a composite analysis is significantly more difficult than for an individual site's performance assessment. Worst-case assumptions also tend to penalize sites with less information.

Therefore, prioritization is usually biased towards sites that lack information. Mass conservation is a useful constraint on sitewide inventories in composite analyses. Worst-case analyses for each individual site will violate mass conservation by systematically inflating inventories across the site.

The DQO process is a flexible and adaptive approach that attempts to match the type, quality, and quantity of data collected to the needs of the decision maker for confidence in decisions that will be based on that data. The DOE specifically directed that DQOs be employed in the Composite Analysis process. For a variety of reasons mentioned in Section 2.5, the standard DQO approach has not and can not be directly applied to the dose forecast problem of a composite analysis. Rather than try to "force fit" the DQO process, a slightly modified approach that involves incorporating the concepts of model uncertainty analysis and Value of Information (VOI) analysis is proposed. In the modified DQO process (described further in Section 2.5.2), decision makers use their assessment of the severity of consequences if model predictions are incorrect to justify the cost of any model improvements. In order to completely implement the modified DQO approach, a probabilistic modeling effort and subsequent cost/benefit and VOI analysis will be required.

2.2 Process Flow Diagram

Figure 2.1 illustrates the process used in the Composite Analysis. The process iterates until the decision makers have their stated level of required confidence in the dose estimate to support their decisions. The modified DQO process is used throughout the Composite Analysis to ensure that the cost of additional information gathering and model improvements are tied to the decisions to be made and the limits on decision error as specified by the decision makers. Decision makers set limits on the type of decision errors that they can accept, based on the actions identified in the decision rules and their assessment of the severity of the consequences that could result from making incorrect decisions based on model results. The seven activities shown in Figure 2.1 (rectangular boxes numbered 1 through 7) are described in detail below.

1. **Select models.** This step involves the selection of process models and uncertainty models. Examples of process models include release models, vadose zone transport models, groundwater transport models, atmospheric transport models, and exposure/dose models. In subsequent iterations of the Composite Analysis, multiple models may be employed for a single process to help address the issue of uncertainty in the process models. Uncertainty models attempt to define the distribution of errors in process model parameters and inputs, as well as to quantify the uncertainty in the predictions from analyses. In the first iteration of the Composite Analysis, process models were limited to readily available models. These models are discussed in detail in Section 4.1. Since the first iteration provided only a deterministic baseline, uncertainty models were not employed. In all subsequent iterations, uncertainty models will be required. A modified DQO process will be used to direct the selection of both process models and uncertainty models.

2. **Select the type and number of scenarios.** This step involves selecting the type and number of scenarios analyzed. Each scenario is described by the manner in which the process models and uncertainty models are combined. Generally, each scenario results in a feasible realization of the estimated composite dose. In some cases, a realization can be eliminated from subsequent

consideration by comparing its estimate of the current state with observed conditions. In the first iteration of the Composite Analysis, only one scenario for the transport models was considered whereas multiple exposure scenarios were considered. In subsequent iterations, multiple scenarios for the process models will be included. Since additional scenarios will result in additional cost and time to complete that specific iteration, the number of scenarios will be defined through the DQO process.

3. **Select sites and radionuclides.** This step allows the analysis to limit the number of sites and radionuclides for which detailed analyses are to be performed. Many sites and radionuclides have only a negligible impact on dose, even under worst-case conditions. Lowering the allowable dose to compensate for the combined worst-case dose from these sites and radionuclides can significantly reduce the analysis effort required. Analysis effort can then be focused on the sites and radionuclides most likely to significantly affect the composite dose estimate. In the first iteration of the Composite Analysis, all sites with reported inventories were analyzed. However, only carbon-14, chlorine-36, selenium-79, technetium-99, iodine-129, and uranium isotopes and all their daughters (which were expected to contribute the most to the composite dose) were completely analyzed. In subsequent iterations, the DQO process will be employed to define the amount the allowable dose will be lowered to compensate for the worst-case dose estimates from less significant sites and radionuclides.

4. **Conduct performance analysis.** This step requires the execution of the process and uncertainty models using the selected scenarios, sites, and radionuclides to assess the performance of the composite waste disposal facilities with respect to the applicable performance objectives. This is generally the most significant element of cost and time in the Composite Analysis process.

5. **Perform screening ALARA assessment.** If composite dose estimates are less than 30 mrem in a year, only a screening ALARA (as low as reasonably achievable) assessment is required. In the first iteration of the Composite Analysis, the dose estimate was less than 30 mrem in a year so a screening ALARA was performed. If subsequent analyses continue to result in a dose estimate less than 30 mrem in a year, screening ALARA assessments will be repeated. Since the dose estimates will be probabilistic estimates, the DQO process will need to define the specific manner in which the 30 mrem in a year standard is defined. For example, the standard could be compared to the mean estimated dose or to the upper 95 percentile value.

6. **Perform options analysis.** If the composite dose estimate exceeds 30 mrem in a year, an options analysis is required. Since the first iteration of the Composite Analysis resulted in doses less than 30 mrem in a year, an options analysis was not performed. If subsequent analyses result in a dose estimate greater than 30 mrem in a year, an options analysis will be performed.

7. **Perform ALARA assessment.** If the composite dose estimate exceeds 30 mrem in a year, an ALARA assessment is required. Since the first iteration of the Composite Analysis resulted in doses less than 30 mrem in a year, a complete ALARA assessment was not performed. If subsequent analyses result in a dose estimate greater than 30 mrem in a year, a complete ALARA assessment will be performed.

The purpose of the full options analysis and ALARA assessment would be to pose and analyze alternate actions. These analyses need to be thorough in order to potentially support a DOE decision to change course.

The process proceeds iteratively. Each iteration helps define the optimal steps to improve the confidence in the subsequent analysis. The process terminates when adequate confidence exists in the model predictions and decisions are made to take action based on the model predictions.

2.3 The Impact of Various Types of Errors on Uncertainty in Dose

Because of its magnitude, uncertainty in environmental systems cannot be neglected in the decision process. Uncertainty in dose predictions from the models can be attributed to many sources of errors. These errors propagate and compound throughout a composite analysis. The four main types of errors, measurement errors, sampling errors, forecast errors, and model errors, are described below.

- Measurement errors are errors that result from inaccuracies in analytical measurements. The precision and accuracy of analytic measurement equipment and procedures are finite. Measurements of the identical sample will not always yield the same exact value. Measurement errors are readily dealt with using well-established statistical methods.

- Sampling errors are those errors that result from the spatial and/or temporal variability of the items being sampled. For instance, numerous samples are required to develop an understanding of the three-dimensional shape of a groundwater plume. Additionally, samples must be taken over time to characterize the migration and evolution of a plume. Geostatistical methods for estimating spatially variable fields and for estimating the errors in these estimates are currently available.

- Forecast errors are those errors that result from the limited ability to predict future conditions. Future climate, future land use, and exposure scenarios are all examples of processes subject to significant forecast errors. Some forecast errors, climate for instance, can be estimated by assuming the historically observed variability will persist into the future. However, other processes, such as land use and exposure scenarios, have no historical analogs.

- Model errors are those errors that result from the conceptual or numerical formulation of the process models. While many numerical method errors can be readily corrected with improved numerical algorithms, conceptual model errors often require expensive laboratory experiments and/or field measurements in order to validate the model's process formulation. Model errors cannot be dealt with using the statistical and geostatistical methods applied to measurement and sampling errors.

Measurement errors and sampling errors are the easiest to quantify. Unfortunately, in a composite analysis, forecast errors and conceptual model errors contribute significantly more to the overall uncertainty than numerical formulation, sampling or measurement errors. Model forecast errors and conceptual errors are also the most difficult to quantify. Monte Carlo methods have been shown to be useful in evaluating model uncertainty (IAEA 1989; NCRP 1996). Stakeholder acceptance, as well as scientific issues, must be considered in selecting models and future scenarios.

Challenges to an accepted conceptual model can even require that a whole new set of measurement and sampling methods be developed to measure critical factors that distinguish among alternate conceptualizations. Deciding which of the feasible process models is valid may require expensive experiments. However, it is not essential to determine which exact process model is valid, if the other feasible process models would not result in a different waste management action or waste site remediation decision.

Each iteration of the Composite Analysis will close confidence gaps in the composite dose estimate. This will require the development of a specific set of analytical procedures to quantify performance and uncertainty.

2.4 Decisions Made Throughout the Composite Analysis

Decisions made throughout the Composite Analysis process will be directed by the decision makers' answers to three questions. These three questions are:

- Will the dose be greater than a prescribed level?
- How should resources for model development and improvement be allocated?
- What set of possible actions are to be considered in the options analysis, if required?

The primary decisions in the Composite Analysis are associated with the first question; the second and third questions support these primary decisions. Whether the actions identified in the third question are taken depends on the outcome of the answer to the first question and the confidence in this answer. The level of confidence in the predicted dose shapes the second decision on the allocation of resources committed to model development and improvement. The three decisions are tied together, and the methods and techniques for dealing with them show a strong interdependency. However, the focus must remain on the primary decision. It is a decision about some unknown, future state.

Making decisions and taking actions based on model predictions begs the question of "How good must a model be in order to make good decisions?" There is a need to balance the desire for more accurate model predictions against the costs of developing and testing new or revised models. Implicit in the need for better models is the potential cost or loss function associated with making either of the following two types of errors based on inaccurate model predictions:

- Taking actions that are not required (e.g., models predict doses greater than the standard when in fact true doses are less than the standard)

- Not taking actions that are required (e.g., models predict doses less than the standard when in fact true doses are greater than the standard).

Before extensive resources are expended, answers are needed to the three questions listed below.

- How reliable are the predicted values from the model?
- What level of effort or expenditure of resources is required to get "better" model results?
- How much "better" does the model need to be?

The answer to the first question depends on the quality of the input data for the predictive models and the reliability of the models themselves. This has not been explicitly addressed in the first iteration of the Composite Analysis. The underlying assumptions to the second question are listed below.

- Better model results can be achieved with greater effort.

- It is possible to quantify the relationship between increased effort and increased probability that the model predictions are correct.

These assumptions have not been explicitly addressed in the first iteration of the Composite Analysis. The answer to the third question requires a DQO approach where all parties that have a stake in the accuracy of model prediction supply input to the decision. The underlying assumption for the third question is that it is possible to quantify the likelihood of making the correct decision when decisions are based on model output. In order to work within this final assumption, it is necessary to take a flexible and somewhat more qualitative approach to the DQO process than the U.S. Environmental Protection Agency (EPA)-sanctioned DQO approach developed for the standard environmental sample collection problem (EPA 1994).

2.5 Data Quality Objectives

The DQO process was applied to the Composite Analysis in a different manner than is generally used. The reasons for this are explained in the next two sections. However, the philosophy of the DQO process was incorporated in the first Composite Analysis iteration, and will be incorporated in future iterations. This philosophy is: before extensive effort is expended on collecting data and, in the case of the Composite Analysis, making improvements to models, it is necessary to identify the specific decisions to be made based on the information and the level of confidence in model results required in those decisions. The intended use of data and model predictions, and consequences associated with decision error, drive the type and quality of information needed. Future improvements to the Composite Analysis will be a function of perceived needs to improve the type and quality of information needed to make the necessary decisions.

2.5.1 The Standard DQO Approach

The DQO process was developed in response to the need for Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) and Resource Conservation and Recovery Act (RCRA) investigations to define the quantity and quality of characterization data required to make cleanup decisions. The DQO process normally involves the following seven steps:

1. Statement of the problem
2. Identification of the decision
3. Identification of inputs to the decision
4. Definition of the boundaries
5. Development of a decision rule
6. Specification of limits on decision errors
7. Optimization of data collection.

The EPA guidance (EPA 1994) provides more detail on the DQO process. For the standard DQO approach, the assumptions, theory, decision-error limits, and relative decision-error consequences combine into a closed-form solution where sample size formulas and equipment quality selection criteria are the outputs of the DQO process. In Step 7, the stakeholders make cost-benefit comparisons that reflect resource constraints and risk versus cost tradeoffs. The result is an "optimized" level of resources to commit to improved sample collection and analysis and hence, improved decision quality. The DQO process does not provide explicit guidance on how to make these tradeoffs, but implicit in Step 7 is the concept that the costs of decision error consequences will be matched against costs of increased sampling, with the assumption that increased sampling will lead to greater confidence in field characterization and fewer decision errors.

2.5.2 Modified DQO Process Applied to Model Predictions

Decisions based on model predictions during a future time for a hypothetical maximally exposed individual, rather than on sample data from a current "true" state, require modifications to the later steps of the standard DQO process. However, the objective of the DQO process remains the same: to balance the desire for more and better data (more reliable dose forecasts) against the cost of obtaining more and better data (more reliable dose forecasts). The decision on the amount of data (more reliable dose forecasts) needed is based on the amount of uncertainty that can be tolerated, which, in turn, is related to the consequences of making an incorrect decision.

For the standard CERCLA/RCRA DQO application, collecting a 100% sample (i.e., complete enumeration) of a current condition results in a 0% risk of making a decision error on the characterization of that condition, given accurate sample collection and analysis equipment. The metric for decision performance is percent of time a randomly selected sample gives an accurate assessment of the "true" condition if the sampling event is repeated over and over. For a model prediction problem with stochastic variables, there is no single future "true" condition against which to measure accuracy of a model forecast. The analogy of increasing sample sizes to achieve a 0% risk of making a decision error must be modified before it can be translated into the prediction problem. Increasing expenditure of resources for the modeling problem can decrease uncertainty, but there is no simple linear relationship between increased expenditure and increased confidence in model results. The literature on model uncertainty captures some of this complexity and refers to Type A and Type B uncertainty for the modeling problem (IAEA 1989).

Thus, the standard DQO approach cannot be applied to the dose forecast problem directly. Rather than try to "force fit" the components, it is better to make a slight paradigm shift by using the terminology

of model uncertainty/model sensitivity analysis coupled with a VOI approach (Morgan and Henrion 1990) to making risk/cost/benefit tradeoffs when trying to decide whether model results are good enough for decision making. The result is a modified approach to the DQO process; decision makers use their assessment of the severity of consequences if model predictions are incorrect and balance that with the cost of model confidence or uncertainty reduction. In order to do that, they need a way to assess uncertainty in model output similar to the way statisticians assess the uncertainty in sampling results.

2.5.2.1 Steps 1 through 4

The first four steps of the DQO process can be followed for the modified approach with very few exceptions from the standard approach. Much of the work that is done in the early steps of the DQO process remains the same for the sampling and the dose forecast problem. For the dose forecast problem, the outputs from Steps 1 through 4 translate directly into what is modeled, what parameter inputs are used, and what ranges are considered in selecting parameters and model formulations. The results of Steps 1 through 4 for the initial iteration of the Composite Analysis are discussed in Section 2.6.

2.5.2.2 Steps 5 through 7

Step 5 is usually derived from a statutory or regulatory requirement, along with what form the modeling output takes. In the composite analysis problem, it is the various dose limits established by the DOE (for radionuclides) and the EPA (for chemicals) as protective of human health and the environment that drive the decisions to be made. It is the scope of the modeling effort that drives how detailed the decision rules can be.

In Step 6, the definition and treatment of model prediction uncertainty becomes a critical component of applying the DQO process to the composite analysis problem. In Step 6, decision makers provide the desired or acceptable levels of decision errors they can accommodate based on their assessment of possible consequences of making decision errors. There is no guarantee that these levels can be achieved within budget and practical constraints; but the decision makers must begin the tradeoff process with a decision quality goal in mind.

In Step 7, decision makers are asked to "optimize" the design. This step usually requires the help of experienced statisticians and optimization experts who can design sampling strategies, refine models, and design metrics to measure and assess potential decision errors. For the model prediction problem, this means the decision makers, along with their technical experts, must decide if spending additional resources on improving model input or the model itself is of "value." If the decision makers use their risk-aversion/risk-taking preferences to guide the expenditure of resources, they have complied with the spirit of Step 7 of the DQO process. Expenditure of resources could refer to the resources spent on model improvements; it could also refer to the resources spent on making changes to the LLW projects if the Composite Analysis indicates dose limits will be exceeded. These resource allocations affect the primary decision of whether dose limits have been exceeded and there is sufficient confidence in the model to act. Progress on Steps 5 and 6 for this first iteration of the Composite Analysis is described in Section 2.6.

2.6 Initial Iteration of Composite Analysis

Because the DOE directed that the first iteration of the Composite Analysis use only available information, no field samples were collected for the first iteration. According to the DOE directive, the Composite Analysis should establish a "base case" or "best estimate" for comparison with the primary dose limit. This case should represent a reasonable, yet conservative, forecast of the future state of the Hanford Site, based on current knowledge. It should include expected remedial activities. Where future disposition of a source is not known, a reasonably conservative assumption should be made.

The initial Composite Analysis described in this document represents a deterministic base case. The results of these calculations represent an initial deterministic assessment based on available models and 'best estimates' of most model parameters. The scope of the analyses performed in subsequent iterations of the Composite Analysis will be directed by these preliminary findings and DOE guidance.

Performance analysis calculations were performed for all sites for which data were available. As mentioned earlier, defining a "worst case" in a composite analysis is problematic because of the superposition of plumes. In this analysis, best available estimates were used for nearly all model parameters. However, wherever uncertainty existed in two inventory estimates, the higher value was generally chosen.

In this initial analysis, sensitivity analyses were performed only to calibrate parameters. In order to ensure that results of the Composite Analysis could affect decision making early in the process, the DOE directive providing guidance for the Composite Analysis eliminated any new measurements from the scope of the initial analysis. As described in Chapter 3, attempts were made in the inventory estimate development process to identify errors in sitewide mass conservation for each of the radionuclides considered.

The first four steps of the DQO process were drafted early in the Composite Analysis, which helped define the problem being addressed. The draft addressing the first four steps of the DQO process was presented to a group referred to in later discussions as the DOE representatives, or "decision makers" for each LLW disposal or remediation activity. This group consisted of DOE and contractor representatives from the various Hanford Site programs. The draft was developed without programmatic input to establish a starting point for discussion. The first four steps are summarized in Table 2.1. The initial group did not have major revisions to the drafted Steps 1 through 4; however, they requested the option of revising Steps 1 through 4 once they reviewed the preliminary model results.

In an effort to complete DQO Steps 5 and 6, a meeting was called on December 13, 1996 with the decision makers to determine the acceptable level of uncertainty in the decisions to be made. The following questions were asked:

- Over what spatial area and what time period should the dose be integrated for comparison to a dose limit that would trigger some action?

- What actions would be taken if doses exceeded limits? This would be broken down into specific actions for specific programs. The representatives may want doses predicted on a smaller scale than currently provided, and integrated over a different range of influence, prior to taking actions that would commit their programs to major redesign expenditures. This is exactly the type of detail, negotiation and discussion that are fostered in the DQO process. Locking in these details prior to seeing the final model runs enables all parties to negotiate upfront and avoids discussions on semantics, assumptions, and meaning once results are final and actions must be taken.

- What dose limits are to be used in the final comparisons? Depending on how the model results deal with and report uncertainty, the operational decision rule may use an upper percentile of the distribution of possible doses to compare to the regulatory limit. A statistician should be involved with this step to make sure double conservatism is not built into the decision making and that uncertainty in model results is accurately compared to the desired limits on uncertainty as provided by the decision makers in Step 6.

At a December 13, 1996 meeting, LLW site representatives provided their best judgements for preliminary limits on decision errors. Specifically, the representatives at the meeting were asked:

> "If the model predicts a dose less than the action limit of 30 mrem in a year (and thus no action is required), but the 'true' dose turns out to be x (values on the x axis as shown in Figure 2.2), what limit would you want to place on making a decision error?"

The error limits that the LLW site representatives provided were specified over a range of possible outcomes. The responses received from the representatives are shown in Figure 2.2. This figure is a "modified" Decision Performance Goal Diagram and is modeled after the EPA structure for representing user-supplied decision error limits. Shown on the y-axis is the probability of deciding the dose is greater than 30 mrem in a year. Thus, 1-y is the probability of making a decision error. This modified diagram shows only one type of decision error: not taking action when it is required.

In this initial iteration of the Composite Analysis, "best estimate" input values and model assumptions are used, with no ranges provided. Therefore, no quantitative measures of the ranges in the dose estimates are possible. Unless decision makers are able to assign a confidence level to input values and model assumptions, there is no way to judge the confidence that should be assigned to model output. A qualitative way to incorporate decision makers' decision error limits into the process is to have the representatives assess their "relative comfort" with the justifications provided that conservative assumptions were used for model input. If the representatives are comfortable with the inputs used and the level of conservatism is consistent with the probabilities provided in Figure 2.2, then on a qualitative basis, the desired decision error limits from the DQO have been achieved and incorporated into the Composite Analysis.

In this initial iteration, maximum predicted doses are about 6 mrem in a year, well below the 30 mrem in a year limit. Only single input values were run through the model. But since there is no way to assess the confidence in the single predicted dose estimate, there is no way to evaluate whether additional model

improvements are justified. In subsequent iterations, quantitative estimates of confidence will be developed so that model improvements can be addressed directly.

2.7 Subsequent Iterations of the Composite Analysis

Several more iterations of the Composite Analysis will be required, before a full probabilistic assessment will be appropriate. Attempting a probabilistic assessment with a conceptual model with which the decision makers have no confidence is not going to increase their confidence. Therefore, uncertainty in the conceptual models will continue to be the dominant concern in early iteration. Monte Carlo methods are not appropriate for assessing uncertainty in the conceptual model, but intercomparisons among alternate conceptual models can bound the impact of model uncertainty. In instances where the impact of conceptual model uncertainty will affect decision making, intercomparisons among alternate conceptual models can suggest laboratory and field experiments that could resolve which conceptual model is appropriate.

One concept in which decision makers have a high degree of confidence is the principle of mass conservation. This is the primary conceptual element of the inventory model/database. While significant uncertainties exist in the exact present and future locations of radionuclides, the total inventory of radionuclides is reasonably well constrained by estimates of the radionuclides produced during reactor operations. Using probabilistic methods, the next iteration of the Composite Analysis will generate multiple equally feasible estimates of inventory for each site that are consistent with the principle of mass conservation. This information is critical to allowing a defensible approach to screening sites and radionuclides from further detailed analysis.

In order to limit the scope of the analyses, sites and radionuclides will be screened by a limited application of worst-case analysis. As discussed earlier, defining a worst-case condition for a composite analysis is considerably more difficult than defining the worst-case conditions for a single site and single radionuclide because of the superposition of plumes. However, by considering just the magnitude of the maximum dose, and not the timing and location of this maximum value, a large number of insignificant contributors to the dose can be placed in a single dose pool. This large number of sites could be simulated as individual releases, and doses could be calculated outside the buffer zone. The maximum dose from each site would be identified, and the sum of all sites accumulated independently of where or when it occurred. The combined dose of this large number of small contributions must be less than the target dose being considered. For instance, if the combined dose of these sites and radionuclides only resulted in a dose of 5 mrem in a year, and 30 mrem in a year was the dose estimate that would result in a different decision, the remaining sites, which would be analyzed in greater detail, would have to equal or exceed 25 mrem in a year before requiring a different decision. Clearly, this approach is biased towards making the decision error of taking actions that are not required. The size of the worst-case reserve, 5 mrem per year in this example, involves a tradeoff between increasing the analysis costs by including more sites and radionuclides in the detailed analysis and decreasing the likelihood of making a decision error of not taking actions that are required.

Subsequent iterations will also provide a basis for completing Steps 5 through 7 of the DQO process for the primary decision described in Section 2.4. The ways in which the subsequent iterations will affect DQO process Steps 5 through 7 are described as follows.

- **Step 5.** The decision rules will be modified to address the revised-model output. The decisions rules will address when specific LLW management actions and disposal practices from the options analysis will be undertaken based on model predictions.

- **Step 6.** Hanford site representatives may want to revisit the inputs provided for the Decision Performance Goal Diagram (Figure 2.2). They will also be asked to provide decision error limits for the second type of error: taking action when none is required.

- **Step 7.** Once results from the bounding or probabilistic assessment are available, and the conceptual model refined, decision makers can make a qualitative attempt at explicitly incorporating DQO limits on decision error into probabilistic analysis by placing upper confidence bounds on model output and comparing these upper confidence intervals to the dose limits. If the upper bound exceeds the limits, the decision makers are tasked with making resource allocation decisions and tradeoffs. Are model improvements required to reduce uncertainty bounds? Are low-level waste project modifications required to reduce dose predictions? The decision makers now have the tools and the input required to address these difficult questions posed in Section 2.4. While incorporating decision error limits into a probabilistic analysis and making resource allocations and tradeoffs are challenging, there are examples where such issues have been addressed along with the methods used to address them (Black et al. 1997; Black et al.1994; Freeze et al. 1992; Gilbert, Bittner, and Essington 1995).

Table 2.1. DQO Steps 1 Through 4 for the Hanford Site Composite Analysis

Step 1. Define the Problem	Predict the maximum annual dose to a hypothetical future member of the public resulting from combined radionuclide releases to groundwater from multiple sources during the 1000-year period following closure of the Hanford Site.
Step 2. Define the Decision(s)	A range of decisions must be made based on the results of the predicted dose to a hypothetical future member of the public during the 1000-year period following closure of the Hanford Site. If the maximum predicted dose is greater than 100 mrem/yr in any year, then an options analysis and ALARA assessment is performed and actions (determined by the options analysis) are taken to reduce the predicted dose below 100 mrem in a year and ALARA (as determined by the ALARA assessment). If the maximum predicted dose is greater than 30 mrem in a year, but less than 100 mrem in a year, then an options analysis and ALARA assessment are performed to identify the most effective actions that could be taken to reduce the predicted maximum dose. However, an alternate action is only recommended if it is feasible and beneficial considering economic, social-cultural, and ecological-resource factors. If the maximum predicted dose is less than 30 mrem in a year, then a screening-type ALARA assessment that weighs the cost of the options analysis and the potential benefit of dose reduction is performed to determine if a full options analysis and ALARA assessment is warranted.
Step 3. Define the Inputs	The calculated composite dose at locations that are accessible to hypothetical future members of the public is the information initially needed to make the decision whether an options analysis and ALARA assessment is required. These composite doses were calculated by a series of models that describe the release of radionuclides from waste sites, transport through the vadose zone, transport through groundwater, and exposure of individuals. The required inputs are categorized below: •Inventory data - total activity of each radionuclide that could contribute to the calculated composite dose •Release model assumptions - chemical and physical form of waste, release mechanism (i.e. dissolution, diffusion, and corrosion) •Release model parameters – water flux through waste site, dissolution rates, diffusion coefficients, temperature •Vadose zone contaminant transport model parameters – depth of waste, cover type and integrity, recharge rate •Vadose zone contaminant transport parameters – porosity, unsaturated flow parameters, moisture content, distribution coefficients

Table 2.1. (contd)

	•Groundwater transport model assumptions – future land use, location of the boundary where public access is assumed •Groundwater transport parameters – porosity, saturated hydraulic conductivity, aquifer thickness, distribution coefficients, recharge rates •Exposure scenario assumptions and parameters. If an options analysis and ALARA assessment are required, more information will be needed regarding treatment and disposal options. Information is also required to support the ALARA assessment of the economic, social-cultural, and ecological-resource impacts of alternate remediation strategies.
Step 4. Define the Boundaries	The analysis of exposure and dose to a member of the public applies to the land area where future members of the public may be exposed to radionuclides that have migrated from final disposal locations at Hanford. This area will exclude a waste management area assumed to remain under DOE control and not be accessible to the public. The decisions will be based on calculated doses during the first 1000 years after Hanford Site closure. However, calculation of doses will be carried out for longer periods of time to fully understand the migration, potential, and longer-term fate of the radionuclides. No accident or intruder scenarios will be considered. Note: It may be determined that the decision unit is each half acre of land in a buffer zone near the boundary, and/or that a separate decision is required for the maximum exposed individual as well as an average dose. These issues remain to be resolved.

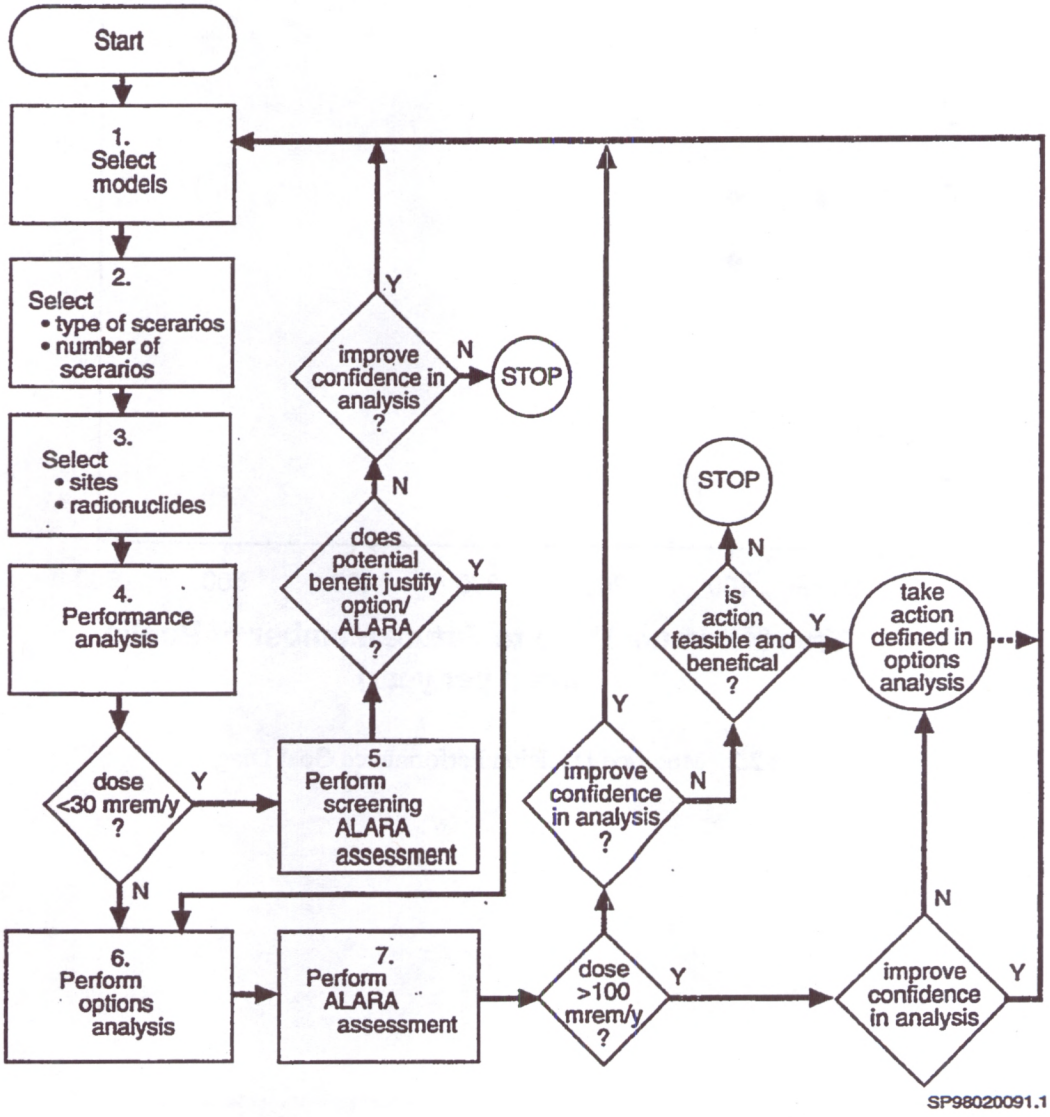


Figure 2.1. Composite Analysis Process Flow Diagram

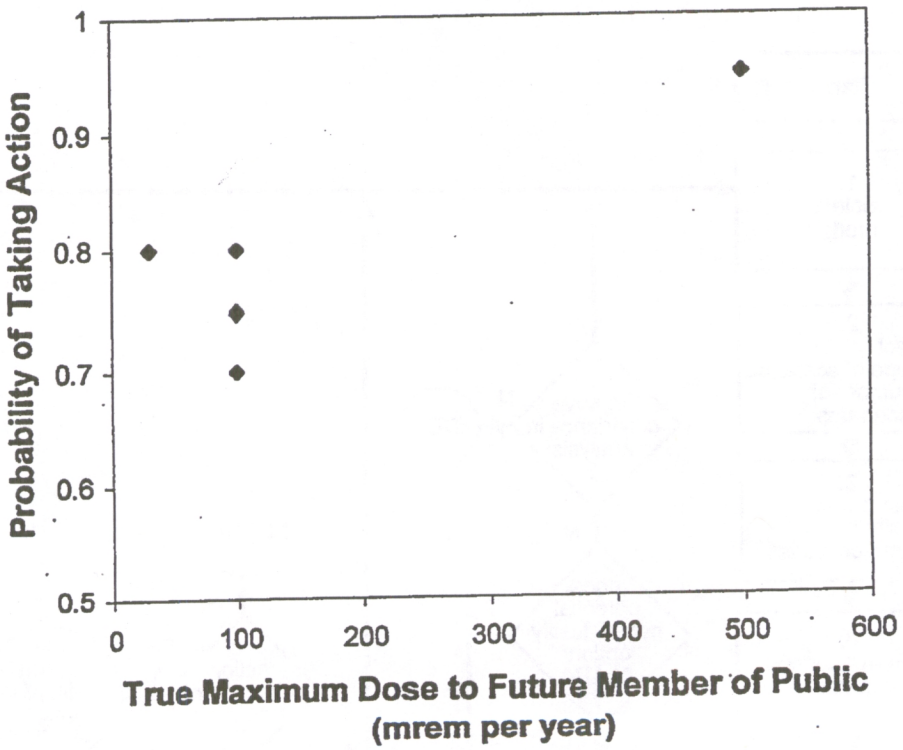


Figure 2.2. Modified Decision Performance Goal Diagram

3.0 Source Term Development

This chapter describes the sources of radioactive material that were considered for the *Composite Analysis for Low-Level Waste Disposal in the 200 Area Plateau of the Hanford Site* (Composite Analysis). Chapter 3 presents both the rationale for selecting these sources as likely to contribute to the dose from the low-level waste (LLW) disposal facilities received by the hypothetical future member of the public as well as the justification for excluding other sources from the analysis. The basis and justification for estimating the radioactive waste source term, (i.e., inventory) for each source included in the Composite Analysis, and the estimated source terms are also provided.

3.1 Sources of Radioactive Material

From 1943 until 1990, the mission of the Hanford Site was to produce special nuclear materials for weapons. After developing the largest site within the U.S. Department of Energy (DOE) complex devoted to production of special nuclear materials, activities at Hanford underwent a series of dramatic changes beginning in 1964. Plutonium production was sharply curtailed in response to the nation's changing defense needs. Eight production reactors were shut down by 1971. In January 1987, the N Reactor was placed in stand-down status for an extensive maintenance and safety enhancements program. In February 1988, the N Reactor was placed on cold standby. In July 1991, the DOE decided to cease preservation of the reactor and proceeded with activities leading to the ultimate decommissioning of the facility.

In July 1989, the Hanford Site was listed on the National Priorities List (NPL) under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) of 1980 as amended by the Superfund Amendments and Reauthorization Act (SARA) of 1986. In 1990, the mission of the Hanford Site changed to the safe cleanup and management of the legacy wastes, and the development and deployment of science and technology.

The vast majority of the radioactive waste inventory at Hanford was created during the production mission. There were three distinct steps in the production process: fuel fabrication, fuel irradiation, and chemical separation. During the first decades of production work at Hanford, it was common to locate waste disposal sites relatively close to waste-generating facilities. This practice resulted in numerous and varied disposal sites. The most dangerous radioactive wastes were stored in large single-shell tanks in the 200 Areas (Agnew et al. 1997; Kupfer et al. 1997). Large volumes of solid waste (e.g., contaminated tools and protective clothing) were disposed in burial grounds, and large volumes of liquid waste were discharged to shallow subsurface cribs, French drains, injection (or reverse) wells, and specific retention trenches.

More recently, all fuel fabrication and reactor operation activities ended and cleanup of past-practice units associated with them began in the 300 Area and 100 Areas. Low-level waste from ongoing operations is disposed in specific burial grounds in 200 West and 200 East Areas, and liquid discharges of radioactive wastes are being discontinued. The Tank Waste Remediation System (TWRS) program

addresses the waste disposal and site cleanup issues for tank wastes and tank farm facilities in the 200 Areas. DOE programs are in place and coordinating with representatives of the State of Washington Department of Ecology (Ecology) and the U.S. Environmental Protection Agency (EPA) to evaluate and decide upon the decontamination, decommissioning, or remediation strategies for reactors, chemical separation plants, and 200 Area past-practice sites (e.g., solid waste burial grounds and liquid discharge sites).

3.1.1 Solid Waste Burial Grounds

For the Composite Analysis, active solid waste burial grounds were defined as those open and receiving waste since September 26, 1988. The radionuclide inventories included in previous performance assessments (Wood et al. 1995; Wood et al. 1996) were those disposed since September 26, 1988. The list includes burial grounds 218-W-3A, 218-W-3AE, 218-W-4C, and 218-W-5 in 200 West Area, and 218-E-10 and 218-E-12B in 200 East Area. These burial grounds continue to receive solid waste (e.g., contaminated tools and clothing) from operations in their respective areas. In addition, some wastes are received from offsite generators within the DOE complex and the U.S. Department of Defense (e.g., U.S. Navy ship reactor compartments in Trench 94 of 218-E-12B).

In the past, wastes from the chemical separations plants were a function of plant operation. Today the wastes that are disposed in solid waste burial grounds at Hanford are from facility deactivation projects. At the end of these projects, the burial grounds will be transitioned to the Environmental Restoration Contractor (ERC). Whatever the source, those wastes containing sufficient inventories of waste that could migrate through the environment and result in potential radiation dose (e.g., technetium-99 and uranium) are stabilized in various grout formulations or disposed in high-integrity containers, or both.

At Hanford, private contractors are becoming involved in the chemical separation of high-level and low-level waste fractions from the tanks, and in the creation of immobilized waste forms (e.g., glass). Secondary low-level waste streams from these private companies were not considered in this analysis. Those secondary waste streams that meet specifications in contracts between the DOE and private companies will be returned to the DOE, and they may be disposed in the solid waste burial grounds at Hanford. Their inventory and volume are unknown at this time, but could include carbon-14, iodine-129, and technetium-99 scrubbed from atmospheric emissions.

3.1.2 Environmental Restoration Disposal Facility

The Environmental Restoration Disposal Facility (ERDF) trench receives waste from the remediation of CERCLA past-practice sites. Debris and excavated materials from these sites contain dangerous and hazardous waste, polychlorinated biphenyl (PCB) and asbestos waste, low-level radioactive waste, and low-level mixed waste containing both dangerous and radioactive waste components.

At present, the remediation efforts for CERCLA sites are focused on those nearest the Columbia River, i.e., those in the 300 Area and 100 Areas (Hartman and Dresel 1997). In the 300 Area, the effort is focused on past-practice solid waste disposal sites and liquid discharge sites associated with research conducted in the facilities and fuel fabrication efforts. In the 100 Areas, the effort is focused on similar

burial ground and liquid discharge sites associated with reactor operation and with the demolition of structures other than of the reactor buildings themselves.

Remediation plans for 200 Area past-practice sites are being developed. These plans require the joint agreement of the DOE, Ecology, and EPA. Facility decommissioning wastes will be disposed in the ERDF trench and not the solid waste burial grounds. The final dispositions of past-practice burial grounds, liquid discharge sites, and canyon facilities are unknown. For example, in the case of canyon buildings, remediation may involve the mounding of facility debris alongside the building prior to placement of a surface barrier or cover designed to limit intrusion and recharge.

3.1.3 Tank Waste Remediation System Waste

Since 1944, high-level wastes from the chemical separation plants have been stored in and transferred between large single-shell and double-shell tanks. These wastes are the result of the variety of processes briefly described in Chapter 1. They include waste streams from the dissolution of cladding materials and irradiated fuel slugs, the original bismuth-phosphate precipitation process, the solvent extraction processes used to recover plutonium and uranium, and the evaporators used to concentrate the waste in the tank farms.

As processes used to capture plutonium and uranium from solutions changed, the characteristics of wastes changed. These tank wastes are characterized as concentrated complexed waste, dilute complexed waste, double-shell slurry and double-shell slurry feed, aging waste, and noncomplexed waste (Hanlon 1997). Because carbon steel tanks were used at Hanford, wastes stored in the tanks were neutralized and often have pH values between 12 and 14. Wastes containing complexants were segregated from those that do not. The Composite Analysis therefore includes a distinction between complexed and noncomplexed waste regarding their mobility in the subsurface environment.

Sixty-seven of 149 single-shell tanks have leaked or are suspected to have leaked a portion of their inventory into the environment (Hanlon 1997). If sluicing is the method adopted for removal of tank wastes, it is anticipated the single-shell tanks will lose more liquid tank waste to the vadose zone. The TWRS program and private contractors will recover the tank waste, separate it into high-level and low-level waste fractions, and immobilize each. The TWRS program has begun the process to have the low-level waste fraction that will be disposed onsite declared incident waste, i.e., not high-level waste[a] (Peterson 1996). This low-activity waste fraction from the tanks will become immobilized low-activity waste (ILAW) and will be disposed at the Hanford Site. The high-level waste will be stored until it can be transferred to a national high-level waste repository. The process to declare past tank leaks, future losses, and tank waste residuals incident waste has not begun.

(a) From a letter, dated June 1997, sent by C. J. Papiello, Director of the Office of Nuclear Material Safety and Safeguards, of the U.S. Nuclear Regulatory Commission, to J. Kinzer, Assistant Manager, Office of Tank Waste Remediation System, DOE, "Classification of Hanford Low-Activity Waste Fraction." This letter may be found in Mann et al. (1997).

The recovery of wastes from both single- and double-shell tanks will not be perfect. The interim retrieval goal in Milestone M-45 of the Hanford Federal Facility Agreement and Consent Order (also known as the Tri-Party Agreement; Ecology, EPA, and DOE 1989) is to leave no more than 360 ft^3 of waste in each 100-series single-shell tank, and no more than 30 ft^3 of waste in each 200-series single-shell tank. This corresponds to 1% of the current single-shell tank waste inventory of 36 million gallons, allocated equally to each of the 149 single-shell tanks in proportion to the cross-sectional area of the tanks. Thus, an estimated 1% of the waste volume will remain in each tank following completion of recovery operations. For single-shell tanks the waste source types include leaks, losses during recovery operations, and a residual in the tanks after recovery. In the Composite Analysis, double-shell tanks were assumed to maintain their integrity during waste recovery, so only the residual left following recovery operations was considered. In addition to tank waste source types listed above, the TWRS program, specifically the privatization contractors, will produce secondary waste streams during their separations and immobilization steps. These wastes will be returned to DOE for final disposal.

3.2 Sources that Could Superimpose

Sources that could superimpose are those likely to contribute to the dose received by hypothetical future members of the public from the four LLW disposal facilities.

Waste disposal at Hanford has been centralized, the vast majority of wastes are to be disposed in the exclusive waste management area. Each of the active or planned LLW disposal facilities is located on the 200 Area Plateau and inside the exclusive waste management area. These wastes are from past operations of the chemical separations plants, from the cleanup and decommissioning of the chemical separation facilities, from the tanks, and from the CERCLA sites (i.e., the 100 Areas and 300 Area) along the Columbia River. The first iteration of the Composite Analysis focused on wastes disposed on the 200 Area Plateau of the Hanford Site because the majority of the low-level radioactive waste disposals at the Hanford Site will reside at this location.

Because the waste disposal sites, liquid discharge sites, chemical separations plants, and tank farms are either within or close to the exclusion area, they were all considered in the Composite Analysis. However, some inventories of radioactive waste are absent, available in insufficient detail to allow simulation, or not of significant magnitude to be included. Thus, many liquid discharge sites and all canyon buildings were omitted from the first iteration of the Composite Analysis. These sites and the reasons for their omission from the analysis are addressed in Section 3.3. While canyon buildings are analyzed, a sensitivity analysis representing the cesium and strontium inventory in a canyon building and filter system is included in Chapter 4.

Wastes in other areas (e.g., 100 Areas, 300 Area, and 400 Area) are located some distance from the 200 Areas. It was assumed that multiple sources within each of the 100 Areas and within the 300 Area will not create significant commingled groundwater plumes with contamination from the 200 Areas. The basis for this assumption is confidence in the CERCLA process to create a safe closure setting for each of these past use areas. If plausible situations are identified where sources from other areas commingle with plumes from 200 Area sources and create a potential threat to human health and safety, they will be analyzed in subsequent iterations of the Composite Analysis.

With regard to the atmospheric pathway, previously completed environmental impact statements and performance assessments were reviewed. Given the known waste sources and assumed conditions of release, the only waste sources potentially capable of making significant atmospheric pathway contributions to all pathways dose were the graphite cores of the production reactors.

3.3 Sources Excluded

This section provides justification for excluding sources from the groundwater or atmospheric pathways for the all-pathways dose estimate in the Composite Analysis.

3.3.1 Chemical Separation Plants (Canyons)

Six canyon buildings, designed for the processing of special nuclear materials, are present on the 200 Area Plateau. Two of these plants, B Plant and the Plutonium Uranium Extraction Plant (PUREX), are located inside 200 East Area. Four plants, T Plant, the Plutonium Finishing Plant (PFP), U Plant and Reduction-Oxidation Plant (REDOX), are located inside 200 West Area.

The canyon buildings will be decontaminated and decommissioned under the CERCLA program. However, the various standards (e.g., for levels of contamination) and final disposition of the canyon buildings (e.g., whether cells are to be filled to provide stability and prevent subsidence, canyon buildings are to be demolished to grade, entombed, and covered with surface barriers to reduce infiltration) have not been defined.

In the case of each canyon building, the major radionuclide sources and waste within the retired plant will be removed, reduced, or stabilized. Radiological contamination within the facility will be removed or fixed in place. The canyon buildings are massive concrete structures, and concrete is an excellent waste form for sorbed radionuclides. Whatever structure is left in place will be stabilized (i.e., filled with soil, gravel, or concrete) and all services (such as water) will be disconnected. Retired filters will be isolated and stabilized to ensure a safe condition. It is likely that these areas and especially any remaining structure will be covered with a protective barrier to further isolate contamination from intrusion and recharge. Final disposals will be dry with minimal driving force to mobilize and transport radionuclides from facilities.

In the absence of an inventory including any mobile and long-lived radionuclides, and with the assurance that all contamination will be removed from or entombed in these substantial structures, these facilities are not analyzed in the first Composite Analysis. It appears unlikely that the canyon buildings will be a significant source of groundwater contamination, especially in the next 1000 years. When more is known about their final inventories (e.g., the quantity and radionuclides known to be fixed in place) and physical state (e.g., whether infiltration barriers will be constructed to minimize infiltration), they could be simulated as contaminated concrete monoliths. A sensitivity analysis case was evaluated to determine whether the cesium and strontium inventory in a canyon building and its retired filters could contribute to the composite dose.

The PUREX storage tunnels (#1 and #2) branch off from the PUREX railroad tunnel and extend southward from the east end of the PUREX plant. The tunnels are used for storage of mixed waste (e.g., spent equipment and tank cars) from the PUREX plant and from other onsite sources. The radiological contamination in the tunnels consists primarily of uranium, transuranics, and/or mixed fission products. Currently, each storage tunnel is isolated from the railroad tunnel by a water-filled shielding door. No electrical utilities, water lines, fire detection or suppression systems, radiation monitoring, or communication systems are provided inside the PUREX storage tunnels. Material selected for storage is typically loaded on railcars modified to serve as both transport and storage platforms. Tunnel #1 is constructed of creosote-treated timber covered by roofing material and 2.4 m of earthen fill. Tunnel #2 is constructed of steel and reinforced concrete covered with 2.4 m of earthen fill.

Final closure of the PUREX storage tunnels will require the evaluation of alternatives. In general, these alternatives will involve either stabilizing the waste in the tunnels, or removing it and then stabilizing the tunnels (DOE 1996c). Alternatives for stabilizing the waste in place include, but are not limited to, backfilling the tunnels, waste, and railcars with gravel, or grout, or a combination of the grout on the bottom and gravel on the top. All means of access to the tunnels would be permanently sealed. Then a final surface barrier that meets Resource Conservation and Recovery Act (RCRA) landfill cover requirements to prevent water from leaching the waste in the tunnels would be constructed. Thus, the tunnels would be left in a stable configuration resistant to consolidation and settlement. The waste would be left in either a grout matrix or a gravel cocoon. Because these options have excellent waste form performance characteristics in the vadose zone when overlain by a surface barrier that significantly limits recharge through the waste emplacement and because of the absence of an inventory including any mobile and long-lived radionuclides, the PUREX tunnels were also excluded from the first iteration of the Composite Analysis.

3.3.2 Atmospheric Pathway

The potential for releases of radionuclides to the atmosphere depends on the final configuration of buried waste surface barriers and the radionuclides present in the waste. In order for an atmospheric release to occur, some mechanism is necessary to transport the radionuclides from the waste through the barrier to the surface. Release of radionuclides may occur by diffusion through surface barriers, by erosion of surface barriers followed by wind suspension, by transport to the surface by burrowing animals or plant roots followed by wind suspension, and as the result of disruptive events (e.g., intruder actions or severe natural phenomena).

Atmospheric releases resulting from disruptive events have been covered in previous performance assessments (Wood et al. 1995; Wood et al. 1996; Mann et al. 1997) and are beyond the scope of the present analysis. The previous studies have assumed that institutional controls prevent intrusion into the waste and atmospheric releases for at least 100 years and that passive controls prevent intrusion for 500 years. These studies have also addressed the issue of barrier erosion and concluded erosion is an unlikely mode of release to the atmosphere.

Barriers are expected to effectively inhibit transport to the surface by plant and animal penetration to the waste layer (DOE 1994b). While roots of plants may penetrate below 2.5 m, the quantity of

radioactive inventory brought to the aboveground plant mass is not expected to be large compared to that for the intrusion scenarios. Material in the aboveground plant mass would not be released to the atmosphere until the plant dies and decays. Barriers designed to limit water infiltration through the use of a capillary break, which reduces the water content below it, would also discourage plant root growth into the waste. Burrowing animals and insects are not expected to penetrate the soil significantly beyond 2 m (Napier et al. 1988). Previous performance assessment studies for LLW have concluded that plant and animal transport is not a significant route of airborne release.

The previous performance assessments have evaluated diffusion of volatile radionuclides through the barriers as a source of airborne release. The radionuclides considered were tritium, carbon-14, and radon-222. The analyses have required the assumption of unlikely and conservative conditions over time and have resulted in very small estimates of release. Doses calculated to an individual living above the waste have also been small. The production of radon-222 from uranium-238 is small during the first 1000 years after placement, and does not peak until about 100,000 years. Even at the peak release rate, the amount of radon-222 reaching the surface is small because of the delay in diffusion through the soil overburden and the short half-life of radon-222 (about 3.8 days). Prior analysis of the graphite cores from the production reactors (DOE 1989, 1992) produced the only source of small but potentially significant airborne release. In that analysis it was assumed that half of the core inventory was available for release and migration via the atmospheric pathway.

Based on the review of past studies, the only atmospheric releases included in the initial Composite Analysis are the releases of the volatile radionuclides tritium and carbon-14 from buried graphite reactor cores in the 200 Areas. The graphite cores do not have a source of radon-222 in their inventory.

3.4 Estimation of Source Inventory and Release Rate

This section provides the basis and justification for estimating the source term for each source to be included in the Composite Analysis. The estimated radionuclide inventory is included.

3.4.1 Selection of Key Radionuclides for Study

The Composite Analysis is the beneficiary of preceding analyses and field observations. It is a companion analysis to the performance assessments for 200 West and 200 East post-1988 burial grounds (Wood et al. 1995; Wood et al. 1996) and the remedial investigation and feasibility study of ERDF (DOE 1994b). It was also preceded by an analysis of a new waste form, the immobilized low-activity waste from Hanford tank farms (Mann et al. 1997). These and other analyses, (e.g., environmental impact statements) included development of inventory data and application of screening or significance criteria to identify those radionuclides that could be expected to significantly contribute to either the dose or risk calculated in the respective analysis.

Clearly, those radionuclides identified as potentially significant in these published analyses are also expected to be key radionuclides in the Composite Analysis. Older studies were reviewed to identify any radionuclides unique to specific wastes or closed facilities. Of greatest interest were the more recently completed studies including those supporting the disposal of immobilized low-activity radioactive waste

originating from the single- and double-shell tanks (Mann et al. 1997) and residing in the shallow land burials (Wood et al. 1995; Wood et al. 1996).

3.4.1.1 Low-Activity Waste from Tanks

The activation products, actinides, and fission products generated in the reactors at the Hanford Site are anticipated components of the low-activity radioactive stream coming from Hanford single- and double-shell tanks. The complete list of these isotopes can be found in Schmittroth et al. (1995) and Watrous and Wooten (1997).

The screen applied by Schmittroth et al. (1995) to identify those radionuclides that could be potentially significant contributors to dose in groundwater pathway scenarios yielded twelve potentially important isotopes. In order of their contribution to drinking water dose, a major component to all-pathways dose, the twelve isotopes were technetium-99, selenium-79, uranium-233, uranium-234, uranium-238, radium-228, niobium-93m, iodine-129, radium-226, uranium-236, curium-245, and uranium-235. To arrive at this list, Schmittroth et al. (1995) used a simple retardation model, and where distribution coefficient data were absent, made the conservative assumption of no sorption. After reviewing the distribution coefficients, the following values were assigned to several of the elements (Kaplan and Serne 1995; Kaplan et al. 1996): technetium and selenium, 0 mL/g; uranium, 0.6 mL/g; radium, 15 mL/g; niobium, 40 mL/g; iodine, 3 mL/g; and curium, 100 mL/g. The radionuclides that were assigned nonzero distribution coefficient values in the study by Schmittroth et al. (1995) failed the screen as significant contributors to dose via the groundwater pathway. Consequently, those elements (i.e., radium, niobium, and curium) assigned the higher values after the initial screen were also eliminated. Accordingly, only the top eight isotopes contributing to drinking water dose were identified as potential key radionuclides for the Composite Analysis: technetium-99; selenium-79; iodine-129; and uranium-233, -234, -235, -236, -238, and their daughters.

3.4.1.2 Solid Waste Burial Grounds

Those radionuclides remaining after the screening process for the 200 East Area burial grounds were long-lived and mobile (Wood et al. 1996). A list of all radionuclides considered in the dose analysis for the 200 East Area burial grounds appears in Wood et al. (1996, Table 4.1). The screening process eliminated all moderately to strongly sorbed radionuclides because they were predicted to have no significant ability to contaminate groundwater in the next 1000 years. Radionuclides passing the screen were tritium, carbon-14, chlorine-36, selenium-79, technetium-99, iodine-129, rhenium-187, and the uranium isotopes. Because of their unique inventory and waste form degradation characteristics, the U.S. Navy ship reactor compartments were treated as a special case. In this special case, the list of radionuclides potentially able to contaminate groundwater is a subset of the above list: carbon-14, chlorine-36, selenium-79, technetium-99, and iodine-129.

One isotope, rhenium-187, that passed the screen was eliminated from further consideration. The screen criteria included potential mobility and decay half-life; however, rhenium-187 is not present at Hanford in sufficient quantity to present a health threat. Rhenium-187 is an activation product of tungsten, and its existence in significant quantities in the DOE radioactive waste would indicate that a

significant quantity of tungsten had been employed in the fuel or its cladding. This was not the case. Schmittroth et al. (1995) estimated the total production of rhenium-187 at 8.6×10^{-6} Ci using the Oak Ridge Isotope Generation and Depletion (ORIGEN2) code (Croff 1980). Based on its potential contribution to drinking water dose, this quantity will not significantly contribute to dose.

3.4.1.3 Radionuclides Selected by Screenings in Other Analyses

The closure plan for the commercial LLW site operated by US Ecology on the Hanford Site (Grant Environmental, Chase Environmental Group, and US Ecology 1996) presents a total inventory to date and a projection for disposal at the site until its closure in 2063. The inventory was screened according to two criteria, total activity greater than 1 Ci and decay half-life in excess of 100 years. Of the radionuclides identified for further analysis, several have distribution coefficients at or only slightly greater than zero, including carbon-14, chlorine-36, iodine-129, potassium-40, technetium-99, and uranium-238. While all the other radionuclides were identified in prior analyses, potassium-40 was identified as a contaminant of potential concern. In the review of the inventory for the ERDF trench, potassium-40 was identified as a potential isotope of concern; however, it was also identified as a radionuclide considered to be derived completely from natural background. Wood et al. (1995) noted that an average background value of ~15 pCi/g supports this hypothesis. Wood et al. (1995) also noted that potassium-40 is not a known fission product, and consequently, its activity was not considered when calculating the potential dose from DOE wastes such as those in the ERDF. Accordingly, for the purposes of the Composite Analysis, potassium-40 was omitted from the calculation of composite dose from either DOE sites or the commercial LLW disposal facility.

3.4.1.4 Uranium Daughters

During release and migration of radionuclides from the vadose zone to the unconfined aquifer, some radionuclides will decay and produce daughter products. However, radioactive decay involving generation of progeny radionuclides can be difficult to model in systems that allow each chain member to move independently. Physical separation of the chain members is not generally accounted for in decay and environmental transport algorithms. In the Composite Analysis, computational resources did not permit modeling individual progeny, so an alternative treatment was used.

Two options were considered for daughter products in the Composite Analysis: 1) daughter products that do not contribute significantly to dose do not need to be simulated; and 2) decay chain members can be simulated as equally mobile as their parent.

Regarding the first option, the regulatory period of interest is short (1000 years), and may provide insufficient time for significant quantities of uranium daughters to be created. In addition, the decay products in the uranium chains, aside from other uranium isotopes, are radium and thorium. In general, both are more highly sorbed in comparison to the parent uranium. In the aquifer the best estimate distribution coefficient values for uranium, radium and thorium are 3, 20 and 1000 mL/g respectively (see Appendix E). For only one waste type (i.e., very high salt and very basic tank wastes) are radium and thorium more mobile than uranium in the vadose zone, and this is true only in the sediments immediately below waste tanks. In the lower portion of the vadose zone these wastes are believed to be buffered by

the vadose zone sediments and soil water, and radium and thorium are again assigned higher distribution coefficients than uranium. For the other five waste types disposed to ground, radium and thorium are always more highly sorbed than uranium. In general, because of their sorption, the radium and thorium daughters will not enter the groundwater from the vadose zone in the 1000-year period. Essentially, radium and thorium found in the aquifer will be a result of uranium entering the aquifer and then undergoing decay to create daughters. The radium and thorium daughters will not move with uranium in the aquifer. Thus, a reasonable treatment for the first 1000 years after Hanford Site closure would be to account for uranium isotopes and uranium daughters, and neglect the radium and thorium daughters in the dose calculation.

Regarding the second option, radioactive chain decay in the subsurface can be separated from the transport calculation if the chain members all travel at the same rate (i.e., without physical separation). If the analysis were conducted with all chain members in the same medium and traveling together, decay could be accounted for based on the elapsed time between initial source definition and the time of interest. In the case of uranium parents, as long as uranium transports as fast or faster than its daughter(s), it is conservative to model the daughter(s) as moving with the parent.

For the Composite Analysis, radioactive chain decay was separated from the transport analysis. To accomplish the separation ORIGEN2 code simulations of irradiated fuels (see Appendix A) were used to define the relative abundance of uranium isotopes in an average Hanford Site waste. The abundance of other uranium isotopes were defined in terms of the uranium-238 level. The grams of uranium isotopes uranium-233, -234, -235, and –236 per gram of uranium-238 were assumed as follows: 1.07E-08, 5.65E-05, 6.70E-03, and 1.46E-04 grams of the isotope per gram of uranium-238. A chain decay calculation was used to determine the relative significance of the uranium progeny contribution to dose when progeny were as mobile as parent. A calculation of the dose resulting from 1 mg/L of uranium-238, other uranium isotopes, and their progeny, shows that after 1000 years the dose from all progeny in the agriculture scenario was <10% of the dose from the uranium parents. The same calculation was performed for the industrial, recreational, and residential scenarios with similar results. Consequently, to be conservative the contribution to composite dose from uranium was based on uranium-238 release and migration, the relative abundance of other uranium isotopes as indicated by ORIGEN2 simulations, and the inclusion of all progeny as though they were as mobile as the parent. This is conservative in light of the greater sorption of the radium and thorium daughters.

3.4.1.5 Radionuclides Included in the Groundwater Pathway

The radionuclides included in the groundwater pathway analysis for future sources were carbon-14; chlorine-36; selenium-79; technetium-99; iodine-129; and uranium-233, -234, -235, -236, -238 and their daughters. This list is the result of merging the two lists from the immobilized low-activity waste from tanks and the solid waste burial grounds. In addition, the remedial investigation and feasibility study (RI/FS) for the ERDF and other environmental impact statements (DOE 1989, 1992, 1994b, and 1996a; DOE and Ecology 1996) were reviewed, and no other radionuclides were identified as potentially significant contributors to groundwater pathway dose. In the first iteration of the Composite Analysis, the

contribution of uranium and its progeny to dose was estimated by simulating uranium-238, approximating the abundance of other uranium isotopes using a single set of isotopic ratios, and assuming uranium daughter products move with the parent.

Plumes of tritium, strontium-90, technetium-99, and iodine-99 exist in the unconfined aquifer at the Hanford Site. While radionuclides with long half-lives, i.e., technetium-99 and iodine-129, are identified as key nuclides in the Composite Analysis, tritium and strontium-90 are not. Neither tritium nor strontium-90 are included as key mobile radionuclides in the study, but both were included in a recent study of existing plumes (Cole et al. 1997), and the Composite Analysis has included the influence of these existing plumes on future dose projections. Thus, while no effort has been made to assemble inventory data and model release and vadose zone migration of either tritium or strontium-90, their effects on dose are included.

Because of its mobility and its disposal to cribs in relatively large volumes of liquid waste, tritium is assumed to be in the aquifer and not significantly retained in the vadose zone. Thus, simulations of the existing plume of tritium and of future disposals of liquid waste at the State-Approved Land Disposal Site have captured the future impact of tritium (Cole et al. 1997). Strontium-90 plumes were simulated by Cole et al. (1997), and those results are also incorporated into the Composite Analysis. Strontium is highly sorbed in the aquifer and does not pose a threat outside the buffer zone when the source is inside the exclusive waste management area and buffer zone. Strontium-90 will be shown to contribute to dose, but only in the immediate vicinity of these releases.

The selection of radionuclides for inclusion in this first iteration Composite Analysis has relied on the results of several disposal studies. Each involved the burial of an essentially dry waste form in the thick vadose zone deposit of the 200 Area Plateau. Future iterations of the Composite Analysis will benefit from ongoing studies of liquid discharge sites and tank leaks. Other radionuclides may be identified in these studies as being sufficiently mobile to reach the aquifer. Their mobility in the vadose zone may be a result of the original waste composition and a lower potential for adsorption to sediments or the precipitation of minerals, or the increased driving force of the liquid discharge. Subsequent iterations of the Composite Analysis will revisit the key radionuclide identification process and take advantage of future findings.

3.4.2 Solid Waste Burial Grounds

Inventories of key mobile radionuclides disposed in each of the 200 East and 200 West solid waste burial grounds were estimated for pre-September 1988 and post-September 1988 amounts using an aged-fuel-ratio methodology and the record of cesium, uranium, or plutonium disposal (Appendix A). The inventories are stored in the "inventory" worksheet within the *Composite Analysis 2.0.XLS* Excel™ workbook, described in Chapter 4. These inventories of the key mobile radionuclides were estimated using radionuclide inventory information from the Solid Waste Information Tracking System (SWITS) database (Clark 1995). In addition, the ORIGEN2 code (Croff 1980) was used to estimate the abundance of key mobile radionuclides potentially present but not reported in the SWITS database.

Activities of cesium-137 and masses of uranium and plutonium disposed were obtained directly from the SWITS database. Two types of SWITS database reports were generated for two periods. The types of reports summarized unsegregated waste and post-1970 non-transuranic (non-TRU) segregated wastes. These reports were generated for startup through September 1988 and startup through December 1996. The inventories of uranium, plutonium, and cesium-137 disposed were totaled between the unsegregated disposal inventory and the segregated non-TRU inventory. This excluded the transuranic (TRU) waste, which was not expected to remain onsite. By subtracting the September 1988 inventory from the December 1996 inventory, an estimate of the post-September 1988 inventory disposed was obtained.

3.4.2.1 Suspect Transuranic Waste and Pre-1988 Inventory

Before 1970, TRU waste at the Hanford Site was not segregated prior to disposal (Wood et al. 1995). After 1970, TRU waste, defined as >10 nCi/g, was segregated prior to disposal so that it could be retrieved and eventually be disposed offsite. In 1984, the definition of TRU waste was changed from >10 nCi/g to >100 nCi/g. Therefore, a portion of segregated TRU waste disposed between 1970 and 1984 may be reclassified as LLW and be disposed on the Hanford Site. The plans for dealing with this type of waste are being developed. For the Composite Analysis, the suspect TRU waste sites are governed by CERCLA, and, therefore, are associated with the pre-September 1988 inventory. The estimated inventory of pre-September 1988 waste was incremented by the estimated suspect TRU waste inventory that will be reclassified as LLW (see Appendix A, Section A.2.1). This is the pre-1988 solid-waste burial ground inventory applied in the Composite Analysis (included in Table 3.1).

3.4.2.2 Future Disposal Inventories

Future disposal inventories are uncertain. In the Composite Analysis the inventory disposed between September 1988 and December 1996 was extrapolated for the planned 30 years of disposal assuming a constant rate of disposal. The inventory values were compared to projections made in the performance assessments for the 200 East and 200 West Area solid waste burial grounds (Wood et al. 1996; Wood et al. 1995). In cases where the solid waste performance projection values exceeded the linear extrapolation of waste disposal over 30 years, the performance assessment values were used. The differences were the result of having a different and longer record of waste disposed since September 1988 to use as the basis of the future forecast. Table 3.1 includes the future inventory of key radionuclides for the post-1988 period of disposal in the solid waste burial grounds. Although key radionuclides in Table 3.1 are listed in association with disposal areas, future waste disposal may not occur in the same locations.

3.4.2.3 Estimation of Non-Reported Radionuclides

While uranium, plutonium, and cesium-137 are relatively well reported within the SWITS database (Clark 1995), a number of radionuclides may also be present but are not consistently reported. Some of these radionuclides are potentially important to performance assessment calculations, (e.g., carbon-14, chlorine-36, selenium-79, technetium-99, and iodine-129) and were therefore also important to the Composite Analysis. In an effort to estimate inventories of these radionuclides, Version 2.1 of the ORIGEN2 code was used to estimate the relative abundance of other radionuclides that are important but not consistently reported, compared to the major radionuclides that were reported. This method was

applied to develop inventory for solid waste burial grounds (see Table 3.1) and those liquid discharge sites that did not receive tank waste (see Section 3.4.5 on CERCLA Sources). This section summarizes major points of the estimation method, which is more fully described in Appendix A.

ORIGEN2 calculations were made for single-pass reactor and N-Reactor irradiation to determine radionuclide concentrations in spent fuel and cladding. Impurities in the fuel and cladding were included in the model. The quantities are presented in Appendix A (Tables A.1 through A.6) and are based on Bergsman (1993). A weighted average between the single pass and N-reactor nuclide concentrations was used to estimate the overall average nuclide composition. About 90% of the fuels processed at Hanford were irradiated in the single-pass reactors.

Inventories of omitted fission products in solid waste were estimated by multiplying the undecayed cesium-137 inventory from SWITS by the ratio of the Ci/kg concentration of the radionuclide of interest to that of cesium-137 from the ORIGEN2 calculation. The ratios were developed for a fuel age of 10 years after discharge from the reactor. Estimates based on fuel decayed for 1 year are more conservative for radionuclides with decay half-lives less than that of cesium-137 (~30 years). The key radionuclides have longer decay half-lives. Estimates based on 10 years of decay prior to disposal were more conservative for radionuclides with longer half-lives. Where the activity of a fission product increased over time beyond 1 year, the maximum activity between 1 and 3000 years was used to calculate the ratio to cesium-137 at 10 years.

The SWITS database reports provide both a mass of uranium disposed, which is not identified by isotope, and quantities of uranium isotopes that are specifically identified. The ORIGEN2 results were used to divide the uranium that was not identified by isotope among the uranium isotopes, and to estimate the quantity of other actinides (except plutonium) that may be present. This was accomplished by multiplying the uranium mass reported in SWITS by the ratio of activity of actinide (or daughter) to uranium mass in discharged fuel. Similar to the fission product case, estimates were provided for fuel with 10 years of decay. As in the case of fission products, the maximum activity between 1 and 3000 years in the ORIGEN2 calculation was used to calculate the ratio to uranium mass. Plutonium reported without isotopic distribution was divided into isotopes based on the relative abundance indicated in the ORIGEN2 results. Quantities of plutonium reported in SWITS as specific isotopes were then added to arrive at total plutonium isotopic values.

Because of its identification as a key mobile radionuclide in the graphite cores, an effort was made to determine the potential significance of chlorine-36 elsewhere in the inventory. Chlorine-36 is a potentially important radionuclide that may be formed by the irradiation of chlorine impurities in the fuel or cladding. No data on the chlorine-35 impurity levels within metallic uranium fuel were available. Because of the uncertainty in chlorine levels, a calculation was performed assuming a 1-ppm by weight impurity in the fuel. The 1-ppm level is an estimate but is believed to be within an order of magnitude of the actual impurity level. However, it may be a factor of 3 over the impurity level allowed in yellow cake. The chlorine-36 abundance in waste was calculated according to the reported cesium-137 content, as was done for other fission products. The purpose of including chlorine-36 in the inventory is to

determine if the nuclide is potentially important. If the 1-ppm level is potentially important, a more in-depth investigation into chlorine-36 may be justified, otherwise the additional effort may not be warranted.

The choice of using ORIGEN2 predicted ratios of key mobile radionuclides to cesium-137, i.e., the aged-fuel-ratio method, was based on previous work by Wood et al. (1996) that provided a proposed breakdown of "time after discharge" to be applied to disposals. In their work, Wood et al. (1996) found it was appropriate to use 1-year fuel ratios for waste disposed from 1945 through 1973. However, disposals in more recent years may originate from waste discharged by the reactor several years prior to disposal. After 10 years the cesium-137 inventory declines by about 20%. As a result, when the inventories of long-lived fission products in wastes were estimated based on cesium-137 content, and using the 10-year fuel age assumption, the values are about 20% higher than when the inventories are estimated using the 1-year assumption. Overall, the sensitivity to using a 1- or 10-year fuel age assumption was small, relative to the uncertainty caused by using a ratio of other radionuclides to cesium-137. The cesium-137 ratio calculation is based on the assumption that the isotopic ratios in the waste were similar to those in the discharged, irradiated fuel.

3.4.3 Environmental Restoration Disposal Facility

A variety of burial grounds and liquid discharge sites in the 300 Area and 100 Areas are undergoing cleanup efforts. The goals are to excavate contaminated soils and clean sites up so that they may support unlimited or unrestricted industrial (300 Area) and residential (100 Areas) use, to control sources of groundwater contamination to protect the Columbia River, and to control future groundwater cleanup costs (DOE 1996g). Wastes from these sites are being disposed in the ERDF trench. The objectives and methods of remediation for 200 Area sites have not yet been negotiated between DOE, Ecology, and the U.S. Environmental Protection Agency (EPA). However, only wastes from CERCLA cleanup activities will be disposed in the ERDF trench.

Two documents describe the environmental consequences of the ERDF disposal facility: the RI/FS report (DOE 1994b) and a performance assessment (Wood et al. 1995). As a result of decisions made by DOE regarding the applicability of DOE Order 5820.2a (DOE 1988b) to the disposal of cleanup wastes from CERCLA sites, the final performance assessment (Wood et al. 1995) was not peer reviewed but was published as a record of work completed and analyses conducted. Based on the RI/FS (DOE 1994b), a record of decision (ROD 1995) was issued January 1995 that authorized the construction and operation of two disposal cells with an expected capacity of 920,000 m^3 (1,200,000 yd^3).

The RI/FS lists the maximum detected concentrations of radionuclides for soils in the waste sites of the 100, 200 and 300 Areas. Overall maximum contaminant concentrations (pCi/g) for soils in all three areas are listed in the RI/FS (DOE 1994b, Table 3.8). Based on the RI/FS, these concentrations of radionuclides were assumed to be disposed in the ERDF. Consequently, in the first iteration of Composite Analysis, these maximum concentrations were assumed to exist in all wastes disposed at the ERDF.

While the ROD describes the initial construction and operation of two cells, planning is currently underway for the disposal of 3.59×10^6 m^3 (4.7×10^6 yd^3) in up to six cells. If approved, extending the disposal pit excavation to the east will create the additional four cells. The volume for a six-cell facility is the current projected waste volume for the cleanup and removal of wastes from all 100 and 300 Area CERCLA sites. The density of these wastes upon delivery to the ERDF is an assumed loose density of 1.66×10^6 g/m^3 (1.4 tons/yd^3). The in-place density compacted to 90% is 2.02×10^6 g/m^3 (1.7 tons/yd^3). Therefore, the in-place compacted volume of the disposal will be 2.96×10^6 m^3 (3.87×10^6 yd^3). Location details (e.g., Washington State Plane coordinates for the disposal cell corners, bottom elevation of the disposal), for the ERDF were taken from the subgrade survey control drawing,[(a)] and the eastward projection of the construction was based on personal communications with contractor staff.[(b)]

The maximum contaminant concentrations from the RI/FS (DOE 1994b, Table 3.8) were applied to the estimated 3.59×10^6 m^3 (4.7×10^6 yd^3) of loose waste to be delivered to the ERDF to produce the total curies of each radionuclide disposed. This assumption is conservative and likely results in an over-estimate of the inventory. These inventory data were stored in the "inventory" worksheet within the *Composite Analysis 2.0.XLS* Excel™ workbook (described in Chapter 4). The key radionuclide inventory of the ERDF is shown in Table 3.2.

3.4.4 Hanford Tanks

Some waste currently stored in tanks at the Hanford Site will remain at Hanford after closure in one of four forms (DOE and Ecology 1996). The majority will be an ILAW created from the incidental waste fraction recovered from tanks. Some will be in the form of a residual left in the tanks after waste recovery operations. For the Composite Analysis, losses to the surrounding soils during recovery operations were assumed to remain in the soil column as well as past tank leaks (i.e., they will not be removed during remediation). These source inventories, immobilized low-activity waste from tanks, leaks and slurry losses from single-shell tanks, residuals in single-shell tanks, and residuals in double-shell tanks, are described in the following four sections.

Since the Composite Analysis began, the TWRS program has established standard inventories for chemicals and radionuclides in the tank wastes (Kupfer et al. 1997). The Kupfer et al. (1997) inventory is a best-basis global inventory. A best basis tank-by-tank estimate was also produced.[(c)] The fourth revision of the Hanford Defined Waste (HDW) model (Agnew et al. 1997) was also issued since the effort to assemble Composite Analysis inventories began. Agnew et al. (1997) is a supporting document to the

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(a) U.S. Department of Energy, Drawing No. 0600X-DD-C0033, Rev. 1. Date: 11/18/96. Record number H-6-14624 SHT 1.

(b) Information received by C. T. Kincaid, PNNL, during a meeting on February 4, 1997 with F. V. Roeck and M. A. Casbon, Bechtel Hanford Inc., ERC. The meeting topic was "Composite Analysis/ERC."

(c) From letter FDH-9757750 dated August 29, 1997 from D. J. Washenfelder (Fluor Daniel Hanford) to J. K. McClusky (DOE), "Contract Number DE-AC06-96RL 13200; Completion of Milestone T24-97-158, Contractor Letter to Department of Energy, Richland Operations Office, Reporting Completion of Standard Inventory Estimates for All Tanks."

more recently published best-basis or standard inventory (Kupfer et al. 1997). While the first iteration of the Composite Analysis was based on data available at the time of the analysis, greater consistency in tank waste inventories will be achieved in future iterations when current editions of the standard or best-basis inventory for tank wastes are employed.

3.4.4.1 Immobilized Low-Activity Waste from Tanks

The source inventory for the incident waste fraction of waste currently stored in single- and double-shell tanks is reported in the interim performance assessment for low-level tank waste (Mann et al. 1997, Table 3.1). Following recovery from the tanks, waste will be separated into high-level waste and incident waste fractions. The incident waste fraction will be immobilized and returned to the DOE for disposal as ILAW. The high-level fraction is to be returned to DOE for storage until it also is immobilized. After immobilization, it will be stored until the national high-level waste repository is opened, and then it will be shipped to the repository and disposed.

The inventory that appears in the interim performance assessment is fully documented (Schmittroth et al. 1995) as one among many data packages (Mann 1995) developed in support of the interim performance assessment. This published inventory and the associated release models have been adopted for the first iteration Composite Analysis. Subsequent iterations of the performance assessment for ILAW will rely on the current standard or best-basis inventory (e.g., Kupfer et al. 1997). Plans call for these wastes to be disposed in two locations in four existing vaults and several new disposal vaults. The inventory of ILAW to be disposed in existing facilities is based on the fraction of the waste volume they can contain, and the total inventory reported by Mann et al. (1997). Table 3.3 shows the key radionuclide inventory assumed for each disposal location.

3.4.4.2 Single-Shell Tank Farms – Tank Leaks and Slurry Losses

There are twelve single-shell tank farms containing 149 tanks on the 200 Area Plateau. Six tank farms (S, SX, T, TX, TY, and U) containing 83 tanks are located in the 200 West Area. Six tank farms (A, AX, B, BX, BY, and C) containing 66 tanks are located in the 200 East Area. Three types of releases from single-shell tanks were included in the Composite Analysis. In chronological order of occurrence they are 1) past tank leaks, 2) future losses from tanks during recovery of wastes, and 3) residuals to remain in tanks. Of the 149 single-shell tanks at Hanford, there are 67 tanks confirmed or assumed to have leaked (Hanlon 1996, Appendix H). As noted in Hanlon's monthly reports, volume estimates have been made for these 67 leaking tanks. However, estimates of inventory lost during tank leaks are incomplete. Hanlon (1996) reports only cesium-137 losses for 17 of the 67 leaks. The second and third types of release are the result of the waste recovery operations.

The TWRS program has published the initial retrieval sequence and blending strategy (Penwell, Grenard, and Wittman 1996). The retrieval operation is projected to occur over a 15-year period beginning in 2004 and ending in 2019. Penwell, Grenard, and Wittman (1996) provided detail on the retrieval sequence of each tank and each tank farm. The Composite Analysis simulated losses during the recovery operation time interval for each tank farm as specified in the retrieval sequence document. The TWRS program is committed to revise annually the single-shell tank retrieval sequence, (e.g., Kirkbride

et al. 1997). Results of this initial Composite Analysis would not differ significantly if the later retrieval sequence had been used. Future Composite Analyses will use the most current retrieval sequence for single-shell tanks.

Using currently available leak detection and mitigation technologies, a tank leak could not be detected before 4000 gallons (15 m^3) has been released and not stopped for most tanks before approximately 8000 gallons (30 m^3) had been released (WHC 1996). Consequently, the TWRS program assumed an average release volume per single-shell tank of 8000 gallons (30 m^3). This is a more current estimate than the 4000 gallons (15 m^3) per tank value assumed in the TWRS environmental impact statement (EIS) (DOE and Ecology 1996). Conservative assumptions to establish an upper bound on the amount of leakage from single-shell tank 241-C-106 and its potential impact resulted in a calculated leak volume of 40,000 gallons (150 m^3) because of hydraulic sluicing of that tank (Lowe 1993). While an average loss volume of 8000 gallons (30 m^3) has been assumed, there are reasons to expect a lower average. For example, some tanks will have better leak detection and mitigation capabilities than others, and tanks that are confirmed leakers (~50 single-shell tanks) are candidates for alternate cleanout technologies that use robotic arms or low-volume liquid methodologies or both.

A significant unknown for both tank leaks and losses during recovery operations is the inventory potentially lost to the subsurface environment. The inventory reported in the TWRS EIS (DOE and Ecology 1996) is a total inventory of radionuclides contained within liquid, sludge, and solid wastes in the tanks. An attempt to use the TWRS EIS inventory data, specifically the average concentration from its total inventory and tank farm volume, combined with Hanlon's (1997) tank leak volumes, failed to qualitatively match the cesium releases noted by Hanlon. This likely resulted from not using an inventory divided among liquid, sludge and solid wastes in the tanks. In other databases and reports (e.g., the Tank Characterization Reports of DiCenso and Simpson [1994] and Winkelman [1996]), liquids are characterized separately from sludge and solids, and they are reported as either supernatant or drainable liquid.

The average concentrations of radionuclides in liquid tank wastes (i.e., including both supernatant and drainable liquid) were calculated using data reported in the Tank Characterization Reports (DiCenso and Simpson 1994; Winkelman 1996). However, insufficient data were found to assemble average values on a tank farm basis. Therefore, average values were assembled for four waste types from data on all single-shell tanks. The four waste types were double-shell-slurry-feed, noncomplexed waste, concentrated complexant waste, and dilute complexant waste. Using Hanlon's (1997) reported volumes and waste types for leaking single-shell tanks, the inventory lost to the subsurface was calculated for each tank farm. Using the estimated loss volume of 8000 gallons (30 m^3) per tank, the same concentration data were used to calculate the losses during recovery operations in each tank farm. Because of potentially significant differences in the mobility of complexed as opposed to noncomplexed tank wastes, these inventories lost to the ground were calculated for complexed and noncomplexed waste within each tank farm.

The inventories for the single-shell tank farms are shown in Table 3.4. All of the single-shell tank farm related inventories are reported in this table.

3.4.4.3 Single-Shell Tank Farm Residuals

Source inventories for the tank wastes were recently compiled and published in the TWRS EIS (DOE and Ecology 1996). The inventory for the no-action alternative of the TWRS EIS (DOE and Ecology 1996, Figure 2.2.2 in Appendix F) was an estimate of the contents of the tanks, and for the Composite Analysis, it was the basis for estimating residuals to be left in the tanks.

Single-shell tanks were originally constructed in tank farms that contained from 4 to 18 tanks each. In the TWRS EIS, single-shell tanks were aggregated into five tank groups that contain tanks from one or more tank farms in physical proximity to one another. For the Composite Analysis, tank waste sources were modeled on the basis of tank farms. The higher resolution of this approach may allow sources and plumes to be associated directly with individual tank farms. In order to be consistent with the inventories reported in the EIS, the Composite Analysis used the same spreadsheet as for the TWRS EIS tank group inventories. This spreadsheet contained inventory data at the tank-farm scale, enabling the Composite Analysis to generate and apply single-shell tank farm inventories consistent with the tank group inventories previously published.

Regarding the residuals remaining after the tank wastes are recovered, the TWRS EIS (DOE and Ecology 1996) states:

> "... The amount and type of waste that would remain in the tanks after retrieval is uncertain. The Hanford Federal Facility Agreement and Consent Order (Tri-Party Agreement) (Ecology et al. 1994) set a goal of no more than 1 percent residuals and the ex situ alternatives have been developed to attempt to achieve that goal. However, achieving this level of tank waste retrieval may require extraordinary efforts and cost and it may not be practical to achieve 99 percent retrieval. Conversely, the contaminants that are not recovered are likely to be those that are insoluble in water since substantial quantities of water would be used in an attempt to dissolve or suspend the waste in water during retrieval. Since neither of these issues can be resolved, a conservative assumption was made to bound the impacts of the residual waste. For purposes of this analysis it was assumed that 99 percent recovery would be achieved but that the residual would contain 1 percent of all the contaminants including the water soluble contaminants."

As in the TWRS EIS, the Composite Analysis estimate of residual was assumed to be 1% of the original inventory. The original inventory of the no-action alternative was used as the inventory for the Composite Analysis. One percent of each radionuclide was assumed to remain in the tank farms following completion of waste recovery (Table 3.4 or the *Inventory* worksheet of the *Composite Analysis 2.0.xls* Excel™ workbook). As noted above, this assumption is believed conservative because it is likely that the recovery operation will preferentially remove the highly soluble chemical compounds and radionuclides. In general, radionuclides with long decay half-lives and potentially significant geochemical mobility have been shown to contribute significantly to long-term dose. The 1% residual is believed to overestimate the inventories of these radionuclides (i.e., carbon-14, selenium-79, and technetium-99) that remain in the tanks following Hanford Site closure.

In the release model for tank residuals, leachate concentrations from residual tank wastes were defined as a function of nitrate dissolution (i.e., a maximum nitrate concentration of 360 g/L is maintained) with congruent release of all radionuclides. Thus, the nitrate inventory, water infiltration rate, and solubility of nitrate define the time required for nitrate to be leached from residual wastes. All radionuclides were assumed to linearly release over the same time. As in the TWRS EIS (DOE and Ecology 1996), the Composite Analysis was based on the assumption that the single-shell tank structure and tank farm remediation (e.g., stabilization fill and surface barriers) present a high-integrity barrier to release; consequently, the release was delayed for 500 years.

3.4.4.4 Double-Shell Tank Farm Residuals

There are six double-shell tank farms in the 200 Areas at the Hanford Site. The SY Tank Farm contains 3 tanks and is located in 200 West Area. The AN, AP, AW, AY, and AZ tank farms contain 25 tanks and are all located on the eastern side of 200 East Area. The source inventories for the double-shell tank wastes were also recently compiled and published in the TWRS EIS (DOE and Ecology 1996). As for the single-shell tanks, the inventory for the no-action alternative (DOE and Ecology 1995, Table F.2.2.2 in Appendix F) was an estimate of double-shell tank contents and is the basis for estimating residuals to be left in these tanks. Because the double-shell tanks provide an ability to detect leaks in the tank annulus, accidental leaks and losses during waste recovery operations were assumed to not occur. As in the case of the single-shell tanks, a 1% residual was assumed in the double-shell tanks upon completion of waste recovery operations. Therefore, the only assumed release from double-shell tanks in the Composite Analysis was the leaching of a 1% residual. The TWRS EIS inventory spreadsheet (DOE and Ecology 1996) contained the necessary tank farm data for carbon-14 and technetium-99, and 1% of the no-action alternative inventory is employed in this release (Table 3.5 or the *Inventory* worksheet of the *Composite Analysis 2.0.xls* Excel™ workbook). Chlorine-36, selenium-79, and uranium-238 inventories were not included for double-shell tanks in the TWRS EIS. Iodine-129 is reported in the published EIS on a tank-farm-group-basis instead of a tank-farm basis, and, therefore, it was omitted from the Composite Analysis. As in the case of the single-shell tanks, the 1% residual is believed to overestimate the inventories of mobile and long-lived radionuclides in the tanks after completion of waste recovery. As in the case of single-shell tank residuals, nitrate dissolution and congruent release of radionuclides was assumed to occur after the high-integrity structure and remediation delay release for 500 years.

3.4.5 CERCLA Sources

The CERCLA source term in the Composite Analysis included past-practice waste sites that are being addressed under the CERCLA process and inactive sites that are being addressed under RCRA. The ERC is responsible for evaluation and remediation of these sites. For administrative purposes, the waste sites have been grouped into Operable Units (OUs) and are designated as either CERCLA past-practice units or RCRA past-practice units. However, the eventual disposition of these sites is similar and in the Composite Analysis, all past-practice waste sites under the jurisdiction of the ERC were grouped as CERCLA sources. The CERCLA source term does not include past-practice waste sites that are under the jurisdiction of tank farm operations or decontamination and decommissioning.

3.4.5.1 Description of CERCLA Sources

A total of 190 separate CERCLA sources were included in the current iteration of the Composite Analysis. The CERCLA source term includes liquid discharge sites such as cribs, trenches, and ponds. It also includes a few solid waste sites (landfills) and storage tanks. Sites that are not suspected to have received radioactive wastes were eliminated from the Composite Analysis source term. These include septic systems and nonradioactive waste landfills. Although portions of the low-level solid waste burial grounds are considered past-practice units, the source term for pre-1988 solid waste burial grounds were described above (Section 3.4.2.1).

In addition to the 190 CERCLA sites, 151 waste sites and more than 200 "unplanned releases" in the 200 Area Plateau that do not have any documented inventory estimates were identified. These were classified as CERCLA sites, but were not included in this iteration of the Composite Analysis. Most of these waste sites and unplanned releases have very low radionuclide inventories, have already been remediated, or have been included in another source inventory.

3.4.5.2 Assumptions

Only CERCLA sites located on the 200 Area Plateau were included in the source term for the Composite Analysis. It was assumed that past-practice waste sites outside this region, including those in the 100 Areas, 300 Area and 1100 Area, have been or will be remediated to the point where they are not significant sources of cumulative all-pathways dose for interaction with plumes originating from the exclusion zone. Cleanup wastes from CERCLA sites outside the buffer zone will be transported to the ERDF, which is treated as a separate source in the Composite Analysis (Section 3.4.3). The reactor cores from the 100 Areas were also treated as a separate source and are described in Section 3.4.7. Several CERCLA sites that were included in the analysis are outside of the exclusive waste management area. These sites will most likely be remediated, as discussed below. However, for the first iteration of the Composite Analysis, a conservative approach was adopted that treated these sources as being left in place. The sources did not affect the results unless significant levels of contaminants reached the water table within the 1000-year period of analysis. If any of the sources located outside the exclusive waste management area appear as significant contributors to the groundwater pathway, then the assumption that they are left in place will be reexamined in the next iteration of the Composite Analysis.

The assumption that sources outside the central plateau will be remediated and not represent significant sources of radionuclide exposure and dose following site closure was based on goals documented in the *Hanford Strategic Plan* (DOE 1996d). This document presents goals for seven geographic areas. Goals for the four areas that currently contain wastes sites are described below.

- **Reactors on the River**. Remove and/or stabilize spent fuel, surplus facilities, and waste sites to protect groundwater and the Columbia River and to ensure protection of people, the environment, and natural/cultural resources. Pending Congressional action on the Wild and Scenic River designation, use will continue to be restricted; sensitive ecological, cultural, and Native American resources will be protected.

- **Central Plateau**: The 200 Areas and central plateau will be used for the management of nuclear materials and the collection and disposal of waste materials that remain onsite and for other related and compatible uses. Cleanup levels and disposal standards that are consistent with these long-term uses will be established.

- **Central Core.** This area will remain in federal ownership consistent with safety analysis boundaries and continued waste management operations in the 200 Area. These areas will be available for other federal programs or leased for nonfederal uses, consistent with appropriate recognition of cultural and ecosystem values.

- **South 600 Area**. The 300 and 400 Area waste sites, materials, and facilities will be remediated to allow industrial and economic diversification opportunities. The federal government will retain ownership of land in and adjacent to the 300 and 400 Areas, but will lease land for private and public uses to support regional industrial and economic development. Excess land in the 1100 and 3000 Areas will be targeted for transition to nonfederal ownership.

These goals are addressed on a site-by-site basis through RODs for CERCLA sites and closure plans for RCRA sites. Although RODs and closure plans are still pending and cleanup actions have not yet been completed at most sites, the *Hanford Strategic Plan* provides a basis for assuming that no significant sources of radionuclides will remain outside of the central plateau region after site closure.

Some form of remediation was assumed for all significant sources in the 200 Area Plateau. According to DOE (1996i), the strategy for remediation of the 200 Area Plateau is:

> "...to cap waste in place for sites with high levels of contamination, to remove contamination at sites that exhibit high levels of spotty contamination or lower levels of persistent contamination over a broad area, and no action at sites where risks are demonstrated to be acceptable or where natural attenuation (e.g., decay of short-lived radionuclides) is an effective remedy. In general, this approach results in placing engineered barriers at sites located within the 200 Area fenceline and removal actions at sites outside the fenceline (i.e., 200 Area buffer zone). Sites that have mobile contaminants deep in the subsurface and have the potential to impact groundwater, may require some level of treatment (preferably in situ)."

Based on this strategy, it was assumed that wastes within the exclusive waste management area zone will remain in place and be capped to limit water infiltration and recharge. It was also assumed that institutional controls within the exclusive waste management area will remain in place as long as necessary to ensure that barriers and waste materials are not excavated or otherwise disturbed.

Additional assumptions were made regarding the inventories of radionuclides, future groundwater recharge conditions, and the timing of remedial activities such as placement of barriers. These assumptions are discussed in the following sections and in Chapter 4.

3.4.5.3 CERCLA Radionuclide Inventories

There were two primary sources of inventory information for CERCLA site radionuclides, *Waste Site Groupings for 200 Areas Soils Investigations* (DOE 1997b, Table A.1), and *Tank Wastes Discharged Directly to the Soil at the Hanford Site* (Waite 1991). When both reports provided an inventory for a specific site, the higher value was used.

In DOE (1997b, Table A.1), there are 23 waste categories based on the type and concentration of both radioactive and chemical contaminants that are likely to be present in the waste. The report lists 662 waste sites located in central plateau area that are under the jurisdiction of the ERC. Of these, 36 are nonradioactive waste burial grounds, 55 are septic tanks or drain fields that are not suspected of having received any radioactive contaminants, and 30 are burial grounds that are already covered under the low-level burial grounds source term. Of the remaining 541 potential sources, partial inventory information was listed for 184 sites. However, radionuclides reported were limited to cesium-137, strontium-90, total uranium, total plutonium, and americium-241. A secondary data source was a spreadsheet provided by the ERC (Appendix B). This spreadsheet contained inventories for additional radionuclides at many of the 184 sites, and at 6 additional sites, which brought the total number of sites with inventory information to 190.

Appendix C contains a list of those sites without inventory data; the available information on the source, the type of waste (radioactive, chemical, or mixed), the effluent volume, and an evaluation of whether the release constitutes a potentially significant source for the Composite Analysis. It was assumed that sites with some radionuclide information in these data sources were the most significantly contaminated sites and that sites without inventory information were generally less significant sources. However, it is recognized that some sites, particularly those that received waste in the early years of Hanford operations, may have received significant quantities of radionuclides that are not recorded.

The radionuclides most significantly affecting the Composite Analysis results are mobile in the subsurface and have relatively long half-lives. Inventory data for most of these radionuclides are not available for most of the waste sites because they were not commonly measured in waste streams. A strategy based on the use of radionuclide ratios in aged fuel was used to estimate the absent inventories of key mobile nuclides. Thus, the estimated inventories of fission products and actinides are based on inventories of cesium-137, total uranium, and total plutonium, which are usually reported. Some sites were missing the inventory of cesium-137, total uranium, and total plutonium. To calculate the mobile radionuclide inventory, the missing cesium-137, total uranium, and total plutonium inventories were first estimated. This estimate was based on the average ratios of total uranium to cesium-137, total plutonium to cesium-137, or total plutonium to total uranium for other waste sites in the same waste site group defined in DOE (1997b). The average ratios of these species for each waste site group are listed in Table 3.6. The spreadsheet provided by the ERC (see Appendix B) contained reported inventory data for some specific radionuclides in addition to cesium-137, total uranium, and total plutonium for some of the waste sites. To be certain that inventories were not underestimated, the inventories calculated using the methodology described above were compared to the reported inventories listed in the ERC spreadsheet and the maximum values were used.

Waite (1991) reported the type, quantity, and characteristics of wastes associated with the single-shell storage tanks and discharged intentionally to the subsurface at the Hanford Site. Wodrich (1991) also described these wastes and their inventories in a presentation, including those wastes discharged from the single-shell tanks directly to ground through cascade overflow and by pumping wastes to cribs or specific retention trenches. Being limited to facilities that received different forms of tank waste, these estimates of liquid waste volumes and inventories were generated for relatively few of the CERCLA liquid discharge sites. However, these discharges contain potentially significant radionuclide inventories, e.g., 930 Ci of technetium-99 and 1.8 Ci of iodine-129. Based on the Track Radioactive Components (TRAC) model (Jungfleisch 1980, 1983), inventories were assigned to individual cribs and specific retention trenches (see Table 3.7).[a] For those sites that received tank waste discharges, the inventories estimated by Coony[b] were applied because they are higher than inventories reported in the Waste Site Groupings report published by the Environmental Restoration program. Inventories of key nuclides for the CERCLA sites are listed in Table 3.8.

3.4.6 US Ecology Commercial LLW Site

The inventory for the commercial low-level waste disposal site operated by US Ecology was derived from the recently completed site stabilization and closure plan. The inventory is reported by Grant Environmental, Chase Environmental Group, and US Ecology (1996, in Volume II, Attachment 3 of Attachment D, subsection "Source Term" in section "Pathways Analysis Report"). A detailed accounting of inventory is presented in the same document on page 3.6, Table 3.1 and page 3.12, Table 3.7. The key radionuclides inventory of the commercial disposal site was used in the Composite Analysis; it includes inventories for carbon-14, chlorine-36, technetium-99, iodine-129, and uranium (see Table 3.9). Of the more mobile radionuclides thought to be of concern in DOE wastes at Hanford, selenium-79 was the only one for which no data were available in the detailed inventory.

After receiving the site stabilization and closure plan for the commercial LLW disposal site, the State of Washington Department of Health (DOH) decided to complete a State Environmental Policy Act (SEPA) Environmental Impact Statement (EIS) for the site. The DOH has developed its own inventory for the commercial disposal site[c]. Minor differences exist between the DOH and Grant Environmental, Chase Environmental Group, and US Ecology (1996) inventories. One similarity is that selenium-79 is also absent from the DOH inventory. Its absence from the commercial inventory is because it is an inconsequential nuclide in the waste streams accepted at the commercial disposal facility. Where there is

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(a) Inventories were developed by F. M. Coony of Waste Management Federal Services of Hanford. Information was received in two electronic mail messages with attached files sent by Coony to C. T. Kincaid: 1) Subject, "Questions on Crib Releases in the 200 Areas," dated November 5, 1997; 2) Subject, "Tc-99 (and I-129)," dated October 29, 1997.

(b) F. M. Coony is the individual responsible for the SWITS database and Hanford input to the complex-wide integrated database.

(c) From two electronic mail messages; Subject, "Comments for Composite Analysis": 1) from A. H. Thatcher (DOH, Olympia, Washington) to R. D. Hildebrand (DOE-RL), dated February 2, 1998; 2) from M. Dunkelman (DOH, Olympia, Washington) to R. D. Hildebrand (DOE-RL), dated January 28, 1998.

a large discrepancy for a key mobile radionuclide, e.g., chlorine-36, the inventory from the stabilization and closure plan is conservative. However, in one instance the DOH inventory is larger. For carbon-14, which is slightly sorbed, it shows an inventory of 4909 Ci while the stabilization and closure plan inventory shows 3850 Ci. While assigning a higher initial inventory, the DOH assumed 55% of the carbon-14 was biodegradable and that the entire inventory was released through the gas phase to the atmosphere. In the Composite Analysis, the atmospheric pathway contribution was found to be negligible, and the entire inventory was released through the liquid phase to the soil water and aquifer.

The DOH and Grant Environmental, Chase Environmental Group, and US Ecology (1996) differ in their estimates of future inventory. The DOH based their projections on recent disposals (i.e., 1994-1996) and included expected inventories from decommissioning of two commercial power stations in the region, (i.e., Trojan and WNP-2). However, because future disposal inventories are small in comparison to past disposals, the total inventories examined in the Composite Analysis were not significantly different than those that will be examined in the SEPA EIS. For example, the DOH cumulative or total inventories for iodine-129 and technetium-99 are approximately 4% and 2% greater than inventories presented in the stabilization and closure plan.

3.4.7 Graphite Cores from Production Reactors

Alternatives for decommissioning the Hanford production reactors were evaluated in a draft EIS (DOE 1989), and its final supplement (DOE 1992). The ROD (1993) states the preferred alternative is for the surplus production reactors to be disposed in the 200 West Area. The EIS evaluated eight of the nine production reactors; omitting the N Reactor because it was not shutdown when the study was done. The B Reactor was included in the EIS; however, since then, the B Reactor has been declared a national historic monument and may be preserved for future public display at its present location (ROD 1993). Thus, the EIS contains information on seven reactors; C, D, DR, F, H, KE, and KW that will be moved to the plateau when the ROD is implemented.

The source inventories for the seven production reactors were derived from Appendix A of the surplus production reactor EIS (DOE 1989, 1992). Twenty radionuclides were included, including tritium, carbon-14, chlorine-36, technetium-99, and uranium-238. Mobile and long-lived radionuclides of interest in other DOE wastes that were not represented in the graphite cores include selenium-79 and iodine-129. The ERC provided an inventory for the graphite core of the N Reactor.[a] The N Reactor core was assumed to be disposed concurrently with the other seven reactor cores in the 200 West Area. Inventories for each of the reactors are shown in Table 3.10.

(a) The N Reactor inventory was provided by V. G. Edens (from Interoffice Memorandum #042809; Subject, "105 N and 107 N Hazardous Assessment [Inventories]"; sent by R. S. Day to V. G. Edens of Hanford Environmental Restoration Contractor; February 11, 1997).

3.5 Inventory Compilation for Composite Analysis

The inventories from the different waste sites were compiled in an Excel™ workbook. These inventory compilations were compared with others that have been made at the Hanford Site to check for inconsistencies.

3.5.1 The Excel™ Workbook

Inventory information was assembled and made available to subsequent Composite Analysis calculations in an Excel™ workbook called *Composite Analysis.xls*. This workbook includes Excel™ macros to extract inventories from a variety of independent workbooks and adjust the inventory estimates to a common date of 2050. These inventory data were stored in the "Inventory" worksheet included in the *Composite Analysis.xls* workbook. Decay was calculated and did not consider ingrowth in adjusting inventory estimates. The dose contributions of uranium and its progeny were captured in the dose calculation.

Only six elements were explicitly considered in the first iteration of the Composite Analysis (i.e., carbon-14, chlorine-36, selenium-79, technetium -99, iodine-129, and five uranium isotopes and their daughters). Nitrate inventories were also included in the workbook for tank waste residuals, because the tank-waste-residual-release model is based on nitrate dissolution. Where available, inventories were provided for each of the 241 unique source sites. Twenty-five of the source sites were subdivided to distinguish between waste inventories released in different modes or between wastes disposed of at different times (e.g., past tank leaks, future tank sluicing losses).

3.5.2 Multiple Sources of Inventory Data, Inconsistencies in Totals

The inventory for the various sources was assembled from several separate efforts to develop inventories for specific wastes and waste forms. Occurring at different times and under different programs, these separate efforts were not coordinated to provide a single and consistent database for wastes that will reside at the Hanford Site after closure. Consequently, the total inventory examined in the Composite Analysis includes significant inconsistencies. Accordingly, uncertainties with respect to cumulative impact result from the inventory analyzed. Estimates of total inventory and several subtotals are shown in Table 3.11.

Some inventory data are from past TWRS program efforts to define the LLW inventory (Schmittroth et al. 1995). Others were taken from the inventory assembled for the TWRS EIS (DOE and Ecology 1996). Portions were also developed from the Tank Characterization Reports. The TRAC model results, used in the TWRS EIS, were also used as the basis for the Waite (1991) report on tank wastes discharge to the subsurface. Finally, estimates of the abundance of mobile isotopes in some wastes (solids and liquids) were based on the abundance of mobile isotopes in aged fuel, (e.g., fission products were defined by their abundance with respect to cesium in 10-year old fuel). It is apparent that inventory data were developed to satisfy a variety of objectives. Often, conservative estimates were developed for and employed in program specific analyses.

The TWRS program is making an effort to define radionuclide and chemical inventories in the single- and double-shell tanks. This includes the effort by Schmittroth et al. (1995) to define the inventory for immobilized low-activity waste. More recently, Kupfer et al. (1997) and Washenfelder[(a)] have provided a coordinated database for all TWRS and privatization activities including recovery operations, chemical separations, vitrification or immobilization of waste, and disposal. However, not all data are available to fully quantify the chemical separations and immobilization steps that will be undertaken by the privatization vendors.

The Kupfer et al. (1997) and Washenfelder[(a)] work builds on the Agnew et al. (1997) Hanford Defined Waste (HDW) model and the isotope production estimates produced by Watrous and Wootan (1997). The Watrous and Wooten report is an extension of work documented by Schmittroth et al. (1995). The results presented in this sequence of documents differ primarily in how they split the radionuclides (e.g., between recovered metal and waste streams, between precipitated solids in tanks and liquid waste) during the processing steps that follow the production of isotopes in the reactors. Differences between the earlier and more recent data compilations will be discussed below.

For the Composite Analysis, the inventories for past tank leaks and future tank losses were derived from Tank Characterization Reports. Data on radionuclide concentrations in liquid tank wastes were used to estimate the concentrations of key radionuclides lost to the subsurface during leak and slurry loss events. Because data are sparse for the highly mobile and long-lived key nuclides of greatest interest, the approach adopted in the Composite Analysis was to average the contaminant concentration over all single-shell tanks of similar waste over all time. Thus, the history of tank contents was effectively smoothed over all time because of the absence of data on liquid waste characteristics at specific moments.

Inventories for residual wastes remaining in the tanks after recovery operations were based on the published inventories in the TWRS EIS (DOE and Ecology 1996). These inventories were estimated with TRAC simulations that account for waste stream delivery to tanks, subsequent waste routing among the tanks, and processing steps to concentrate the waste or remove specific radionuclides such as cesium-137 and strontium-90. When the Composite Analysis began, the EIS contained the most recently assembled inventory data on a tank farm basis. For some programs (e.g., TWRS), the TRAC model has been replaced by the HDW model of Agnew et al. (1997). The Kupfer et al. (1997) database supersedes that effort and should be employed in future iterations of the Composite Analysis to increase consistency and better quantify uncertainty.

Inventories of key isotopes in tank waste discharged to ground (e.g., to cribs and specific retention trenches) were taken from Waite (1991). This inventory was based on TRAC simulations and represents only a portion of the liquid discharges to the subsurface. The extent to which process knowledge embedded in TRAC is different than that contained in the more recent HDW model (Agnew et al. 1997)

(a) From letter FDH-9757750 dated August 29, 1997 from D. J. Washenfelder (Fluor Daniel Hanford) to J. K. McClusky (DOE), "Contract Number DE-AC06-96RL 13200; Completion of Milestone T24-97-158, Contractor Letter to Department of Energy, Richland Operations Office, Reporting Completion of Standard Inventory Estimates for All Tanks."

needs to be determined. If differences exist, then new and consistent estimates of tank waste inventories discharged to the subsurface are needed to develop and maintain consistency between tank inventories and discharged waste inventories.

For those solid waste burial grounds and liquid discharge sites where the disposal records were limited to the mass of cesium, uranium, and plutonium, an aged-fuel-ratio method was used to estimate the mass of mobile fission products and actinides that might be codisposed with these elements. This method of estimating nonreported isotopes did not take into account process and operational knowledge that could alter the estimate. For example, high cesium and strontium levels in solid waste from B Plant are a result of the separation processes that were used to extract cesium and strontium. These same processes may have also acted to minimize the amount of mobile isotopes in many waste streams. For example, in the B Plant waste, it is expected that the ratio of cesium or strontium to other radionuclides has been increased relative to aged fuel ratios because of the separation process. Thus, high cesium and strontium levels may not imply large inventories of iodine or technetium. Given the significance of B Plant as a source of cesium and strontium activity, estimates of mobile radionuclides in the solid waste burial grounds using the aged-fuel scaling factors may be conservative.

The inventory assigned to the ERDF was based on field data (i.e., maximum measured contamination levels) from environmental restoration sites and the estimated total volume of the ERDF trench. Thus, the ERDF inventory for specific elements such as uranium may be largely independent of reactor operations and chemical separation factors that define the inventory in tanks. However, the ERDF inventories should be related through the Hanford Site inventory to other inventories onsite. For example, the uranium brought onto the Hanford Site for production of special nuclear materials should be accounted for as wastes to remain onsite, as a special nuclear material exported from the Hanford Site, or as a component of high-level or TRU waste to be disposed offsite. Through such a mass balance check the magnitude of the uranium inventory assigned to the ERDF may be called into question, and the decision may be made to improve the ERDF inventory estimate by using more representative estimates of contamination levels in CERCLA cleanup wastes. Certainly, as disposals continue in the ERDF trench, disposal records may provide an alternative inventory for analysis in future iterations of the Composite Analysis.

The inventories for the graphite cores of the production reactors and the commercial LLW disposal site were based on data and models that are substantially independent of the tank waste estimates.

3.5.2.1 Differences in the Kupfer et al. (1997), Agnew et al. (1997), and Schmittroth et al. (1995) Totals

Kupfer et al. (1997) and Agnew et al. (1997) present global estimates of waste inventories in the single- and double-shell tanks. In developing their estimate of the low-level fraction of tank wastes for immobilization and disposal, Schmittroth et al. (1995) present an estimate of total tank wastes in both double- and single-shell tanks. However, significant differences appear in the estimates of key radionuclides carbon-14, technetium-99, and uranium-238 because different split factors were applied in these studies for the chemical processing steps that followed production of isotopes in the reactors.

In the case of carbon-14, the difference may be related to the assumption in the more recent model (Agnew et al. 1997; Kupfer et al. 1997) that all carbon-14 was routed to the tanks. A portion is suspected to have been lost to the atmosphere during fuel dissolution. Differences with regard to technetium-99 are related the assumed amount exported with uranium to other facilities in the DOE complex. Finally, the amount of uranium-238 is similar in Schmittroth et al. (1995) and Kupfer et al. (1997), 296 and 322 Ci, respectively, but different than in Agnew et al. (1997), 906 Ci in tanks. The apparent overprediction of the HDW model (Agnew et al. 1997) for uranium in the tanks is attributed to the use of a conservative factor for the fraction of uranium metal waste that was not recovered.

3.5.2.2 Carbon-14

By far the greatest inventory of carbon-14 at Hanford (42,200 Ci) is in the graphite cores of the production reactors. Significant inventories of carbon-14 are also associated with the ERDF (3800 Ci) and the commercial LLW disposal facilities (3850 Ci).

Significant differences exist between the Schmittroth et al. (1995) estimate of 769 Ci and those of Agnew et al. (1997) and Kupfer et al. (1997), 4910 Ci and 4808 Ci, respectively. Global estimates of carbon-14 by Agnew et al. (1997) and Kupfer et al. (1995) were based on an assumed 100% delivery of carbon-14 in fuel to the waste tanks. Consequently, their estimates of carbon-14 may be high.

Regardless of the inventory in the tanks, the future location of 99% of the tank inventory after chemical separation into high-level and low-activity waste streams and immobilization is not clearly identified. One percent (1%) of the tank inventory is assigned to the ILAW. Ninety-nine percent (99%) is assigned to the immobilized high-level waste. However, the high-level waste may be a vitrified glass waste form and it may not capture volatile iodine isotopes. Furthermore, the integrated database for spent fuel and radioactive waste (ORNL 1997) shows 4.42 Ci of carbon-14 in ILAW and only 0.0911 Ci in high-level waste glass canisters at the Hanford Site following completion of the chemical separation and immobilization campaigns.

The Composite Analysis accounted for 194 Ci of carbon-14 in the tanks, solid waste burial grounds and liquid discharges of DOE wastes. However, considerably more than that was not accounted for if 99% if the carbon-14 in the tanks is not retained in the immobilized high-level waste. Ninety-nine percent of the Schmittroth et al. (1995) inventory is 761 Ci, a factor of 4 more carbon-14 than modeled as remaining at Hanford. Ninety-nine percent of the Agnew et al. (1997) and Kupfer et al. (1997) inventories are approximately 4760 Ci; a factor of 25 more carbon-14 than modeled.

3.5.2.3 Chlorine-36

As with carbon-14, the graphite cores are the dominant source of chlorine-36 at Hanford (302 Ci). In order to investigate the potential significance of chlorine-36 in other Hanford Site wastes, a 1-ppm level of chlorine-35 contamination was introduced in the ORIGEN2 simulations of irradiated fuel. There are no data on the actual chlorine-35 impurity levels in DOE fuel irradiated in the graphite core production

reactors at Hanford. However, it is believed the 1-ppm level is within an order of magnitude of the true value. This level of impurity has been used to forecast the level of chlorine-36 in aged fuel. Fuel ratios and the inventory of cesium-137 were used to build chlorine-36 inventory into inventories for solid waste burial grounds and liquid discharge sites. If significant impacts from chlorine-36 are forecast, it is important to remember they may not be real. If such a forecast results, it will be important to determine chlorine impurity levels in DOE fuels and develop a true estimate of its potential contribution to dose.

3.5.2.4 Iodine-129

Total inventory values for iodine-129 are fairly consistent among the past and present TWRS inventories. However, while ~65 Ci were projected to reside in Hanford tanks, fewer than 11 Ci were accounted for in the Composite Analysis as remaining at the Hanford Site after closure. Of this amount, the majority could reside in the ILAW from the tanks. While little of the highly volatile iodine-129 inventory may remain in the ILAW to bound the effect of iodine-129 dose from this waste form an estimated 10% of the original tank inventory, or 6.6 Ci of iodine-129, was assigned to ILAW. While ~5 Ci of iodine-129 are distributed among the liquid discharge sites and solid waste burial grounds, it is not clear where 90% of the original tank contents (~58 Ci of iodine-129) reside after chemical separation and immobilization of tank wastes. It is assumed that it will be contained in the immobilized high-level waste.

The total inventory is based on the assumption that all iodine-129 was routed to the tanks. Such an assumption neglects losses of iodine to the atmosphere, disposals of iodine to solid waste burial grounds and cribs, and the storage of two silver reactors in the second PUREX tunnel.[a] Kupfer et al. (1997) estimated that 71% of the iodine may have been routed to tanks, and the remainder (i.e., 29%, or ~18 Ci) to the atmosphere or ground.

The volatile character of iodine implies it will not be captured in a vitrified high-level waste and subsequently exported from the Hanford Site. Some may be identified as leaving the site in TRU waste. With this exception, an upper bound for the final disposal of iodine-129 at Hanford could include the entire inventory generated at the Hanford Site (~65 Ci). This is approximately a factor of 6 more iodine-129 than was accounted for in the first iteration of the Composite Analysis.

3.5.2.5 Selenium-79

The global inventories of selenium-79 in the tanks were relatively consistent among the assembled inventories (i.e., Agnew et al. [1997], 773 Ci; Kupfer et al. [1997], 773 Ci; Schmittroth et al. [1995], 1030 Ci). It was assumed the entire selenium-79 inventory in the tanks will be contained in the ILAW (Mann et al. 1997). Fewer than 20 Ci were assigned to the other tank inventories, e.g., tank leaks, solid waste burial grounds, and liquid discharges.

(a) From a letter, dated September 29, 1993, from J. Reddick of Los Alamos Technical Associates, Kennewick, Washington, to D. Washenfelder of Westinghouse Hanford Company, Richland, Washington; Subject: "PUREX and UO3 Plant Inventory Estimates."

It is anticipated that selenium-79 inventories for the Hanford Site will be reduced by a factor of eight in the near future based on a recent update of the decay half-life of this isotope (Kupfer et al. 1997). The significance of selenium-79 as a contributor to dose should decrease proportionately.

3.5.2.6 Technetium-99

The estimates produced by Schmittroth et al. (1995) for the ILAW disposal were used in this analysis to represent the ILAW. Global estimates of tank waste inventory by Agnew et al. (1997) and Kupfer et al. (1997) were not published when the Composite Analysis inventory was assembled. Schmittroth et al. (1995) estimated a total 27,200 Ci of technetium-99 in the tanks. Of that total, 22,300 Ci are to go into ILAW and the remaining 4900 Ci are to go to high-level waste glass. Agnew et al. (1997) and Kupfer et al. (1997) present global estimates of the amount of technetium-99 produced at the Hanford Site and stored in the single- and double-shell tanks. The Agnew et al. (1997) and Kupfer et al. (1997) estimate of 32,600 Ci technetium-99 in the tanks is higher than the Schmittroth et al. (1995) estimate because they decided to show a bounding inventory value and, therefore, neither took into account the technetium-99 exported from the site. Schmittroth et al. (1995) documented that an estimated 20% of the technetium-99 produced at the site was lost from the tank waste. Most of this 5000- to 6000-Ci inventory was coprocessed with the uranium oxide metal and sent offsite.

While the Schmittroth et al. (1995) estimate shows ~5000 Ci of technetium-99 going to high-level waste glass, any technetium-99 produced as a separate waste stream may require special treatment. Privatization contractors that perform the separation and immobilization steps for tank waste may find it advantageous to remove technetium-99 to ensure waste form performance and product acceptance by the DOE. The final disposition of a special waste stream containing technetium-99 is not known. Alternatives include its inclusion in immobilized high-level waste leaving the site, disposal in special packages in solid waste burial grounds, or disposal onsite or offsite as a special waste form.

Based on TRAC model results, it was estimated that liquid discharge sites have received ~930 Ci of technetium-99 (Waite 1991). Based on data in the tank characterization reports for liquid tank wastes, the tanks were estimated to have leaked ~460 Ci and to lose ~470 Ci of technetium-99 during retrieval. Based on the TWRS EIS database (DOE and Ecology 1996) and the assumption of 1% volume remaining following recovery operations, ~320 Ci of technetium-99 will be in tank residuals. Based on aged-fuel ratios and the inventory of cesium, another 325 Ci of technetium-99 are assumed to reside in the solid waste burial grounds. These dispositions, which total ~2500 Ci, are based on a number of different models. While each method of estimating technetium-99 disposal has been useful, not all are consistent.

Ultimately, aside from the ILAW, the 2500 Ci inventory of technetium-99 lost to or disposed in the subsurface environment at the Hanford Site is less than 10% of the total technetium-99 inventory. An effort to generate a fully consistent inventory estimate could yield a lower estimate of losses and disposals. For example, because of its solubility, most of the technetium-99 should be removed from the tanks during the tank waste recovery campaigns, and less than the 320 Ci estimated here should remain in the tank residuals. Similarly, if sluicing methods are used to recover tank wastes, it is likely that contaminant concentrations in sluicing losses from the tanks will be lower than contaminant concentrations in tank wastes. Thus, the estimated 470 Ci of technetium-99 lost during tank waste recovery

operations, that was based on tank waste radionuclide concentrations, would decrease. Finally, the Agnew et al. (1997) model provides an estimate of only 107 Ci of technetium-99 lost in past tank leaks compared to the 460 Ci estimated here. Clearly, a lower inventory of loss and disposal could result from a consistent or best-estimate inventory estimate. However, there is also uncertainty in the future technetium-99 waste streams that private contractors may generate and return to the DOE for disposal.

3.5.2.7 Uranium-238

Kupfer et al. (1997) reconciled the HDW model results for uranium (906 Ci of uranium-238) and tank sample data (322 Ci), and decided in favor of the sample data. The discrepancy among TWRS total inventory estimates of uranium is attributed to the factor used to describe the fraction of metal waste not recovered. However, estimates in Waite (1991) for uranium in tank waste discharges to cribs and specific retention trenches, and estimates provided by Coony[(a)] are much lower than estimates that appear in Agnew et al. (1997). Coony estimated 47.5 Ci of uranium-238 as compared to 1,310 Ci estimated by Agnew et al. (1997). The Agnew et al. (1997) inventory of uranium-238 sent to ground in liquid discharges may also be an overestimate because it is based on the factor assumed for uranium metal recovery.

A clearly unrealistic and high estimate of uranium-238 is included in the ERDF inventory (i.e., 54,300 Ci). This inventory estimate is based on maximum observed uranium-238 concentrations in sediments at CERCLA sites. The composition of uranium in ERDF has the signature of enriched uranium, but this is an artifact of using maximum observed concentrations of uranium isotopes to estimate the total inventory disposed. The commercial LLW disposal facility also contains a considerable inventory of uranium-238 (10,900 Ci).

(a) From an electronic mail message with attached files regarding "Questions on Crib Releases in the 200 Areas." Sent by F. M. Coony of Waste Management Federal Services of Hanford to C. T. Kincaid on November 5, 1997.

Table 3.1. Inventory of Key Radionuclides for the Solid Waste Burial Grounds

	Radionuclide Inventories* in Curies**					
Site Name	**C-14**	**Cl-36**	**I-129**	**Se-79**	**Tc-99**	**U-238**
218-EC-9(a)+	2.29E-03	1.51E-04	1.23E-05	1.84E-04	6.22E-03	0.00E+00
218-EC-9(b)++	2.79E-05	1.83E-06	1.49E-07	2.24E-06	7.57E-05	0.00E+00
218-E-1(b)	1.39E-04	9.15E-06	7.45E-07	1.12E-05	3.77E-04	1.35E-01
218-E-10(b)	7.73E+01	5.08E+00	4.14E-01	6.19E+00	2.10E+02	2.69E-01
218-E-10(a)	1.15E-01	7.58E-03	6.17E-04	9.23E-03	3.13E-01	0.00E+00
218-E-12A(b)	1.24E-03	8.14E-05	6.63E-06	9.92E-05	3.36E-03	3.33E-01
218-E-12B(b)	2.03E+00	1.34E-01	1.09E-02	1.63E-01	5.51E+00	6.57E-02
218-E-12B(a)	1.73E-02	1.14E-03	4.14E-02	1.38E-03	4.69E-02	6.68E-02
218-E-2(b)	3.48E-02	2.29E-03	1.86E-04	2.79E-03	9.44E-02	1.01E-01
218-E-4(b)	1.39E-05	9.15E-07	7.45E-08	1.12E-06	3.77E-05	3.36E-04
218-E-5(b)	1.04E-02	6.86E-04	5.59E-05	8.36E-04	2.83E-02	4.04E-02
218-E-5A(b)	2.30E-02	1.51E-03	1.23E-04	1.84E-03	6.23E-02	4.04E-02
218-E-8(b)	1.39E-05	9.15E-07	7.45E-08	1.12E-06	3.77E-05	6.73E-04
218-W-1(b)	2.78E-04	1.83E-05	1.49E-06	2.23E-05	7.55E-04	2.35E-01
218-W-11(b)	1.39E-07	9.15E-09	7.45E-10	1.12E-08	3.77E-07	0.00E+00
218-W-1A(b)	6.68E-02	4.40E-03	3.58E-04	5.36E-03	1.81E-01	3.03E-01
218-W-2(b)	6.96E-04	4.58E-05	3.72E-06	5.58E-05	1.89E-03	4.71E-01
218-W-2A(b)	3.63E-01	2.39E-02	1.94E-03	2.91E-02	9.84E-01	9.05E-01
218-W-3(b)	1.25E-03	8.24E-05	6.70E-06	1.00E-04	3.40E-03	2.35E+01
218-W-3A(b)	1.99E+01	1.31E+00	1.06E-01	1.59E+00	5.39E+01	1.99E+01
218-W-3A(a)	6.62E-01	4.36E-02	3.68E-03	5.31E-02	2.89E+00	4.23E-01
218-W-3AE(b)	8.15E-01	5.36E-02	4.36E-03	6.53E-02	2.21E+00	8.93E+00
218-W-3AE(a)	1.10E+01	7.25E-01	5.47E-02	8.83E-01	3.58E+01	1.87E+02
218-W-4A(b)	4.61E-03	3.03E-04	2.47E-05	3.70E-04	1.25E-02	1.33E+02
218-W-4B-c(b)	2.35E-01	1.55E-02	1.26E-03	1.88E-02	6.37E-01	1.00E-01
218-W-4B-n(b)	5.13E-01	3.37E-02	5.00E-01	4.11E-02	1.39E+00	0.00E+00
218-W-4B-c(a)	5.68E-02	3.74E-03	3.04E-04	4.55E-03	1.54E-01	0.00E+00
218-W-4C(a)	4.10E+00	9.42E-03	1.13E-02	6.24E-02	9.88E+00	1.39E+02
218-W-4C(b)	2.90E+00	1.25E-02	1.02E-03	1.61E-02	6.07E-01	7.90E-01
218-W-5(b)	4.09E+00	2.73E-03	3.00E-03	3.33E-03	1.13E-01	3.99E+00
218-W-5(a)	1.51E+00	5.09E-02	1.40E-01	6.20E-02	2.77E+00	1.98E+01
218-W-7(b)	5.61E-03	3.69E-04	3.00E-05	4.49E-04	1.52E-02	2.35E-04
218-W-8(b)	1.07E-03	7.05E-05	5.73E-06	8.58E-05	2.91E-03	1.01E-04
218-W-9(b)	1.39E-07	9.15E-09	7.45E-10	1.12E-08	3.77E-07	0.00E+00

* See Appendix A for greater detail in the development of solid waste burial ground inventories.

** Inventories are decayed to a common date of 2050.

\+ (a) refers to waste disposed after September 30, 1988.

++ (b) refers to waste disposed before September 30, 1988.

Table 3.2. Inventory of Key Radionuclides for ERDF

	Radionuclide Inventories* in Curies**					
Site Name	**C-14**	**Cl-36**	**I-129**	**Se-79**	**Tc-99**	**U-238**
ERDF	3.80E+03				6.57E+00	5.43E+04

* Total inventory was calculated using waste volumes for a full six-cell ERDF trench and maximum concentrations reported in the ERDF RI/FS (DOE 1994b). Chlorine, iodine, and selenium values were not reported.

** Inventories are decayed to a common date of 2050.

Table 3.3. Inventory of Key Radionuclides for TWRS Low-Activity Waste

	Radionuclide Inventories* in Curies**					
Site Name	**C-14**	**Cl-36**	**I-129**	**Se-79**	**Tc-99**	**U-238**
TWRS glass grout vault	4.54E-01		3.91E-01	6.07E+01	1.32E+03	1.05E+00
TWRS glass new site	7.24E+00		6.23E+00	9.69E+02	2.10E+04	1.67E+01

* The waste inventory in each site is based on the fraction of waste volume in each site and the total inventory.

** Inventories are decayed to a common date of 2050.

Table 3.4. Inventory of Key Radionuclides for TWRS Single-Shell Tanks

	Radionuclide Inventories in Curies*					
Site Name	**C-14**	**Cl-36**	**I-129**	**Se-79**	**Tc-99**	**U-238**
TK-A-S**	9.43E-02	0.00E+00	2.81E-02	1.68E-02	3.31E+01	7.36E-04
TK-A-L***	1.10E+00	0.00E+00	1.83E-01	1.96E-01	1.25E+02	8.57E-03
TK-A-R+	2.11E+00		1.71E-03	8.33E-02	1.15E+00	1.52E-01
TK-AX-S-1++	8.00E-02	0.00E+00	0.00E+00	0.00E+00	1.68E+00	0.00E+00
TK-AX-S-2	3.14E-02	0.00E+00	9.38E-03	5.60E-03	1.10E+01	2.45E-04
TK-AX-L-1	1.50E-02	0.00E+00	0.00E+00	0.00E+00	0.3156146	0.00E+00
TK-AX-L-2	3.14E-02	0.00E+00	5.14E-03	5.60E-03	3.43E+00	2.45E-04
TK-AX-R-1	2.75E-02		3.90E-05	1.95E-03	2.68E-02	2.13E-03
TK-AX-R-2	1.37E-01		1.95E-04	9.74E-03	1.34E-01	1.07E-02
TK-B-S	5.03E-01	0.00E+00	8.23E-02	8.96E-02	5.48E+01	3.92E-03
TK-B-L	2.12E-01	0.00E+00	3.46E-02	3.77E-02	2.31E+01	1.65E-03
TK-B-R	4.98E+00		2.02E-02	1.01E+00	1.39E+01	3.41E-01
TK-BX-S	3.77E-01	0.00E+00	6.17E-02	6.72E-02	4.11E+01	2.94E-03
TK-BX-L	3.79E-01	0.00E+00	6.20E-02	6.75E-02	4.13E+01	2.96E-03
TK-BX-R	9.18E+00		4.78E-02	2.39E+00	3.28E+01	4.87E-01
TK-BY-S	3.77E-01	0.00E+00	6.17E-02	6.72E-02	4.11E+01	2.94E-03
TK-BY-L	1.61E-01	0.00E+00	2.64E-02	2.88E-02	1.76E+01	1.26E-03
TK-BY-R	2.18E+00		1.76E-02	8.83E-01	1.22E+01	7.93E-01
TK-C-S-1	5.80E-02	0.00E+00	2.16E-03	5.60E-03	3.27E+00	2.35E-04
TK-C-S-2	3.46E-01	0.00E+00	5.66E-02	6.16E-02	3.77E+01	2.70E-03
TK-C-L-1	1.50E-03	0.00E+00	1.80E-04	4.67E-04	2.02E-01	1.96E-05
TK-C-L-2	1.07E-01	0.00E+00	1.75E-02	1.91E-02	1.17E+01	8.35E-04
TK-C-R-1	9.49E-01		3.53E-03	1.68E-01	2.32E+00	3.05E-01
TK-C-R-2	8.79E-01		3.27E-03	1.55E-01	2.15E+00	2.83E-01
TK-S-S	3.14E-01	0.00E+00	5.99E-02	5.60E-02	4.95E+01	2.45E-03
TK-S-L	9.43E-02	0.00E+00	1.54E-02	1.68E-02	1.03E+01	7.36E-04
TK-S-R	3.82E+00		2.38E-02	1.19E+00	1.65E+01	1.82E-01
TK-SX-S-1	5.99E-03	0.00E+00	7.21E-04	1.87E-03	8.08E-01	7.84E-05
TK-SX-S-2	3.46E-01	0.00E+00	6.93E-02	6.16E-02	6.05E+01	2.70E-03
TK-SX-L-2	6.30E-01	0.00E+00	1.06E-01	1.12E-01	7.45E+01	4.92E-03
TK-SX-R-1	1.94E-01		2.17E-03	1.09E-01	1.50E+00	1.69E-02
TK-SX-R-2	1.68E+00		1.88E-02	9.50E-01	1.30E+01	1.47E-01
TK-T-S	5.03E-01	0.00E+00	8.23E-02	8.96E-02	5.48E+01	3.92E-03
TK-T-L	5.28E-01	0.00E+00	8.65E-02	9.41E-02	5.76E+01	4.12E-03
TK-T-R	1.50E-01		5.09E-04	2.57E-02	3.51E-01	8.05E-02
TK-TX-S	1.89E-01	0.00E+00	3.09E-02	3.36E-02	2.06E+01	1.47E-03

Table 3.4. (contd)

	Radionuclide Inventories in Curies*					
Site Name	C-14	Cl-36	I-129	Se-79	Tc-99	U-238
TK-TX-L	2.30E-01	0.00E+00	3.76E-02	4.09E-02	2.51E+01	1.79E-03
TK-TX-R	2.91E+00		1.35E-02	6.76E-01	9.34E+00	1.56E+00
TK-TY-R	4.81E-01		5.34E-03	2.68E-01	3.68E+00	7.78E-02
TK-U-S	4.40E-01	0.00E+00	8.05E-02	7.84E-02	6.32E+01	3.43E-03
TK-U-L	3.99E-01	0.00E+00	6.53E-02	7.11E-02	4.35E+01	3.11E-03
TK-U-R	1.35E-01		1.32E-03	6.52E-02	9.08E-01	3.10E-01

* Inventories are decayed to a common date of 2050.

** "S" refers to sluicing losses during recovery of tank wastes. The inventory is based on an 8,000-gallon-per-tank loss and radionuclide concentrations developed from tank characterization reports.

*** "L" refers to past tank leaks as identified in Hanlon (1997). The inventories are based on leak volumes from Hanlon and radionuclide concentrations developed from tank characterization reports.

\+ "R" refers to residual wastes remaining in tank after tank waste recovery. Inventories are based on 1% of tank farm inventory reported in the TWRS EIS (DOE and Ecology 1996).

++ "1" and "2" refer to complexed and non-complexed waste, respectively.

Table 3.5. Inventory of Key Radionuclides for TWRS Double-Shell Tanks

	Radionuclide Inventory* in Curies**					
Site Name	**C-14**	**Cl-36**	**I-129*** **	**Se-79**	**Tc-99**	**U-238**
TK-AN-R-1+,++	8.28E+00				5.56E+01	
TK-AN-R-2	1.14E+01				7.64E+01	
TK-AP-R-1	1.00E-03				2.63E-01	
TK-AP-R-2	2.80E-02				7.35E+00	
TK-AW-R	2.31E-02				8.38E+00	
TK-AY-R-1	3.57E-04				2.77E+00	
TK-AY-R-2	3.29E-04				2.55E+00	
TK-AZ-R	3.48E+00				2.10E+01	
TK-SY-R-1	6.03E-03				2.75E+01	
TK-SY-R-2	1.98E-03				9.05E+00	

* Chlorine-36, selenium-79, and uranium-238 were not reported in the TWRS EIS (DOE and Ecology 1996).

** Inventories decayed to a common date of 2050.

*** Iodine-129 is reported in the TWRS EIS, but on a tank-farm-group basis, instead of a tank-farm basis. Therefore iodine-129, which has a total inventory of 22.3 Ci (DOE and Ecology 1996) all in double-shell tanks, is not reported here.

\+ "R" refers to residual wastes remaining in the tank after the tank waste recovery. Inventories are based on 1% of the tank farm inventory reported in the TWRS EIS (DOE and Ecology 1996).

++ "1" and "2" refer to complexed and noncomplexed waste, respectively.

Table 3.6. Ratios of Cesium-137, Uranium (Total), and Plutonium (Total) for Waste Site Groups

Waste Site Groups*	U/Pu	U/Cs-137 (g/Ci)	Pu/Cs-137 (g/Ci)	Notes
Group 2	4604	2773		
Groups 3 & 4	5.18		7.19	
Group 5				U, Pu, and Cs-137 reported for all sites
Group 6			0.371	
Group 7		348	9.89	
Group 8		970	31.8	
Group 9	400	101		
Group 10			4.07	
Group 11				U, Pu, and Cs-137 reported for all sites
Groups 12-16		46,200	54.7	
Group 17		66,300		
Group 18			138	
Group 19	1,000			assumed
Groups 21 and 23		21,000	6.08	

* Groups 2 through 23 refer to waste site groups defined in DOE (1997b).

Table 3.7. Inventories of Uranium-238, Technetium-99, and Iodine-129 for Liquid Discharge (216) Sites from the SWITS Database

	Radionuclide Inventories in Curies*		
Site Name	**U-238****	**Tc-99+**	**I-129+**
216-A-1	5.12E-02		
216-A-10	8.09E-02		
216-A-18	4.69E-01		
216-A-19	1.30E+01		
216-A-2	2.60E-02		
216-A-20	1.35E-01		
216-A-21	6.49E-02		
216-A-24	1.66E-02		
216-A-25	4.24E+00		
216-A-27	2.26E-02		
216-A-28	2.11E-01		
216-A-3	5.59E-01		
216-A-30	9.98E-02		
216-A-31	6.99E-03		
216-A-36A	4.83E-02		
216-A-36B	3.99E-02		
216-A-37	1.10E-02		
216-A-37-2	1.73E-02		
216-A-39	0.00E+00		
216-A-4	1.33E-01		
216-A-40	0.00E+00		
216-A-45	2.33E-03		
216-A-5	8.75E-02		
216-A-6	5.49E-02		
216-A-7	2.33E-03		
216-A-8	1.23E-01		
216-A-9	0.00E+00		
216-B-10	3.00E-03		
216-B-10B	0.00E+00		
216-B-11	4.66E-03		
216-B-12	6.96E+00		
216-B-14	7.25E-02	6.44E+00	2.24E-02
216-B-15	3.49E-02	5.20E+00	1.81E-02
216-B-16	1.07E-01	1.67E+01	5.83E-02
216-B-17	1.18E-01	5.65E+00	1.97E-02
216-B-18	7.85E-02	6.44E+00	2.24E-02
216-B-22	1.39E-01	1.19E+00	1.88E-03

Table 3.7. (contd)

	Radionuclide Inventories in Curies[a]		
Site Name	U-238[**]	Tc-99[+]	I-129[+]
216-B-23	5.19E-02	2.88E+00	4.56E-03
216-B-24	8.19E-02	3.33E+00	5.28E-03
216-B-25	5.09E-02	1.47E+00	2.33E-03
216-B-26	1.96E-01	2.48E+01	3.92E-02
216-B-27	1.14E-01	9.04E-01	1.43E-03
216-B-28	9.98E-02	6.22E-01	9.84E-04
216-B-29	1.15E-01	1.53E+00	2.42E-03
216-B-3	2.10E+00		
216-B-30	2.93E-02	8.87E+01	1.40E-01
216-B-31	4.06E-02	7.35E-01	1.16E-03
216-B-32	3.66E-03	3.33E+00	5.28E-03
216-B-33	6.66E-03	7.18E+00	1.14E-02
216-B-34	2.83E-02	4.52E-01	7.16E-04
216-B-35	5.66E-03	1.05E+01	1.66E-02
216-B-36	5.32E-03	1.90E+01	3.01E-02
216-B-37	1.33E-03	7.63E+01	1.21E-01
216-B-38	1.40E-02	1.25E+01	1.98E-02
216-B-39	2.00E-03	1.09E+01	1.72E-02
216-B-40	1.16E-02	8.65E+00	1.37E-02
216-B-41	2.66E-03	2.18E+01	3.45E-02
216-B-42	2.27E-01	2.43E+00	3.85E-03
216-B-43	4.66E-03	7.35E+00	2.56E-02
216-B-44	6.66E-04	1.75E+01	6.08E-02
216-B-45	2.33E-03	3.76E+01	1.31E-01
216-B-46	6.36E-02	5.03E+00	1.75E-02
216-B-47	2.33E-03	3.79E+00	1.32E-02
216-B-48	6.66E-04	1.13E+01	3.94E-02
216-B-49	1.06E-01	1.03E+01	3.58E-02
216-B-5	0.00E+00		
216-B-50	0.00E+00		
216-B-52	9.98E-03	9.04E+00	1.43E-02
216-B-55	2.66E-02		
216-B-57	3.33E-04		
216-B-58	3.00E-03		
216-B-59	0.00E+00		
216-B-60	2.39E-01		
216-B-62	9.98E-03		
216-B-63	1.50E-01		

Table 3.7. (contd)

	Radionuclide Inventories in Curies*		
Site Name	U-238**	Tc-99+	I-129+
216-B-7	6.06E-02	2.43E+00	8.47E-03
216-B-8	1.50E-02	1.13E+00	3.94E-03
216-B-9	1.50E-02		
216-C-1	9.82E-02		
216-C-10	0.00E+00		
216-C-3	1.50E-02		
216-C-4	9.98E-04		
216-C-5	1.80E-02		
216-C-6	0.00E+00		
216-C-7	0.00E+00		
216-C-9	3.33E-04		
216-N-2	0.00E+00		
216-N-3	0.00E+00		
216-N-4	1.66E-03		
216-N-5	0.00E+00		
216-N-6	1.66E-03		
216-N-7	0.00E+00		
216-S-1&2	7.55E-01		
216-S-10	6.72E-02		
216-S-11	6.99E-03		
216-S-12	1.66E-03		
216-S-13	3.03E-02		
216-S-16	1.05E+00		
216-S-17	4.53E-02		
216-S-19	5.19E-02		
216-S-20	1.26E-02		
216-S-21	1.33E-03		
216-S-3	0.00E+00		
216-S-5	9.05E-02		
216-S-6	9.05E-02		
216-S-7	8.62E-01		
216-S-8	6.49E-02		
216-S-9	1.13E-02		
216-T-1	1.66E-03		
216-T-12	1.50E-02		
216-T-14	9.98E-03	1.15E+01	1.83E-02
216-T-15	8.99E-03	2.54E+01	4.03E-02
216-T-16	7.32E-03	1.28E+01	2.03E-02

Table 3.7. (contd)

	Radionuclide Inventories in Curies*		
Site Name	U-238**	Tc-99+	I-129+
216-T-17	6.66E-03	9.15E+00	1.45E-02
216-T-18	8.99E-03	1.36E+00	4.73E-03
216-T-19	3.33E-03	9.89E+00	3.45E-02
216-T-20	1.66E-03		
216-T-21	3.33E-04	9.83E+00	1.56E-02
216-T-22	6.66E-04	4.54E+01	7.19E-02
216-T-23	3.33E-04	3.26E+01	5.16E-02
216-T-24	2.66E-03	3.49E+01	5.52E-02
216-T-25	3.33E-04	2.18E+02	3.45E-01
216-T-26	4.99E-02	4.29E+00	1.50E-02
216-T-27	2.33E-03		
216-T-28	1.30E-01	1.09E+01	3.80E-02
216-T-3	0.00E+00		
216-T-30	1.66E-03		
216-T-32	7.56E-01	5.65E-01	1.97E-03
216-T-33	1.66E-03		
216-T-34	1.33E-03		
216-T-35	1.63E-02		
216-T-36	3.33E-04		
216-T-4	2.32E-01		
216-T-5	1.66E-03	1.75E+00	2.77E-03
216-T-6	7.65E-03		
216-U-10	1.88E+00		
216-U-12	6.77E-01		
216-U-13	0.00E+00		
216-U-15	6.66E-04		
216-U-16	5.99E-03		
216-U-17	3.33E-04		
216-U-3	5.99E-03		
216-U-4A	3.00E-03		
216-U-4B	0.00E+00		
216-U-5	1.21E-01		
216-U-6	1.21E-01		
216-U-8	8.00E+00		
216-W-LWC	6.66E-04		
216-Z-1&2	2.70E-02		
216-Z-10	0.00E+00		
216-Z-12	0.00E+00		

Table 3.7. (contd)

	Radionuclide Inventories in Curies*		
Site Name	U-238**	Tc-99+	I-129+
216-Z-16	0.00E+00		
216-Z-17	0.00E+00		
216-Z-18	0.00E+00		
216-Z-1A	0.00E+00		
216-Z-1A A	0.00E+00		
216-Z-1A B	0.00E+00		
216-Z-1A C	0.00E+00		
216-Z-20	0.00E+00		
216-Z-3	0.00E+00		
216-Z-4	0.00E+00		
216-Z-5	0.00E+00		
216-Z-6	0.00E+00		
216-Z-7	1.66E-03		
216-Z-8	0.00E+00		
216-Z-9	0.00E+00		

* Inventories decayed to a common date of 2050.

** Inventory was developed by F. M. Coony. From an electronic mail message with attached files regarding "Questions on Crib Releases in the 200 Areas." Sent by F. M. Coony of Waste Management Federal Services of Hanford to C. T. Kincaid on November 5, 1997.

\+ Inventories were developed by F. M. Coony. From an electronic mail message with attached files regarding Tc-99 (and I-129). Sent by F. M. Coony of Waste Management Federal Services of Hanford to C. T. Kincaid on October 29, 1997.

Table 3.8. Inventories of Key Radionuclides for CERCLA Sites

	Radionuclide Inventories* in Curies**					
Site Name	**C-14**	**Cl-36**	**I-129**	**Se-79**	**Tc-99**	**U-238**
207-U	6.38E-05	4.2E-06	3.42E-07	5.11E-06	1.73E-04	1.51E-02
216-A-1	2.91E-06	1.91E-07	1.56E-08	2.33E-07	7.89E-06	5.12E-02
216-A-10	5.27E-03	3.47E-04	1.07E-01	4.23E-04	1.43E-02	8.09E-02
216-A-18	2.91E-06	1.91E-07	1.56E-08	2.33E-07	7.89E-06	4.69E-01
216-A-19	2.91E-06	1.91E-07	1.56E-08	2.33E-07	7.89E-06	1.30E+01
216-A-2	9.5E-05	6.25E-06	5.08E-07	7.61E-06	2.58E-04	2.60E-02
216-A-20	2.91E-06	1.91E-07	1.56E-08	2.33E-07	7.89E-06	1.35E-01
216-A-21	5.14E-03	3.38E-04	2.75E-05	4.12E-04	1.40E-02	6.49E-02
216-A-24	1.76E-02	1.15E-03	9.4E-05	1.41E-03	4.76E-02	1.66E-02
216-A-25	1.34E-02	8.79E-04	7.15E-05	1.07E-03	3.63E-02	4.24E+00
216-A-27	2.12E-03	1.40E-04	1.14E-05	1.70E-04	5.76E-03	2.26E-02
216-A-28	1.48E-02	9.75E-04	7.93E-05	1.19E-03	4.02E-02	2.11E-01
216-A-3	2.98E-06	1.96E-07	1.6E-08	2.39E-07	8.09E-06	5.59E-01
216-A-30	7.66E-03	5.04E-04	4.1E-05	6.14E-04	2.08E-02	9.98E-02
216-A-31	5.37E-03	3.53E-04	2.88E-05	4.31E-04	1.46E-02	6.99E-03
216-A-36A/B	7.84E-02	5.16E-03	4.20E-04	6.28E-03	2.13E-01	8.82E-02
216-A-37-1	6.2E-06	4.08E-07	4.26E-03	4.97E-07	1.68E-05	1.10E-02
216-A-37-2	1.34E-05	8.79E-07	7.15E-08	1.07E-06	3.63E-05	1.73E-02
216-A-4	4.54E-04	2.99E-05	2.43E-06	3.64E-05	1.23E-03	1.33E-01
216-A-45	6.35E-07	4.18E-08	1.10E-02	5.09E-08	1.72E-06	2.33E-03
216-A-5	7.93E-04	5.21E-05	4.24E-06	6.35E-05	2.15E-03	8.75E-02
216-A-6	6.88E-03	4.52E-04	3.68E-05	5.51E-04	1.87E-02	5.49E-02
216-A-7	1.51E-04	9.95E-06	8.1E-07	1.21E-05	4.11E-04	2.33E-03
216-A-8	3.42E-02	2.25E-03	1.83E-04	2.74E-03	9.28E-02	1.23E-01
216-A-9	3.05E-04	2E-05	1.63E-06	2.44E-05	8.26E-04	8E-05
216-B-10A	2.63E-05	1.73E-06	1.41E-07	2.11E-06	7.13E-05	3.00E-03
216-B-10B	6.55E-09	4.31E-10	3.51E-11	5.25E-10	1.78E-08	2.23E-06
216-B-11A&B	1.40E-03	9.18E-05	7.47E-06	1.12E-04	3.79E-03	4.66E-03
216-B-12	4.69E-02	3.09E-03	2.51E-04	3.76E-03	1.27E-01	6.96E+00
216-B-14	7.47E-03	4.91E-04	2.24E-02	5.99E-04	6.44E+00	7.25E-02
216-B-15	6.05E-03	3.98E-04	1.81E-02	4.85E-04	5.20E+00	3.49E-02
216-B-16	1.94E-02	1.28E-03	5.83E-02	1.55E-03	1.67E+01	1.07E-01
216-B-17	6.55E-03	4.31E-04	1.97E-02	5.25E-04	5.65E+00	1.18E-01
216-B-18	7.47E-03	4.91E-04	2.24E-02	5.99E-04	6.44E+00	7.85E-02
216-B-19	8.25E-03	5.43E-04	2.48E-02	6.62E-04	7.12E+00	6.06E-02
216-B-20	4.48E-02	2.95E-03	6.12E-02	3.59E-03	3.86E+01	1.17E-01
216-B-21	1.11E-02	7.28E-04	1.51E-02	8.87E-04	9.55E+00	2.25E-01

Table 3.8. (contd)

	Radionuclide Inventories* in Curies**					
Site Name	C-14	Cl-36	I-129	Se-79	Tc-99	U-238
216-B-2-1	6.13E-03	4.03E-04	3.28E-05	4.91E-04	1.66E-02	1.45E+00
216-B-2-2	2.06E-05	1.35E-06	1.1E-07	1.65E-06	5.58E-05	4.88E-03
216-B-23	3.33E-03	2.19E-04	4.56E-03	2.67E-04	2.881494	5.19E-02
216-B-2-3	2.06E-05	1.35E-06	1.1E-07	1.65E-06	5.58E-05	4.88E-03
216-B-24	3.84E-03	2.53E-04	5.28E-03	3.08E-04	3.33E+00	8.19E-02
216-B-25	1.67E-03	1.10E-04	2.33E-03	1.34E-04	1.47E+00	5.09E-02
216-B-26	2.87E-02	1.89E-03	3.92E-02	2.30E-03	2.47E+01	1.96E-01
216-B-27	1.04E-03	6.81E-05	1.43E-03	8.3E-05	9.04E-01	1:14E-01
216-B-28	7.01E-04	4.61E-05	9.84E-04	5.62E-05	6.21E-01	9.98E-02
216-B-29	1.80E-03	1.18E-04	2.42E-03	1.44E-04	1.53E+00	1.15E-01
216-B-3	6.13E-03	4.03E-04	3.28E-05	4.91E-04	1.66E-02	2.10E+00
216-B-30	1.03E-01	6.77E-03	1.40E-01	8.24E-03	8.87E+01	2.93E-02
216-B-31	7.80E-02	5.13E-03	1.16E-03	6.25E-03	7.34E-01	4.06E-02
216-B-32	3.84E-03	2.53E-04	5.28E-03	3.08E-04	3.33E+00	3.66E-03
216-B-33	8.32E-03	5.47E-04	1.14E-02	6.67E-04	7.18E+00	6.66E-03
216-B-34	5.18E-04	3.41E-05	7.16E-04	4.15E-05	4.52E-01	2.83E-02
216-B-35	1.21E-02	7.97E-04	1.66E-02	9.71E-04	1.05E+01	5.66E-03
216-B-36	2.20E-02	1.45E-03	3.01E-02	1.76E-03	1.90E+01	5.32E-03
216-B-37	8.84E-02	5.82E-03	1.21E-01	7.09E-03	7.63E+01	1.33E-03
216-B-38	1.45E-02	9.52E-04	1.98E-02	1.16E-03	1.25E+01	1.40E-02
216-B-39	1.26E-02	8.27E-04	1.72E-02	1.01E-03	1.08E+01	2.00E-03
216-B-40	1.00E-02	6.59E-04	1.37E-02	8.03E-04	8.64E+00	1.16E-02
216-B-41	2.53E-02	1.66E-03	3.45E-02	2.03E-03	2.18E+01	2.66E-03
216-B-42	2.80E-03	1.84E-04	3.85E-03	2.24E-04	2.43E+00	2.27E-01
216-B-43	8.52E-03	5.60E-04	2.56E-02	6.83E-04	7.34E+00	4.66E-03
216-B-44	2.02E-02	1.33E-03	6.08E-02	1.62E-03	1.75E+01	6.66E-04
216-B-45	4.36E-02	2.87E-03	1.31E-01	3.50E-03	3.76E+01	2.33E-03
216-B-46	5.82E-03	3.83E-04	1.75E-02	4.67E-04	5.03E+00	6.36E-02
216-B-47	4.36E-03	2.87E-04	1.32E-02	3.50E-04	3.79E+00	2.33E-03
216-B-48	1.31E-02	8.62E-04	3.94E-02	1.05E-03	1.13E+01	6.66E-04
216-B-49	1.19E-02	7.84E-04	3.58E-02	9.56E-04	1.03E+01	1.06E-01
216-B-5	1.91E-03	1.26E-04	1.02E-05	1.53E-04	5.19E-03	9.52E-03
216-B-50	3.35E-03	2.21E-04	1.8E-05	2.69E-04	9.10E-03	1E-04
216-B-52	1.05E-02	6.89E-04	1.43E-02	8.40E-04	9.04E+00	9.98E-03
216-B-53A	3.66E-06	2.41E-07	1.96E-08	2.94E-07	9.93E-06	7.65E-03
216-B-53B	2.42E-04	1.59E-05	1.3E-06	1.94E-05	6.58E-04	3.00E-03
216-B-54	3.58E-06	2.36E-07	1.92E-08	2.87E-07	9.72E-06	3.00E-03
216-B-55	8.98E-04	5.9E-05	4.8E-06	7.19E-05	2.43E-03	2.66E-02

Table 3.8. (contd)

	Radionuclide Inventories* in Curies**					
Site Name	C-14	Cl-36	I-129	Se-79	Tc-99	U-238
216-B-57	1.48E-02	9.74E-04	7.92E-05	1.19E-03	4.02E-02	3.33E-04
216-B-58	2.88E-04	1.9E-05	1.54E-06	2.31E-05	7.82E-04	3.00E-03
216-B-59	7.86E-07	5.17E-08	4.21E-09	6.3E-08	2.13E-06	1.86E-04
216-B-60	1.70E-02	1.12E-03	9.1E-05	1.36E-03	4.61E-02	2.39E-01
216-B-62	8.84E-03	5.82E-04	4.73E-05	7.09E-04	2.40E-02	9.98E-03
216-B-63	4.09E-05	2.69E-06	2.19E-07	3.28E-06	1.11E-04	1.50E-01
216-B-7A&B	2.83E-03	1.86E-04	8.47E-03	2.27E-04	2.43E+00	6.06E-02
216-B-8	1.30E-03	8.53E-05	3.94E-03	1.04E-04	1.13E+00	1.50E-02
216-B-9	2.57E-04	1.69E-05	1.37E-06	2.06E-05	6.97E-04	1.50E-02
216-C-1	2.98E-06	1.96E-07	1.6E-08	2.39E-07	8.09E-06	9.82E-02
216-C-10	5.6E-06	3.68E-07	3E-08	4.49E-07	1.52E-05	1.68E-05
216-C-3	2.78E-06	1.83E-07	1.49E-08	2.23E-07	7.53E-06	1.50E-02
216-C-4	2.84E-06	1.87E-07	1.52E-08	2.27E-07	7.69E-06	9.98E-04
216-C-5	2.91E-06	1.91E-07	1.56E-08	2.33E-07	7.89E-06	1.80E-02
216-C-6	3.05E-06	2E-07	1.63E-08	2.44E-07	8.26E-06	1E-04
216-C-7	3.5E-06	2.3E-07	1.87E-08	2.8E-07	9.49E-06	3.36E-06
216-C-9	4.61E-05	3.03E-06	2.46E-07	3.69E-06	1.25E-04	3.33E-04
216-N-2	5.14E-06	3.38E-07	2.75E-08	4.12E-07	1.4E-05	1.22E-03
216-N-3	5.77E-06	3.8E-07	3.09E-08	4.63E-07	1.57E-05	1.37E-03
216-N-4	5.33E-06	3.5E-07	2.85E-08	4.27E-07	1.44E-05	1.66E-03
216-N-5	5.77E-06	3.8E-07	3.09E-08	4.63E-07	1.57E-05	1.37E-03
216-N-6	5.33E-06	3.5E-07	2.85E-08	4.27E-07	1.44E-05	1.66E-03
216-N-7	5.77E-06	3.8E-07	3.09E-08	4.63E-07	1.57E-05	1.37E-03
216-S-1&2	7.21E-02	4.74E-03	3.86E-04	5.78E-03	1.95E-01	7.55E-01
216-S-10D	8.12E-05	5.34E-06	4.35E-07	6.51E-06	2.20E-04	6.72E-02
216-S-11	5.37E-05	3.53E-06	2.88E-07	4.31E-06	1.46E-04	6.99E-03
216-S-12	2.84E-05	1.87E-06	1.52E-07	2.28E-06	7.71E-05	1.66E-03
216-S-13	1.81E-04	1.19E-05	9.71E-07	1.45E-05	4.92E-04	3.03E-02
216-S-16P	1.97E-03	1.29E-04	1.05E-05	1.58E-04	5.33E-03	1.05E+00
216-S-17	8.32E-04	5.47E-05	4.45E-06	6.67E-05	2.26E-03	4.53E-02
216-S-19	8.45E-05	5.56E-06	4.52E-07	6.77E-06	2.29E-04	5.19E-02
216-S-20	3.70E-03	2.43E-04	1.98E-05	2.97E-04	1.00E-02	1.26E-02
216-S-21	5.77E-03	3.79E-04	3.09E-05	4.62E-04	1.56E-02	1.33E-03
216-S-22	3.13E-05	2.06E-06	1.68E-07	2.51E-06	8.49E-05	1.68E-05
216-S-23	2.27E-04	1.5E-05	1.22E-06	1.82E-05	6.17E-04	9.75E-05
216-S-25	4.24E-06	2.79E-07	2.27E-08	3.4E-07	1.15E-05	5.56E-02
216-S-26	2.02E-07	1.33E-08	1.08E-09	1.62E-08	5.49E-07	6.89E-05
216-S-3	1.43E-03	9.44E-05	7.68E-06	1.15E-04	3.89E-03	9.75E-05

Table 3.8. (contd)

	Radionuclide Inventories* in Curies**					
Site Name	C-14	Cl-36	I-129	Se-79	Tc-99	U-238
216-S-5	1.73E-03	1.14E-04	9.26E-06	1.39E-04	4.69E-03	9.05E-02
216-S-6	7.53E-03	4.96E-04	4.03E-05	6.04E-04	2.04E-02	9.05E-02
216-S-7	4.61E-02	3.03E-03	2.46E-04	3.69E-03	1.25E-01	8.62E-01
216-S-8	3.22E-04	2.12E-05	1.73E-06	2.58E-05	8.74E-04	6.49E-02
216-S-9	1.90E-02	1.25E-03	1.02E-04	1.52E-03	5.15E-02	1.13E-02
216-T-1	2.54E-06	1.67E-07	1.36E-08	2.03E-07	6.88E-06	1.66E-03
216-T-12	2.84E-04	1.87E-05	1.52E-06	2.28E-05	7.71E-04	1.50E-02
216-T-14	1.34E-02	8.79E-04	1.83E-02	1.07E-03	1.15E+01	9.98E-03
216-T-15	2.95E-02	1.94E-03	4.03E-02	2.36E-03	2.54E+01	8.99E-03
216-T-16	1.49E-02	9.78E-04	2.03E-02	1.19E-03	1.28E+01	7.32E-03
216-T-17	1.06E-02	6.98E-04	1.45E-02	8.51E-04	9.15E+00	6.66E-03
216-T-18	1.59E-03	1.04E-04	4.73E-03	1.27E-04	1.36E+00	8.99E-03
216-T-19	1.15E-03	7.54E-05	3.45E-02	9.19E-05	9.89E+00	3.33E-03
216-T-20	2.88E-05	1.9E-06	1.54E-07	2.31E-06	7.82E-05	1.66E-03
216-T-21	1.14E-02	7.50E-04	0.01557	9.14E-04	9.83E+00	3.33E-04
216-T-22	5.26E-02	3.46E-03	7.19E-02	4.22E-03	4.54E+01	6.66E-04
216-T-23	3.78E-02	2.49E-03	5.16E-02	3.03E-03	3.26E+01	3.33E-04
216-T-24	4.04E-02	2.66E-03	5.52E-02	3.24E-03	3.49E+01	2.66E-03
216-T-25	2.53E-01	1.66E-02	3.45E-01	2.03E-02	2.18E+02	3.33E-04
216-T-26	4.95E-03	3.26E-04	1.50E-02	3.97E-04	4.29E+00	4.99E-02
216-T-27	3.66E-03	2.41E-04	1.96E-05	2.94E-04	9.93E-03	2.33E-03
216-T-28	1.26E-02	8.32E-04	3.80E-02	1.01E-03	1.09E+01	1.30E-01
216-T-3	1.40E-03	9.18E-05	7.47E-06	1.12E-04	3.79E-03	6.95E-03
216-T-32	6.36E-04	4.18E-05	1.97E-03	5.1E-05	5.65E-01	7.56E-01
216-T-33	1.75E-05	1.15E-06	9.36E-08	1.4E-06	4.74E-05	1.66E-03
216-T-34	1.03E-02	6.77E-04	5.5E-05	8.24E-04	2.79E-02	1.33E-03
216-T-35	7.66E-04	5.04E-05	4.1E-06	6.14E-05	2.08E-03	1.63E-02
216-T-36	2.48E-04	1.63E-05	1.33E-06	1.99E-05	6.74E-04	3.33E-04
216-T-4B	4.08E-04	2.68E-05	2.18E-06	3.27E-05	1.11E-03	2.32E-01
216-T-5	2.04E-03	1.34E-04	2.77E-03	1.63E-04	1.75E+00	1.66E-03
216-T-6	7.21E-03	4.74E-04	3.86E-05	5.78E-04	1.95E-02	7.65E-03
216-T-7	1.39E-03	9.14E-05	4.14E-03	1.11E-04	1.19E+00	3.00E-03
216-T-8	2.63E-06	1.73E-07	1.41E-08	2.11E-07	7.13E-06	1.66E-03
216-U-1&2	2.86E-04	1.88E-05	1.53E-06	2.29E-05	7.75E-04	7.02E-01
216-U-10	7.21E-04	4.74E-05	3.86E-06	5.78E-05	1.95E-03	1.88E+00
216-U-12	3.71E-06	2.44E-07	1.98E-08	2.97E-07	1.01E-05	6.77E-01
216-U-13	2.91E-06	1.91E-07	1.56E-08	2.33E-07	7.89E-06	1.20E-04
216-U-15	3.05E-06	2E-07	1.63E-08	2.44E-07	8.26E-06	6.66E-04

Table 3.8. (contd)

	Radionuclide Inventories* in Curies**					
Site Name	C-14	Cl-36	I-129	Se-79	Tc-99	U-238
216-U-16	1.08E-06	7.11E-08	5.79E-09	8.66E-08	2.93E-06	5.99E-03
216-U-17	2.67E-04	1.76E-05	1.43E-06	2.14E-05	7.24E-04	3.33E-04
216-U-3	2.84E-05	1.87E-06	1.52E-07	2.28E-06	7.71E-05	5.99E-03
216-U-4A	1.21E-05	7.97E-07	6.49E-08	9.71E-07	3.29E-05	3.00E-03
216-U-4B	1.29E-05	8.49E-07	6.91E-08	1.03E-06	3.5E-05	4.39E-03
216-U-5	8.57E-03	5.64E-04	4.59E-05	6.87E-04	2.32E-02	1.21E-01
216-U-6	8.57E-03	5.64E-04	4.59E-05	6.87E-04	2.32E-02	1.21E-01
216-U-7	4.37E-04	2.87E-05	2.34E-06	3.5E-05	1.18E-03	4.71E-02
216-U-8	2.98E-06	1.96E-07	1.6E-08	2.39E-07	8.09E-06	8.00E+00
216-Z-1&2	2.62E-06	1.72E-07	1.4E-08	2.1E-07	7.11E-06	2.70E-02
216-Z-10	4.55E-04	2.99E-05	2.44E-06	3.65E-05	1.24E-03	8.71E-05
216-Z-12	3.47E-06	2.28E-07	1.86E-08	2.78E-07	9.42E-06	1.7E-05
216-Z-16	3.42E-05	2.25E-06	1.83E-07	2.74E-06	9.27E-05	1.16E-02
216-Z-17	2.37E-05	1.56E-06	1.27E-07	1.9E-06	6.44E-05	5E-05
216-Z-18	2.09E-01	1.38E-02	1.12E-03	1.68E-02	5.68E-01	4.01E-02
216-Z-1A	1.05E-05	6.89E-07	5.61E-08	8.4E-07	2.84E-05	0.00E+00
216-Z-20	5.66E-06	3.72E-07	3.03E-08	4.54E-07	1.54E-05	1.34E-03
216-Z-3	3.14E-06	2.07E-07	1.68E-08	2.52E-07	8.53E-06	1.7E-05
216-Z-4	2.29E-06	1.51E-07	1.23E-08	1.84E-07	6.22E-06	1.7E-05
216-Z-5	2.36E-04	1.55E-05	1.26E-06	1.89E-05	6.40E-04	1.7E-05
216-Z-6	2.29E-06	1.51E-07	1.23E-08	1.84E-07	6.22E-06	1.7E-05
216-Z-7	1.31E-02	8.62E-04	7.01E-05	1.05E-03	3.55E-02	1.66E-03
216-Z-8	1.82E-05	1.2E-06	9.75E-08	1.46E-06	4.94E-05	3.48E-06
216-Z-9	3.41E-06	2.24E-07	1.82E-08	2.73E-07	9.24E-06	1.7E-05

* Refer to Sections 3.4.5 for a detailed discussion of the development of CERCLA radionuclide inventories.

** Inventories decayed to a common date of 2050.

Table 3.9. Inventory of Key Radionuclides for US Ecology

	Radionuclide Inventories* in Curies**					
Site Name	**C-14**	**Cl-36**	**I-129**	**Se-79+**	**Tc-99**	**U-238**
US Ecology current	3.66E+03	3.44E+01	5.63E+00		6.17E+01	1.08E+04
US Ecology future	1.91E+02	6.00E-02	1.40E-01		3.91E+00	1.21E+02

* Total inventories were taken from the *Site Stabilization and Closure Plan for Low-Level Radioactive Waste Management Facility, US Ecology, Inc., Richland, Washington* (Grant Environmental, Chase Environmental Group, and US Ecology 1996).

** Inventories decayed to a common date of 2050.

\+ The absence of selenium-79 from the commercial low-level waste disposal is a result of commercial waste not having a significant source of this radionuclide.

Table 3.10. Inventory of Key Radionuclides for the Decommissioned Reactor Cores

	Radionuclide Inventories* in Curies**					
Site Name	**C-14**	**Cl-36**	**I-129+**	**Se-79+**	**Tc-99**	**U-238**
C Reactor	4.47E+03	1.20E+01			2.00E-03	4.00E-03
D Reactor	4.27E+03	3.40E+01			2.00E-03	0.00E+00
DR Reactor	3.18E+03	2.60E+01			2.00E-03	0.00E+00
F Reactor	3.68E+03	3.30E+01			2.00E-03	0.00E+00
H Reactor	3.48E+03	1.70E+01			2.00E-03	0.00E+00
KE Reactor	6.95E+03	5.40E+01			3.30E-02	0.00E+00
KW Reactor	6.66E+03	5.20E+01			3.30E-02	0.00E+00
N Reactor	9.49E+03	7.50E+01			3.30E-02	0.00E+00

* Inventories were from Appendix A of the draft EIS *Decommissioning of Eight Surplus Production Reactors at the Hanford Site, Richland, Washington* (DOE 1989) for all reactors except N Reactor. The N Reactor inventory was provided by V. G. Edens (from Interoffice Memorandum #042809; Subject, "105N and 107N Hazardous Assessment [Inventories]"; sent by R. S. Day to V. G. Edens of Hanford Environmental Restoration contractor; February 11, 1997).

** Inventories were decayed to a common date of 2050.

\+ Neither iodine-129 nor selenium-79 were reported in the inventories for the decommissioned reactor cores.

Table 3.11. Summary Table of Inventories Considered in the Composite Analysis

Site Name	Radionuclide Inventories in Curies*					
	C-14	Cl-36	I-129	Se-79	Tc-99	U-238
Agnew** All Tanks	4.78E+03		6.30E+01	7.73E+02	3.26E+04	9.06E+02
Agnew** Cribs	1.24E+02		1.64E+00	2.63E+01	8.68E+02	1.31E+03
Agnew** Leaks	1.44E+01		2.04E-01	1.85E+00	1.07E+02	4.63E-01
Agnew** Total Site	4.91E+03		6.48E+01	8.01E+02	3.35E+04	2.22E+03
Kupfer*** Global Tank Inventories	4.78E+03		6.61E+01	7.73E+02	3.26E+04	3.22E+02
Schmittroth**** Total	7.69E+02		6.61E+01	1.03E+03	2.72E+04	2.96E+02
Total+	5.00E+04	3.45E+02	1.71E+01	1.05E+03	2.49E+04	6.60E+04
Total minus US Ecology	4.62E+04	3.11E+02	1.13E+01	1.05E+03	2.48E+04	5.50E+04
Total minus (cores + US Ecology)	3.95E+03	7.60E+00	1.13E+01	1.05E+03	2.48E+04	5.50E+04
Total minus (cores + US Ecology + ERDF)	1.50E+02	7.60E+00	1.13E+01	1.05E+03	2.48E+04	8.00E+02
TWRS ILAW	7.69E+00	0.00E+00	6.62E+00	1.03E+03	2.23E+04	1.78E+01
TWRS SST Leaks – cmplx++	3.15E-01	0.00E+00	5.99E-02	5.60E-02	5.22E+01	2.45E-03
TWRS SST Leaks – ncmplx+++	4.11E+00	0.00E+00	6.78E-01	7.32E-01	4.59E+02	3.21E-02
TWRS SST Losses - cmplx	1.44E-01	0.00E+00	2.88E-03	7.47E-03	5.76E+00	3.14E-04
TWRS SST Losses - ncmplx	3.52E+00	0.00E+00	6.23E-01	6.27E-01	4.67E+02	2.75E-02
TWRS SST Residuals - cmplx	1.17E+00	0.00E+00	5.74E-03	2.79E-01	3.84E+00	3.24E-01
TWRS SST Residuals - ncmplx	2.86E+01	0.00E+00	1.54E-01	7.70E+00	1.06E+02	4.42E+00
TWRS DST Residuals - cmplx	8.28E+00	0.00E+00	0.00E+00	0.00E+00	8.62E+01	0.00E+00
TWRS DST Residuals - ncmplx	1.49E+01	0.00E+00	0.00E+00	0.00E+00	1.25E+02	0.00E+00
216£ liquid discharges + 241££	3.65E+00	2.40E-01	1.94E+00	2.93E-01	9.37E+02	1.57E+02
218✓ 200 W✓✓ pre-1988	2.89E+01	1.45E+00	6.18E-01	1.77E+00	6.01E+01	1.92E+02
218 200 E pre-1988	7.94E+01	5.22E+00	4.25E-01	6.36E+00	2.15E+02	9.85E-01
218 200 W post-1988	1.74E+01	8.33E-01	2.10E-01	1.07E+00	5.15E+01	3.46E+02
218 200 E post-1988	1.35E-01	8.87E-03	4.21E-02	1.08E-02	3.66E-01	6.68E-02
ERDF	3.80E+03	0.00E+00	0.00E+00	0.00E+00	6.57E+00	5.43E+04
Production Reactor Cores	4.22E+04	3.03E+02	0.00E+00	0.00E+00	1.09E-01	4.00E-03
US Ecology	3.85E+03	3.44E+01	5.77E+00	0.00E+00	6.56E+01	1.09E+04

* Inventories have been decayed to a common date of 2050.

** See Agnew et al. (1997).

*** See Kupfer et al. (1997).

**** See Schmittroth et al. (1995).

\+ Sum of estimated inventories of sites included in the first iteration of the Composite Analysis.

++ cmplx = complexed wastes.

+++ ncmplx = noncomplexed wastes.

£ 216 refers to past-practice liquid disposals.

££ 241 refers to tanks associated with reverse wells.

✓ 218 refers to solid waste burial grounds.

✓✓ W and E refer to the 200 West Area and the 200 East Area, respectively.

4.0 Performance Analysis

The Composite Analysis included calculations for source release, vadose zone transport, groundwater transport, atmospheric transport, and dose for the radionuclides of concern identified in Chapter 3. The performance analysis was completed for each of the existing or planned waste sites with radionuclide inventories within the 200 Area Plateau. This chapter describes the assumptions, implementation, results, and sensitivity analyses associated with each component of the performance analysis. Results from the Composite Analysis are compared with earlier performance assessments conducted for sites within the 200 Area. The results are summarized and compared to the dose limits in Chapter 5.

4.1 Methodology and Results

The performance analysis involved estimating cumulative radionuclide doses from both subsurface and atmospheric pathways. The surface pathway was not considered because surface water transport within the 200 Area Plateau rarely occurs. The points of assessment for the Composite Analysis were located on the Hanford Site between the buffer zone and the Columbia River. The area inside the buffer zone (see Figure 1.4) was excluded from the bulk of this analysis because in current land use plans, this portion of the Hanford Site will be used exclusively for waste management to minimize human exposure (DOE 1996a). Dose impacts inside the buffer zone are shown only for the industrial exposure scenario. Although the atmospheric pathway was included in the analysis, the primary exposure route for contaminants from the Hanford Site was through the groundwater pathway, involving source term release, transport through the vadose zone and groundwater, and exposure from pumping and using the contaminated groundwater in a variety of exposure pathways. The transport and exposure pathways considered in the Composite Analysis are illustrated in Figure 4.1.

Radiological doses from the subsurface transport pathway were analyzed for each source site considered in the Composite Analysis. The radionuclide inventory for each waste site was released to the vadose zone according to its release model. Transport within the vadose zone was estimated with a transient one-dimensional variably saturated vadose zone transport model. Travel times for annual releases of unit mass were defined by arrival of 50% of each unit mass. These travel times were used to translate annual releases from the waste into releases to the water table of the aquifer. The resulting fluxes into the water table were transported in the unconfined aquifer with a transient three-dimensional saturated groundwater transport model. The concentrations in the groundwater plumes for each radionuclide were translated into doses associated with agricultural, residential, recreational, and industrial exposures using dose conversion factors. Doses from the various source locations and various radionuclides were combined to estimate the cumulative dose. Uranium toxicity was also considered in the Composite Analysis.

Radiological doses from the atmospheric pathway only considered releases from the graphite cores of surplus production reactors that are planned to be relocated to the 200 West Area solid waste burial

grounds (ROD 1993) prior to Hanford Site closure. The radionuclide inventory contained in the reactor cores was released based on the atmospheric release model. The doses at different locations were estimated with spatial distribution functions for unit releases and the predicted atmospheric transport developed from historical wind profiles at the Hanford Site.

The sequence of calculations required to estimate the cumulative dose was performed with a suite of software elements that were integrated across two computational environments. These software elements included: 1) an Excel™ workbook; 2) a dynamically linked library version of the Subsurface Transport Over Multiple Phases (STOMP) code (White and Oostrom 1996; White and Oostrom 1997; Nichols et al. 1997); 3) the Coupled Fluid, Energy, and Solute Transport (CFEST-96) code (Gupta 1997); and 4) the ARC-INFO™ Geographic Information System.[a] Elements 1 and 2 were implemented on personal computers running either Windows 95™ or Windows NT™. Elements 3 and 4 were implemented on UNIX workstations. Figure 4.2 illustrates the relationship among the software elements.

The methodologies for calculating source release, vadose zone transport, groundwater transport, atmospheric transport, and cumulative dose are described in the following sections. The key assumptions (e.g., geometry, initial conditions, boundary conditions, and parameters) for each calculation are identified and discussed. The implementation of each model for the base case and the sensitivity analyses are also described.

4.1.1 Source Release Models

Because of the variety of waste sources within the exclusive waste management area that have released to the atmosphere or subsurface environment (or are expected to release in the future), a variety of source release models were used. For the first iteration of the Composite Analysis, seven idealized source release models were applied. Of the seven release models, one was for liquid releases to vadose zone, five were for leaching from various solid waste forms to the vadose zone, and one was for atmospheric releases.

4.1.1.1 Background

Each of the release models in the Composite Analysis involved different assumptions. The assumptions for each of the release models are discussed below. Each source was characterized in terms of its generic waste form type, contaminant inventories, volume, duration of disposal, and geometry to facilitate calculation of release. The liquid source release model was the simplest and the most common. The five models for leaching from solid waste forms are more complex and are discussed in Appendix D which contains a detailed discussion of the conceptual model and mathematical approach for each type of source and the rationale for choosing parameter values in the release model equations. The atmospheric release model followed the approach defined in the draft environmental impact statement (EIS),

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(a) ARC/INFO is a registered trademark of Environmental Systems Research Institute, Inc., Redlands, California.

Decommissioning of Eight Surplus Production Reactors at the Hanford Site, Richland, Washington (DOE 1989) and in the final EIS (DOE 1992). Table 4.1 describes the critical assumptions of the source release models and the likely impact of each assumption on the overall performance analysis.

4.1.1.1.1 Liquid Release Model

In the liquid source release model, contaminants were assumed uniformly distributed in the liquid effluent. Therefore, the remaining fraction of the undecayed inventory was assumed to be equal to the fraction of liquid remaining at any time. Releases were assumed to have occurred uniformly over the period of the specific site's operation and discharge of waste to the vadose zone. Based on the type of disposal facility, different flux rates were used. Once the liquid source enters the soil, it was assumed to move vertically downward through the vadose zone to the water table. Liquid releases were the most common release mechanism in the Composite Analysis and included sources from tank leaks, tank sluicing losses, trenches, ditches, ponds, reverse wells, French drains, and cribs.

4.1.1.1.2 Soil-Debris Release Model

In the soil-debris waste model, wastes are assumed to be mixed with soils. Waste sources included in this model were assumed to be permeable to percolating water. Thus, all surfaces of the waste were assumed to come into contact with percolating water. If contaminant inventories in the source were high enough, leaching of contaminant through the bottom of the source was controlled by the solubility of the contaminant in soil water. Otherwise, leaching was controlled by partitioning the radionuclides between aqueous and sorbed phases. The inventory was assumed to be perfectly mixed throughout the source volume during the entire release period. Assuming perfectly mixed conditions reduced the likelihood that solubility would control the release. The soil-debris model was the second most frequently used release model. It was employed for all the solid waste burial grounds, including the Environmental Restoration Disposal Facility (ERDF) and the commercial low-level waste (LLW) burial ground operated by US Ecology, except those involving grouted waste or high-integrity containers for waste stabilization.

4.1.1.1.3 Cake Release Model

In the cake release model consolidated tank wastes were assumed to be permeable to water and dissolved over time because a major structural component of the waste (in this case nitrate salt) dissolved in the water percolating through the waste form. As the solid waste dissolved at a constant rate controlled by the aqueous solubility of nitrate, the contaminants associated with the dissolved portion of the waste form were assumed to be released into the percolating water congruently at constant rates related to their concentration in the waste form. The cake model was employed for residual wastes remaining in both single-shell and double-shell tanks after tank waste recovery operations have been completed. This release model was applied in Tank Waste Remediation System (TWRS) EIS (DOE and Ecology 1996).

4.1.1.1.4 Glass Release Model

In the glass release model, vitrified wastes are assumed to release contaminants into pore water through corrosion of the glass. For glass, the aqueous permeability was assumed to be sufficiently low such that aqueous transport within the waste form itself was essentially zero. Because of the rectangular box called for in contract specifications and the likelihood of glass fracturing, this waste form was assumed to be roughly cubical in shape. Release was assumed to occur with time by slow dissolution from the exterior surfaces of the glass. All of the contaminants associated with the dissolved portion of the waste form undergo congruent release into the surrounding pore water at rates related to their concentration in the waste form and the overall waste form dissolution rate at the given time. The dissolution rate for vitrified waste was taken from the contract specification as it appears in the interim performance assessment for immobilized low-activity waste (Mann et al. 1997). The glass release model was applied for both of the proposed TWRS glass waste disposal sites.

4.1.1.1.5 Cement Release Model

In the cement release model, the waste form is assumed to have permeability much lower than that of the surrounding soil. The pore space connectivity in the cementitious waste form is sufficiently high to allow contaminant mobility within the waste form by diffusion. Percolating water was assumed to surround this waste form, and contaminants inside the waste form were assumed to diffuse to the outer surface and enter the percolating water. Therefore, overall contaminant release from the source zone was assumed to be controlled by the contaminant's effective diffusion coefficient in the waste form. The cement release model was only used for two soil waste burial grounds that contained cementitious waste forms (e.g., caissons).

4.1.1.1.6 Reactor Block Release Model

In the reactor release model, irradiated solids were assumed to release contaminants into the water percolating past them by unspecified loss processes from the solid matrix and by corrosion of the solid components themselves over time. Because of the absence of information regarding the conceptual and mathematical description of the processes occurring, release of contaminants from the reactor blocks was assumed to be described by rates calculated from experimental leach test data. The reactor block release model was used to simulate release from each of the surplus reactors. This release model was first developed and applied to the reactor blocks in the draft EIS, *Decommissioning of Eight Surplus Production Reactors at the Hanford Site, Richland, Washington* (DOE 1989).

4.1.1.1.7 Atmospheric Release Model

Atmospheric releases were only estimated for tritium and carbon-14 inventories in the surplus production reactor cores that are scheduled to be relocated to the 200 Area West solid waste burial

grounds. The method was based on the same experimental leach rate data used for the reactor block model. This approach was also described in the draft environmental impact statement for the surplus reactors (DOE 1989).

4.1.1.2 Source Term Release Model Implementation

Each waste source considered in the Composite Analysis was categorized as one of the generic waste form types described in the previous section. The inventories of radioactive contaminants for each waste source were compiled (as described in Chapter 3). Models for liquid, solid, and atmospheric releases were implemented separately to facilitate calculations. All three groups of release models were implemented within the *Composite Analysis.xls* Excel™ workbook. Figure 4.3 illustrates the implementation of the release models. Table 4.2 describes each of the primary worksheets in the *Composite Analysis.xls* workbook.

The approach used to estimate the temporal distribution of radionuclide fluxes to the water table does not require any specific implementation of a release model for liquid releases because these releases were assumed to occur uniformly over the release period specified in the *Source Site* worksheet of the *Composite Analysis.xls* Excel™ workbook. Atmospheric releases were estimated independently in the *Composite Analysis.xls* Excel™ workbook. The spatial distributions of unit atmospheric releases were calculated separately and were provided for processing by the geographic information system.

The release models for solid waste forms (soil-debris, cake, glass, cement, and reactor block) were implemented within several worksheets and Excel™ macros in the *Composite Analysis.xls* Excel™ workbook. The *Nuclides & Release Model Data* worksheet provided release parameters (such as fractional release from glass, cement diffusion coefficient, fractional release from reactor) and general nuclide data (such as decay half-life and specific activity) for each nuclide for each of the various solid waste sites. For the soil-debris waste form, the overall volume of the source zone was used to estimate contaminant concentrations from inventories. For cake, glass, and cement waste forms, their actual volumes were used. The release model associated with the reactor block type of waste form did not contain volume and concentration considerations. These and other waste site and waste form geometry data required for the release models were retrieved from the *Source Site* worksheet (see Table 4.3). The chemical classification of the waste stream for each waste site is listed in Table 4.4.

The estimates of the volumetric water content of the source zone and the sorption coefficient required for the soil-debris waste form model were obtained from the *K_d and Release Model Classes* worksheet. The recharge rates and release periods used in the source release and vadose zone transport models are summarized in Table 4.5. For many waste sites, the total inventory was assumed in place at the midpoint in the operational period. For those sites, the second and fourth columns in Table 4.5 represent the midpoint of the disposal or discharge operations and the end of operation. The *MyRelease* macro estimated the annual releases for 1500 years, beginning in 1944 when Hanford Site operations began, and stored these values in the *Temp* worksheet for later integration with results from the vadose zone simulation to achieve water table releases.

The values of some waste site and release model parameters were specific to the conditions at a particular source site. In those cases, where it was believed that reasonable "Hanford Site-specific" values were known, they were used in the calculations. Most waste site and release/transport model parameter values were based on actual data. However, some were based on an assumed similarity in behavior with other radionuclides, and some values were set equal to "default" values when no other information was available.

The source term release models are closely linked to the vadose zone transport models. Results from the combined components of the model are summarized as cumulative release (Ci/yr) to groundwater in the vadose zone transport (see Section 4.1.2.3). The sensitivity of results to the source term release models was investigated by varying the type of release model applied. As in the case of the results from the source-term release models, the results of sensitivity analyses will be summarized in the vadose zone transport (see Section 4.1.2.4).

4.1.2 Vadose Zone Model

Contaminants released from the various Hanford Site waste sources were transported downward through the vadose zone to the water table. The primary mechanism for transport in the vadose zone was water flow in response to gravitational and capillary forces. The radionuclide influx from each waste site release was accounted for in the Composite Analysis. Dry disposals such as the burial grounds, the immobilized low-activity waste (ILAW) disposal, the ERDF, and the reactor cores were placed at the assumed depths of disposal. After the waste disposal operations ceased, transient hydraulic conditions from different surface covers (including revegetation) that affect recharge were represented in the model. Recharge directly from precipitation or snowmelt infiltrates into the vadose zone. The recharge rate varies with operations and the placement of any covers for each of the waste sites. The geology and soils in the vadose zone are heterogeneous. Geochemical conditions in the vadose zone are similarly heterogeneous, with conditions near some waste sources more strongly influenced by the chemical nature of the waste itself. Because of the uncertainty in hydraulic and geochemical properties in the vadose zone, the uncertainties in the vadose zone model itself (DOE 1997a), and because the end states are not well defined for all waste sites at Hanford, vadose zone flow and transport predictions in the Composite Analysis are also uncertain. The data used in the vadose zone model are described in the remainder of this section.

4.1.2.1 Background

The vadose zone was modeled as a stratified one-dimensional column. In the first iteration of the Composite Analysis, it was not appropriate to represent the vadose zone with a multidimensional model because of the large number of waste sites modeled and the limited characterization of the vadose zone. Multidimensional modeling of the vadose zone has been determined to be important and has been performed for some waste forms (Mann et al. 1997; DOE 1997b), but is not practical for the first iteration of the Composite Analysis. The multidimensional effects will be accounted for in detailed modeling of

individual waste sites and used to adjust the recharge rates and cross-sectional areas used for the one-dimensional model in future iterations of the Composite Analysis. Multidimensional modeling will be considered in future iterations of the Composite Analysis as well.

In the remainder of this section, the stratigraphy, hydraulic properties, recharge, and geochemical conditions used in the first iteration of the Composite Analysis are described.

4.1.2.1.1 Stratigraphy

The stratigraphy used in the model was consistent with the major geologic formations found in the vadose zone beneath the 200 Area Plateau and was based on work documented in Thorne and Chamness (1992), Thorne et al. (1993), and Thorne et al. (1994). The geology at each site was defined as a set of strata consistent with nearest available well log. Each of the well logs included location, ground surface elevation, and the thickness of the various major sediment types. A summary of the geologic well logs used in the Composite Analysis appears in Table 4.6.

Seven sediment types and one rock type (basalt) were identified and used to define the stratigraphy at each profile location. The sediment types are: East Hanford Gravel, East Hanford Sand, East Ringold, West Hanford Sand, West Early Palouse, Plio Pleistocene, and West Ringold. The definitions of "east" or "west" were used to distinguish sediment types found only in the 200 East or 200 West Areas, respectively. The East Hanford Gravel also appears in the spreadsheet as Lower East Hanford Gravel, but the same soil moisture characteristics are applied to both. At most, four different sediment types occurred above the basalt at any location. In the vadose zone model, the basalt rock type was regarded as impermeable and was used to define the default bottom of the vadose zone profile. If the water table fell below the top of the basalt, the vadose zone was still assumed to be limited to the basalt surface.

4.1.2.1.2 Hydraulic Characteristics

Modeling water flow and radionuclide transport through the vadose zone required a description of the relationship between moisture content, pressure head, and unsaturated hydraulic conductivity. These relationships, called soil moisture characteristics, are highly nonlinear. In the Composite Analysis, nonhysteretic relationships were assumed for Hanford Site soils because few measurements have been made for Hanford Site soils to characterize hysteresis, and it is believed to be of secondary importance. The hydraulic properties of Hanford Site soils are highly variable, both between the Hanford and Ringold formations and within each of the formations (Khaleel and Freeman 1995).

In the Composite Analysis, different sediment types were used to define the one-dimensional columns beneath the waste sites. The hydraulic properties of the sediment types were assumed to be uniform with each sediment layer. Preferential flow paths in the form of wells and clastic dikes were not considered in the Composite Analysis because use of one-dimensional models can not represent their local influence in a three-dimensional environment. The potential influence of preferential flow paths, especially clastic dikes, have been addressed in the performance assessments for the solid waste burial

grounds (Wood et al. 1995; Wood et al. 1996) and more recently by Ward, Gee, and White (1997). Wood et al. (1995) and Wood et al. (1996) concluded that clastic dikes were insufficiently large and insufficiently continuous to provide a true preferential pathway.

The model of soil hydraulic properties based on the van Genuchten (1980) and Mualem (1976) analytical expressions was used as the basis for the relationships between moisture content, pressure head, and unsaturated hydraulic conductivity. This model has been applied in previous vadose zone studies at the Hanford Site. Parameters for the van Genuchten and Mualem models have been determined by fitting experimental data for Hanford Site sediments to the classic analytic expressions of these models. These results are described in several Hanford Site documents, but the parameters used in the initial iteration of the Composite Analysis were compiled by Khaleel and Freeman (1995).

For the Composite Analysis, unsaturated flow parameters were established for each of the vadose zone sediment types defined above. The sediment types and associated sets of parameters used in the Composite Analysis unsaturated flow modeling are shown in Table 4.7. It should be noted that the laboratory-measured moisture retention and saturated conductivity data in Table 4.7 have been corrected for the gravel fraction (>2 mm) present in the bulk sample.

4.1.2.1.3 Recharge Rates

Initial investigations in the Composite Analysis demonstrated that the significant changes in the recharge rates throughout the 1000-year study period require an analysis of transient vadose zone flow and transport. At the Hanford Site, data on the current distribution of soil moisture and contaminants in the vadose zone at the majority of waste sites are inadequate to define present initial conditions for modeling, so simulations were begun at the initiation of each waste source site's release to the vadose zone. Therefore, initial conditions in the Composite Analysis were based on expected conditions before operations started in the 200 Area; i.e., based on steady-state recharge under natural recharge conditions with no contaminants in the vadose zone. The recharge rate was allowed to vary, representing a range of surface cover conditions, from undisturbed surfaces with natural vegetation, to disturbed surfaces maintained free of vegetation, to engineered surface barriers designed for long-term service.

The current recharge rate into coarse surface sediments maintained free of vegetation was estimated as 75 mm/yr, based on data from a nonvegetated gravel-covered lysimeter on the Hanford Site.[a] For a revegetated site, the recharge rate was estimated by Wood et al. (1996) to drop to 5 mm/yr. If a Hanford Protective Barrier was installed, the recharge was estimated to drop to 0.5 mm/yr (Wing 1994). A variety of end states was proposed for the different waste sites by the different U.S. Department of Energy (DOE) programs queried for information supporting the first iteration of the Composite Analysis. For example, the solid waste burial grounds were assumed to have a long-term surface barrier limiting

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(a) From an electronic mail message dated July 30, 1997 sent by M. J. Fayer, Pacific Northwest National Laboratory to C. T. Kincaid and L.W. Vail, Pacific Northwest National Laboratory; subject, "Recharge in Tank Farms."

annual average recharge to no more than 5 mm/yr while the ERDF trench, TWRS ILAW disposal facility, tank farms, and surplus reactor cores were assumed to employ a Hanford Protective Barrier, with a 0.5-mm/yr recharge rate. Based on guidance from the *Hanford Strategic Plan* (DOE 1996d), it was assumed that liquid disposal sites will be closed in place with surface barriers such as the Hanford Protective Barrier.

Infiltration rates for liquid discharge sites during their active disposal period were estimated based on the type of disposal facility. For ponds, the recharge rate was assumed to be the maximum infiltration rate that sediments beneath the pond would allow under unit gradient conditions, i.e., the saturated hydraulic conductivity of the least conductive geohydrologic unit in the vadose zone profile. For example, infiltration from such facilities in the 200 West Area were governed by the saturated hydraulic conductivity of the Early Palouse sediment, or 3040 cm/yr. For cribs, ditches, specific retention trenches, reverse wells, and French drains that received lesser quantities of liquid discharge, the flux rate was assumed to be one third of the saturated hydraulic conductivity. Thus, the wetted cross section of the one-dimensional column was assumed to be three times that defined by the saturated hydraulic conductivity and the assigned discharge rate of the facility. A reduced flux rate over a larger area was employed to represent the spreading or lateral dispersion that would occur during migration of contaminants in the vadose zone.

4.1.2.1.4 Distribution Coefficients

In the initial iteration of the Composite Analysis, the linear sorption isotherm model was used in transport calculations. This model was selected because it was the only approach for which model parameters (distribution coefficients) were available for a broad range of waste sites and radionuclides. At some waste sites the chemistry of the waste streams disposed to ground at the Hanford Site appreciably altered the geochemistry of the near-field sediments. Such changes in the geochemistry likely altered the sorption properties of the altered sediment. An approach was used in the Composite Analysis that allows the distribution coefficient to vary with depth. Both near-field and far-field[a] distribution coefficients were defined for six waste types (Appendix E) representing the waste chemistries disposed to the subsurface. The waste type is listed for each source site in Table 4.4. The location of the transition from near- to far-field was estimated from information available in post-mortem studies of waste sites (Fecht, Last, and Price 1977).

The depths at which distribution coefficients change were estimated from the maximum penetration depth of beta- and gamma-emitting radionuclides in or adjacent to facilities. These measurements mainly reflect cesium-137 and strontium-90. If measurements were available for a facility, then the measured penetration depth was used. If no measurements were available, then the depth was estimated from measurements at facilities that received the same types of waste. The assumption was made that cesium is essentially mobile to the transition depth and immobile after the transition depth is reached.

(a) "Near-field" and "far-field" are referred to as "high impact" and "intermediate impact" zones in Appendix E.

However, total volume discharged was also examined, and for sites with relatively large discharge volumes, the transition depth was taken to be something less than the maximum depth of measured gamma and beta. The selection of distribution coefficients is discussed in detail in Appendix E.

Assumptions for the vadose zone model, the rationale for the assumptions, and the expected impacts are listed in Table 4.8.

4.1.2.2 Vadose Zone Model Implementation

The vadose zone flow and transport model was implemented within the *Composite Analysis.xls* workbook. Figure 4.3 illustrates implementation of the vadose model in the Excel™ spreadsheet. The STOMP code (White and Oostrom 1996; White and Oostrom 1997; Nichols et al. 1997) was accessed from the workbook to perform the fate and transport portion of the calculation. Implementation of the vadose zone model resulted in estimates of the annual contaminant flux to the water table.

The STOMP code was developed under the Volatile Organic Compounds (VOC) Arid Demonstration Project through the DOE Office of Technology Development (White and Oostrom 1997). STOMP is based on the numerical solution of the three-dimensional Richard's equation for fluid flow and the advection-dispersion equation for contaminant transport. While STOMP is capable of three-dimensional simulations, it is also designed to be efficient in performing one- and two-dimensional simulations. By selecting STOMP for the Composite Analysis, the same code can be used in subsequent iterations, even if dimensionality of the simulations change. The code is based on an integral-volume, finite-difference method and is designed to simulate a wide variety of multidimensional, nonlinear, nonisothermal, and multiphase situations. STOMP was selected for the Composite Analysis because of computational efficiency and flexibility, its prior application to the Hanford Site vadose zone (Ward, Gee, and White 1997), and its thorough documentation (Nichols et al. 1997; White and Oostrom 1997; White and Oostrom 1996). STOMP is a candidate code for future performance assessment simulations in support of the TWRS ILAW.

Vadose zone stratigraphy for the Composite Analysis was defined at nine locations in and near the 200 East Area and at nine locations in and near the 200 West Area (Table 4.6). All but one of the stratigraphic profiles were defined at well locations from the geologic log and supporting information for the well. One of the stratigraphic profiles, labeled 218-W-5, was defined from a suite of wells located around low-level waste (LLW) burial ground 218-W-5. This was the same stratigraphic profile applied in the performance assessment for this burial ground.

Water table elevations for future conditions at each waste site location were calculated with the groundwater flow model. This information was used in the vadose zone transport calculations to define the bottom of the vadose zone. The elevation of the top of the vadose zone at each source was calculated from land surface elevations and depth to the bottom of the source, which was tabulated for each waste site. Because the elevation for the top of the vadose zone at a particular source generally did not match the elevation at the top of the stratigraphic profile applied to that source, an adjustment was made. If the

elevation at the source was less than the top of the stratigraphic column, the portion of the column above that elevation was ignored. If the elevation at the source was greater than the top of the stratigraphic column, then the upper stratigraphic layer thickness was increased to make up the difference. A similar adjustment was made if the bottom of the defined stratigraphic column was below the water table elevation at the source. In this case the thickness of the lowest stratigraphic layer was increased.

Figure 4.4 illustrates the method used to estimate the flux to the water table. The source with an inventory of 6 units was assumed to completely release in three years. Three units leave the source and enter the upper vadose zone in the first year. Two units were assumed to leave the source in the second year, and one unit was assumed to leave the source in the third year. In the STOMP simulation, a single unit was assumed to enter the upper boundary of the simulated domain each year. After 3 years, half of the first unit released was predicted by STOMP to have passed through the lower boundary into the aquifer. The entire mass that was estimated by the release model to enter the vadose zone in the first year was assumed to have transported through the vadose zone and entered the water table at this time. The cumulative release to water table curve illustrates this for each of the three years' releases. Taking the derivative over time of the cumulative release curve provides an estimate of the instantaneous release. If the time between changes in the cumulative release is greater than the time periods used in the CFEST-96 simulations, the instantaneous inputs to CFEST-96 can become sharp peaks.

4.1.2.3 Vadose Zone Model Results

Existing plumes in the unconfined aquifer are the first measure of the expected response of the vadose zone transport model in the Composite Analysis. In an effort to match the response of the vadose zone model to field observation, the mass of the technetium-99 plume in 1996 was compared to the release forecast from liquid discharge sites and past leaks from single-shell tanks. These two types of sources represent the logical origin of existing plumes. For some liquid discharge sites, a considerable volume of waste was discharged over a relatively short period of time. The theory of vadose zone hydraulics implies that infiltration of these wastes into the vadose zone is limited by the saturated hydraulic conductivity of the vadose zone sediments. Because the model is one-dimensional, the least conductive of the sediment layers underlying the discharge site will define the infiltration rate.

Based on the discharge volume and duration for a given facility, and the governing saturated hydraulic conductivity, the cross section of the one-dimensional model was calculated. To account for lateral dispersion or spreading of the contaminant plume in the vadose zone, a sensitivity case examined the effect of increasing the cross section. It was determined that increasing the cross section by a factor of three produced a release by 1996 of 181 Ci of technetium-99. Greater cross sections and larger factors have a diminishing affect on the estimated amount of nuclide breaking through to the water table. Estimates of the observed mass of technetium-99 in the aquifer vary from 15.8 to 37.6 Ci. Use of a sufficiently high factor to cause the estimated release to drop to approximately 37.6 Ci is not reasonable. Therefore, the factor of three was applied to all liquid discharge site releases. Additional adjustments of the technetium-99 release to the aquifer to result in an improved match with the existing plume are described in Section 4.3.

Using the above model, the estimated releases of key mobile radionuclides into the water table are shown in Figures 4.5 through 4.13. The releases are shown for the periods 1940-2150 and 1940-3000. Releases for US Ecology, pre-1988 solid waste burial grounds, post-1988 solid waste burial grounds, tank leaks, tank sluicing losses, and other liquid release are shown. In general, liquid releases arrive first, followed by tank leaks and sluicing losses. Radionuclides leached from pre-1988 solid waste burial grounds and US Ecology arrive later. Finally, post-1988 solid waste burial grounds reach the water table. Primarily because of the surface cover or barrier applied to each, the ERDF waste and TWRS ILAW do not reach the water table within the 1500-year period simulated. All the releases are undecayed estimates for inventories estimated for 2050.

Figures 4.5a and 4.5b show the cumulative release of technetium-99 to the water table for the period 1940 through 2150 and 1940 through 3000. Liquid discharge sites, tank leaks and pre-1988 solid waste burial grounds dominated releases prior to 2150. Some tank sluicing losses also contributed in this period. Initial technetium-99 release from post-1988 solid waste burial grounds began in approximately 2200. Shortly after that, first release occurred from the commercial low-level waste disposal facility.

The bulk of the technetium-99 inventory at the Hanford Site will be encapsulated in the TWRS ILAW. Of the inventory in single- and double-shell tanks, any not in the ILAW will be encapsulated in the immobilized high-activity waste from the tanks and will eventually be shipped to the national high-level waste repository. Of the 1900 Ci assigned to liquid discharges, tank leaks, and tank sluicing losses, and the 275 Ci assigned to pre-1988 solid waste burial grounds, in excess of 1200 Ci of technetium-99 is forecast by the model to be in the unconfined aquifer by 2150. However, the rate at which it is predicted to enter the aquifer is lower than the rate that created the present technetium-99 plumes, and predicted concentrations in groundwater would be lower than in the current plumes.

Figures 4.6a and 4.6b show the cumulative release of iodine-129 to the water table for the period 1940 through 2150 and 1940 through 3000. Liquid discharge sites and tank leaks dominated releases prior to 2150. Tank sluicing losses are a relatively minor contributor to releases by 2150, and remain a minor contributor through the year 3000. Best-estimate distribution coefficients for iodine are small but nonzero, and prevent releases from other disposals of iodine-129 from reaching the water table in the 1000 years following Hanford Site closure.

Of the total inventory of 66 Ci of iodine-129 estimated to be at the Hanford Site, only 4.3 Ci of iodine-129 are included in liquid discharges, tank leaks, tank sluicing losses, and pre-1988 solid waste disposals. Of that, the Composite Analysis projected approximately 0.5 Ci were released to the aquifer by 1996. This compares with an estimate of between 1.2 and 7 Ci based on an integration of field observations. These estimates of iodine-129 in the aquifer are highly dependent on the assumed distribution coefficient for iodine in that they take into account both the aqueous and adsorbed masses of the isotope. Potentially, more significant than the apparent underestimate of existing contamination in the aquifer, is the fact that present and planned disposals account for less than 11 Ci of the total 66 Ci estimated as generated in the production reactors.

Figures 4.7a and 4.7b show the cumulative release of carbon-14 to the water table for the period 1940 through 2150 and 1940 through 3000. Tank leaks and liquid discharge sites dominated releases prior to 2150, and tank sluicing losses are a relatively minor contributor. The best-estimate distribution coefficients for carbon are small but nonzero, and as in the case of iodine, they prevent other disposals from releasing carbon-14 to the water table in the 1000 years following Hanford Site closure.

Nearly 5000 Ci of carbon-14 were estimated to have been generated in the Hanford Site production reactors. However, estimates of the carbon-14 in liquid discharges (3.7 Ci), tank leaks (4.4 Ci), tank sluicing losses (3.7 Ci), pre-1988 solid waste burials (<110 Ci), and post-1988 solid waste burials (<20 Ci) total to a much lower inventory. The estimated solid waste inventories are based on cesium-137 inventory and isotopic ratios in 10-year old fuel, and therefore, are highly uncertain. Clearly, the inventory that was originally generated is not accounted for in estimated current and future disposals. It is important to note that the vast majority of carbon-14 to remain at Hanford Site resides in the graphite cores of the production reactors and the Composite Analysis indicates they do not release to groundwater in the 1000 years following Hanford Site closure.

Figures 4.8a and 4.8b show the cumulative release of chlorine-36 to the water table for the period 1940 through 2150 and 1940 through 3000. Releases from the pre-1988 solid waste burial grounds dominate prior to 2150. Small inventories for chlorine-36 estimated in liquid discharges produce releases that can barely be observed. Both of these sources are hypothetical. They are based on an assumed impurity level of 1 ppm chlorine-35 in fuel irradiated in the production reactors, on cesium-137 levels in disposals, and on isotope ratios in 10-year old fuel. The release of chlorine-36 shown for the commercial LLW disposal site occurs later and is real in the sense that the inventory is based on shipment manifest records. The greatest inventory of chlorine-36 resides in the graphite reactor cores and the Composite Analysis indicates it does not release to groundwater in the 1000 years following Hanford Site closure.

The pronounced steps in the cumulative release curve for chlorine-36 are an artifact of the methodology used to translate releases from waste sources to the water table. The commercial LLW disposal facility operated by US Ecology contains over 82% of the total inventory of sites expected to have any release to the water table within the first 1500 years. Because of the high solubility and low sorption ($K_d = 0$) of chlorine-36, nearly 20% of US Ecology's total chlorine-36 inventory is predicted to have entered the aquifer by 3000.

Figures 4.9a and 4.9b show the cumulative release of selenium-79 to the water table for the period 1940 through 2150 and 1940 through 3000. Pre-1988 solid waste burial grounds dominated releases prior to 2150. Secondary contributions were from tank leaks and liquid discharge sites with a very minor contribution from tank sluicing losses. The high mobility of selenium-79 allows both solid waste and liquid disposals to contribute to the cumulative release.

Selenium-79 generation in the production reactors was estimated at 800 Ci by Agnew et al. (1997) and 1030 Ci by Schmittroth et al. (1995). However, this isotope was only recently identified as potentially significant with respect to long-term dose, and previously was not included in inventory estimates for liquid discharges, leaks, or solid wastes. The isotopic ratio of selenium-79 to cesium-137 in 10-year old fuel was used to estimate the quantity of this isotope in these waste discharges. Accordingly, the significance of sources is directly related to the inventories assigned them. Pre-1988 solid waste burials were assigned ~8.1 Ci, tank leaks were assigned ~0.78 Ci, tank sluicing losses were assigned ~0.63 Ci, liquid discharge sites were assigned ~0.3 Ci, and post-1988 solid waste burial grounds were assigned ~1.1 Ci. The total of these inventories is less than 11 Ci and the Composite Analysis indicates slightly more than 6 Ci release prior to 2150. In an effort to be conservative or bounding with respect to future tank wastes, the TWRS program has assumed the entire inventory of selenium-79 is in the tanks and will be contained in the ILAW. However, if selenium-79 were assumed to be as abundant as other highly mobile radionuclides (e.g., technetium-99) in liquid discharges, then because of its mobility, a greater near-term release of this radionuclide would result.

Figures 4.10a and 4.10b show the cumulative release of uranium-238 to the water table for the periods 1940 through 2150 and 1940 through 3000. The small but nonzero distribution coefficient assigned to uranium for all waste forms was sufficient to retard its migration and result in no release to the water table from solid waste burial grounds or the commercial LLW disposal facility. Liquid discharge sites, especially ponds, are among the largest sources of uranium-238, and the Composite Analysis indicates fewer than 9 Ci released from these liquid discharge sites. These releases are forecast to occur in the next decade. The model did not predict the significant release of uranium from the 216-U-1 and 2 crib site. This is a result of the unique events (e.g., mobilization of deposits, flushing by new crib discharges, preferential flow down an unsealed reverse well) that created this particular release (Baker et al. 1988) compared to the generic approach taken in the Composite Analysis to analyze all key radionuclide disposals and discharges to the environment.

A significant inventory of approximately 54,300 Ci of uranium (total) is assumed to reside in the ERDF, but none is forecast to reach the water table in the next 1000 years. The ERDF is assigned a substantial and perhaps unrealistic inventory of uranium. It is a conservative or bounding inventory estimate based on the maximum observed uranium concentration for contaminated soils or sediments at the Hanford Site, and on the total disposal volume forecast for the ERDF trench.

In addition to the key mobile radionuclides, releases of cobalt-60, americium-241, and neptunium-237 were evaluated for potential release and migration from the vadose zone. Cobalt-60 is of interest because there is an existing plume; however, this radionuclide's short decay half-life greatly diminishes its mass and health impact by the time of Hanford Site closure. While there are no plumes of americium-241 in the aquifer today, this radionuclide's potential mobility in chelated wastes was of interest. Finally, neptunium-237 was included because it is a uranium daughter of some interest and generally appears as a contributor to dose in longer-term assessments (e.g., 10,000 years and beyond).

Figures 4.11a and 4.11b show the cumulative release of cobalt-60 to the water table for the period 1940 through 2150 and 1940 through 3000. The cumulative flux of cobalt-60 is less than 0.004 Ci by 3000. However, inventories are for a decay date 2050. Cobalt has a half-life of 5.27 years, and it experiences significant decay prior to Hanford Site closure. All cobalt-60 released to groundwater is from cribs and specific retention trenches.

Figures 4.12a and 4.12b show the forecast of cumulative release of americium-241 to the water table for the period 1940 through 2150 and 1940 through 3000. Liquid discharge sites that received wastes containing organic complexants and radionuclides were shown to release approximately 130 Ci of americium-241 to the aquifer. The model indicates releases dating back to the 1950s and 1960s, and a cumulative release in 1996 of more than 100 Ci. Americium-241 has not been found in the aquifer. Obviously, this release is being overestimated. Dominant physicochemical processes governing the release, migration, and fate of americium-241 in the presence of organic complexants are not appropriately represented in the release and vadose zone models.

Figures 4.13a and 4.13b show the cumulative release of neptunium-237 to the water table for the period 1940 through 2150 and 1940 through 3000. The Composite Analysis indicates liquid discharge sites release less than 0.012 Ci of neptunium-237 to the groundwater by 3000. Most of the neptunium-237, in excess of 0.01 Ci, was released to the water table by 2010. The model indicated cribs and ponds, notably 216-A-8, 216-A-25, 216-B-3, and 216-B-7A & B were the dominant sources of neptunium.

4.1.2.4 Vadose Zone Model Sensitivity

The sensitivity of the vadose zone model was investigated by varying the cross-sectional areas of the one-dimensional columns, the recharge rates, initial conditions, and distribution coefficients. For liquid discharges a relationship between the cross-sectional area of the column and volume and duration of the discharge was developed. Different area factors were applied to illustrate the sensitivity of the results to this cross-sectional area. Figures 4.14a and b show the sensitivity of the cumulative release to the area factor for all liquid discharge sites (see Tables 4.3 and 4.5 for further description of these sites) releasing technetium-99 for the years 1940 through 2040 and 1940 through 3000. Increasing the area factor (i.e., reducing the recharge rate) delays and reduces the cumulative release to the water table.

The impact of different initial soil moisture conditions, consistent with three different steady recharge rates, on the cumulative flux from liquid and solid waste sites for a $K_d = 0$ was also investigated. Figure 4.15 shows the results of the sensitivity analysis for initial soil moisture profiles consistent with steady recharge rates of 75, 5, and 0.5 mm/yr. A dry site, 218-E-10, and a wet site, 216-B-37, were analyzed based on the inventory estimates of technetium-99 for each site (see Tables 4.3 and 4.5 for further description of these sites). The recharge values used in the Composite Analysis for disturbed and coarse surface sediments maintained free of vegetation, a 2- or more-meter-thick surface barrier with natural vegetation, and a Hanford Protective Barrier were 75, 5, and 0.5 mm/yr, respectively.

For both the dry and wet sites, increasing the initial soil moisture (i.e., higher recharge rate) results in earlier breakthroughs. However, by 2020 any difference in cumulative release as a result of the initial soil moisture condition is undetectable.

The effect of different assumed distribution coefficients was investigated for both liquid and solid waste sites. Figure 4.16 shows the response of the release and vadose zone model to varying distribution coefficients for technetium-99 release from the liquid discharge and solid waste burial sites. (See Tables 4.3 and 4.5 for further description of these sites.) Inventory estimates of technetium-99 for each site were used in this analysis. (Note, the analysis is generic and could use any nuclide. Technetium-99 was not modeled as adsorbed in the environment in any other case in the Composite Analysis.) Hypothetical distribution coefficient values of 0.0, 0.1, and 0.15 mL/g were analyzed. It is important to remember that both sites are subjected to time-varying recharge rates. In both cases, as the distribution coefficient increases, less of the contaminant breaks through. The dry site shows the most profound decrease with no breakthrough estimated for the distribution coefficient of 0.15 mL/g within the 1500-year period simulated. The release from the liquid site is decreased by over three orders of magnitude. The results demonstrate that cumulative releases of adsorbed radionuclides are very sensitive to the selection of the distribution coefficient.

4.1.3 Groundwater Flow Model

The Composite Analysis used an existing three-dimensional numerical model for groundwater flow and solute transport in the Hanford Site unconfined aquifer (Wurstner et al. 1995; Barnett et al. 1997; Cole et al. 1997). This three-dimensional model was developed and enhanced as part of the Hanford Groundwater Project (HGWP) (Thorne and Chamness 1992; Thorne et al. 1993; Thorne et al. 1994; Wurstner et al. 1995; Cole et al. 1997). The three-dimensional model was developed to increase the understanding of future changes in water levels and to enhance predictions of contaminant plume movement being monitored by the HGWP (Cole et al. 1997). Applications and developments made on the HGWP's three-dimensional sitewide model of the Hanford Site unconfined aquifer are routinely reported in the Hanford Site's annual groundwater monitoring reports (e.g., Hartman and Dresel 1997).

The geologic and hydrologic data used in the sitewide model used in this Composite Analysis are discussed and summarized in the conceptual model report by Thorne et al. (1994) and the status report on the three-dimensional model implementation by Wurstner et al. (1995). As discussed in Thorne et al. (1994), the data needed to develop the three-dimensional conceptual model were derived from a variety of previous studies and ongoing Hanford Site investigations, as well as from work conducted specifically to support the sitewide model.

Hydraulic property data were obtained from the results of hydraulic tests documented in Bierschenk (1959); Kipp and Mud (1973); Deju (1974); Lindberg and Bond (1979); Graham et al. (1981); DOE (1988a); Liikala and Aaberg (1988); Thorne and Newcomer (1992); Peterson (1992); Connelly, Ford, and Lindberg (1992); Connelly, Ford, and Borghese (1992); Swanson (1992); Thorne et al. (1993); Connelly (1994); and Swanson (1994). Information was also obtained from new tests and tests that were

previously undocumented. Information on the subsurface geologic framework came primarily from interpreting geologic descriptions of samples acquired during well drilling. These interpretations were based on work by Lindsey, Bjornstad, and Connelly (1991); Lindsey (1992); Lindsey et al. (1992); Lindsey and Jaeger (1993); Lindberg (1993a, 1993b); Hartman and Lindsey (1993); and Swanson (1992) in the 100, 200, and 300 Areas of the Hanford Site, which use the lithofacies units outlined in Lindsey (1991).

Many of the wells used to define the geologic framework were drilled to basalt as part of a study for a proposed nuclear power plant (PSPL 1982). Other information used in defining the top of basalt came from wells drilled for the Basalt Waste Isolation Project (DOE 1988a), which studied the basalts underlying the Hanford Site for disposal of high-level nuclear waste. Approximately 550 wells were used to define the three-dimensional hydrogeologic structure of the unconfined aquifer system. Many of these wells were used to determine the elevation of the top of basalt, and not all have been interpreted over their entire depth. Information on the southern part of the Hanford Site and the Richland area came from studies conducted by the U.S. Geological Survey (Ebbert et al. 1993), from Liikala (1994), and from private well logs filed with the State of Washington Department of Ecology (Ecology). Information on the construction of Hanford Site wells was obtained from Chamness and Merz (1993) and from the Hanford Environmental Information System (HEIS) database.

4.1.3.1 Background

The Hanford Site lies within the Pasco Basin, a structural depression that has accumulated a relatively thick sequence of fluvial, lacustrine, and glacio-fluvial sediments. The geologic and hydrologic data used in the model were summarized in Wurstner et al. (1995) and are based on a number of reports published for the Hanford Site. The Pasco Basin and nearby anticlines and synclines initially developed in the underlying Columbia River Basalt Group, a sequence of continental flood basalts covering more than 160,000 km^2. Overlying the basalt within the Pasco Basin are fluvial and lacustrine sediments of the Ringold Formation and the glacio-fluvial Hanford formation. Together, these sedimentary deposits comprise the Hanford Site unconfined aquifer system. The saturated thickness of this unconfined aquifer system is greater than 61 m in some areas but the thickness decreases and pinches out along the flanks of the basalt ridges. Depth to the groundwater ranges from less than 0.3 m near the Columbia River to more than 100 m near the 200 Areas. Groundwater in this unconfined aquifer system generally flows from recharge areas in the west to the Columbia River to the north and east.

Natural recharge to the unconfined aquifer system occurs from infiltration of 1) runoff from elevated regions along the western boundary of the Hanford Site, 2) spring discharges originating from the confined basalt aquifer system, and 3) precipitation falling across the Hanford Site. Some recharge to the unconfined aquifer also occurs along the Yakima River in the southern portion of the Hanford Site. Natural recharge from runoff and irrigation in the Cold Creek and Dry Creek valleys, upgradient of the Hanford Site, provides a source of groundwater inflow where these valleys enter the area of interest. Areal recharge from precipitation falling on the Hanford Site is highly variable, both spatially and

temporally, and depends on local climate, soil type, and vegetation as discussed in Fayer and Walters (1995). The spatial variability in recharge resulting from the sitewide variation of these controlling parameters is illustrated in Figure 4.17. This figure shows ranges in recharge to make it easier to see the different recharge patterns and to relate them to specific features. For example, note the high recharge in the sand dunes area in the central part of the Hanford Site near the Columbia River. When overlaid on the computational grid, the actual distribution of recharge can vary on a grid-by-grid basis. This same recharge estimate (Figure 4.17) was used in the earlier three-dimensional model development efforts (Wurstner et al. 1995) as well as in the current Composite Analysis. Fayer developed this distribution for the 1979 time period using the same methods discussed in Fayer and Walters (Section 4.4.1, 1995). However, this recharge distribution was based on a different vegetation distribution. The recharge distribution developed in Fayer and Walters (1995) is not appropriate for this analysis because it reflects the effects of a large fire on the vegetation distribution. This altered vegetation distribution was not appropriate for the 1979 time period for which the model was calibrated, nor for long-term future conditions since the Hanford Site is expected to return to more natural vegetation patterns.

The other source of recharge to the unconfined aquifer is artificial recharge from wastewater disposal. Over the past 50 years the large volume of wastewater discharged to disposal facilities at the Hanford Site has significantly affected groundwater flow and contaminant transport in the unconfined aquifer. The volume of artificial recharge has decreased significantly during the past 10 years and is continuing to decrease. The major discharge facilities considered in this analysis are summarized in Wurstner et al. (1995). The major wastewater discharges from both past and future sources are summarized in Cole et al. (1997).

The boundaries for the Hanford Site unconfined aquifer system are the Columbia River to the north and east and basalt ridges on the south and west. The Columbia River represents the regional discharge for the unconfined aquifer. The amount of groundwater discharging to the river at any location and time is a function of the local hydraulic gradient and the local aquifer properties (specifically the hydraulic conductivity and saturated thickness). The hydraulic gradient is highly variable at any given time, since it is affected directly by the river stage which changes on a seasonal basis in response to precipitation and temperatures within the entire Columbia River basin upstream of the Hanford Site. The river stage, and thus hydraulic gradient, are also affected by weekly and daily changes in river flows at dams on the river, as determined by electric power generation needs, fisheries resources management, and other dam operations.

Hydraulic properties important to the conceptual model include both horizontal and vertical hydraulic conductivity, storativity, and specific yield. To apply a numerical model, the distribution of these parameters must be specified for each hydrogeologic unit. Hydraulic properties have been measured for the unconfined aquifer during pumping tests and from laboratory permeability tests. The results of these tests have been documented in published and unpublished reports over the past 50 years and in more recent summaries (DOE 1988a; Thorne and Newcomer 1992). As indicated in these documents, the quality of results from aquifer tests at the Hanford Site varies widely and has been affected by both aquifer conditions and analysis procedures. Thorne and Newcomer (1992) and Wurstner et al.

(1995) analyzed the aquifer tests, many of which were single-well pumping tests, and selected the set of aquifer transmissivity calibration data used in the two-dimensional inverse model. The locations of wells that were tested to provide hydraulic properties used for model calibration are illustrated in Figure 4.18. The values illustrated in the figure are aquifer test interpretations of transmissivity in m^2/d.

The model of the unconfined aquifer system was calibrated to match 1979 water-table conditions. This time period was assumed by Jacobson and Freshley (1990) to approximate steady-state conditions during Hanford Site operations based on the fact the well hydrographs were steady and site discharges were relatively constant during this time period.

Key assumptions made for development of the groundwater flow model are listed in Table 4.9.

4.1.3.2 Groundwater Flow Model Selection, Chronology, and Implementation

The three-dimensional groundwater flow and transport model developed for the Hanford Groundwater Project and used in the Composite Analysis was implemented numerically using the CFEST code (Gupta et al. 1987; Cole, Yabusaki, and Kincaid 1988; Gupta 1997). The CFEST code was originally designed to support the radioactive waste repository investigations under DOE's Civilian Radioactive Waste Management Program (Gupta et al. 1987). It has also been effectively used by the chemical waste management community for conducting exposure assessments, evaluating remediation alternatives, and designing extraction and control systems for aquifers.

Selection. Descriptions of the capabilities and approach used in the CFEST code and its selection for the Hanford Groundwater Project are included in Evans et al. (1988) and Wurstner et al. (1995). The chronology in the continuing development of the Pacific Northwest National Laboratory (PNNL) sitewide model of the unconfined aquifer is outlined below. CFEST is an approved code for working on Hanford Federal Facility Agreement and Consent Order (also known as the Tri-Party Agreement; Ecology, EPA, and DOE 1989) milestones related to risk assessment (DOE 1991). The CFEST software library was extensively tested and brought under strict software quality assurance/quality control procedures by the Office of Nuclear Waste Isolation (ONWI) when it was developed by ONWI for DOE's Civilian Radioactive Waste Management Program. The supercomputer version (CFEST-SC), developed to run on all major UNIX work stations (Cole, Yabusaki, and Kincaid 1988), was used for all flow and transport modeling prior to FY 1997. In FY 1997, the refinement of sitewide three-dimensional model continued with its application to contaminant transport of selected contaminant plumes (Cole et al. 1997). An updated version of the CFEST code called CFEST96 (Gupta 1997) was used in this effort and in the Composite Analysis.

Composite Analysis results from CFEST are graphically displayed using the ARC/INFO® geographic information system (GIS). The ARC/INFO® GIS package is also used to store fundamental hydrogeologic data and information used to represent the three-dimensional conceptual model and to

construct the three-dimensional numerical model. The three-dimensional visualization software package known as EarthVision®[a] is used to manipulate hydrogeologic data for the conceptual model.

Chronology. Summarizing from the chronology discussed in Wurstner et al. (1995), a sitewide flow and transport model has been under continuous development by Pacific Northwest National Laboratory staff since the early 1960s as part of PNNL's continuing involvement the Hanford Site's groundwater monitoring efforts. The sitewide flow model and transport model capability has been and continues to be refined and updated as additional information is gathered and as conditions and application needs change at the Hanford Site. Pacific Northwest National Laboratory's Hanford Site unconfined aquifer model consists of a conceptual model and database that defines current system understanding.

Early flow models were two dimensional (i.e., the Variable Thickness Transient [VTT] code, Kipp et al. 1972) and transport modeling, depending on the application, was either of the advective type (i.e., the Hanford Pathline Calculation code [Friedrichs, Cole, and Arnett 1977]); quasi-three-dimensional particle tracking type (i.e., the Multicomponent Mass Transport [MMT] code [Alhstrom et al. 1977]); or multiple streamtube type (i.e., the TRANSS code [Simmons, Kincaid, and Resienauer 1986]). Early flow model calibration was carried out using a streamtube approach that used available field measurements of transmissivity, river stage, disposal rates to ground, and head in an iterative approach to determine the Hanford Site unconfined aquifer transmissivity distribution (Transmissivity Iterative Calculation Routine [Cearlock, Kipp, and Friedrichs 1975]). Applications of the VTT, MMT, and TRANSS codes at the Hanford Site are described by Freshley and Graham (1988).

In the mid 1980s, the CFEST code was selected for upgrading of Pacific Northwest National Laboratory's two-dimensional modeling capability. CFEST has been used to model the Hanford Site and a number of other sites in three dimensions (Dove et al. 1982; Cole et al. 1984; Gale et al. 1987; Foley et al. 1995). Evans et al. (1988), in a Hanford Site groundwater monitoring report for 1987, discuss selection of CFEST code for application to modeling flow and transport in the Hanford Site's unconfined aquifer.

Initial flow modeling with the CFEST code was two-dimensional as it had been with the previous VTT code. New data were used to recalibrate the CFEST two-dimensional groundwater flow model of the Hanford Site unconfined aquifer. A steady-state finite-element inverse calibration method developed by Neuman and Yakowitz (1979) and modified by Jacobson (1985) was used in this effort. All available information on aquifer hydraulic properties (e.g., transmissivities), hydraulic heads, boundary conditions, and discharges to and withdrawals from the aquifer were included in this inverse calibration. Initial inverse calibration efforts are described by Evans et al. (1988), final calibration results are described by Jacobson and Freshley (1990), and the calibrated two-dimensional model of the unconfined aquifer is described in Wurstner and Devary (1993).

(a) EarthVision is a registered trademark of Dynamic Graphics, Inc., Alameda, California.

Two-dimensional flow models used extensively at the Hanford Site prior to cessation of disposal operations were generally adequate for predicting aquifer head changes and directions of groundwater flow. This is because groundwater levels were somewhat stable through time across the Hanford Site. However, in the early 1990s it was recognized that a three-dimensional model was needed for accurate calculation of future aquifer head changes, directions of groundwater flow, mass transport, and predictions of contaminant concentrations. The three-dimensional model was needed because there is significant vertical heterogeneity in the unconfined aquifer and the water table is dropping over most of the Hanford Site in response to cessation of large liquid disposals to ground. Development of a three-dimensional model began in 1992 (Thorne and Chamness 1992) and was completed in 1995 (Wurstner et al. 1995). In the interpretation of the geohydrology of the Hanford Site unconfined aquifer, Thorne et al. (1994) indicate that it is composed of alternating series of transmissive units that are separated from each other in most places by less transmissive or mud units. Accounting for this vertical heterogeneity is particularly important for unconfined aquifer predictions at the Hanford Site as future water-table changes result in the dewatering of hydrogeologic layers. The water table is near the contact between the Hanford formation and the underlying, and much less permeable, Ringold Formation over a large part of the Hanford Site. Water level declines caused by decreased discharge at disposal facilities is causing and will continue to cause dewatering of the highly permeable Hanford formation sediments in some areas (Wurstner and Freshley 1994). This may result in aquifer transmissivity changes of an order of magnitude or more that would not be properly accounted for by two-dimensional flow and transport models that average vertical properties at each spatial location. As a result, changes in groundwater levels, groundwater flow direction, and contaminant transport can not be accurately simulated by a two-dimensional model because the three-dimensional routing of groundwater flow and contaminant mass resulting from the vertical heterogeneity can not be properly accounted for. These three-dimensional effects are especially important to the Composite Analysis because the purpose of a composite analysis is to add together different plumes by accounting for the superposition of plumes from different sources through time. Changes along the migrating front of desaturating sediments can provide the means for plumes emanating from different places and at different times to composite. Such issues can not be properly addressed by a two-dimensional model or even a two- or three-layer, three-dimensional model because there is no ability (two-dimensional model) or limited ability (simple three-dimensional model) for one plume to migrate under another.

The initial three-dimensional model of the Hanford Site unconfined aquifer (Wurstner et al. 1995) was calibrated in a two-step process. In the first step the two-dimensional model was recalibrated with a steady-state, statistical inverse method implemented with the CFEST-INV computer code Devary (1987). The two-dimensional transmissivity distribution from this inverse modeling was preserved during the calibration of the three-dimensional model as is described in Wurstner et al. (1995).

The final improvements and calibration of the Pacific Northwest National Laboratory sitewide model used in this Composite Analysis were carried out during FY 1996 and FY 1997 as part of the HGWP. The purpose of this effort was to assist the HGWP in interpreting monitoring data; to investigate contaminant mass transport issues and evaluate the future movement of existing contaminant plumes; and to identify and quantify potential groundwater quality problems for onsite and offsite use. The report

on this effort (Cole et al. 1997) describes the improvements to the three-dimensional model, the model recalibration, and the application of the model to predict the future transport of existing contaminant plumes in the unconfined aquifer. The Cole et al. (1997) report presents predicted changes in transient-flow conditions in the unconfined aquifer to the year 4000. This provided the hydrologic basis for simulating migration of existing contaminant plumes presented in the Cole et al. (1997) report as well as the future contaminant plume migration simulated as part of the Composite Analysis. The contaminant migration results used in the Composite Analysis that are described in the Cole et al. (1997) report include: the transport of the tritium plume resulting from future operations of the State-Approved Land Disposal Site (SALDS), and the transport of the existing tritium, iodine-129, technetium-99, uranium, and strontium-90 plumes originating from the 200 Areas. Tritium plume migration resulting from future operations of the SALDS is presented in more detail in Barnett et al. (1997).

Implementation. The lateral extent and relationships between the nine hydrogeologic units of the Ringold Formation and Hanford formation determined to be sufficient to adequately represent the unconfined aquifer were defined by determining geologic contacts between these layers at as many wells as possible. These interpreted distributions and thicknesses were integrated into EarthVision®, which was used to construct a database for formulation of the three-dimensional Hanford Site conceptual model. The resulting numerical model contains nine hydrogeologic units above the top of the underlying basalt. A brief summary of each of these units, based on descriptions in Wurstner et al. (1995), is provided in Table 4.10.

A depiction of the surface finite-element grid and boundary conditions used in the three dimensional flow (and transport) model is illustrated in Figure 4.19. The finite-element grid was designed for the Composite Analysis to increase the overall effectiveness of the three-dimensional model in simulating transport problems. Most of the interior surface elements are regular elements that are 375 m on a side. Surface elements away from the 200 Area Plateau are larger. The total number of surface elements in the three-dimensional model is 2991 elements. The three-dimensional model, based on this surface grid, comprises a total of 23128 elements (2991 surface and 20137 subsurface elements) and 23668 nodes.

The Columbia River boundary in the updated three-dimensional model extended from the Hanford Site shore of the river to the middle of the river channel to reflect the hydraulic interaction of the unconfined aquifer and the river. The surface node at the river boundary was simulated as a prescribed head boundary condition reflective of the assumed river stage that was based on a long-term river stage average. The Columbia River was represented as a constant head boundary along half of its width by having the constant head nodes at both the edge and centerline of the river. Nodes below the surface, along the centerline of the river, were simulated as no-flow boundaries. This design provides a more accurate approximation of the upward movement of groundwater controlled by the hydraulic gradient between the aquifer and the river. The CHARIMA river simulation model (Walters, Richmond, and Gilmore 1994) was used to generate long-term average water surface elevations for the Columbia River based on 1979 conditions.

At the Cold Creek and Dry Creek valleys (Figure 4.19), the unconfined aquifer extends westward beyond the boundary of the Hanford Site groundwater flow model and as a result the unconfined aquifer is recharged from these sources. Additionally the unconfined aquifer also is recharged from springs and runoff that infiltrate the aquifer along the northern side of the Rattlesnake Hills along the western edge of the model (Figure 4.19). To approximate the groundwater flux entering the modeled area from these valleys and the Rattlesnake Hills, both prescribed head and prescribed flux boundary conditions were defined. For the steady-state model calibration runs, a prescribed head boundary condition was specified for Cold Creek and Dry Creek Valleys as well as along the Rattlesnake Hills. Once calibrated, the steady-state model was used to calculate the flux condition that was then used in the transient simulations. The prescribed flux boundary was used because it better represents the response of the boundary to a declining water table than a prescribed head boundary.

Since the description of the sitewide model provided in Wurstner et al. (1995), a number of changes have been made to the extent of the model, model boundary conditions, and model grid. These changes were made to reflect the most recent understanding and interpretation of the unconfined aquifer system by the Hanford Groundwater Project. The most significant changes incorporated in the current version of the sitewide models were derived from reinterpretation of the 1979 water table surface of the unconfined aquifer and the top of the basalt. The reinterpretation led to changes in both internal and lateral boundary conditions, including:

- Moving the model boundary inward along Rattlesnake Ridge and the Yakima River to more closely approximate the location where basalt intersects the water table surface

- Increasing the extent of basalt subcrops above the water table surface in areas south and east of Gable Mountain and northwest of Gable Butte, to more closely approximate the location where basalt intersects the water table surface.

Simulations of Hanford Site water table conditions for the Composite Analysis focused on predicting the impact of ceasing the wastewater discharges that have been used extensively as a part of waste management practices. Previous analyses of post-Hanford Site unconfined aquifer conditions have considered land uses such as large-scale irrigation on the Hanford Site that could significantly alter the long-term behavior of the unconfined aquifer beneath the Hanford Site. The potential for large-scale agricultural irrigation at on the Hanford Site in the future was examined for the Composite Analysis. Consultations with staff from the Agricultural Research Service at the Agricultural Experiment Station in Prosser, Washington, resulted in the conclusion that the prospect of large-scale irrigation occurring on the Hanford Site is unlikely for the following reasons.

- Public acceptance of food products grown on the Hanford Site, regardless of the actual risk associated with agricultural development, is uncertain.

- Sufficient water rights within the Pasco Basin for development of crops requiring large-scale irrigation on the Hanford Site are unavailable. If agriculture should develop on the Hanford Site, it is

likely that the crops to be planted will use the efficient and focused irrigation methods (e.g., drip irrigation) that are used in fruit orchards or vineyards.

- New technologies and advanced resource management practices will likely eliminate or significantly curtail over-irrigation of crops.

- The availability of sufficient water rights and land in the East High portion of the Columbia Basin Project suggests, in the event of a developing national or international need for increased agricultural production, that other areas of the State of Washington would be developed before the Hanford Site would be used.

Prior to conducting contaminant transport simulations with the three-dimensional model, the previous steady-state, two-dimensional model of the unconfined aquifer system was calibrated to 1979 water table conditions with a statistical inverse method implemented in the CFEST-INV computer code Devary (1987). The three-dimensional model was calibrated by preserving the spatial distribution of transmissivity from the two-dimensional inverse modeling. The vertical distribution of hydraulic conductivity at each spatial location was interpreted based on the inverse transmissivity value and the available three-dimensional hydraulic property data, that included: data on the geologic structure, facies data, generic property values based on facies descriptions. A complete description of the seven-step process used to distribute the transmissivity distribution derived from the inverse calibration among the major conductive hydrogeologic units is described in Cole et al. (1997).

The transient behavior of the three-dimensional flow model was calibrated by adjusting model storage properties (specific yield) until transient water-table predictions approximated observed water-table elevations between 1979 and 1996. Following the steady-state and transient calibrations, the three-dimensional model was applied to predict the future response of the water table to postulated changes in Hanford Site operations.

4.1.3.3 Groundwater Flow Model Results

The three-dimensional model was used to simulate transient-flow conditions from 1996 through the year 4000, based on the distribution of hydraulic conductivity from the steady-state calibration and the distribution of specific yields developed from the transient calibration (0.25 for Hanford formation layers and 0.1 for the Ringold Formation layers). The water table contours estimated for the years 2000 (Figure 4.20), 2100 (Figure 4.21), 2200 (Figure 4.22), and 2350 (Figure 4.23) with the three-dimensional model, predict an overall decline in the water table and hydraulic gradient across the entire site. The different areas approach steady state at varying rates, as illustrated in Cole et al. (1997). The areas north of the gap between Gable Butte and Gable Mountain along the Columbia River have the shortest time constants, and water levels in this region reach steady state by the year 2100. The area between the Gable Butte and Gable Mountain reach steady-state conditions sometime between the years 2200 and 2300. The rest of the Hanford Site, including the area south of Gable Mountain and east of the 200 West Area, all are predicted to reach steady-state conditions by the year 2350.

Over about a 300-year period following elimination of wastewater discharges to the ground at the Hanford Site, the water table is predicted by the model to decline significantly and return to near pre-Hanford Site conditions that were estimated to exist in 1944, Kipp and Mudd (1974). Over this period, the water table is predicted to drop as much as 11 m beneath the 200 West Area near U Pond and 7 to 8 m beneath the 200 East Area near B Pond. The areas of the model predicted to be different from the estimated 1944 conditions include: the area west of the 200 West Area, where higher predicted hydraulic heads reflect the effects of increased irrigation from upgradient regions; and the area of the North Richland well field, where annual injection and withdrawal sequences are assumed to continue.

Flow modeling results also suggested that as water levels drop in the vicinity of central areas in the model, the saturated thickness of the unconfined aquifer greatly decreases and may eventually dry out south of Gable Mountain along the south east extension of the Gable Butte anticline. This could cause the unconfined aquifer to the north and south of this line to become hydrologically separated. As a result, flow paths from the 200 West Area and the northern half of 200 East Area that currently extend through the gap between Gable Butte and Gable Mountain, effectively may be cut off in the future. In time, the overall water table, including groundwater mounds near the 200 East Area will decline. As a result, the groundwater movement from the 200 Area Plateau would shift to a more west-to-east pattern of flow toward points of discharge along the Columbia River between the old Hanford town site and the Washington Public Power Supply System facility.

4.1.4 Groundwater Transport Model

A groundwater transport model based on the CFEST-96 code, discussed above, was developed and implemented for the Composite Analysis. This model was used to evaluate the future migration and fate of existing contaminant plumes (Cole et al. 1997) as well as the development and migration of plumes from future sources of unconfined aquifer contamination predicted by the source term release and vadose zone transport model discussed earlier.

4.1.4.1 Background

Transport simulations of both existing plumes and plumes from future sources were based on the previously described three-dimensional flow model. Transient flow conditions were used to provide the basis for all Composite Analysis modeling transport predictions.

Additional model parameters are required to model the contaminant transport processes of dispersion and adsorption. These additional model parameters include longitudinal and transverse dispersivities (D_l and D_t) and contaminant retardation factors (R_f). The key assumptions made in the development of the contaminant transport model are listed in Table 4.11.

4.1.4.2 Groundwater Transport Model Implementation

Dispersivity "the most elusive of the solute transport parameters" (Freeze and Cherry 1979) cannot be directly measured in the field or laboratory. It is determined by inverse modeling of tracer test breakthrough curves from tests performed at the transport scale of interest and in the geohydrologic system of interest (Farmer 1986). Freeze and Cherry (1979) indicate that values of longitudinal and transverse dispersivities are significantly larger than values obtained in laboratory experiments on homogeneous materials and materials with simple heterogeneity. No field tests have been conducted at the Hanford Site to develop an estimate for this parameter at the scale of transport appropriate for the Composite Analysis.

General studies indicate that dispersivity is a function of both time and transport distance because of unaccounted for temporal changes and unaccounted for heterogeneities. The U.S. Environmental Protection Agency (EPA), in their guidance for water quality assessment screening for toxic and conventional pollutants in surface and groundwater (Mills et al. 1985), indicates "A rough estimate of longitudinal dispersivity in saturated porous media may be made by setting D_l (cm) equal to 10% of the mean travel distance." This rule of thumb is based on analysis of tracer tests performed over a large range of laboratory and field scales and for a wide variety of aquifers.

The original work was performed by Lallemand-Barres and Peaudecerf (1978) and expanded by Gelhar and Axness (1981). Later in 1992, Gelhar, Welty, and Rehfeldt reexamined the data and indicated that because of the potential unreliability of the data that no definite conclusion regarding the rule could be reached beyond transport distances of 100 m. However, this was later refuted by Neuman (1993).

Dispersivity is theoretically expected to have an asymptotic value that can be related to the scale of uncharacterized aquifer heterogeneity (Farmer 1986). In contaminant transport simulations, large values of dispersivity result in lower peak concentration estimates, but give rise to earlier first arrival times that can increase arrival concentrations of radionuclides with short half-lives. For the Composite Analysis, a longitudinal dispersivity, D_l, of 95 m was selected. This is not inconsistent with observations made in Freeze and Cherry (1979) that longitudinal dispersivities as large as 100 m and lateral dispersivities as large as 50 m have been used in migration studies of large contaminant plumes. As discussed in Wurstner et al. (1995), the 1/10 approach has generally been used in the past for determining dispersivity values for Hanford Site transport modeling. Law (1992) used values of $D_l = 43$ m and $D_t = 12$ m for a scale of 9500 m based on values compiled in Gelhar et al. (1985). An earlier model (Golder Associates 1990) used values of 15 m and 1.5 m, respectively, for longitudinal and transverse dispersivity, which were also based on Gelhar et al. (1985).

It should be also recognized that the dispersivity values, determined from field tests at 59 different sites compiled by Gelhar, Welty, and Rehfeldt (1992), included results from two investigations at the Hanford Site. The first was a 1950s tracer test that resulted in values of $D_l = 6$ m for the Hanford formation and $D_l = 460$ m for the Ringold Formation, as reported by Bierschenk (1959). Also included

are values of $D_l = 30.5$ m and $D_t = 18.3$ m for a scale of 20,000 m. These were calculated from two-dimensional transport modeling of the 200 East Area tritium plume as reported in Ahlstrom et al. (1977).

Dispersivity is likely to vary across the Hanford Site depending on the degree of heterogeneity and the temporal variability of flow gradients. Ahlstrom et al. (1977) noted that the ratio of D_l to D_t calculated from their model of the Hanford Site was much higher than the ratio expected. They attributed the high ratio to heterogeneity. However, horizontal dispersion may have been enhanced by temporal variations in flow gradients caused by disposal practices. The flow paths for the tritium transport from the 200 East Area have gradually shifted from due east to a south-easterly direction, in response to wastewater discharges to B Pond and the 200 East Area. This shift in the flow path has enhanced the apparent dispersion of the tritium plume emanating from the 200 East Area. More recent sitewide modeling studies (Law et al. 1996) used values of D_l and D_t of 30.5 m and 3 m respectively, which appear to be related to the transport grid spacing of 100 m. In the recent *Hanford Low-level Tank Waste Interim Performance Assessment* (Mann et al. 1997) the horizontal dispersivity for aquifer transport was set at 10% of the travel length in the direction of flow and in the vertical direction at 1% of the travel length. The Draft Hanford Remedial Action EIS (DOE 1996a) set transverse dispersivity at 1/5 of the longitudinal value. Longitudinal dispersivities were based on the scale dependency relationships between longitudinal dispersivity and mean travel distance discussed in Walton (1985).

While the value of $D_l = 95$ m is not based on any Hanford Site data, it satisfies all three of the following constraints on its value:

1. The numerical constraint is related to the grid Peclet number, P_e=(grid spacing)/ D_l. For finite element transport simulations $P_e < 4$ are required for acceptable solutions (Campbell, Longsine, and Reeves 1981). The 95-m dispersivity estimate is approximately one quarter of the grid spacing in the finest part of the model grid in the 200 Area Plateau where the smallest grid spacing is on the order of about 375 m by 375 m (Figure 4.19).

2. At the grid scale of 375 m used in this modeling, the modeled system is homogeneous. Heterogeneities at scales less than 375 meters are uncharacterized. The 95-m dispersivity value selected satisfies this constraint.

3. Finally, because it is more than 10 km from the closest source in the 200 East Area to the Columbia River, a nonasymptotic value of 1000 m for the longitudinal dispersivity could be appropriate. Because large values of dispersivity are not conservative in transport simulations, the 95-m dispersivity value selected for use in the Composite Analysis is the smallest value that could be used with the grid spacing selected. Applying the rule of thumb, discussed above, estimates of concentration 950 m from the source should be accurate and for greater distances they should be conservative.

With regard to transverse dispersivity the following is noted:

- EPA guidance (Mills et al. 1985) is 1/3 for the ratio of D_l / D_t.
- Freeze and Cherry (1979) indicate transverse dispersivity is lower by a range of 5 to 20 (i.e., 0.2 to 0.05).
- Walton (1985) states that reported ratios of D_l / D_t vary from 1 to 24 but that common values are 1/5 and 1/10.

The transverse dispersivity, D_t used in these simulations was assumed to be approximately 20% of the longitudinal dispersivity. Thus, a transverse dispersivity of 20 m was used in all simulations.

With regard to sensitivity, a 45-m grid spacing was used in the recent 200-West Effluent Treatment Facility study (Barnett et al. 1997) with dispersivities of 20 m and 2 m (longitudinal and lateral respectively). Comparing these results with the Composite Analysis results the peak values and resolution were less because of the larger grids but the general character of the predicted plumes was much the same (see Section 4.3).

The vertical grid spacing for the transport (as well as the flow) model consisted of multiple transport layers that subdivided the nine hydrostratigraphic units. The basic thickness of these transport layers was 8 m. The transport layers were defined from the water table surface to the basalt to account for the overall declining water table and to adequately represent contaminant concentrations in the three-dimensional model. At every model node each of the nine hydrostratigraphic units below the water table was represented by at least one transport model layer. Nonconductive (e.g., mud units) below the water table were always represented by at least 2 transport model layers regardless of their saturated thickness in order to assure the vertical flow and transport through these units was appropriately represented. For units whose saturated thickness was <12 m thick, the layer thickness was set to the actual saturated thickness of the unit. Nonconductive and conductive units whose saturated thickness was >12 m were divided into multiple transport model layers in the same manner. For all units with thickness >12 m, the transport layering algorithm is as follows: create as many uniform 8-m transport layers as possible until the remaining unaccounted for saturated thickness is >12 m but <=16 m, then create two additional transport layers set to half of the remaining saturated thickness of the hydrostratigraphic unit being layered.

Calculation of the effective contaminant retardation factors required estimates of contaminant-specific distribution coefficients as well as estimates of effective bulk density and porosity of the aquifer materials. Detail on contaminant-specific distribution coefficients measured or estimated for the unconfined aquifer is summarized in Appendix E. No adsorption was accounted for in simulation of the tritium and technetium-99 plumes in the Composite Analysis. However, for the iodine-129, uranium, and strontium-90 plumes, best-estimate distribution coefficients were developed and applied.

In addition to the estimated distribution coefficient, calculation of contaminant-specific retardation factors used in the transport model required estimates of the effective bulk density and porosity. A bulk density of 1.9 g/cm^3 was used for the calculation of retardation factors in all groundwater transport simulations in the Composite Analysis. The effective porosity was estimated from specific yields obtained from multiple-well aquifer tests, which ranged from 0.01 to 0.37 cm^3/cm^3. Laboratory measurements of porosity available for samples from a few Hanford Site wells, which ranged from 0.19 to 0.41 cm^3/cm^3, were also considered. The few tracer tests conducted at the Hanford Site indicated a range in effective porosity from 0.1 to 0.25 cm^3/cm^3. Based on the ranges of values considered, a best-estimate value of the effective porosity of 0.25 cm^3/cm^3 was used for the calculation of retardation factors in all groundwater transport simulations in the Composite Analysis.

Transport simulations were developed to evaluate the future migration and fate of selected existing contaminant plumes, and to identify and quantify potential radiological impacts of offsite use of groundwater. Monitoring of groundwater in the unconfined aquifer has detected a number of radioactive contaminant plumes emanating from various operational areas (Hartman and Dresel 1997). The most widespread plumes are from tritium and iodine-129. Smaller plumes of strontium-90, technetium-99, and plutonium contain concentration levels of these constituents exceeding EPA and the State of Washington interim drinking water standards (DWS). Uranium concentrations are also found at levels greater than the proposed DWS. In recent years, areas of groundwater contaminated by cesium-137 and cobalt-60 have also been found at or exceeding the DWS.

The existing contaminant plumes in the unconfined aquifer simulated for the Composite Analysis included the tritium, iodine-129, technetium-99, uranium, and strontium-90 plumes. Each of the transport simulations was based on the predicted future transient-flow conditions and the high-resolution finite-element grid designed to resolve areas of future plume transport. Interpreted plume maps for 1996 (Hartman and Dresel 1997) were used to represent initial conditions for the existing plume simulations. The initial conditions for the existing tritium, iodine-129, technetium-99, uranium, and strontium-90 plumes are illustrated in Cole et al. (1997).

Transport of future contaminant releases to the unconfined aquifer for source areas in the exclusive waste management area were evaluated to examine the future movement of contaminant plumes resulting from these releases to areas outside of the buffer zone. Radionuclides evaluated include future releases of technetium-99, iodine-129, carbon-14, chlorine-36, selenium-79, and uranium.

4.1.4.3 Groundwater Transport Model Results

Groundwater transport simulation results used in the Composite Analysis were performed in two steps. Transport of the tritium plume resulting from future operations of the SALDS, and the transport of the existing tritium, iodine-129, technetium-99, uranium, and strontium-90 plumes originating from the 200 Areas were simulated as part of the Hanford Groundwater Project effort (Cole et al. 1997) discussed above. All other plumes related to future sources were simulated as part of the Composite Analysis

using the same model presented in Cole et al. (1997). The existing contaminant plumes in the unconfined aquifer were transported from their current distributions with the hydraulic gradients that are projected for the future as the groundwater system responds to cessation of wastewater discharges. As discussed in Cole et al. (1997), simulations for all existing plumes except for tritium began in 1996. The initial conditions for these simulations were based on the plumes presented in the Hanford Site groundwater monitoring report for FY 1996 (Hartman and Dresel 1997). The tritium plume simulation was run from 1979 through 2100 and started with initial conditions interpreted from 1979 monitoring data and presented in Cole et al. (1997). Cole et al. (1997) compare simulation results for the 1996 tritium plume with interpretations from monitoring observations reported in Hartman and Dresel (1997).

Separating the analysis of plumes resulting from future leaching of contaminants from the vadose zone, from the analysis of the migration of existing plumes, facilitated interpretation of results. The existing contaminant plumes superimpose with the plumes generated by future releases of contaminants considered in the Composite Analysis. Radiological doses resulting from the separate simulations were simply added together in ARC/INFO® to produce the final results. To illustrate the fate and transport of contaminants considered in the Composite Analysis, the predicted distributions of the contaminant plumes are shown at their times of peak concentration in the unconfined aquifer (which is prior to the start of the compliance period).

The plan-view, maximum-concentration plots discussed in this subsection were prepared from the three-dimensional model results through a sampling process that determined the maximum at each location in space. This process involved sampling the vertical stack of nodes at each plan view location in the grid (Figure 4.19) in order to find the maximum concentration calculated at any depth in the profile. The contour plots of concentration shown represent the spatial distribution of maximum concentration values. The radiological doses resulting from the separate plume simulations were constructed from these maximum plan-view concentration distributions and added together in ARC/INFO® to produce the final results.

Figure 4.24a illustrates the predicted distribution of tritium in the unconfined aquifer in 1997, and Figure 4.24b illustrates the predicted tritium distribution in 2050, the start of the compliance period. All of the tritium considered in the Composite Analysis is from existing plumes or SALDS disposal. Figure 4.25a illustrates the distribution of technetium-99 from existing sources in 1996, the time of peak concentration, and Figure 4.25b illustrates the predicted technetium-99 distribution in 2049, approximately the start of the compliance period. Figure 4.26 illustrates the distribution of technetium-99 from all sources in 2036, at a time when the technetium-99 produces a secondary peak in the groundwater. Figure 4.27a illustrates the distribution of iodine-129 in groundwater in 2036, and Figure 4.27b illustrates the predicted iodine-129 distribution in 2049, approximately the start of the compliance period. Strontium-90 peaks from existing plumes in 1996; carbon-14 from future sources peaks in 2027; chlorine-36 from future sources peaks in 2019; selenium-79 from future sources peaks in 2005, and uranium (total) from existing sources peaks in 1996. Concentration plots at time of peak concentration

and at 2049, approximately the start of the compliance period (i.e., 2050) are shown for strontium-90; carbon-14; chlorine-36; selenium-79, and uranium (total) in Figures 28 (a-b), 29 (a-b), 30 (a-b), 31 (a-b), and 32 (a-b), respectively.

4.1.4.4 Groundwater Transport Model Sensitivity

Wastes from some sites will be released to the groundwater pathway in the far future. To investigate this issue, a series of nine transport model sensitivity runs were made. These runs examined the expected variation in transport model response to source location in the far future to determine if plume formation at various waste sites was significantly different once the water table reached steady state. In each of these transport sensitivity runs 1 Ci per year of a hypothetical long-lived radionuclide was released each year for a 20-year period starting in 3899. The total release over the 20-year period would thus be 20 Ci. The year 3899 has no particular significance. This time period was chosen for these sensitivity runs because transient flow simulation results for this far future time period were available, and it was believed that results for this time period would better represent future steady-state conditions when effects of previous Hanford Site discharge mounds would be minimal.

Four node locations were selected in the 200 East Area to represent hypothetical releases from the AX and AY Tank Farms, the BX and BY Tank Farms, the C Tank Farm, and the future TWRS ILAW. Similarly four node locations were selected in the 200 West Area to represent release from the T Tank Farm, the TX and TY Tank Farm, the U Tank Farm, and the S and SX Tank Farm. The ninth location selected was the node that would best represent release from the US Ecology site. Results of these runs in the form of maximum concentration versus time plots are shown in Figure 4.33. These plots show the predicted maximum concentration (at any depth) versus time at each of the nine source location nodes. Analysis of these results indicates that the time required to reach the maximum concentration at a source node is generally shorter in the 200 East Area (3 years at BX-BY Tank Farm source node, 5 years at C Tank Farm source node, and approximately 10 years at AX-AY Tank Farm and TWRS ILAW disposal site source nodes) compared to more than 20 years at all four nodes representing losses from tank farm sites in the 200 West Area. Additionally, source node peak concentrations in the 200 East Area are lower (i.e., 679 pCi/L at the BX-BY Tank Farm source node, 2051 pCi/L at TWRS ILAW source node, 2713 pCi/L at AX-AY Tank Farm source node, and 2980 pCi/L at C Tank Farm source node) than 200 West Area source node peaks (i.e., 12866 pCi/L at S-SX Tank Farm node, and between 15000 and 16000 pCi/L at T, TX-TY, and U Tank Farm source nodes). The response at US Ecology is somewhat in between the 200 East Area and 200 West Area responses, although it is closer to the 200 West Area results. These results can be scaled up or down to investigate the effect of different postulated future release rates at sites in these areas.

4.1.5 Atmospheric Model

The atmospheric pathway was evaluated for a single suite of sources in the Composite Analysis. Based on a review of previously completed analyses that showed minimal contribution to all-pathways

dose from the atmospheric pathway, only the graphite cores from the production reactors were assumed to release contaminants that could be transported via the atmospheric pathway.

4.1.5.1 Background

The evaluation of the atmospheric pathway in the Composite Analysis only considered potential exposures to individuals living in the vicinity of the releases. Radionuclides released to the atmosphere were transported downwind from the solid waste burial ground that contained the graphite cores. The location employed in this analysis was assumed and simply placed the cores in the northwestern portion of the 200 West Area.

The key assumptions made for development of the atmospheric transport model are listed in Table 4.12.

4.1.5.2 Atmospheric Model Implementation

Unit transport factors (UTFs) were calculated for the postulated release originating within the exclusive waste management area. The atmospheric transport of gaseous radionuclides was evaluated with the Multimedia Environmental Pollutant Analysis System (MEPAS). Buck et al. (1995) and Droppo and Buck (1996) describe the MEPAS code. The MEPAS code is based on the sector-averaged Gaussian model, which is the method recommended for dose calculations performed for releases from Hanford Site facilities (Schreckhise et al. 1993).

The UTFs provide estimates or air concentration and deposition rate to soil as a function of distance and direction from each source area. The UTFs were normalized to an annual release of 1 pCi of each radionuclide and provided air concentration estimates in units of pCi/m^3 and deposition rates in units of $pCi/m^2/yr$. The emission was assumed to occur uniformly over an area source 100 m by 600 m. Recommended atmospheric data from Schreckhise et al. (1993) were used to perform the atmospheric transport calculations. The environmental settings for the transport calculations used for the Composite Analysis are described by Holdren et al. (1995).

4.1.5.3 Atmospheric Model Results

For simplicity, atmospheric transport away from the eight surplus reactor cores, which for the purposes of this Composite Analysis were located at a hypothetical burial site in the northwestern part of the 200 West Area, was treated as a radial transport directed away from the center of the source area. Because the source is a distributed source based on either the actual size of the reactor cores or the size of the burial ground cover under which the cores would be placed, the peak values for dose rate and concentration estimated at the actual source location center are not very meaningful resulting from the radial nature to the fall off. The model predictions at the source should be ignored at points inside the 100-m by 600-m source areas. No method was developed to partition the gas versus liquid phase for carbon-14 and tritium as it is released from the reactor cores. As a result, the 2050 inventories of tritium

(7,300 Ci) and carbon-14 (42,000 Ci) associated with these eight reactor cores were accounted for twice in the Composite Analysis because these inventories were released both to air and vadose zone pathways. The estimated release rate for carbon-14 was taken from DOE (1989), which indicates a maximum potential release rate for carbon-14 from water-saturated graphite cores of < 1.5 Ci per year per reactor, or 12 Ci per year. Tritium release was derived using the reactor core release model used for all the vadose zone transport calculations (Appendix D). The tritium release rate, using this model, was estimated at 0.0073 Ci per year in 2050, the time when it was assumed that the cores would be placed in their hypothetical disposal area.

4.1.6 Exposure and Dose Model

Four exposure scenarios were used in the Composite Analysis to evaluate the potential impact on individuals from radionuclide releases to water and air. The exposure scenarios used in the Composite Analysis are those defined for the Hanford Site Risk Assessment Methodology (HSRAM) (DOE 1995). The HSRAM exposure scenarios were developed for the Hanford Site to facilitate evaluations of dose and risk related to Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) remedial investigations and Resource Conservation and Recovery Act (RCRA) facility investigations. The four HSRAM exposure scenarios are referred to as recreational, industrial, residential, and agricultural. These scenarios are summarized in this section of the report as described by Strenge and Chamberlain (1994). Additional detail on the exposure scenarios and unit dose factors is provided in Appendix F.

4.1.6.1 Background

The radiological dose impacts[a] considered in the Composite Analysis were predicted with unit dose factors (UDFs) that relate concentration of a radionuclide in an environmental medium to the resulting radiation dose. The UDFs were evaluated for the radionuclides of interest, and for chemical effects of uranium, as specified in the Composite Analysis guidance (DOE 1996b). The UDFs were evaluated for each exposure scenario and environmental medium appropriate to the exposure scenarios. The environmental media considered include groundwater, air, and soil contaminated by airborne deposition. The contributions to dose from all exposure pathways defined for each scenario were included in the UDFs. Key assumptions for the exposure and dose model are listed in Table 4.13.

The industrial scenario was intended to represent potential exposures to workers in a commercial industrial setting. The industrial scenario primarily involved indoor activities, but outdoor activities (e.g., soil contact) were also included. The workers were assumed to wear no protective clothing; the scenario was not intended to represent exposure of remediation workers. The specific exposure pathways included in the industrial scenario are listed in Table 4.14 for both radionuclides and chemicals, and for each transport medium.

(a) All doses in the Composite Analysis (except where noted) are in units of mrem effective dose equivalent (EDE) in a year.

The recreational scenario was intended to represent exposure to individuals engaging in recreational activities on the Hanford Site. Exposure pathways include soil contact, ingestion of water, and inhalation. The individuals were also assumed to hunt and eat game (deer) meat. The specific exposure pathways included in the recreational scenario are listed in Table 4.15 for both radionuclides and chemicals, and for each transport medium.

The residential scenario was intended to represent potential exposures to future individuals who may take up residence on the Hanford Site. The exposures were assumed to be continuous throughout the year. The specific exposure pathways included in the residential scenario are listed in Table 4.16.

The agricultural scenario was very similar to the residential scenario, with the addition of meat, game, and milk ingestion. The individual was assumed to take up residence on the Hanford Site and grow vegetables, fruit, and raise meat and milk animals. These food products were assumed to be consumed on the family farm. The specific exposure pathways included in the agricultural scenario are listed in Table 4.17.

4.1.6.2 Exposure and Dose Model Implementation

Unit dose factors for radionuclides were used to calculate the annual radiation dose received by an individual exposed in each of the defined HSRAM scenarios (DOE 1995). The dose is expressed in units of rem per year and represents the committed effective dose equivalent for one year of intake or exposure. The UDFs were evaluated for a unit concentration in a specific exposure medium. For example, with groundwater as the transport medium, the UDF was expressed per pCi/L in the groundwater. When air was the transport medium, the UDF was expressed per pCi/m^3 in air.

The evaluation of annual radiation dose as the endpoint in the analysis represents a deviation from the HSRAM (DOE 1995). The HSRAM report describes evaluation of the lifetime cancer incidence risk from radionuclides using slope factors. The slope factors relate intake (pCi) to the lifetime cancer incidence risk. However, the guidance for the Composite Analysis specifies evaluation of annual radiation dose (DOE 1996b). Therefore, the use of slope factors in the HSRAM guidance was replaced with radiation-dose-conversion factors in the Composite Analysis.

The evaluation of annual radiation dose in the Composite Analysis was based on radiation-dose-conversion factors published in Federal Guidance Reports No. 11 and 12 (Eckerman,Wolbarst, and Richardson 1988; Eckerman and Ryman 1993). These dose factors are based on recommendation of the International Commission on Radiological Protection as given in ICRP (1979a, b). The resulting doses represent the effective-dose-equivalent received over a commitment period of 50 years following intake in the first year.

Consistent with the HSRAM scenarios, the radionuclide concentrations in transport media were assumed to be constant over the exposure duration. The concentrations were also assumed to be constant for a period of time prior to an exposure period in which deposited radionucides (from irrigation or

atmospheric deposition, if appropriate to the scenario) were allowed to reach equilibrium with the soil. Equilibrium was assumed reached when the deposition rate was equal to leaching and radioactive decay losses from the soil. An analysis was performed to determine the time necessary for each radionuclide to reach equilibrium in the surface soil layer (see Appendix D). Mobile and short-lived radionuclides would reach equilibrium within a year. However, for the Composite Analysis, the longer-lived radionuclides and radionuclides that generate progeny radionuclides did not come to equilibrium within the 1000 years considered. Therefore, all UDFs were evaluated for 50 years of prior deposition and accumulation in the soil from air or irrigation water deposition. This assumption will represent near-equilibrium conditions for most radionuclides.

For uranium, the UDF was represented by the hazard quotient. The hazard quotient is defined by EPA as the average daily intake of a chemical (in this case uranium) divided by the Reference Dose (RfD) for that chemical. The hazard quotient was evaluated for both inhalation exposures and ingestion exposures with RfDs determined for each route.

The UDFs used in the Composite Analysis are summarized in Table 4.18.

4.1.6.3 Exposure and Dose Model Results

The radiological dose results consist of doses from individual radionuclides and the composite doses from all sources for the four exposure scenarios considered in the analysis. The multiple-step compositing process developed both the spatial distribution of composite dose rate outside the buffer area and the maximum composite dose rate versus time. For each time step calculated and for each contaminant plume for which calculations were performed (e.g., tritium, technetium-99 from tanks, technetium-99 from liquid discharge sources, technetium-99 from existing plumes, chlorine-36 from all sources) a plan-view representation of maximum concentration was prepared as discussed in Section 4.1.4.3. Once each of these spatial distributions of maximum concentration were prepared for each and every plume and time step, the spatial distributions of dose rate for each of the four scenarios was prepared for each time step. The spatial distribution of composite dose rate for a given scenario and at a given time step was calculated from these maximum concentration distributions. The composite dose rate at each plan-view location was calculated as the sum (over all contributing contaminant plumes such as tritium, technetium-99 from tanks, technetium-99 from liquid discharge sources, technetium-99 from existing plumes, chlorine-36 from all sources) of the product of maximum concentration for the contributing nuclide times the appropriate dose conversion factor. The individual dose results are presented as the maximum dose rates versus time outside the buffer zone for the agricultural exposure scenario, which resulted in the highest dose rates.

A review of existing radionuclide plumes in the unconfined aquifer revealed the presence of a strontium-90 plume beneath the decommissioned Gable Mountain Pond. The observed peak concentration of strontium-90 in the vicinity of the retired pond was 1500 pCi/L in 1996 (Hartman and Dresel, 1997; Figure 6.10-10). Using the unit dose factor for strontium-90 from the agricultural scenario, this concentration in groundwater converts to a dose rate of ~470 mrem in a year. If the site is not

remediated to remove the strontium-90 in groundwater and in the overlying vadose zone, it is recommended the exclusive waste management area be expanded to include this decommissioned pond. Furthermore, it is also recommended a buffer zone of ~1000 m be established as a region of relatively clean groundwater surrounding the existing strontium-90 plume such that monitoring can detect movement of the strontium. Strontium is highly sorbed on aquifer sediments (K_d = 20 mL/g) and its decay half-life is relatively short, 28.78 years (Parrington et al. 1996). It is anticipated the declining water table will cause strontium in the upper sediments of the aquifer to be suspended in the vadose zone, and, thereby, act to further isolate the contamination. To simplify the discussion of results in the Composite Analysis, it is assumed the exclusive waste management area and buffer zone will be expanded as recommended. Hence, discussion of dose outside the buffer zone assumes the region surrounding Gable Mountain Pond is included inside the exclusive waste management area and buffer zone.

Figure 4.34 illustrates individual maximum dose rate results outside the buffer zone for the agricultural scenario for a) maximum dose, b) all key nuclide contributions, c) tritium, d) strontium-90 from existing plumes, e) carbon-14, f) chlorine-36, g) selenium-79, h) technetium-99 from existing plumes, i) technetium-99 from liquid discharges, j) technetium-99 from tank sources, k) technetium-99 from solid waste sources, l) iodine-129 from existing plumes, m) iodine-129 from future sources, n) total uranium from existing plumes, and o) total uranium from future sources. These graphs illustrate the maximum dose rates for each radionuclide in the unconfined aquifer outside the buffer zone regardless of location.

Figure 4.35 depicts the composite dose rates from all radionuclides and all sources presented as maxima versus time outside the buffer zone for the a) agricultural, b) residential, c) recreational, and d) industrial exposure scenarios. These graphs illustrate the maximum dose rates wherever they occur in the unconfined aquifer outside the buffer zone. The area of the unconfined aquifer predicted to be above the dose rate of 4 mrem in a year for the agricultural scenario decreases from more than 100 km^2 in 1996 to 40 km^2 in 2050 and zero by 2085.

Comparison of the maximum composite dose rate versus time and the maximum dose rates from individual radionuclides shows that the dose rates from 1996 to 2020 are dominated by the contributions of tritium and iodine-129. The peak composite dose rate occurs in 1996, primarily from the existing tritium and iodine-129 plumes. After the tritium concentrations in the unconfined aquifer are reduced by dispersion and decay, and the iodine-129 concentrations are reduced by dispersion, the largest contribution to the composite dose rate is technetium-99. Figure 4.34 shows this will occur very near the end of the 1000-year period and result in a maximum dose rate of ~1 mrem in a year.

Secondary peaks (beyond 1996) occur in the maximum composite dose rate in 2020 (23 mrem in a year for the agricultural scenario) and 2031 (14 mrem in a year), primarily from technetium-99 and iodine-129. The sources of the technetium-99 in these future peaks are tank leaks and contributions from liquid discharge waste sites. The primary source of the iodine-129 in the future peaks is predicted to be liquid discharge waste sites.

Before site closure, the maximum composite dose rates are predicted to be above 30 mrem in a year. However, because the Composite Analysis (DOE 1996b) is a post-closure analysis, maximum dose rates after 2050 were compared to the dose limit of 100 mrem in a year and the dose constraint of 30 mrem in a year. By site closure in 2050, the maximum composite dose for the agricultural scenario is predicted to be less than 6 mrem in a year and by 2150 (loss of institutional control), the maximum composite dose rate is predicted to be ~4 mrem in a year.

The predicted distributions of composite dose rate for each of the exposure scenarios are illustrated for the time of peak dose rate (1996), near site closure (2049), and near the time of loss of institutional control outside the buffer zone (2159). Model results for the exact times of site closure (2050) and loss of institutional control (2150) were not shown because dose rate results were not modeled at those specific time planes. Figures 4.36 through 4.39 illustrate the distribution of composite dose rate in 1996 for the agricultural, residential, recreational, and industrial exposures, respectively. Figures 4.40 through 4.42 illustrate the predicted distribution of composite dose rate in 2049 for the agricultural, residential, and industrial exposures, respectively. Figures 4.43 and 4.44 illustrate the predicted distribution of composite dose rate in 2159 for the agricultural and residential exposure scenarios, respectively. The dose rate results for the recreational scenario at 2049 and 2159 are not illustrated because the predicted dose rates were less than 0.4 mrem in a year. Similarly, the dose rate results for the industrial scenario are not included for 2159 because those predicted doses were below 0.4 mrem in a year.

The radiological dose rate results are presented for lands outside the buffer zone because the exposure scenarios (agricultural, residential, recreational, and industrial) are assumed to not apply inside the buffer zone. These portions of the Hanford Site will remain in exclusive use for waste management with a surrounding buffer area for protection of the public. It is assumed these lands will remain under federal control until they are determined to be safe for release to the public. To provide an indication of the potential impacts if groundwater inside the buffer zone was used, radiological dose rates resulting from the industrial exposure scenario were calculated for the area inside the buffer and exclusion zones. If groundwater inside the zone were used in the industrial scenario, the peak dose rate inside the buffer zone in 1996 (time of peak dose) would be 124 mrem in a year. The maximum dose rate at 2049 (i.e., the approximate time assumed for Hanford Site closure in 2050) would be 32 mrem in a year, and the maximum dose rate at 2139 (i.e., the approximate time assumed for the end of institutional control in 2150) would be 3.6 mrem in a year. These dose rates are from strontium-90 in the groundwater at the 216-B-5 reverse well site. Strontium-90 also appears in groundwater beneath Gable Mountain Pond. Strontium dominated all exposure and dose scenario calculations inside the buffer zone during this period. The DOE intends to maintain the exclusive waste management area and buffer zone until they can be released to the public. The DOE has acknowledged that many low-level radioactive waste facilities may never be suitable for unconditional release to the public, and that deed restrictions on the future use of groundwater resources may be necessary. Consequently, these future doses will not be realized.

The results for uranium treated as a hazardous chemical do not show any impacts outside the exclusion zone and are therefore not illustrated in a figure. These results were produced by estimating uranium impacts with a hazard quotient calculation.

Results of this analysis indicate that for all times the peak air and soil dose rates for tritium are more than 4 orders of magnitude below lowest dose estimate that is contoured (i.e., 0.4 mrem in a year). Results of the carbon-14 modeling indicate that peak air transport medium dose rates of 4.6 mrem in a year at the source occur at the time of disposal and remain essentially constant through time, decreasing only as a result of carbon-14 decay. No separate plots of air transport medium dose rate are shown because the 0.4 mrem in a year contour essentially occurs at the reactor-core-disposal-area boundary and lower doses occur outside the buffer zone. In the soil (air/deposition) transport medium, soil concentrations are created by the continuous air releases, their subsequent deposition, and leaching by infiltration. Dose from contact with contaminated soils is virtually constant over the 1000-year analysis period for the long-lived radionuclides like carbon-14. For short-lived radionuclides like tritium, the maximum soil dose occurs at the beginning of the release. Figure 4.45 illustrates the maximum dose rate for atmospheric release from both the air transport medium and the soil (air deposition) transport medium. The values shown are for the agricultural scenario because it was the scenario showing the greatest impact. The 4 mrem in a year dose rate contour is immediately above the source, and a dose rate of 0.4 mrem in a year barely extends into the buffer zone. The dose rate falls off spatially very quickly and is well below the 0.4 mrem in a year level outside the buffer zone. The industrial scenario, the only viable scenario inside the exclusive waste management area and buffer zone, yielded an 0.2 mrem in a year closed contour immediately above the source and also decreased very quickly at points away from the source.

4.1.6.4 Exposure and Dose Model Sensitivity

The sensitivity of the exposure and dose model was evaluated by considering different unit dose factors for the key radionuclides contributing to dose. The TWRS ILAW interim performance assessment (Mann et al. 1997) used somewhat different dose conversion factors than those used in the Composite Analysis. Table 4.19 provides a comparison of the dose factors. In the table, the unit dose factors for the radionuclides contributing the greatest amount to dose, (e.g., tritium, iodine-129, and technetium-99), are not appreciably different for the two analyses. Therefore, variations of the unit dose factor within the range presented would not produce significantly different dose rate results.

4.2 Comparison with Other 200 Area Modeling Analyses

Several independent modeling analyses have been performed as part of other environmental assessments for specific existing or proposed facilities within the exclusive waste management area. This section briefly compares the salient features of these independent assessments with the analysis performed for these specific sites in the Composite Analysis.

Only three of the independent assessments estimate breakthrough from the waste site to the water table within the 1500-year period modeled in the Composite Analysis. One of these three assessments was for past tank leaks from a specific tank farm. The other two assessments with breakthroughs within the 1500-year period involved shallow land burial of wastes. Three other dry disposal assessments that estimated travel times to the water table in excess of 1500 years are discussed briefly.

Work toward the decontamination and decommissioning of canyon buildings and associated facilities has begun at the Hanford Site. However, this assessment has not obtained key mobile radionuclide inventories in canyon buildings and related facilities, and therefore, has not analyzed their migration and fate. The work has examined the potential migration of large inventories of cesium-137 and strontium-90 from the B Plant and its sand and high-efficiency particulate air (HEPA) filters.

Besides using different models, each of these assessments employed different inventories, model parameters, and assessment points and times of compliance. Generally, it was found that site-specific assessments were more likely to use more conservative parameters than the "best-estimate" values employed in the Composite Analysis. While the results are not necessarily identical, they do suggest fundamental consistency between the site-specific analyses and the Composite Analysis.

4.2.1 Hanford Tanks Initiative

Recent interest in subsurface environmental impacts arising from past leaks and future losses from tanks has resulted in an ongoing analysis of leaks and losses from the tanks in the AX tank farm as part of the Hanford Tanks Initiative (HTI). Liquid losses from single-shell tanks may occur during the recovery of tank waste. This section compares the preliminary unpublished results [(a)] of the HTI analysis with the Composite Analysis.

To estimate the cumulative release of an 8,000-gallon (30-m^3) liquid waste loss from a single-shell tank to the water table, the HTI analysis employed a two-dimensional model of a vertical plane running from the AX tank farm to the water table. The Composite Analysis employed a one-dimensional model. Considerably more detail has been included in the spatial discretization of the soil properties of the HTI model than could be incorporated in the one-dimensional soil column of the Composite Analysis. Whereas the Composite Analysis released the liquid source over the entire tank bottom, the HTI analysis released from a much smaller area representing a header leak and allowed the hydrostratigraphic layers and subsurface properties to spread the plume during its downward migration. The technetium-99 inventories in the two analysis were 4.52 Ci for the HTI assessment and 3.43 Ci for the Composite Analysis assessment. The background recharge rates used were 10 cm/yr for the HTI assessment and 7.5 cm/yr for the Composite Analysis assessment.

(a) From two electronic mail messages, both dated December 30, 1997, sent by P. Rogers, Jacobs Engineering, Richland, Washington to L. W. Vail, Pacific Northwest National Laboratory, Richland, Washington; subjects, "Past Leak Flux" and "Cumulative Mass Files".

Figures 4.46 and 4.47 compare the predicted cumulative release of technetium-99 from the AX Tank Farm for time periods from 1940-2150 and 1940-3000. Figure 4.46 shows generally earlier breakthrough for the Composite Analysis assessment. Figure 4.47 shows that the higher inventory used in the HTI assessment eventually releases, and its cumulative release surpasses that of the Composite Analysis during the 1000 years of the analysis.

As observed, a two-dimensional model should result in later breakthroughs and a more gradual draining of the soil column. This is likely because of the more complex patterns of spreading resulting from the more complex and heterogeneous representation of soil properties. The differences in total curies released at year 3000 are consistent with the different inventory estimates. The multidimensional HTI model has overtaken the one-dimensional Composite Analysis model in cumulative curies released to the water table. However, over 90% of the Composite Analysis inventory has been purged from the one-dimensional column while less than 75% of the HTI inventory has released. This may be attributed to the greater lateral dispersal permitted by the multidimensional model. It may also be a function of the combined effects of lateral dispersal and structural features in the multidimensional analysis that act to shelter some fraction of the release from direct leaching by recharge.

4.2.2 200 Area Solid Waste Burial Ground Performance Assessments

Performance assessments have been performed for the solid waste burial grounds in both the 200 East Area (Wood et al. 1996) and 200 West Area (Wood et al. 1995). These assessments were required to demonstrate that the disposal practice is in compliance with performance objectives in DOE Order 5820.2a (DOE 1998b). As part of these performance assessments, it was required to estimate the temporal distribution of contaminant flux to the water table. An identical calculation was made in the Composite Analysis for nonsorbed radionuclides. Similar calculations for sorbed radionuclides appear in both analyses; however, different distribution coefficients were employed.

Results from the Composite Analysis and the performance assessments for low-level burial grounds in the 200 West and 200 East Areas are somewhat but not remarkably different. Because of the difference in the stratigraphic profiles, and, hence, the unsaturated hydraulic properties of the vadose zone sediments, the transport of contaminants is generally slower through the vadose zone beneath the 200 West Area than for 200 East Area. For 200 West Area, the mean travel time for an advective (unit pulse) release reported in the performance assessment is approximately 1070 years. The Composite Analysis methodology estimates a mean travel time of 1054 using the same recharge rate of 0.5 cm/yr. For 218-E-10 and 218-E-12 burial grounds in 200 East Area, the estimated mean travel times using the performance assessment methodology were approximately 1150 and 650 years, respectively. For a variety of reasons, the Solid Waste Program plans to place the majority of future solid waste in the 200 West Area burial grounds. Therefore, a mean travel time of approximately 900 to 1000 years is indicative of the environmental response for these wastes.

Releases to the aquifer from the post-1988 solid waste burial grounds occur well after the peak releases from other sources that occur in the next few decades, and after the resulting maximum

individual dose outside the buffer zone at the time of Hanford Site closure. Contributions to dose from burial ground releases outside the buffer zone occur later in the 1000-year period and contribute to lower doses.

4.2.3 Commercial/Low Level Waste Site Assessment

Analyses have been performed to demonstrate that the commercial low-level waste (LLW) disposal facility on the Hanford Site will meet the license requirements established by the State of Washington and the U.S. Nuclear Regulatory Commission. These analyses are detailed in the site stabilization and closure plan for the commercial LLW disposal site (Grant Environmental, Chase Environmental Group, and US Ecology 1996). The commercial LLW disposal site assessment assumed a steady recharge rate of 0.5 cm/yr, whereas the transient simulation of the Composite Analysis assumed a change in recharge rates. In the Composite Analysis, a recharge rate of 7.5 cm/yr was assumed until site closure. Until that time it was assumed the cover soils were coarse and maintained free of vegetation. The Composite Analysis did not take any credit for the integrity of the packaging of the disposed waste and allowed leaching to occur during the period prior to cover placement. The Composite Analysis assumed the recharge rate dropped to 0.127 cm/yr after closure of the presently used trenches in 2000.

The commercial LLW disposal site assessment reported travel times of 140, 1110, and 3575 years for steady recharge rates of 5, 0.5, and 0.127 cm/yr, respectively. These all assumed the current depth to the water table is 81 m. However, they estimated the water table beneath the site to drop as much as 13 m as a result of the end of significant liquid disposals from Hanford Site production operations. The Composite Analysis assumed the water table had already dropped to pre-Hanford Site levels before the plume reached the water table resulting in an estimated depth to water table of 87 m. The commercial LLW disposal site assessment reported an estimated travel time of 4288 yr with a recharge of 0.127 cm/yr and a depth to water table of 96 m. The Composite Analysis estimated breakthrough of a nonsorbed radionuclide in the present inventory to occur after 246 years. This time estimate reflects the impact of transient hydrology. Specifically, it reflects the assumed relatively dry initial conditions based on 0.5 cm/yr, 21 years of relatively high recharge of 7.5 cm/yr, followed by low recharge of 0.127 cm/yr. Despite a relatively early breakthrough of nonsorbed radionuclides (e.g., chlorine-36 and technetium-99) in the 1000-year period, these releases do not coincide with the releases of the immediate future. Those occurring now and during the next few decades are associated with liquid discharge sites, tank leaks, losses from tanks, and pre-1988 solid waste burial grounds. These are the sources responsible for the maximum dose outside the buffer zone during the 1000-year period following Hanford Site closure. Releases from the commercial LLW disposal site occur later in the 1000-year period and contribute to lower doses.

4.2.4 Remedial Investigation and Feasibility Study for the Environmental Restoration Disposal Facility

Analyses were performed to evaluate alternatives for the placement of wastes in an ERDF. All wastes disposed in such a facility are to be generated during the remediation of past-practice sites at the

Hanford Site. The analyses and their assumptions are documented in the ERDF Remedial Investigation and Feasibility Study (RI/FS) (DOE 1994b). Travel times for wastes leached from the ERDF and arriving at the water table were estimated using a simple analytical approach in the RI/FS. Several facility designs (i.e., various surface barrier and liner options) and two climate conditions were examined. The Composite Analysis simulated a single case that represented the facility design described as the preferred alternative in the record of decision for the ERDF (Amended ROD September 1997).

In the ERDF RI/FS analyses. travel times from the waste form to the water table were estimated using user-prescribed recharge rates and moisture contents, whereas in the Composite Analysis the moisture contents throughout the soil column were estimated using a physically based model and specific recharge rates. For the preferred alternative the RI/FS employed recharge rates of 0.01 cm/yr for the base climate and 0.4 cm/yr for the wetter climate conditions, respectively. The Composite Analysis assumed a recharge rate of 0.05 cm/yr. Because of the presence of a double liner, leaching was assumed to begin after site closure.

The ERDF RI/FS estimated travel times of 13,000 and 500 years for the base and wetter climate, respectively. The Composite Analysis simulated a period of 1500 years without detecting any breakthrough to the water table.

4.2.5 Environmental Assessment of Surplus Production Reactors

The record of decision on decommissioning the surplus production reactors at the Hanford Site states the preferred alternative is for disposal on the central plateau in the 200 West Area after up to 75 years of continued storage in their respective 100 Areas (ROD 1993). Once disposed within the exclusive waste management area, a potential pathway for environmental impact is the transport of radionuclides through the vadose zone to the water table. Analyses of the vadose zone and groundwater pathway are discussed in Appendix C of the environmental impact statement (DOE 1989). The EIS analysis assumed a recharge rate of 0.1 cm/yr for the Hanford protective surface barrier. Since the late 1980s, the Hanford Site Permanent Isolation Barrier Development Program adopted a design standard of 0.05 cm/yr for allowable recharge rate. Accordingly, the Composite Analysis assumed a recharge rate of 0.05 cm/yr.

The draft EIS (DOE 1989) reported a travel time of 4,200 years. The Composite Analysis simulated a period of 1500 years without detecting any breakthrough to the water table from the production reactors.

4.2.6 TWRS ILAW Performance Assessment

The *Hanford Low-Level Tank Waste Interim Performance Assessment* (Mann et. al., 1997) examined the long-term environmental effects associated with the disposal of the low-level fraction of the Hanford single- and double-shell tank waste in a disposal facility located within the 200 East Area. A three-dimensional computer code was used to simulate the flow and transport of contaminants from the

waste form through the vadose zone to the groundwater. Sensitivity analyses included in this interim performance assessment considered uncertainty in the depth to water table, hydraulic parameters, geochemical parameters, and recharge rates.

The base case of the performance assessment assumed an initial recharge rate of 0.05 cm/yr followed by a recharge rate of 0.3 cm/yr after 1000 years. The Composite Analysis assumed a recharge rate of 0.05 cm/yr throughout the 1500 year period simulated. The interim performance assessment reported a mean travel time of approximately 3,000 years. The Composite Analysis simulation stopped after 1,500 years without detecting any breakthrough to the water table from the immobilized low-activity waste. These wastes are the subject of a formal performance assessment with a planned submittal date of March 1998[a].

4.2.7 Canyon Buildings

As a screening analysis of possible releases from canyon buildings on the Hanford Site, releases of cesium-137 and strontium-90 from the B-plant and its permanent filters were considered. These facilities have a combined inventory of approximately 2.1×10^6 Ci of cesium-137 and 4.2×10^5 Ci of strontium-90. The combined information for the B Plant and its sand and HEPA filters was used to estimate a conservative value for the depth of the source to the water table. Assuming a Hanford Protective Barrier with a recharge rate of 0.05 cm/yr, the Composite Analysis methodology estimated no breakthrough to the water table within 1500 years for fully mobile radionuclides (i.e., distribution coefficient = 0 mL/g). This is a conservative representation for these nuclides because cesium and strontium in the most mobile waste forms have a finite nonzero distribution coefficient.

4.3 Model Calibration and Comparisons of Results with Observations

The first iteration of the Composite Analysis required complex calculations of contaminant release and transport through the vadose zone, groundwater, and atmosphere. This section discusses available information on the relationships among liquid discharge sites, inventory estimates for these sites, and existing plumes were used to perform a limited calibration or history match of the vadose zone model. The section also discusses the calibration of the Composite Analysis aquifer model and compares predicted contaminant concentrations with observations.

4.3.1 Background

At the Hanford Site, there are few specific data sets suitable for aquifer or vadose-zone transport model calibration and comparison of results with observations. The data sets potentially the most useful

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(a) Mann, F. M., R. P. Puigh II, C. R. Eiholzer, Y. Chen, N. W. Kline, A. H. Lu, B. P. McGrail, P. D. Rittmann, G. F. Williamson, J. A. Voogd, N. R. Brown, and P. E. LaMont. 1998. *Hanford Immobilized Low-Activity Tank Waste Performance Assessment.* DOE/RL-97-69, Rev 0, U.S. Department of Energy, Richland, Washington.

for calibration or comparison with observations would be isolated liquid disposal sites receiving large amounts of liquids containing highly mobile nuclides (e.g., tritium and technetium-99). For these type of sites, movement through the vadose zone would be rapid and the plume created in the aquifer may be unique and identifiable, yet large enough to adequately characterize. Other sites, such as past-practice landfills, are unlikely candidates for calibration or comparison with observations because of the uncertainty associated with the waste inventory, waste containment, and waste leaching.

While there are more than 175 liquid discharge sites in the 200 Areas at Hanford, none are completely adequate for calibration or comparison with observations. This is because early records on liquid disposals are spotty and the information on radiological content of the highly mobile radionuclides was often limited to gross alpha and beta counts.

There are no specific, liquid-discharge site data sets available for use in vadose-zone model calibration and the subsequent comparison of model predictions with observations. As a result, the vadose-zone model calibrations and comparisons used in this Composite Analysis were done through a more global, mass-balance approach described in Section 4.3.2.

The best data for a limited calibration of transport in the aquifer is information on the tritium plume. Information on liquid disposals to ground and the tritium content of these liquids is available starting in the mid 1970s. There are also yearly estimates of the "near-water-table" concentrations of the tritium plume based on groundwater monitoring data. However, there is only very limited information on the vertical distribution of the tritium or any other contamination in the aquifer. This lack of information on the vertical distribution of the contamination poses an additional calibration difficulty, because tritium disposal prior to 1979 is the largest contributor to the total inventory estimated to be in the existing plume. Therefore, the lack of a good inventory for the tritium disposal that gave rise to the plume, and the lack of knowledge of the spatial variation of concentration with depth poses a problem in developing initial conditions for the existing plume simulations. This uncertainty in the initial conditions poses a problem when trying to compare model results with observations because the effects of the disposals after 1979 on future plumes cannot be separated from the problem with the initial conditions. As a result, it can not be determined whether the inability to match future plumes (post-1979) is related to a poor aquifer model or an inappropriate vertical distribution of initial conditions. In the process of simulating the existing plumes (Cole et al. 1997), a limited calibration effort was undertaken to address the issue related to the vertical depth of contamination assigned to existing plumes when imposing initial conditions.

Two depths for assigning initial conditions were examined. In the initial model, concentrations, as interpreted from monitoring reports, were assigned to all calculational nodes within 6 m of the water table. This depth corresponds to the screen height of most monitoring wells as a result of an assessment by Eddy, Myers, and Raymond (1978) that the bulk of the contamination was believed to located in the uppermost 5 to 10 m of the aquifer. In the final model of existing plumes, initial condition concentrations were applied to all nodes within 25 m of the water table. Comparison of these modeling results with observations indicated that the 25 m depth provided a better match. This is the depth that was used

to model all existing plumes (Cole et al. 1997). Comparison of the sitewide aquifer model results with observations for the tritium plume is discussed in Section 4.3.3 to provide information on the quality and uncertainty in sitewide aquifer model predictions.

4.3.2 Predicted Contaminant Releases to Groundwater from the Vadose Zone

Contaminant releases to the groundwater in the Composite Analysis were evaluated as a combined waste form release and vadose zone transport calculation. The information on vadose zone transport presented in Section 4.1 consisted of cumulative releases of the various radionuclides from the vadose zone to the groundwater. The STOMP code was used to predict the one-dimensional transport of contaminants through the vadose zone and determine the time of release to the water table of the contaminant mass leaving the source during each time step.

The Composite Analysis results demonstrate that pre-1988 solid waste burial grounds can be expected to release to the water table in the coming decades. Significant portions of their inventories are predicted to release within the next 100 years. However, the active and planned disposal actions are dry disposals that include placement of surface covers to reduce recharge, and thus their releases occur over a much longer time frame. The uncertainty in container integrity, and thus in the actual contaminant quantity released, makes pre-1988 burial ground data useless for model calibration. As a result the data available for determining how realistic predicted vadose zone contaminant releases to the water table are, are restricted to data from past-practice or liquid discharge sites.

At the Hanford Site, there are only a limited number vadose-zone data sets that could be used to compare vadose-zone models with observations. Sisson and Lu (1984) and Fayer et al. (1995) report on model comparisons with a field injection experiment conducted in the 200 East Area. In this experiment, a dilute, mixed-salt solution containing radionuclides was injected 4.5 m belowgrade and migration was monitored through a collection of 32 wells surrounding the injection point to a depth of 18 m. This solution contained calcium, chloride, nitrate, and trace amounts of barium, cesium-134, rubidium, and strontium-85. Water contents and gamma scanning data were collected during the experiment and Fayer et al. (1995) reported on logging of the wells with a high-resolution spectral gamma logging system. Because of the scale of the experiment and the specific radionuclides examined, the experiment was not applicable to the Composite Analysis model calibration problem. Field studies in response to tank leaks (Freeman-Pollard, Caggiano, and Trent 1994) and liquid discharges (DOE 1993a, 1994a) are also incomplete with respect to data requirements for model calibration. As a result, data on the existing technetium-99 plumes, technetium-99 inventory associated with liquid discharges, data on liquid discharge breakthrough (including those from tank leaks), and the uncertainty in these estimates were used for adjusting vadose-zone model parameters and for comparison with model results.

The basic assumption used in the vadose zone model calibration was that contaminant mass estimates for existing plumes combined with spatial and temporal knowledge on the first appearance and suspected source of these various plumes could be used to adjust vadose-zone model assumptions and/or parameters. The existing radionuclide plumes in the unconfined aquifer characterized by groundwater

monitoring (Hartman and Dresel 1997) include strontium-90, uranium, iodine-129, tritium, and technetium-99. Data on other important Composite Analysis radionuclides (e.g. carbon-14, chlorine-36, selenium-79) can not be compared with observations because monitoring data either do not exist or are limited. The release of significant amounts of uranium to groundwater resulted from unique events that caused the mobilization of uranium in the vadose zone beneath one crib, by flushing of water from another crib and preferential flow down the unsealed annulus of a reverse well (Baker et al. 1988). A significant release of strontium-90 to the aquifer has created a plume beneath 200 East Area, however, it resulted from discharge to a reverse well that was completed in the aquifer. The generic approach used in the Composite Analysis does not account for this level of detail in the conceptual and numerical models and as a result, the data on these existing plumes can not be used for model calibration. The iodine-129 inventory and iodine's retardation factor are both uncertain. No credible inventory of iodine-129 discharge to ground during the last PUREX campaign (1984-1986) was found. However, the existing iodine-129 plume appears to be well correlated with this PUREX operation and the absence of release data makes this data set useless for calibration of the release and vadose zone contaminant migration model. The tritium data are not useful for vadose zone model calibration because the various plumes have commingled and there have been so many sources it is impossible to relate specific sources to specific plumes. As a result of the available existing plume data, only the technetium-99 data set was found to be appropriate for vadose-zone model calibration. In an effort to match the response of the release and vadose zone transport models to field observations, the predicted release of technetium-99 from all sources in the Composite Analysis was compared with the mass estimated to be in the aquifer. Mass in the aquifer was estimated from the 1996 groundwater concentration contours interpreted from groundwater monitoring data and presented in Hartman and Dresel (1997). The release and vadose zone transport models were then adjusted to match the observed mass of technetium-99 in the unconfined aquifer with the mass of technetium-99 predicted to be released to the water table before 1996. However, as discussed below, both model parameters and the uncertain inventory estimates for technetium-99 disposed at liquid-discharge sites had to be reconciled during the history matching process.

Mean cross-sectional area associated with the liquid discharge or tank leak was varied in the model calibration, because this parameter directly affects the travel time of the contaminant through the vadose zone and it is a highly uncertain parameter. In the early modeling of tank leaks and liquid discharge sites, a conservative approximation was made to estimate this parameter. At each site where a one-dimensional model was developed, the infiltration rate was assumed to be limited by the saturated hydraulic conductivity of the least conductive of the sediment layer in the hydrostratigraphic column assigned to that site in the Composite Analysis. With this approximation, the cross-sectional area for each discharge facility or leak is estimated based on the saturated hydraulic conductivity of the limiting layer, an assumption of a unit gradient, and a liquid discharge volume and discharge duration for each respective site. The cross-sectional area was very small, except for ponds, producing results that were not consistent with observations, both in terms the observed spatial distribution of contaminants and the total inventory estimated for plumes in 1996.

Using the initial cross-sectional area approximation, all mobile constituents from tank leaks arrived at the water table within a few years and even iodine-129 with a distribution coefficient (K_d) of 0.5 mL/g was predicted to arrive where no iodine-129 plumes have been observed. During the model calibration effort a cross-sectional area equal to the area of a tank bottom yielded results that were most consistent with field observations. This revised cross-sectional area approximation for tank leaks and sluicing losses (as discussed earlier in Section 4.1.2.3) predicted 0.5 Ci of iodine-129 would be released to the aquifer prior to 1996 compared to estimates, based on monitoring data, of 7 Ci of iodine-129 in the aquifer, most of which is believed to be from PUREX operations in the mid-1980s.

Based on the above model, the amount of technetium-99 predicted to release before 1996 was 5 Ci. The upper estimate of the observed mass of technetium-99 in the aquifer is 37.6 Ci, based on integration of the existing plume distribution. This left 32.6 or ~33 Ci of technetium-99 attributed to liquid discharge sites. Prior to the discovery that the technetium-99 inventory data for the liquid-discharge sites from the Waste Site Groupings report (DOE 1997b) and the Environmental Restoration program (i.e., 5.1 Ci total) were significantly lower than the Waite (1991) inventory estimates for these sites (i.e., 930 Ci), there was no inventory estimate that could justify the existing technetium-99 plume which is estimated to contain between 15.8 and 37.6 Ci. All the technetium-99 was predicted to release before 1996, but the entire inventory of 5.1 Ci was less than the ~16 to ~38 Ci of technetium-99 estimated to be in the existing plume. However, the Waite (1991) inventory estimates created the opposite dilemma; with the initial cross-sectional area approximation much more technetium-99 (i.e., ~300 Ci) was predicted to be released than could be accounted by the existing plumes.

To delay the arrival of the technetium-99 at the water table and account for lateral dispersion, or spreading of the contaminant plume in the vadose zone for the liquid discharges, other than ponds, the effect of increasing the cross-sectional area was examined (see Section 4.1.2.4). These studies indicated increasing cross-sectional areas had a diminishing effect on the estimated amount of technetium-99 released to the water table. Increasing the cross-sectional area by a factor of three reduced the predicted release of technetium-99 prior to 1996 from 300 Ci to ~181 Ci. The release of technetium-99 from past tank leaks was calculated to be approximately 5 Ci, which left ~33 Ci of technetium-99 to be associated with liquid-discharge sites prior to 1996. The cross-sectional area required to match the 33 Ci of technetium-99 was unreasonably high (e.g., greater than 10). Therefore, the three-fold increase in cross-sectional area was adopted. This factor was applied to all liquid discharge sites, except ponds, for all radionuclides.

Based on these modeling results, the Waite (1991) estimated inventory of technetium-99 released to liquid discharge sites was believed to be too high, so the inventory for the base case was scaled from 930 Ci to ~167 Ci in order for the predicted pre-1996 release to the water table of ~181 Ci to match the ~33 Ci estimated to be in the unconfined aquifer based on monitoring data. A sensitivity case was also used to demonstrate the effects of using the higher inventory estimate. In this case the pre-1996 release was the same as the base case. However, the post-1996 release rate was scaled up so that the full Waite (1991) estimate of 930 Ci would be achieved. Figure 4.48 shows the cumulative release of technetium-99 from all sources to the water table from 1940 to 3000 for the three inventory and release scenarios

described for the liquid-discharge sites. The plot shows the results for the full Waite inventory for liquid discharge sites (930 Ci) (as was shown in Figure 4.5b), the scaled or base-case inventory estimate (~167 Ci), and the sensitivity case with the enhanced post-1996 release rate that achieves the full Waite (1991) inventory estimate for liquid discharge sites.

Using the adjusted parameters resulting from the qualitative calibration of the high-volume liquid discharges, the Composite Analysis model predict rapid release to the water table that has already occurred or will occur in the near future, consistent with observations. The model results for the past tank leaks show current impacts (releases) and future impacts to the aquifer, consistent with recent observations at several of the tank farms documented by Johnson and Chou (1998) and Hodges (1998).

In comparison with the liquid disposals, few if any observations are available for model comparison and parameter adjustment for the dry disposals. As previously described, the dry disposals include placement of surface covers to reduce recharge and their releases occur over a much longer time frame. A mean travel time of approximately 1000 years was associated with burial grounds that will receive the majority of future solid waste disposals. Forecasts of release from the pre-1988 burial indicate these sites have not released yet. Therefore, data are not available for determining how realistic the predicted vadose zone contaminant releases are for the dry disposals.

One method for establishing confidence in the models used to predict radionuclide releases from dry disposals was to compare Composite Analysis predictions with other performance assessments. These comparisons were made in Section 4.2 and demonstrated that dry disposal sites will release in the future. The time frames for release predicted with the Composite Analysis model for post-1988 disposals of low-level waste are consistent with those in other performance assessment calculations.

4.3.3 Predicted Groundwater Contaminant Concentrations in the Aquifer

Prior to conducting simulations of the contaminant transport summarized in this report, confidence in the three-dimensional model of the unconfined aquifer system was established by calibration of the model to 1979 water table conditions, which was a time of quasi-steady state, as described in Cole et al. (1997). The resulting distribution of hydraulic properties developed for the three-dimensional model were derived from the original transmissivity distributions developed for the two-dimensional version of the sitewide aquifer system and a statistical inverse method described in Jacobson and Freshley (1990). A seven-step process, described in Cole et al. (1997), was used to derive the three-dimensional distribution of hydraulic properties. This seven-step process used hydrostratigraphic and facies descriptions while preserving the calibrated spatial distribution of transmissivities determined from the two-dimensional inverse modeling.

Confidence in the transient behavior of the three-dimensional flow model was established by evaluating its ability to approximate changes in the water table in response to transient liquid discharges to the unconfined aquifer between 1979 and 1996. The evaluation examined a range of model storage properties (specific yield) until transient water table predictions approximated observed water table

changes during this period. Transient simulation results and comparisons of predicted and observed transient water table changes are presented in Cole et al. (1997). These results indicate that the best approximation was achieved when a specific yield of 0.1 was used for units in the Ringold Formation and a specific yield of 0.25 was used for the Hanford formation.

Model simulations of projected declines in artificial discharges at the site presented in Cole et al. (1997) showed that, over about a 300-year period, the water table would decline significantly and return to near pre-Hanford water table conditions that were estimated to exist in 1944. The predicted water table was estimated to be very close to steady state within 100 years. Over the 300-year period, model results show that the water table will drop as much as 11 m in the 200 West Area near the retired U Pond and 10 m in the 200 East Area near B Pond. Modeled areas that differed from the estimated 1944 hindcast included:

- the area west of the 200 Area Plateau, where higher predicted hydraulic heads reflect boundary conditions that consider the effect of increased irrigation from areas upgradient of the modeled region

- the area north of Richland, where the model included the hydraulic effect of the North Richland well field.

Results generated by the Composite Analysis three-dimensional model (Cole et al. 1997) were consistent with the post-Hanford analysis of the water table changes reported by Chiaramonte et al. (1997).

Prior to simulating the future transport of existing plumes and future source of contaminants, confidence in the three-dimensional transport model was evaluated by examining the ability of the model to simulate the transient behavior of the existing plume of tritium from 1979 to 1996. The tritium plume was selected for evaluation because estimates of tritium discharges were available and the plume was monitored during this period (1979 to 1996). A comparison of predicted and observed tritium plume transport, presented in Cole et al. 1997, suggests that the three-dimensional model provides a reasonable approximation of the overall transport of the tritium plume during the period of concern. Results of simulation were also in reasonable agreement with the transport behavior of the tritium plume over the same period performed by Chiaramonte et al. (1997).

Initial conditions used in the transport simulations of existing plumes (tritium, technetium-99, iodine-129, uranium, and strontium-90), were derived from interpreted areal distributions of existing plumes presented in Hartman and Dresel (1997). As discussed above, contamination was assumed to be uniformly spread from the water table to 25 m below the water table. The existing plumes model (Cole et al. 1997) and the groundwater model used in the Composite Analysis are exactly the same except for initial conditions and radionuclide source terms. The SALDS model (Barnett et al. 1997) not only has different initial conditions and radionuclide source terms, but a different grid resolution and assigned dispersivity. Since the SALDS tritium plume was modeled at lower resolution with the coarse grid

model (Cole et al. 1997) and with the locally refined grid model and smaller dispersivity discussed in Barnett et al. (1997), a comparison of these results allows the effect of grid resolution and dispersivity on predicted results to be examined. The SALDS model used a local-scale horizontal grid spacing of 45 m by 45 m in the vicinity of the SALDS and a ~6 m vertical grid all the way to the basalt. Lateral and transverse dispersivities were set to 20 m and 2 m, respectively. The existing plumes analysis used a horizontal grid spacing of 375 m by 375 m and the vertical grid spacing was variable (minimum thickness of 8 m). Lateral and transverse dispersivities were 95 m and 20 m, respectively. The SALDS model contours for the tritium plume from Barnett et al. (1997) are shown in Figures 4.49a and b for the years 2020 and 2045, while existing plumes modeling results for the SALDS from Cole et al. (1997) for these same times are shown in Figure 4.50 and Figure 4.24b respectively. From a comparison of the general shape and movement of both predicted plumes one can conclude that the results are very similar. Plumes of the two models were compared by measuring the width of the plumes at their widest point for a given contour level (e.g. concentration). In 2020, the coarse-grid, large-dispersivity model predicted the plume diameter above 2,000 pCi/L to be 2.4 km and the high-resolution, small-dispersivity model prediction was 1.6 km. Comparisons of high- and low-resolution results for the 20,000 pCi/L contour were 1.1 km and 1.2 km respectively. Similar comparisons for the 2,000 pCi/L contour in 2050, after the centroid of the plume had moved 0.7 km from the disposal site, were 1.5 km for the high resolution model and 1.6 km for the low resolution model. A comparison of all the results of these two models would show that the low-resolution, large-dispersivity model missed the estimated peak values directly below the SALDS during the disposal phase. Small areas (100 m in diameter) were predicted to be above 2 $x10^6$ pCi/L by the high-resolution model while no concentrations above that level were predicted by the low-resolution model. However, a comparison of results through time indicates that the overall areal extent and concentration levels predicted for the SALDS tritium plume using the low-resolution model from the start of operations through site closure and until 2100, when all predicted levels by both models were below 500 pCi/L, were very consistent with results produced by the high-resolution local-scale model.

In Cole et al. (1997), model-predicted concentrations of selected contaminants were also evaluated with respect to observations. As illustrated in the above high- and low-resolution comparison, the 375-m grid resolution being used in the Composite Analysis model means that model-estimated concentration levels near small individual source locations are expected to be lower than observations made in wells near contaminant sources. However, the dispersion predicted by the model away from the sources and outside the buffer zone is likely to be consistent with the amount of dispersion that has been observed in monitoring data. Since, the Composite Analysis model predicts relatively fast reduction of plume concentrations as they migrate from the source, it is important to evaluate the reasonableness of the Composite Analysis model predicted fall off in concentration levels with migration distance. This can be accomplished by comparing simulated reduction of modeled concentrations to the observed reduction of groundwater concentrations at different migration distances from the source. Tritium groundwater concentrations measured in wells near the PUREX facility during its early operations and more recent measurements in observation wells located within the tritium plume outside of the buffer zone provide the data sets for evaluating the reasonableness of model predicted plume dispersion with distance.

Process condensate liquid waste containing tritium from PUREX operations was discharged to ground at the 216-A-10 crib south of PUREX in the 200 East area. The crib was initially operated for a 4-month period in 1956. In 1961, the crib received PUREX effluent continuously until 1973; it then received waste sporadically in 1977, 1978, and 1981. In 1982, effluent discharges resumed on a continuous basis until the crib was taken out of service and replaced by the 216-A-45 crib in 1987. The effect of the effluent discharges on groundwater near the 216-A-10 crib have been monitored in two wells, 299-E17-1 and 299-E24-2 since the 1961 start of operations. Long-term concentration histories at these two wells demonstrate that groundwater concentrations of tritium were at their highest within 1 to 2 years after the start of operations. A maximum tritium concentration of 4.6×10^7 pCi/L was measured in well 299-E24-2 in 1963 (Figure 4.51).

Approximately 10 to 12 km downgradient from the PUREX facility, maximum tritium levels observed in the plume, which has now moved toward the Columbia River, are just above 300,000 pCi/L. One example of these observations is the tritium levels in well 699-42-12A (Figure 4.52) where concentrations between 300,000 and 360,000 pCi/L were observed between 1976 and 1988. The peak values are approximately 150 times lower than levels that were originally observed near the PUREX facility in 1963. If decay of tritium is considered (i.e., a factor of 2), concentration levels of tritium following its migration to this area over a 12- to 13-year period would be about 75 times lower than maximum levels originally observed near PUREX.

The increases in tritium levels suspected to originate from near the PUREX facilities have also been observed in numerous wells within 5 to 6 km downgradient of the PUREX facilities just outside the buffer zone. Concentration histories for two wells, 699-31-31 and 699-34-39A (Figure 4.53) illustrate the rise and fall of elevated tritium concentrations with time in the area just outside of the 200 East Area southeast of PUREX. At these locations, tritium concentrations rose to levels of 4 to 5 million pCi/L in the early 1960s. These levels are about a factor of about ten lower than levels observed near PUREX.

Composite Analysis results simulated with the current model (Cole et al. 1997) are consistent with the early observations of dispersion of the tritium plume resulting from early PUREX discharges. Composite Analysis existing plume results of tritium transport for the period from 1979 to 1996, which incorporated the restart of discharges to PUREX in the mid-1980s, were compared with the well observations made 5 to 6 km downgradient of the PUREX facilities discussed above. Model transport results from Cole et al. (1997) for 1985 (Figure 4.54), the period of maximum simulated tritium concentrations at PUREX, show approximately an order of magnitude decline of tritium concentrations as the resultant tritium plume migrates outside the buffer zone boundary southeast of PUREX. This result is generally consistent with order of magnitude decrease in tritium levels that were observed in wells 5 to 6 km downgradient of PUREX in the early 1960s (Figures 4.51 and 4.53).

Table 4.1. Summary of Key Assumptions for the Source Release Models

Assumption	Rationale	Impact
Instantaneous response to changes in recharge rates.	Sites are generally shallow and should respond quickly to changes in recharge relative to the 1000-year study period.	Changes in recharge at deeper sites will occur gradually over many years. Since decreased recharge results in decreased release from the waste form for each of the release models, when recharge rates decrease the model will underestimate the release for the next few years.
Uniform release of contaminants in liquid releases.	Insufficient data were available to justify distributing the mass of contaminants released in liquid discharges in any specific distribution.	If the majority of mass releases occurred early in the operation of the liquid disposals, the approach employed in the Composite Analysis would underestimate the cumulative mass release at the water table. However, within a few hundred years it can be expected that the cumulative releases would be approximately equal.
Water content in soil-debris waste form is constant and equal to estimated pre-Hanford soil moisture content of surrounding soil.	Soil hydraulic properties of soil-debris waste forms are generally unavailable.	In the soil-debris release model, given a specific recharge rate, lowering the soil moisture would result in earlier cumulative releases. Using a low moisture content (estimated from the hydraulic properties of adjacent soil and a steady infiltration rate of 5 mm/yr) would result in earlier cumulative releases except in cases where a barrier reduces the recharge to below 5 mm/yr. However, none of the solid waste disposals with barriers considered in the Composite Analysis provide breakthrough within the 1000 years.

Table 4.1. (contd)

Assumption	Rationale	Impact
Only a single release model was considered for each site.	Inadequate data were available to estimate inventories that may have been disposed in different waste types at the same site. However, tanks were treated as three separate sites: tank leaks; tank losses; and tank residuals.	Highly mobile wastes may be handled separately from less-mobile wastes. For instance, highly mobile waste may be packaged differently (e.g., cement waste forms) and disposed in a solid waste burial ground with less-mobile wastes. The Composite Analysis selected the release model that would result in the earliest cumulative release.
Soil-debris release models assumed the waste form was continuously mixed.	The parameters and distributions of inventories within the waste forms were highly uncertain. Using a completely stirred tank reactor model is a reasonable approximation.	Completely mixing the waste form can result in earlier releases by sufficiently diluting the inventory to prevent any local controls on the release (e.g., solubility controls around a hot spot in the waste form).
Soil-debris release models assumed the waste form was continuously mixed.	The parameters and distributions of inventories within the waste forms were highly uncertain. Using a completely stirred tank reactor model is a reasonable approximation.	Completely mixing the waste form would decrease the early cumulative releases by continuously redistributing the mass into the upper portions of the waste form. Therefore, this is not a conservative assumption. The magnitude of the impact varies for each site. It is most likely to affect releases of highly mobile wastes by delaying their release.

Table 4.2. Description of Worksheets in the *Composite Analysis.xls Workbook*

Worksheet	Function	Primary Fixed Fields	Primary Derived Fields
Source Site	Contains most of the primary data regarding geometry, geochemistry, and timing of releases and recharge for all of the sources considered.	Location • Northing (m) • Easting (m) Depth of Source (m) Release Model Class Waste Type Area (m^2) K_d Switch Depth (m) Volume (m^3) Recharge Dates (yr) Recharge Rates (cm/yr) Water Table Elevation	Column Name Layer thicknesses (m) Corrected Area (m^2)
Soil	Contains soil hydraulic parameters for each of the soils considered.	van Genuchten alpha (-) van Genuchten n (L/cm) Residual water content (-) Porosity (-) Saturated hydraulic conductivity (cm/s) Bulk Density (g/cm^3) Gravel Fraction (%)	Initial Saturation[1]
Column	Contains description of stratigraphy of Hanford from available columns.	Location • Northing (m) • Easting (m) Stratigraphy • Thickness (m) • Soil Type	
Recharge	Contains actual values for various recharge classes.	Recharge Rates (cm/yr)	
Ground Surface & Water Table	Contains gridded ground surface and gridded water table elevations based on CFEST simulation for 1979.	Location • Northing (m) • Easting (m) Elevation • Ground surface (m) • Water Table (m)	
K_d and Release Model Classes	Contains best estimates of K_d for both near-field and far-field for each waste class	Waste Classes K_d for both near-field and far-field	

Table 4.3. (contd)

Worksheet	Function	Primary Fixed Fields	Primary Derived Fields
Nuclides & Release Data	Contains the parameters for each radionuclide for each of the release models.	Atomic Number (-) Aqueous Solubility (Ci/L) Fractional release from glass (%) Cement diffusion coefficient (cm^2/yr) Fractional release from reactor (%) Half-life (yr)	Specific activity
Inventory	Contains radionuclide inventories for each site assembled from a variety of independent Excel™workbooks.		Inventory decayed to 2050 (Ci)
CFEST-time-step-ends	Contains the time steps for which CFEST is set to accept estimates of flux to the water table.	Times (yr)	
Source CFEST-nodes map	Contains the distribution of each site's instantaneous flux to one or more CFEST nodes.	Fractional distribution of flux (%)	
CFEST input	Contains the decayed instantaneous fluxes to the water table at each of the respective CFEST nodes for each of the CFEST time steps.		Decayed instantaneous fluxes (Ci)
Flux	Contains the undecayed (2050) annual cumulative flux for each site that breaks through to the water table within 1500 years.		Annual cumulative flux (Ci)
Temp	Contains the unit release breakthrough times from STOMP simulation and the annual releases from waste form to upper vadose zone predicted with the appropriate inventory and release model for the site.	Cumulative unit flux predicted by STOMP (-)	Annual cumulative release to upper vadose zone (Ci) Annual cumulative release to water table (Ci)

(1) Estimate based on steady-state flux of 0.5/cm/yr using algorithm developed by Rockhold, Simmons, and Fayer (1997).

Table 4.3. Source Geometry Data Required for Release Models in the *Source Site* Worksheet

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
207-U	135,044	566,973	3.00	140	208	299-W14-7	Liquid	Retention Basin	5.0E+03
216-A-1	136,082	575,522	4.57	122	214	299-E25-2	Liquid	Crib	9.8E+01
216-A-10	135,440	574,978	13.72	122	221	299-E24-7	Liquid	Crib	3.2E+06
216-A-18	136,236	575,580	4.57	122	209	299-E25-2	Liquid	Trench	4.9E+02
216-A-19	136,278	575,665	4.57	122	203	299-E25-2	Liquid	Trench	1.1E+03
216-A-2	135,529	575,180	8.23	122	221	299-E25-2	Liquid	Trench	2.3E+02
216-A-20	136,249	575,707	4.57	122	203	299-E25-2	Liquid	Crib	9.6E+02
216-A-21	135,462	575,215	5.79	122	221	299-E25-2	Liquid	Crib	7.8E+04
216-A-24	136,397	575,852	4.57	122	197	299-E26-8	Liquid	Crib	8.2E+05
216-A-25	139,654	574,935	10.00	123	169	218-E-12B	Liquid	Pond	3.1E+08
216-A-27	135,401	575,197	4.27	122	221	299-E25-2	Liquid	Crib	2.3E+04
216-A-28	135,779	575,083	3.35	122	216	299-E25-2	Liquid	Crib	3.0E+01
216-A-3	135,820	575,100	4.88	122	216	299-E25-2	Liquid	Crib	3.1E+03
216-A-30	135,508	575,981	3.66	122	210	299-E25-2	Liquid	Crib	7.1E+06
216-A-31	135,484	575,166	7.32	122	221	299-E25-2	Liquid	Crib	1.0E+01
216-A-36A/B	135,345	575,106	6.71	122	221	299-E24-7	Liquid	Crib	3.2E+05
216-A-37-1	135,679	575,842	3.35	122	211	299-E25-2	Liquid	Crib	3.8E+05
216-A-37-2	135,526	576,170	4.57	122	210	299-E25-2	Liquid	Crib	1.1E+06
216-A-4	135,529	575,217	7.92	122	221	299-E25-2	Liquid	Crib	6.2E+03
216-A-45	135,161	574,908	11.43	122	221	299-E24-7	Liquid	Crib	1.0E+05
216-A-5	135,493	575,048	9.75	122	221	299-E24-7	Liquid	Crib	1.6E+06
216-A-6	135,648	575,591	5.79	122	214	299-E25-2	Liquid	Crib	3.4E+06
216-A-7	136,044	575,506	4.57	122	214	299-E25-2	Liquid	Crib	3.3E+02
216-A-8	136,194	575,780	4.27	122	203	299-E25-2	Liquid	Crib	1.2E+06
216-A-9	136,036	575,099	3.96	122	216	299-E25-2	Liquid	Crib	9.8E+05
216-B-10A	136,340	573,473	6.10	122	220	299-E28-16	Liquid	Crib	1.0E+04
216-B-10B	136,340	573,451	6.10	122	220	299-E28-16	Liquid	Crib	2.8E+01
216-B-11A&B	137,419	573,851	12.19	122	197	218-E-12B	Liquid	Reverse Well	3.0E+04

Table 4.3. (contd)

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
216-B-12	136,600	573,128	7.92	122	220	299-E28-16	Liquid	Crib	5.2E+05
216-B-14	134,405	573,649	3.66	122	229	299-E13-20	Liquid	Crib	8.7E+03
216-B-15	134,432	573,607	4.57	122	229	299-E13-20	Liquid	Crib	6.3E+03
216-B-16	134,366	573,625	3.66	122	229	299-E13-20	Liquid	Crib	5.6E+03
216-B-17	134,390	573,583	4.27	122	229	299-E13-20	Liquid	Crib	3.4E+03
216-B-18	134,323	573,601	4.27	122	229	299-E13-20	Liquid	Crib	8.5E+03
216-B-19	134,347	573,559	4.27	122	229	299-E13-20	Liquid	Crib	6.4E+03
216-B-20	134,376	573,417	3.05	122	229	299-E13-20	Liquid	Trench	4.7E+03
216-B-21	134,376	573,383	3.05	122	229	299-E13-20	Liquid	Trench	4.7E+03
216-B-2-1	137,089	574,524	1.83	122	203	218-E-12B	Liquid	Ditch	1.5E+08
216-B-22	134,380	573,349	3.66	122	229	299-E13-20	Liquid	Trench	4.7E+03
216-B-2-2	137,068	574,517	2.44	122	203	218-E-12B	Liquid	Ditch	5.0E+04
216-B-23	134,235	573,289	2.44	122	226	299-E13-20	Liquid	Trench	4.5E+03
216-B-2-3	137,036	574,468	2.44	122	203	218-E-12B	Liquid	Ditch	1.9E+03
216-B-24	134,205	573,289	2.44	122	226	299-E13-20	Liquid	Trench	4.7E+03
216-B-25	134,174	573,289	3.05	122	226	299-E13-20	Liquid	Trench	3.8E+03
216-B-26	134,144	573,289	2.44	122	226	299-E13-20	Liquid	Trench	5.9E+03
216-B-27	134,113	573,289	2.44	122	225	299-E13-20	Liquid	Trench	4.4E+03
216-B-28	134,081	573,289	3.96	122	225	299-E13-20	Liquid	Trench	5.1E+03
216-B-29	134,439	573,089	3.05	122	231	299-E13-20	Liquid	Trench	4.8E+03
216-B-3	136,687	576,899	10.00	127	178	299-E26-8	Liquid	Pond	2.4E+08
216-B-30	134,402	573,089	3.35	122	231	299-E13-20	Liquid	Trench	4.8E+03
216-B-31	134,361	573,089	3.05	122	229	299-E13-20	Liquid	Reactor	4.7E+03
216-B-32	134,325	573,089	3.05	122	229	299-E13-20	Liquid	Trench	4.8E+03
216-B-33	134,286	573,089	3.05	122	229	299-E13-20	Liquid	Trench	4.7E+03
216-B-34	134,250	573,089	3.05	122	229	299-E13-20	Liquid	Trench	4.9E+03
216-B-35	137,274	573,439	3.05	122	206	218-E-10	Liquid	Trench	1.1E+03
216-B-36	137,292	573,439	3.05	122	206	218-E-10	Liquid	Trench	1.9E+03
216-B-37	137,318	573,439	3.05	122	206	218-E-10	Liquid	Trench	4.3E+03

Table 4.3. (contd)

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
216-B-38	137,345	573,439	3.05	122	206	218-E-10	Liquid	Trench	1.4E+03
216-B-39	137,373	573,439	3.05	122	206	218-E-10	Liquid	Trench	1.5E+03
216-B-40	137,400	573,439	3.05	122	201	218-E-10	Liquid	Trench	1.6E+03
216-B-41	137,427	573,439	3.05	122	201	218-E-10	Liquid	Trench	1.4E+03
216-B-43	137,614	573,625	4.57	122	201	218-E-10	Liquid	Crib	2.1E+03
216-B-44	137,640	573,625	4.57	122	193	218-E-10	Liquid	Crib	5.6E+03
216-B-45	137,666	573,625	4.57	122	193	218-E-10	Liquid	Crib	4.9E+03
216-B-46	137,692	573,625	4.57	122	193	218-E-10	Liquid	Crib	6.7E+03
216-B-47	137,614	573,582	4.57	122	201	218-E-10	Liquid	Crib	3.7E+03
216-B-48	137,640	573,582	4.57	122	193	218-E-10	Liquid	Crib	4.1E+03
216-B-49	137,666	573,582	4.57	122	193	218-E-10	Liquid	Crib	6.7E+03
216-B-5	136,732	573,781	92.05	122	215	299-E28-22	Liquid	Reverse Well	3.1E+04
216-B-50	137,692	573,582	4.57	122	193	218-E-10	Liquid	Crib	5.5E+04
216-B-52	134,271	573,296	3.05	122	226	299-E13-20	Liquid	Trench	8.5E+03
216-B-53A	134,441	573,235	3.05	122	229	299-E13-20	Liquid	Trench	5.5E+02
216-B-53B	134,423	573,241	2.44	122	229	299-E13-20	Liquid	Trench	1.5E+01
216-B-54	134,379	573,242	2.44	122	229	299-E13-20	Liquid	Trench	1.0E+03
216-B-55	136,495	573,092	3.66	122	221	299-E28-16	Liquid	Crib	1.2E+06
216-B-57	137,579	573,499	3.05	122	201	218-E-10	Liquid	Crib	8.4E+04
216-B-58	134,349	573,242	2.44	122	226	299-E13-20	Liquid	Trench	4.1E+02
216-B-59	136,636	573,851	3.66	122	215	299-E28-22	Liquid	Retention Basin	2.5E+02
216-B-60	136,470	573,365	12.19	122	220	299-E28-16	Liquid	crib	1.9E+01
216-B-62	136,815	573,075	5.49	122	215	299-E28-16	Liquid	Crib	2.8E+05
216-B-63	137,199	574,189	3.05	122	196	218-E-12B	Liquid	Ditch	7.2E+06
216-B-7A&B	137,393	573,799	4.27	122	197	218-E-12B	Liquid	Crib	4.4E+04
216-B-8	137,505	573,808	7.01	122	197	218-E-12B	Liquid	Crib	2.7E+04
216-B-9	136,850	573,852	9.14	122	215	299-E28-22	Liquid	Crib	3.6E+04
216-C-1	136,304	574,580	3.96	122	215	299-E28-22	Liquid	Crib	2.3E+04
216-C-10	136,314	574,697	2.13	122	214	299-E28-22	Liquid	Crib	9.0E+02

Table 4.3. (contd)

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
216-C-3	136,300	574,534	3.05	122	215	299-E28-22	Liquid	Crib	5.0E+03
216-C-4	136,305	574,522	4.88	122	215	299-E28-22	Liquid	Crib	1.7E+02
216-C-5	136,292	574,543	4.88	122	215	299-E28-22	Liquid	Crib	3.8E+01
216-C-6	136,288	574,632	4.88	122	214	299-E28-22	Liquid	Crib	5.3E+02
216-C-7	136,283	574,448	3.66	122	215	299-E28-22	Liquid	Crib	6.0E+01
216-C-9	136,478	574,585	7.62	122	215	299-E28-22	Liquid	Pond	1.0E+06
216-N-2	140,380	569,829	2.13	127	175	299-W6-1	Liquid	Trench	7.6E+03
216-N-3	140,371	569,818	1.83	127	175	299-W6-1	Liquid	Trench	7.6E+03
216-N-4	139,933	570,754	0.91	123	177	218-E-10	Liquid	Pond	9.5E+05
216-N-5	140,374	570,635	1.83	123	177	218-E-10	Liquid	Trench	7.6E+03
216-N-6	139,895	571,643	0.91	122	176	218-E-10	Liquid	Pond	9.5E+05
216-N-7	140,384	571,434	1.83	122	173	218-E-10	Liquid	Trench	7.6E+03
216-S-1&2	134,260	566,980	10.67	139	206	299-W22-24	Liquid	Crib	1.6E+05
216-S-10D	133,440	566,650	1.83	139	203	299-W22-24	Liquid	Ditch	4.3E+06
216-S-11	133,270	566,473	3.00	139	203	299-W22-24	Liquid	Pond	2.2E+06
216-S-12	134,120	567,531	3.05	138	210	299-W22-24	Liquid	Trench	6.8E+01
216-S-13	134,011	567,155	10.36	138	209	299-W22-24	Liquid	Crib	5.0E+03
216-S-16P	133,254	565,033	0.91	141	191	299-W18-21	Liquid	Pond	4.1E+07
216-S-17	133,248	565,991	3.05	140	199	299-W18-21	Liquid	Pond	6.4E+06
216-S-19	133,435	567,678	10.00	137	206	299-W22-24	Liquid	Pond	1.3E+06
216-S-20	133,917	567,554	9.14	138	210	299-W22-24	Liquid	Crib	1.4E+05
216-S-22	133,989	567,608	3.05	138	210	299-W22-24	Liquid	Crib	9.8E+01
216-S-23	134,692	567,114	8.23	140	208	299-W22-24	Liquid	Crib	3.4E+04
216-S-25	134,287	566,570	3.05	139	204	299-W18-21	Liquid	Crib	2.9E+05
216-S-26	133,760	567,595	3.66	138	211	299-W22-24	Liquid	Crib	1.6E+05
216-S-3	134,438	566,893	1.83	139	207	299-W22-24	Liquid	French Drain	4.2E+03
216-S-5	133,440	566,430	4.57	139	199	299-W22-24	Liquid	Crib	4.1E+06
216-S-6	133,596	566,217	4.57	139	202	299-W18-21	Liquid	Crib	4.5E+06
216-S-7	134,176	567,168	6.71	139	208	299-W22-24	Liquid	Crib	3.9E+05

Table 4.3. (contd)

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
216-S-8	134,223	566,926	7.62	139	206	299-W22-24	Liquid	Trench	1.0E+04
216-S-9	134,493	567,175	9.14	139	212	299-W22-24	Liquid	Crib	5.0E+04
216-T-1	137,083	567,574	3.05	139	218	299-W11-2	Liquid	Ditch	1.8E+05
216-T-12	136,737	566,993	2.44	140	212	299-W11-2	Liquid	Trench	5.0E+03
216-T-14	136,839	566,948	3.05	140	212	299-W11-2	Liquid	Trench	1.0E+03
216-T-15	136,836	566,976	3.05	140	212	299-W11-2	Liquid	Trench	1.0E+03
216-T-16	136,836	567,003	3.05	140	212	299-W11-2	Liquid	Trench	1.0E+03
216-T-17	136,836	567,018	3.05	140	212	299-W11-2	Liquid	Trench	1.0E+03
216-T-18	136,460	566,949	4.57	140	209	299-W11-2	Liquid	Crib	1.0E+03
216-T-19	135,974	566,849	10.00	141	204	299-W14-7	Liquid	Crib	4.6E+05
216-T-20	136,074	567,119	1.22	140	209	299-W14-7	Liquid	Trench	1.9E+01
216-T-21	136,119	566,555	3.05	141	208	299-W15-15	Liquid	Trench	4.6E+02
216-T-22	136,146	566,555	3.05	141	211	299-W15-15	Liquid	Trench	1.5E+03
216-T-23	136,174	566,555	3.05	141	211	299-W15-15	Liquid	Trench	1.5E+03
216-T-24	136,201	566,555	3.05	141	211	299-W15-15	Liquid	Trench	1.5E+03
216-T-25	136,228	566,546	3.05	141	211	299-W15-15	Liquid	Trench	3.0E+03
216-T-26	136,399	566,932	4.57	140	209	299-W11-2	Liquid	Crib	1.2E+04
216-T-27	136,373	566,933	4.57	140	207	299-W11-2	Liquid	Crib	7.2E+03
216-T-28	136,347	566,933	4.57	140	207	299-W11-2	Liquid	Crib	4.2E+04
216-T-3	136,671	567,261	62.80	139	220	299-W11-2	Liquid	Reverse Well	1.1E+04
216-T-32	136,696	566,719	7.92	140	210	299-W11-2	Liquid	Crib	2.9E+04
216-T-33	136,898	567,462	3.35	139	218	299-W11-2	Liquid	Crib	1.9E+03
216-T-34	137,111	567,265	4.88	139	215	299-W6-1	Liquid	Crib	1.7E+04
216-T-35	137,108	567,168	4.57	139	215	299-W6-1	Liquid	Crib	5.7E+03
216-T-36	136,596	566,702	4.57	140	208	299-W11-2	Liquid	Crib	5.2E+02
216-T-4B	137,271	566,523	1.22	139	207	299-W6-1	Liquid	Pond	2.4E+04
216-T-5	136,727	566,667	3.66	140	210	299-W11-2	Liquid	Trench	2.6E+03
216-T-6	136,663	567,188	7.62	139	220	299-W11-2	Liquid	Crib	4.5E+04
216-T-7	136,660	566,685	10.00	140	210	299-W11-2	Liquid	Crib	1.1E+05

Table 4.3. (contd)

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
216-T-8	136,727	567,651	6.10	138	223	299-W11-2	Liquid	Crib	5.0E+02
216-U-1&2	135,002	567,243	7.32	140	212	299-W14-7	Liquid	Crib	4.6E+04
216-U-10	134,624	566,372	10.00	140	201	299-W18-21	Liquid	Pond	1.7E+08
216-U-12	134,502	567,592	3.96	139	212	299-W22-24	Liquid	Crib	1.5E+05
216-U-15	135,127	567,371	4.57	140	215	299-W14-7	Liquid	Trench	6.8E+01
216-U-16	134,861	567,236	10.00	139	211	299-W22-24	Liquid	Crib	4.1E+05
216-U-17	134,904	567,839	10.00	139	216	299-W22-24	Liquid	Crib	2.1E+03
216-U-3	134,928	566,845	3.66	140	202	299-W18-21	Liquid	French Drain	7.9E+02
216-U-4A	135,111	567,580	22.86	139	213	299-W22-24	Liquid	Reverse Well	5.5E+02
216-U-4B	135,121	567,615	22.86	139	213	299-W14-8A	Liquid	Reverse Well	3.3E+01
216-U-5	135,359	567,673	3.05	139	220	299-W14-8A	Liquid	Trench	4.5E+03
216-U-7	135,204	567,611	5.18	139	221	299-W14-8A	Liquid	French Drain	7.0E+00
216-U-8	134,698	567,617	9.45	139	212	299-W22-24	Liquid	Crib	3.8E+05
216-Z-1&2	135,469	566,547	6.40	141	211	299-W14-7	Liquid	Crib	3.4E+04
216-Z-10	135,897	566,567	45.72	141	208	299-W15-15	Liquid	Reverse Well	1.0E+03
216-Z-12	135,423	566,365	6.10	141	212	299-W15-15	Liquid	Crib	2.8E+05
216-Z-16	135,991	566,430	4.57	141	208	299-W15-15	Liquid	Crib	1.0E+05
216-Z-17	135,863	566,603	2.44	141	208	299-W14-7	Liquid	Ditch	3.7E+04
216-Z-18	135,286	566,440	10.00	140	208	299-W18-21	Liquid	Crib	3.9E+03
216-Z-1A	135,419	566,549	10.00	141	211	299-W14-7	Liquid	Crib	5.3E+03
216-Z-20	135,299	566,624	10.00	140	208	299-W14-7	Liquid	Crib	3.8E+06
216-Z-3	135,459	566,577	7.62	141	211	299-W14-7	Liquid	Crib	1.8E+05
216-Z-4	135,921	566,586	4.57	141	208	299-W14-7	Liquid	Trench	1.1E+01
216-Z-5	135,949	566,555	10.00	141	208	299-W15-15	Liquid	Crib	3.1E+04
216-Z-6	135,876	566,579	2.44	141	208	299-W15-15	Liquid	Crib	9.8E+01
216-Z-7	135,927	566,701	1.52	141	204	299-W14-7	Liquid	Crib	7.9E+04
216-Z-8	135,653	566,654	5.18	141	204	299-W14-7	Liquid	Crib	1.0E+01
216-Z-9	135,611	566,758	6.40	141	205	299-W14-7	Liquid	Crib	4.1E+03
218-EC-9(a)#	136,465	574,658	6.71	122	212	299-E28-22	Soil/debris	Burial Site	1.9E+03

Table 4.3. (contd)

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
218-EC-9(b)##	136,465	574,658	6.50	122	212	299-E28-22	Soil/debris	Burial Site	5.7E+03
218-E-1(b)	135,575	574,755	6.50	122	222	299-E24-7	Soil/debris	Burial Site	3.0E+03
218-E-10(b)	137,268	572,945	6.50	122	210	218-E-10	Soil/debris	Burial Site	2.1E+04
218-E-10(a)	137,268	572,945	6.50	122	210	218-E-10	Soil/debris	Burial Site	3.6E+03
218-E-12A(b)	136,803	574,938	6.50	122	202	218-E-12B	Soil/debris	Burial Site	1.5E+04
218-E-12B(b)	137,447	574,796	6.50	122	188	218-E-12B	Soil/debris	Burial Site	5.1E+04
218-E-12B(a)	137,447	574,796	6.50	122	188	218-E-12B	Soil/debris	Burial Site	3.7E+04
218-E-2(b)	137,078	573,511	6.50	122	209	299-E28-16	Soil/debris	Burial Site	9.0E+03
218-E-4(b)	136,891	573,497	6.50	122	209	299-E28-16	Soil/debris	Burial Site	1.6E+03
218-E-5(b)	137,080	573,417	6.50	122	209	299-E28-16	Soil/debris	Burial Site	3.2E+03
218-E-5A(b)	137,088	573,356	6.50	122	211	299-E28-16	Soil/debris	Burial Site	6.2E+03
218-E-8(b)	137,225	575,116	6.50	122	189	218-E-12B	Soil/debris	Burial Site	2.3E+03
218-W-1(b)	136,222	566,205	6.50	140	212	299-W15-15	Soil/debris	Burial Site	7.2E+03
218-W-11(b)	136,319	566,205	6.50	140	212	299-W15-15	Soil/debris	Burial Site	1.2E+03
218-W-1A(b)	137,184	567,060	6.50	139	214	299-W6-1	Soil/debris	Burial Site	1.4E+04
218-W-2(b)	136,062	566,205	6.50	141	208	299-W15-15	Soil/debris	Burial Site	8.2E+03
218-W-2A(b)	136,891	566,425	6.50	140	210	218-W-5	Soil/debris	Burial Site	5.0E+04
218-W-3(b)	136,745	566,166	6.50	140	213	218-W-5	Soil/debris	Burial Site	2.2E+04
218-W-3A(b)	137,282	566,226	6.50	140	210	218-W-5	Soil/debris	Burial Site	9.5E+04
218-W-3A(a)	137,282	566,226	6.50	140	210	218-W-5	Soil/debris	Burial Site	2.4E+04
218-W-3AE(b)	137,391	566,616	6.50	139	210	299-W6-1	Soil/debris	Burial Site	1.1E+04
218-W-3AE(a)	137,391	566,616	6.50	139	210	299-W6-1	Soil/debris	Burial Site	6.7E+04
218-W-4A(b)	136,491	566,228	6.50	140	210	299-W15-15	Soil/debris	Burial Site	1.8E+04
218-W-4B-c(b)	135,881	566,191	6.50	141	208	299-W15-15	Cement	Burial Site	1.0E+04
218-W-4B-c(a)	135,881	566,191	6.50	141	208	299-W15-15	Cement	Burial Site	2.7E+01
218-W-4C(a)	135,086	566,458	6.50	140	207	299-W18-21	Soil/debris	Burial Site	2.7E+04
218-W-4C(b)	135,086	566,458	6.50	140	207	299-W18-21	Soil/debris	Burial Site	1.0E+04
218-W-5(b)	137,165	565,870	6.50	140	219	218-W-5	Soil/debris	Burial Site	6.3E+03
218-W-5(a)	137,165	565,870	6.50	140	219	218-W-5	Soil/debris	Burial Site	1.8E+05

Table 4.3. (contd)

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
218-W-7	133,865	567,485	6.50	138	211	299-W22-24	Soil/debris	Burial Site	1.6E+02
218-W-8	136,775	567,638	6.50	138	223	299-W11-2	Soil/debris	Burial Site	6.8E+01
218-W-9(b)	134,307	567,189	6.50	139	208	299-W22-24	Soil/debris	Burial Site	4.9E+02
TWRS glass grout vault	135,787	576,019	6.5	122	222	299-E24-7	Glass	Burial Site	n/a
TWRS glass new site	135,298	574;371	6.5	122	222	299-E24-7	Glass	Burial Site	n/a
US Ecology current	134,188	572,175	13.7	123	224	299-E19-1	Soil/debris	Burial Site	4.2E+05
US Ecology future	134,188	572,175	13.7	123	224	299-E19-1	Soil/debris	Burial Site	1.0E+06
ERDF	134,422	568,900	14	135	222	299-W21-1	Soil/debris	Burial Site	3.0E+06
C Reactor	136,852	567,570	22	139	222	299-W11-2	Reactor	Reactor	
D Reactor	136,852	567,570	22	139	222	299-W11-2	Reactor	Reactor	
DR Reactor	136,852	567,570	22	139	222	299-W11-2	Reactor	Reactor	
F Reactor	136,852	567,570	22	139	222	299-W11-2	Reactor	Reactor	
H Reactor	136,852	567,570	22	139	222	299-W11-2	Reactor	Reactor	
KE Reactor	136,852	567,570	22	139	222	299-W11-2	Reactor	Reactor	
KW Reactor	136,852	567,570	22	139	222	299-W11-2	Reactor	Reactor	
N Reactor	136,852	567,570	22	139	222	299-W11-2	Reactor	Reactor	
TK-A-S	136,060	575,353	16.6	122	217	299-E25-2	Liquid	Tank	1.8E+02
TK-A-L	136,060	575,353	16.6	122	217	299-E25-2	Liquid	Tank	1.1E+03
TK-A-R	136,060	575,353	16.6	122	217	299-E25-2	Cake	Tank	
TK-AN-R-1	136,423	575,381	17	122	198	299-E26-8	Cake	Tank	
TK-AN-R-2	136,423	575,381	17	122	198	299-E26-8	Cake	Tank	
TK-AP-R-1	135,822	575,556	17	122	214	299-E25-2	Cake	Tank	
TK-AP-R-2	135,822	575,556	17	122	214	299-E25-2	Cake	Tank	
TK-AW-R	135,858	575,356	17	122	210	299-E25-2	Cake	Tank	
TK-AX-S-1	136,189	575,409	16.6	122	209	299-E25-2	Liquid	Tank	6.1E+01
TK-AX-S-2	136,189	575,409	16.6	122	209	299-E25-2	Liquid	Tank	6.1E+01
TK-AX-L-1	136,189	575,409	16.6	122	209	299-E25-2	Liquid	Tank	1.1E+01

Table 4.3. (contd)

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
TK-AX-L-2	136,189	575,409	16.6	122	209	299-E25-2	Liquid	Tank	3.0E+01
TK-AX-R-1	136,189	575,409	16.6	122	209	299-E25-2	Cake	Tank	
TK-AX-R-2	136,189	575,409	16.6	122	209	299-E25-2	Cake	Tank	
TK-AY-R-1	136,188	575,312	17	122	210	299-E25-2	Cake	Tank	
TK-AY-R-2	136,188	575,312	17	122	210	299-E25-2	Cake	Tank	
TK-AZ-R	136,311	575,397	17	122	209	299-E25-2	Cake	Tank	
TK-B-S	137,299	573,826	10.3	122	204	218-E-12B	Liquid	Tank	4.8E+02
TK-B-L	137,299	573,826	10.3	122	204	218-E-12B	Liquid	Tank	2.0E+02
TK-B-R	137,299	573,826	10.3	122	204	218-E-12B	Cake	Tank	
TK-BX-S	137,347	573,614	12.5	122	206	218-E-10	Liquid	Tank	3.6E+02
TK-BX-L	137,347	573,614	12.5	122	206	218-E-10	Liquid	Tank	3.7E+02
TK-BX-R	137,347	573,614	12.5	122	206	218-E-10	Cake	Tank	
TK-BY-S	137,501	573,613	14.3	122	201	218-E-10	Liquid	Tank	3.6E+02
TK-BY-L	137,501	573,613	14.3	122	201	218-E-10	Liquid	Tank	1.6E+02
TK-BY-R	137,501	573,613	14.3	122	201	218-E-10	Cake	Tank	
TK-C-S-1	136,559	575,151	10.3	122	203	299-E26-8	Liquid	Tank	1.2E+02
TK-C-S-2	136,559	575,151	10.3	122	203	299-E26-8	Liquid	Tank	3.3E+02
TK-C-L-1	136,559	575,151	10.3	122	203	299-E26-8	Liquid	Tank	7.6E+00
TK-C-L-2	136,559	575,151	10.3	122	203	299-E26-8	Liquid	Tank	1.0E+02
TK-C-R-1	136,559	575,151	10.3	122	203	299-E26-8	Cake	Tank	
TK-C-R-2	136,559	575,151	10.3	122	203	299-E26-8	Cake	Tank	
TK-S-S	134,236	566,804	14.3	139	202	299-W22-24	Liquid	Tank	3.6E+02
TK-S-L	134,236	566,804	14.3	139	202	299-W22-24	Liquid	Tank	9.1E+01
TK-S-R	134,236	566,804	14.3	139	202	299-W22-24	Cake	Tank	
TK-SX-S-1	134,456	566,804	16.6	140	203	299-W22-24	Liquid	Tank	3.0E+01
TK-SX-S-2	134,456	566,804	16.6	140	203	299-W22-24	Liquid	Tank	4.2E+02
TK-SX-L-2	134,456	566,804	16.6	140	203	299-W22-24	Liquid	Tank	6.3E+02
TK-SX-R-1	134,456	566,804	16.6	140	203	299-W22-24	Cake	Tank	
TK-SX-R-2	134,456	566,804	16.6	140	203	299-W22-24	Cake	Tank	

Table 4.3. (contd)

Site Name	Northing (m)*	Easting (m)**	Depth (m)***	Water Table Elevation(m)+	Ground Surface Elevation (m)	Column Name++	Release Model Class	Source Type Name	Volume (m^3)+++
TK-SY-R-1	134,541	566,883	17	139	207	299-W22-24	Cake	Tank	
TK-SY-R-2	134,541	566,883	17	139	207	299-W22-24	Cake	Tank	
TK-T-S	136,719	566,806	10.3	140	210	299-W11-2	Liquid	Tank	4.8E+02
TK-T-L	136,719	566,806	10.3	140	210	299-W11-2	Liquid	Tank	5.1E+02
TK-T-R	136,719	566,806	10.3	140	210	299-W11-2	Cake	Tank	
TK-TX-S	136,217	566,759	14.3	140	207	299-W14-7	Liquid	Tank	5.5E+02
TK-TX-L	136,217	566,759	14.3	140	207	299-W14-7	Liquid	Tank	2.2E+02
TK-TX-R	136,217	566,759	14.3	140	207	299-W14-7	Cake	Tank	
TK-TY-S	136,416	566,758	14.3	140	208	299-W11-2	Liquid	Tank	1.8E+02
TK-TY-L	136,416	566,758	14.3	140	208	299-W11-2	Liquid	Tank	2.3E+02
TK-TY-R	136,416	566,758	14.3	140	208	299-W11-2	Cake	Tank	
TK-U-S	135,058	566,812	10.3	140	202	299-W14-7	Liquid	Tank	4.8E+02
TK-U-L	135,058	566,812	10.3	140	202	299-W14-7	Liquid	Tank	3.8E+02
TK-U-R	135,058	566,812	10.3	140	202	299-W14-7	Cake	Tank	

* Refers to north coordinate in Washington State Plane NAD83 coordinate system.

** Refers to east coordinate in Washington State Plane NAD83 coordinate system.

*** Refers to the depth of the source below the ground surface.

\+ Water table elevation estimated for 1979 using CFEST groundwater model. Because of a reduction in liquid disposals, water table elevations are predicted to decline further.

++ See Table 4.6 for description of columns.

+++ For liquid disposals, "volume" refers to the volume of the liquid released. For a solid waste site, "volume" refers to the volumetric capacity of the site.

\# (a) refers to waste disposed after September 30, 1988.

(b) refers to waste disposed before September 30, 1988.

Table 4.4. Chemical Classification of Waste Sites

Site Name	Waste Type Name	K_d Switch Depth (m)*
207-U	Low Organic - Low Salts - Near Neutral	0.0
216-A-1	Low Organic - Low Salts - Near Neutral	0.0
216-A-10	Low Organic - Low Salts - Near Neutral	0.0
216-A-18	Low Organic - Low Salts - Near Neutral	0.0
216-A-19	Low Organic - Low Salts - Near Neutral	0.0
216-A-2	High Organic - Near Neutral	0.0
216-A-20	Low Organic - Low Salts - Near Neutral	0.0
216-A-21	Low Organic - Low Salts - Near Neutral	0.0
216-A-24	High Organic - Near Neutral	0.0
216-A-25	Low Organic - Low Salts - Near Neutral	0.0
216-A-27	Low Organic - Low Salts - Near Neutral	0.0
216-A-28	Low Organic - Low Salts - Near Neutral	0.0
216-A-3	Low Organic - Low Salts - Near Neutral	0.0
216-A-30	Low Organic - Low Salts - Near Neutral	0.0
216-A-31	High Organic - Near Neutral	0.0
216-A-36A/B	Low Organic - Low Salts - Near Neutral	0.0
216-A-37-1	Low Organic - Low Salts - Near Neutral	0.0
216-A-37-2	Low Organic - Low Salts - Near Neutral	0.0
216-A-4	Low Organic - Low Salts - Near Neutral	0.0
216-A-45	Low Organic - Low Salts - Near Neutral	0.0
216-A-5	Low Organic - Low Salts - Near Neutral	0.0
216-A-6	Low Organic - Low Salts - Near Neutral	0.0
216-A-7	High Organic - Near Neutral	0.0
216-A-8	High Organic - Near Neutral	0.0
216-A-9	Low Organic - Low Salts - Near Neutral	0.0
216-B-10A	Low Organic - Low Salts - Near Neutral	0.0
216-B-10B	Low Organic - Low Salts - Near Neutral	0.0
216-B-11A&B	Low Organic - Low Salts - Near Neutral	0.0
216-B-12	Low Organic - Low Salts - Near Neutral	0.0
216-B-14	Chelates - High Salts	0.0
216-B-15	Chelates - High Salts	0.0
216-B-16	Chelates - High Salts	0.0
216-B-17	Chelates - High Salts	0.0
216-B-18	Chelates - High Salts	0.0
216-B-19	Chelates - High Salts	0.0
216-B-20	Chelates - High Salts	0.0
216-B-21	Chelates - High Salts	0.0
216-B-2-1	Low Organic - Low Salts - Near Neutral	0.0

Table 4.4. (contd)

Site Name	Waste Type Name	K_d Switch Depth (m)*
216-B-22	Chelates - High Salts	0.0
216-B-2-2	Low Organic - Low Salts - Near Neutral	0.0
216-B-23	Chelates - High Salts	0.0
216-B-2-3	Low Organic - Low Salts - Near Neutral	0.0
216-B-24	Chelates - High Salts	0.0
216-B-25	Chelates - High Salts	0.0
216-B-26	Chelates - High Salts	0.0
216-B-27	Chelates - High Salts	0.0
216-B-28	Chelates - High Salts	0.0
216-B-29	Chelates - High Salts	0.0
216-B-3	Low Organic - Low Salts - Near Neutral	0.0
216-B-30	Chelates - High Salts	0.0
216-B-31	Chelates - High Salts	0.0
216-B-32	Chelates - High Salts	0.0
216-B-33	Chelates - High Salts	0.0
216-B-34	Chelates - High Salts	0.0
216-B-35	Very High Salts - Very Basic	7.0
216-B-36	Very High Salts - Very Basic	7.0
216-B-37	Very High Salts - Very Basic	7.0
216-B-38	Very High Salts - Very Basic	7.0
216-B-39	Very High Salts - Very Basic	7.0
216-B-40	Very High Salts - Very Basic	7.0
216-B-41	Very High Salts - Very Basic	7.0
216-B-43	Chelates - High Salts	0.0
216-B-44	Chelates - High Salts	0.0
216-B-45	Chelates - High Salts	0.0
216-B-46	Chelates - High Salts	0.0
216-B-47	Chelates - High Salts	0.0
216-B-48	Chelates - High Salts	0.0
216-B-49	Chelates - High Salts	0.0
216-B-5	Very High Salts - Very Basic	0.0
216-B-50	Low Organic - Low Salts - Near Neutral	0.0
216-B-52	Chelates - High Salts	0.0
216-B-53A	Low Organic - Low Salts - Near Neutral	0.0
216-B-53B	Low Organic - Low Salts - Near Neutral	0.0
216-B-54	Low Organic - Low Salts - Near Neutral	0.0
216-B-55	Low Organic - Low Salts - Near Neutral	0.0
216-B-57	Low Organic - Low Salts - Near Neutral	0.0

Table 4.4. (contd)

Site Name	Waste Type Name	K_d Switch Depth (m)*
216-B-58	Low Organic - Low Salts - Near Neutral	0.0
216-B-59	Low Organic - Low Salts - Near Neutral	0.0
216-B-60	Low Organic - Low Salts - Near Neutral	0.0
216-B-62	Low Organic - Low Salts - Near Neutral	0.0
216-B-63	Low Organic - Low Salts - Near Neutral	0.0
216-B-7A&B	Very High Salts - Very Basic	35.7
216-B-8	Very High Salts - Very Basic	33.0
216-B-9	Very High Salts - Very Basic	0.9
216-C-1	Very High Salts - Very Basic	0.0
216-C-10	Low Organic - Low Salts - Near Neutral	0.0
216-C-3	Low Organic - Low Salts - Near Neutral	0.0
216-C-4	High Organic - Near Neutral	0.0
216-C-5	Low Organic - Low Salts - Near Neutral	0.0
216-C-6	Low Organic - Low Salts - Near Neutral	0.0
216-C-7	Low Organic - Low Salts - Near Neutral	0.0
216-C-9	Low Organic - Low Salts - Near Neutral	0.0
216-N-2	Low Organic - Low Salts - Near Neutral	0.0
216-N-3	Low Organic - Low Salts - Near Neutral	0.0
216-N-4	Low Organic - Low Salts - Near Neutral	0.0
216-N-5	Low Organic - Low Salts - Near Neutral	0.0
216-N-6	Low Organic - Low Salts - Near Neutral	0.0
216-N-7	Low Organic - Low Salts - Near Neutral	0.0
216-S-1&2	Low Organic - Low Salts - Acidic	35.3
216-S-10D	Low Organic - Low Salts - Near Neutral	0.0
216-S-11	Low Organic - Low Salts - Near Neutral	0.0
216-S-12	Low Organic - Low Salts - Near Neutral	0.0
216-S-13	High Organic - Near Neutral	0.0
216-S-16P	Low Organic - Low Salts - Near Neutral	0.0
216-S-17	Low Organic - Low Salts - Near Neutral	0.0
216-S-19	Low Organic - Low Salts - Near Neutral	0.0
216-S-20	Low Organic - Low Salts - Near Neutral	0.0
216-S-22	Low Organic - Low Salts - Near Neutral	0.0
216-S-23	Low Organic - Low Salts - Near Neutral	0.0
216-S-25	Low Organic - Low Salts - Near Neutral	0.0
216-S-26	Low Organic - Low Salts - Near Neutral	0.0
216-S-3	Low Organic - Low Salts - Near Neutral	0.0
216-S-5	Low Organic - Low Salts - Near Neutral	0.0
216-S-6	Low Organic - Low Salts - Near Neutral	0.0

Table 4.4. (contd)

Site Name	Waste Type Name	K_d Switch Depth (m)*
216-S-7	Low Organic - Low Salts - Near Neutral	0.0
216-S-8	Low Organic - Low Salts - Near Neutral	0.0
216-S-9	Low Organic - Low Salts - Acidic	32.9
216-T-1	Low Organic - Low Salts - Near Neutral	0.0
216-T-12	Low Organic - Low Salts - Near Neutral	0.0
216-T-14	Very High Salts - Very Basic	7.0
216-T-15	Very High Salts - Very Basic	7.0
216-T-16	Very High Salts - Very Basic	7.0
216-T-17	Very High Salts - Very Basic	7.0
216-T-18	Very High Salts - Very Basic	5.4
216-T-19	High Organic - Near Neutral	0.0
216-T-20	Low Organic - Low Salts - Near Neutral	0.0
216-T-21	Very High Salts - Very Basic	7.0
216-T-22	Very High Salts - Very Basic	7.0
216-T-23	Very High Salts - Very Basic	7.0
216-T-24	Very High Salts - Very Basic	7.0
216-T-25	Very High Salts - Very Basic	17.0
216-T-26	Chelates - High Salts	0.0
216-T-27	Low Organic - Low Salts - Near Neutral	0.0
216-T-28	Low Organic - Low Salts - Near Neutral	0.0
216-T-3	Very High Salts - Very Basic	0.0
216-T-32	Very High Salts - Very Basic	32.1
216-T-33	Low Organic - Low Salts - Near Neutral	0.0
216-T-34	Low Organic - Low Salts - Near Neutral	0.0
216-T-35	Low Organic - Low Salts - Near Neutral	0.0
216-T-36	Low Organic - Low Salts - Near Neutral	0.0
216-T-4B	Low Organic - Low Salts - Near Neutral	0.0
216-T-5	Very High Salts - Very Basic	26.3
216-T-6	Very High Salts - Very Basic	12.4
216-T-7	Very High Salts - Very Basic	20.0
216-T-8	Low Organic - Low Salts - Near Neutral	0.0
216-U-1&2	Low Organic - Low Salts - Acidic	22.7
216-U-10	Low Organic - Low Salts - Near Neutral	0.0
216-U-12	Low Organic - Low Salts - Near Neutral	0.0
216-U-15	High Organic - Near Neutral	0.0
216-U-16	Low Organic - Low Salts - Near Neutral	0.0
216-U-17	Low Organic - Low Salts - Near Neutral	0.0
216-U-3	Low Organic - Low Salts - Near Neutral	0.0

Table 4.4. (contd)

Site Name	Waste Type Name	K_d Switch Depth (m)*
216-U-4A	Low Organic - Low Salts - Near Neutral	0.0
216-U-4B	Low Organic - Low Salts - Near Neutral	0.0
216-U-5	Low Organic - Low Salts - Near Neutral	0.0
216-U-6	Low Organic - Low Salts - Near Neutral	0.0
216-U-7	Low Organic - Low Salts - Near Neutral	0.0
216-U-8	Low Organic - Low Salts - Acidic	20.6
216-Z-1&2	High Organic - Near Neutral	0.0
216-Z-10	Low Organic - Low Salts - Near Neutral	0.0
216-Z-12	High Organic - Near Neutral	0.0
216-Z-16	Low Organic - Low Salts - Near Neutral	0.0
216-Z-17	Low Organic - Low Salts - Near Neutral	0.0
216-Z-18	High Organic - Very Acidic	10.0
216-Z-1A	High Organic - Very Acidic	1.0
216-Z-20	Low Organic - Low Salts - Near Neutral	0.0
216-Z-3	High Organic - Very Acidic	92.4
216-Z-4	Low Organic - Low Salts - Near Neutral	0.0
216-Z-5	Low Organic - Low Salts - Near Neutral	0.0
216-Z-6	Low Organic - Low Salts - Near Neutral	0.0
216-Z-7	High Organic - Very Acidic	98.5
216-Z-8	Low Organic - Low Salts - Near Neutral	0.0
216-Z-9	High Organic - Very Acidic	7.6
218-EC-9(a)**	Low Organic - Low Salts - Near Neutral	0.0
218-EC-9(b)***	Low Organic - Low Salts - Near Neutral	0.0
218-E-1(b)	Low Organic - Low Salts - Near Neutral	0.0
218-E-10(b)	Low Organic - Low Salts - Near Neutral	0.0
218-E-10(a)	Low Organic - Low Salts - Near Neutral	0.0
218-E-12A(b)	Low Organic - Low Salts - Near Neutral	0.0
218-E-12B(b)	Low Organic - Low Salts - Near Neutral	0.0
218-E-12B(a)	Low Organic - Low Salts - Near Neutral	0.0
218-E-2(b)	Low Organic - Low Salts - Near Neutral	0.0
218-E-4(b)	Low Organic - Low Salts - Near Neutral	0.0
218-E-5(b)	Low Organic - Low Salts - Near Neutral	0.0
218-E-5A(b)	Low Organic - Low Salts - Near Neutral	0.0
218-E-8(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-1(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-11(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-1A(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-2(b)	Low Organic - Low Salts - Near Neutral	0.0

Table 4.4. (contd)

Site Name	Waste Type Name	K_d Switch Depth (m)*
218-W-2A(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-3(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-3A(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-3A(a)	Low Organic - Low Salts - Near Neutral	0.0
218-W-3AE(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-3AE(a)	Low Organic - Low Salts - Near Neutral	0.0
218-W-4A(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-4B-c(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-4B-c(a)	Low Organic - Low Salts - Near Neutral	0.0
218-W-4C(a)	Low Organic - Low Salts - Near Neutral	0.0
218-W-4C(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-5(b)	Low Organic - Low Salts - Near Neutral	0.0
218-W-5(a)	Low Organic - Low Salts - Near Neutral	0.0
218-W-7	Very High Salts - Very Basic	0.0
218-W-8	Low Organic - Low Salts - Near Neutral	0.0
218-W-9(b)	Low Organic - Low Salts - Near Neutral	0.0
TWRS glass grout vault	Low Organic - Low Salts - Near Neutral	0.0
TWRS glass new site	Low Organic - Low Salts - Near Neutral	0.0
US Ecology current	Low Organic - Low Salts - Near Neutral	0.0
US Ecology future	Low Organic - Low Salts - Near Neutral	0.0
ERDF	Low Organic - Low Salts - Near Neutral	0.0
C Reactor	Low Organic - Low Salts - Near Neutral	0.0
D Reactor	Low Organic - Low Salts - Near Neutral	0.0
DR Reactor	Low Organic - Low Salts - Near Neutral	0.0
F Reactor	Low Organic - Low Salts - Near Neutral	0.0
H Reactor	Low Organic - Low Salts - Near Neutral	0.0
KE Reactor	Low Organic - Low Salts - Near Neutral	0.0
KW Reactor	Low Organic - Low Salts - Near Neutral	0.0
N Reactor	Low Organic - Low Salts - Near Neutral	0.0
TK-A-S	Very High Salts - Very Basic	23.4
TK-A-L	Very High Salts - Very Basic	23.4
TK-A-R	Very High Salts - Very Basic	23.4
TK-AN-R-1	Chelates - High Salts	0.0
TK-AN-R-2	Very High Salts - Very Basic	23.0
TK-AP-R-1	Chelates - High Salts	0.0
TK-AP-R-2	Very High Salts - Very Basic	23.0
TK-AW-R	Very High Salts - Very Basic	23.0
TK-AX-S-1	Chelates - High Salts	0.0

Table 4.4. (contd)

Site Name	Waste Type Name	K_d Switch Depth (m)*
TK-AX-S-2	Very High Salts - Very Basic	23.4
TK-AX-L-1	Chelates - High Salts	0.0
TK-AX-L-2	Very High Salts - Very Basic	23.4
TK-AX-R-1	Chelates - High Salts	0.0
TK-AX-R-2	Very High Salts - Very Basic	23.4
TK-AY-R-1	Chelates - High Salts	23.0
TK-AY-R-2	Very High Salts - Very Basic	23.0
TK-AZ-R	Very High Salts - Very Basic	23.0
TK-B-S	Very High Salts - Very Basic	29.7
TK-B-L	Very High Salts - Very Basic	29.7
TK-B-R	Very High Salts - Very Basic	29.7
TK-BX-S	Very High Salts - Very Basic	27.5
TK-BX-L	Very High Salts - Very Basic	27.5
TK-BX-R	Very High Salts - Very Basic	27.5
TK-BY-S	Very High Salts - Very Basic	25.7
TK-BY-L	Very High Salts - Very Basic	25.7
TK-BY-R	Very High Salts - Very Basic	25.7
TK-C-S-1	Chelates - High Salts	29.7
TK-C-S-2	Very High Salts - Very Basic	29.7
TK-C-L-1	Chelates - High Salts	29.7
TK-C-L-2	Very High Salts - Very Basic	29.7
TK-C-R-1	Chelates - High Salts	0.0
TK-C-R-2	Very High Salts - Very Basic	29.7
TK-S-S	Very High Salts - Very Basic	25.7
TK-S-L	Very High Salts - Very Basic	25.7
TK-S-R	Very High Salts - Very Basic	25.7
TK-SX-S-1	Chelates - High Salts	0.0
TK-SX-S-2	Very High Salts - Very Basic	23.4
TK-SX-L-2	Very High Salts - Very Basic	23.4
TK-SX-R-1	Chelates - High Salts	0.0
TK-SX-R-2	Very High Salts - Very Basic	23.4
TK-SY-R-1	Chelates - High Salts	0.0
TK-SY-R-2	Very High Salts - Very Basic	23.0
TK-T-S	Very High Salts - Very Basic	29.7
TK-T-L	Very High Salts - Very Basic	29.7
TK-T-R	Very High Salts - Very Basic	29.7
TK-TX-S	Very High Salts - Very Basic	25.7
TK-TX-L	Very High Salts - Very Basic	25.7

Table 4.4. (contd)

Site Name	Waste Type Name	K_d Switch Depth (m)*
TK-TX-R	Very High Salts - Very Basic	25.7
TK-TY-S	Very High Salts - Very Basic	25.7
TK-TY-L	Very High Salts - Very Basic	25.7
TK-TY-R	Very High Salts - Very Basic	25.7
TK-U-S	Very High Salts - Very Basic	29.7
TK-U-L	Very High Salts - Very Basic	29.7
TK-U-R	Very High Salts - Very Basic	29.7

* Refers to depth below ground surface at which the site's K_d is assumed to switch from near-field to far-field values.

** (a) refers to waste disposed after September 30, 1988.

*** (b) refers to waste disposed before September 30, 1988.

Table 4.5. Recharge Rates Applied to Waste Sites

Site Name	Time 1*	Time 2	Time 3	Time 4	Recharge 1**	Recharge 2	Recharge 3	Recharge 4
207-U	1952	1994	2017	2050	1	4	6	7
216-A-1	1955	1955.1	2018	2050	1	4	6	7
216-A-10	1956	1987.2	2015	2050	1	2	6	7
216-A-18	1955	1956.2	2018	2050	1	4	6	7
216-A-19	1955	1956.2	2018	2050	1	4	6	7
216-A-2	1956	1963	2015	2050	1	4	6	7
216-A-20	1955	1955.2	2018	2050	1	4	6	7
216-A-21	1957	1964.7	2015	2050	1	4	6	7
216-A-24	1958	1965.7	2018	2050	1	4	6	7
216-A-25	1957	1986.1	2024	2050	1	4	6	7
216-A-27	1965	1970.1	2015	2050	1	4	6	7
216-A-28	1958	1966.9	2015	2050	2	4	6	7
216-A-3	1956	1981.3	2015	2050	1	4	6	7
216-A-30	1961	1995	2015	2050	1	4	6	7
216-A-31	1964	1966.3	2015	2050	3	4	6	7
216-A-36A/B	1965	1987	2015	2050	1	4	6	7
216-A-37-1	1977	1994.8	2015	2050	1	4	6	7
216-A-37-2	1983	1995	2015	2050	1	4	6	7
216-A-4	1955	1958	2015	2050	1	4	6	7
216-A-45	1987	1990.8	2015	2050	1	4	6	7
216-A-5	1955	1965.8	2015	2050	1	4	6	7
216-A-6	1955	1969.2	2015	2050	1	4	6	7
216-A-7	1955	1966	2018	2050	1	4	6	7
216-A-8	1955	1990.2	2018	2050	1	4	6	7
216-A-9	1956	1969.4	2015	2050	1	4	6	7
216-B-10A	1949	1951.1	2025	2050	1	4	6	7
216-B-10B	1969	1973.3	2020	2050	2	4	6	7
216-B-11A&B	1952	1955	2021	2050	1	4	6	7
216-B-12	1952	1973	2019	2050	1	4	6	7
216-B-14	1956	1956.1	2019	2050	1	4	6	7
216-B-15	1956	1957.7	2019	2050	1	4	6	7
216-B-16	1956	1956.3	2019	2050	1	4	6	7
216-B-17	1956	1956.1	2019	2050	1	4	6	7
216-B-18	1956	1956.1	2019	2050	1	4	6	7
216-B-19	1957	1957.7	2019	2050	1	4	6	7
216-B-20	1956	1956.1	2018	2050	1	4	6	7
216-B-21	1956	1956.1	2019	2050	1	4	6	7
216-B-2-1	1945	1963.6	2018	2050	1	4	6	7

Table 4.5. (contd)

Site Name	Time 1*	Time 2	Time 3	Time 4	Recharge 1**	Recharge 2	Recharge 3	Recharge 4
216-B-22	1956	1997.1	2019	2050	2	4	6	7
216-B-2-2	1963	1969.5	2018	2050	1	4	6	7
216-B-23	1956	1997.1	2019	2050	2	4	6	7
216-B-2-3	1970	1987	2018	2050	1	4	6	7
216-B-24	1956	1956.1	2019	2050	1	4	6	7
216-B-25	1956	1956.1	2019	2050	1	4	6	7
216-B-26	1956	1956.2	2019	2050	1	4	6	7
216-B-27	1957	1957.2	2019	2050	1	4	6	7
216-B-28	1957	1957.2	2019	2050	1	4	6	7
216-B-29	1957	1957.1	2019	2050	1	4	6	7
216-B-3	1945	1997.5	2018	2050	1	4	6	7
216-B-30	1957	1957.1	2019	2050	1	4	6	7
216-B-31	1957	1957.1	2019	2050	1	4	6	7
216-B-32	1957	1957.1	2019	2050	1	4	6	7
216-B-33	1957	1957.1	2019	2050	1	4	6	7
216-B-34	1957	1957.1	2019	2050	1	4	6	7
216-B-35	1954	1954.1	2020	2050	1	4	6	7
216-B-36	1954	1954.1	2020	2050	1	4	6	7
216-B-37	1954	1954.1	2020	2050	1	4	6	7
216-B-38	1954	1954.2	2020	2050	1	4	6	7
216-B-39	1953	1954.9	2020	2050	1	4	6	7
216-B-40	1954	1954.2	2020	2050	1	4	6	7
216-B-41	1954	1954.1	2020	2050	1	4	6	7
216-B-43	1954	1954.1	2013	2050	1	4	6	7
216-B-44	1954	1954.3	2013	2050	1	4	6	7
216-B-45	1955	1955.2	2013	2050	1	4	6	7
216-B-46	1955	1955.2	2013	2050	1	4	6	7
216-B-47	1955	1955.1	2013	2050	1	4	6	7
216-B-48	1955	1955.1	2013	2050	1	4	6	7
216-B-49	1955	1955.1	2013	2050	1	4	6	7
216-B-5	1945	1947.5	2025	2050	1	4	6	7
216-B-50	1965	1974	2013	2050	1	4	6	7
216-B-52	1957	1957.1	2019	2050	1	4	6	7
216-B-53A	1965	1965.1	2019	2050	1	4	6	7
216-B-53B	1962	1962.3	2019	2050	1	4	6	7
216-B-54	1963	1965.6	2019	2050	2	4	6	7
216-B-55	1967	1994.4	2021	2050	1	4	6	7
216-B-57	1968	1973.3	2013	2050	1	4	6	7

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Table 4.5. (contd)

Site Name	Time 1*	Time 2	Time 3	Time 4	Recharge 1**	Recharge 2	Recharge 3	Recharge 4
216-B-58	1965	1966.6	2019	2050	1	4	6	7
216-B-59	1967	1967.1	2025	2050	1	4	6	7
216-B-60	1967	1967.1	2025	2050	1	4	6	7
216-B-62	1973	1990.8	2021	2050	1	4	6	7
216-B-63	1970.3	1993	2017	2050	1	4	6	7
216-B-7A&B	1946	1976.6	2020	2050	1	4	6	7
216-B-8	1948	1953.3	2020	2050	1	4	6	7
216-B-9	1948	1950.9	2025	2050	1	4	6	7
216-C-1	1953	1957.4	2017	2050	1	4	6	7
216-C-10	1964	1968.9	2017	2050	1	4	6	7
216-C-3	1953	1954.2	2017	2050	1	4	6	7
216-C-4	1955	1964.8	2017	2050	2	4	6	7
216-C-5	1955	1955.3	2017	2050	1	4	6	7
216-C-6	1955	1964	2017	2050	1	4	6	7
216-C-7	1961	1982.7	2017	2050	3	4	6	7
216-C-9	1953	1983.5	2017	2050	1	4	6	7
216-N-2	1947	1947.1	2021	2050	1	4	6	7
216-N-3	1952	1952.1	2021	2050	1	4	6	7
216-N-4	1944	1951.8	2021	2050	1	4	6	7
216-N-5	1952	1952.1	2021	2050	1	4	6	7
216-N-6	1944	1951.8	2021	2050	1	4	6	7
216-N-7	1952	1952.1	2021	2050	1	4	6	7
216-S-1&2	1952	1956	2026	2050	1	4	6	7
216-S-10D	1951	1990.2	2019	2050	1	4	6	7
216-S-11	1954	1965.3	2019	2050	1	4	6	7
216-S-12	1954	1954.1	2025	2050	1	4	6	7
216-S-13	1952	1972.5	2026	2050	1	4	6	7
216-S-16P	1957	1975.1	2019	2050	1	4	6	7
216-S-17	1951	1953.5	2019	2050	1	4	6	7
216-S-19	1952	1984.7	2019	2050	1	4	6	7
216-S-20	1952	1973.3	2025	2050	1	4	6	7
216-S-22	1957	1966.3	2024	2050	2	4	6	7
216-S-23	1969	1972.5	2025	2050	1	4	6	7
216-S-25	1973	1994.2	2019	2050	1	4	6	7
216-S-26	1984	1994.3	2025	2050	1	4	6	7
216-S-3	1953	1955.9	2026	2050	1	4	6	7
216-S-5	1954	1957	2019	2050	1	4	6	7
216-S-6	1954	1971.7	2019	2050	1	4	6	7

Table 4.5. (contd)

Site Name	Time 1*	Time 2	Time 3	Time 4	Recharge 1**	Recharge 2	Recharge 3	Recharge 4
216-S-7	1956	1965.5	2026	2050	1	4	6	7
216-S-8	1951	1951.3	2026	2050	1	4	6	7
216-S-9	1965	1968.5	2026	2050	1	4	6	7
216-T-1	1944	1995.1	2021	2050	1	4	6	7
216-T-12	1954	1954.1	2025	2050	1	4	6	7
216-T-14	1954	1954.1	2025	2050	1	4	6	7
216-T-15	1954	1954.1	2025	2050	1	4	6	7
216-T-16	1954	1954.1	2025	2050	1	4	6	7
216-T-17	1954	1954.3	2025	2050	1	4	6	7
216-T-18	1953	1953.1	2024	2050	1	4	6	7
216-T-19	1951	1979.9	2024	2050	1	4	6	7
216-T-20	1952	1952.1	2024	2050	1	4	6	7
216-T-21	1954	1954.2	2024	2050	1	4	6	7
216-T-22	1954	1954.1	2024	2050	1	4	6	7
216-T-23	1954	1954.1	2024	2050	1	4	6	7
216-T-24	1954	1954.1	2024	2050	1	4	6	7
216-T-25	1954	1954.1	2024	2050	1	4	6	7
216-T-26	1955	1956.3	2024	2050	1	4	6	7
216-T-27	1965	1965.2	2024	2050	1	4	6	7
216-T-28	1960	1966	2024	2050	1	4	6	7
216-T-3	1945.5	1946.8	2021	2050	1	4	6	7
216-T-32	1946	1951.5	2024	2050	1	4	6	7
216-T-33	1963	1963.1	2021	2050	1	4	6	7
216-T-34	1966	1966.8	2021	2050	1	4	6	7
216-T-35	1967	1967.8	2021	2050	1	4	6	7
216-T-36	1967	1968.8	2024	2050	1	4	6	7
216-T-4B	1972	1995.6	2025	2050	1	4	6	7
216-T-5	1955	1955.1	2024	2050	1	4	6	7
216-T-6	1946	1950.8	2025	2050	1	4	6	7
216-T-7	1948	1955.6	2024	2050	1	4	6	7
216-T-8	1950	1951.3	2021	2050	1	4	6	7
216-U-1&2	1951	1966.6	2017	2050	1	4	6	7
216-U-10	1944	1986.5	2017	2050	1	4	6	7
216-U-12	1960	1987.9	2017	2050	1	4	6	7
216-U-15	1957	1957.1	2017	2050	1	4	6	7
216-U-16	1984	1986.5	2017	2050	1	4	6	7
216-U-17	1988	1994.6	2017	2050	2	4	6	7
216-U-3	1954	1955.3	2017	2050	1	4	6	7

Table 4.5. (contd)

Site Name	Time 1*	Time 2	Time 3	Time 4	Recharge 1**	Recharge 2	Recharge 3	Recharge 4
216-U-4A	1955	1970	2017	2050	1	4	6	7
216-U-4B	1960	1968.7	2017	2050	1	4	6	7
216-U-5	1952	1952.1	2017	2050	1	4	6	7
216-U-6	1952	1952.1	2017	2050	1	4	6	7
216-U-7	1952	1957.3	2017	2050	2	4	6	7
216-U-8	1952	1959.8	2017	2050	1	4	6	7
216-Z-1&2	1949	1968.8	2019	2050	1	4	6	7
216-Z-10	1945	1945.3	2019	2050	1	4	6	7
216-Z-12	1959	1973.2	2019	2050	1	4	6	7
216-Z-16	1968	1976.8	2019	2050	1	4	6	7
216-Z-17	1967	1968	2019	2050	1	4	6	7
216-Z-18	1969	1973.1	2019	2050	2	4	6	7
216-Z-1A	1949	1969	2019	2050	3	4	6	7
216-Z-20	1981	1994.3	2017	2050	1	4	6	7
216-Z-3	1952	1958.8	2019	2050	1	4	6	7
216-Z-4	1945	1945.1	2019	2050	1	4	6	7
216-Z-5	1945	1946.7	2019	2050	1	4	6	7
216-Z-6	1945	1945.1	2019	2050	1	4	6	7
216-Z-7	1947	1967	2019	2050	1	4	6	7
216-Z-8	1955	1961.8	2019	2050	2	4	6	7
216-Z-9	1955	1961.9	2019	2050	1	4	6	7
218-EC-9(a)+	1989	2018			4	6		
218-EC-9(b)++	1987	2018			4	6		
218-E-1(b)	1949	2018			4	6		
218-E-10(b)	1974	2018			4	6		
218-E-10(a)	2002	2018			4	6		
218-E-12A(b)	1960	2018			4	6		
218-E-12B(b)	1978	2018			4	6		
218-E-12B(a)	2002	2018			4	6		
218-E-2(b)	1949	2018			4	6		
218-E-4(b)	1956	2018			4	6		
218-E-5(b)	1955	2018			4	6		
218-E-5A(b)	1958	2018			4	6		
218-E-8(b)	1959	2018			4	6		
218-W-1(b)	1948	2018			4	6		
218-W-11(b)	1960	2018			4	6		
218-W-1A(b)	1949	2018			4	6		
218-W-2(b)	1955	2018			4	6		

Table 4.5. (contd)

Site Name	Time 1*	Time 2	Time 3	Time 4	Recharge 1**	Recharge 2	Recharge 3	Recharge 4
218-W-2A(b)	1970	2018			4	6		
218-W-3(b)	1959	2018			4	6		
218-W-3A(b)	1979	2018			4	6		
218-W-3A(a)	2003	2018			4	6		
218-W-3AE(b)	1986	2018			4	6		
218-W-3AE(a)	2000	2018			4	6		
218-W-4A(b)	1965	2018			4	6		
218-W-4B-c(b)	1978	2018			4	6		
218-W-4B-c(a)	1999	2018			4	6		
218-W-4C(a)	1998	2018			4	6		
218-W-4C(b)	1983	2018			4	6		
218-W-5(b)	1987	2018			4	6		
218-W-5(a)	2002	2018			4	6		
218-W-7	1956				6			
218-W-8	1948				6			
218-W-9(b)	1954	2018			4	6		
TWRS glass grout vault	2007				7			
TWRS glass new site	2018				7			
US Ecology current	1979	2000			4	9		
US Ecology future	2025.5				9			
ERDF	2021				7			
C Reactor	2050				7			
D Reactor	2050				7			
DR Reactor	2050				7			
F Reactor	2050				7			
H Reactor	2050				7			
KE Reactor	2050				7			
KW Reactor	2050				7			
N Reactor	2050				7			
TK-A-S	2004	2016	2025		1	4	7	
TK-A-L	1963	1987	2025		1	4	7	
TK-A-R	2525				7			
TK-AN-R-1	2525				7			
TK-AN-R-2	2525				7			

Table 4.5. (contd)

Site Name	Time 1*	Time 2	Time 3	Time 4	Recharge 1**	Recharge 2	Recharge 3	Recharge 4
TK-AP-R-1	2525				7			
TK-AP-R-2	2525				7			
TK-AW-R	2525				7			
TK-AX-S-1	2003	2008	2025		1	4	7	
TK-AX-S-2	2005	2008	2025		1	4	7	
TK-AX-L-1	1988	1989	2025		1	4	7	
TK-AX-L-2	1977	1978	2025		1	4	7	
TK-AX-R-1	2525				7			
TK-AX-R-2	2525				7			
TK-AY-R-1	2525				7			
TK-AY-R-2	2525				7			
TK-AZ-R	2525				7			
TK-B-S	2012	2018	2025		1	4	7	
TK-B-L	1974	1984	2025		1	4	7	
TK-B-R	2525				7			
TK-BX-S	2012	2018	2025		1	4	7	
TK-BX-L	1971	1984	2025		1	4	7	
TK-BX-R	2525				7			
TK-BY-S	2014	2019	2025		1	4	7	
TK-BY-L	1972	1984	2025		1	4	7	
TK-BY-R	2525				7			
TK-C-S-1	2007	2016	2025		1	4	7	
TK-C-S-2	1998	2009	2025		1	4	7	
TK-C-L-1	1984	1985	2025		1	4	7	
TK-C-L-2	1968	1988	2025		1	4	7	
TK-C-R-1	2525				7			
TK-C-R-2	2525				7			
TK-S-S	2012	2018	2025		1	4	7	
TK-S-L	1968	1969	2025		1	4	7	
TK-S-R	2525				7			
TK-SX-S-1	2018	2019	2025		1	4	7	
TK-SX-S-2	2004	2020	2025		1	4	7	
TK-SX-L-2	1962	1988	2025		1	4	7	
TK-SX-R-1	2525				7			
TK-SX-R-2	2525				7			
TK-SY-R-1	2525				7			
TK-SY-R-2	2525				7			
TK-T-S	2017	2019	2025		1	4	7	

Table 4.5. (contd)

Site Name	Time 1*	Time 2	Time 3	Time 4	Recharge 1**	Recharge 2	Recharge 3	Recharge 4
TK-T-L	1973	1992	2025		1	4	7	
TK-T-R	2525				7			
TK-TX-S	2012	2019	2025		1	4	7	
TK-TX-L	1974	1984	2025		1	4	7	
TK-TX-R	2525				7			
TK-TY-S	2016	2018	2025		1	4	7	
TK-TY-L	1959	1981	2025		1	4	7	
TK-TY-R	2525				7			
TK-U-S	2007	2019	2025		1	4	7	
TK-U-L	1959	1980	2025		1	4	7	
TK-U-R	2525				7			

* "Time" refers to the year that the application of the corresponding recharge begins. Therefore, recharge 1 is assume to begin at time 1 and end at time 2.

** "Recharge" refers to the index of recharge rate (cm/yr) applied over the time interval specified in the "time" field. An index volume of 1 indicates the recharge rate will be calculated as discussed in Section 4.1.2.1. Index values of 2, 3, 4, 5, 6, 7, 8, and 9 specify recharge rates of 50, 20, 7.5, 5, 0.5, 0.05, 0.01, and 0.172 cm/yr, respectively.

\+ (a) refers to waste disposed after September 30, 1988.

++ (b) refers to waste disposed before September 30, 1988.

Table 4.6. Geologic Well Logs for the Vadose Zone Model

Column	Surface Elevation (m)	Northing (m)*	Easting (m)**	Soil 1+	Thickness (m)	Soil 2	Thickness (m)	Soil 3	Thickness (m)	Soil 4++	Thickness (m)
218-W-5	737.7	137,024	565658	WHS	19	WEP	4	WPP	7	WR	85
218-E-12B	629.5	137,238	574643	EHG	10	EHS	6	LEHG	54	ER	0.01
218-E-10	625.7	137,468	572924	EHG	10	EHS	6	LEHG	59	ER	0.01
299-E13-20	742.9	134,313	573610	EHG	10	EHS	6	LEHG	80	ER	60
299-E19-1	735.4	135,086	572820	EHG	10	EHS	6	LEHG	91	ER	51
299-E24-7	716.0	135,561	574407	EHG	10	EHS	6	LEHG	60	ER	56
299-E25-2	675.5	136,062	575514	EHG	10	EHS	6	LEHG	60	ER	36
299-E26-8	619.4	136,687	575522	EHG	10	EHS	6	LEHG	44	ER	14
299-E28-16	703.1	136,562	573135	EHG	10	EHS	6	LEHG	71	ER	12
299-E28-22	700.3	136,321	574041	EHG	10	EHS	6	LEHG	83	ER	17
299-W6-1	702.5	137,510	567214	WHS	14	WPP	4	WR	121		
299-W11-2	714.5	136,671	567407	WHS	34	WEP	4	WPP	7	WR	110
299-W14-7	677.7	135,655	567034	WHS	38	WPP	2	WR	118		
299-W14-8A	725.2	135,688	568013	WHS	47	WEP	5	WPP	5	WR	106
299-W15-15	698.0	135,752	566089	WHS	42	WEP	3	WPP	8	WR	100
299-W18-21	668.6	134,979	566098	WHS	36	WEP	5	WPP	3	WR	100
299-W21-1	699.3	134,397	568141	WHS	53	WEP	8	WPP	8	WR	100
299-W22-24	692.3	134,411	567648	WHS	42	WEP	13	WPP	12	WR	104

* Refers to north coordinate in Washington State Plane NAD83 coordinate system.

** Refers to east coordinate in Washington State Plane NAD83 coordinate system.

\+ "Soil 1" refers to the upper soil layer.

++ "Soil 4 " refers to the lowest soil layer simulated.

Table 4.7. Sediment Types and Unsaturated Flow Model Parameters Used in the Composite Analysis

Soil Name	Code	van Genuchten alpha (-)	van Genuchten n (1/cm)	Residual Water Content (cm^3/cm^3)	Saturated Water Content (cm^3/cm^3)	Saturated Hydraulic Conductivity (cm/s)	Bulk Density (g/cm^3)	Gravel %*
East Hanford Gravel	EHG	8.11E-03	1.58	0.0146	0.119	1.76E-03	1.97	41.70%
Lower East Hanford Gravel	LEHG	8.11E-03	1.58	0.0146	0.119	1.76E-03	1.97	41.70%
East Hanford Sand	EHS	1.30E-01	2.10	0.0257	0.337	1.19E-02	1.78	17.30%
East Ringold	ER	8.19E-03	1.53	0.0262	0.124	3.97E-04	2.04	43.30%
West Hanford Sand	WHS	1.44E-02	2.20	0.0519	0.382	3.98E-04	1.64	3.60%
Early Palouse	WEP	6.27E-03	2.53	0.0300	0.379	9.69E-05	1.68	2.00%
Plio-Pleistocene	WPP	1.55E-02	1.78	0.0616	0.337	5.79E-02	1.65	8.40%
West Ringold	WR	3.14E-02	1.65	0.0236	0.226	5.76E-02	2.04	43.30%

* Only fine particles were assumed to contribute to sorption of radionuclides. The impact of larger particles was corrected using Gravel %.

Data are from Khaleel and Freeman (1995). A normal distribution was assumed for the parameters "van Genuchten n," "Residual Water Content," and "Saturated Water Content," and the mean was calculated accordingly. A log-normal distribution was assumed for the parameters "van Genuchten alpha" and "Saturated Hydraulic Conductivity," and the mean was calculated accordingly. If the sample size was less than 10, the parameters "van Genuchten alpha" and "Saturated Hydraulic Conductivity" were determined using the geometric mean.

Table 4.8. Summary of Key Assumptions for the Vadose Zone Model

Assumption	Rationale	Impact
Mass released from a waste site was assumed to enter the aquifer directly beneath the site.	Data to characterize the multidimensional flow patterns beneath most sites are inadequate.	Sites with significant horizontal migration within the vadose zone may enter the aquifer at some other location than directly beneath the site.
The vadose zone was represented as a vertical soil column.	Data to characterize the multidimensional flow patterns beneath most sites are inadequate.	In order to ensure simulations with the one-dimensional model do not predict ponding, the infiltration rate was not allowed to exceed the infiltration capacity of the strata with the lowest infiltration rate. These specified infiltration rates were generally much less than for other layers. Lowering the infiltration rates, particularly in the upper layers, delays the predicted cumulative breakthrough to the water table. Additionally, the increased volume in the column simulated, provides additional volume subject to gradually draining which also delays the cumulative breakthrough. This is not a conservative assumption.
For cribs, trenches, and ditches, the simulated area of the discharge was assumed to equal three times the area required to pass the recharge through the strata with the lowest saturated hydraulic conductivity without.	Plumes spread significantly from these sources as they move downward through the vadose zone.	Increasing the simulated area delays the calculated cumulative breakthrough to the water table. Sensitivity of cumulative release to assumed area is discussed in Section 4.1.2.4.
For ponds, the simulated area of the discharge was assumed to equal the area required to infiltrate the recharge through the strata with the lowest saturated hydraulic conductivity without ponding.	The area of ponds was large enough to limit spreading to a relatively small area around the edges.	Increasing the simulated area delays the calculated cumulative breakthrough to the water table.
For tank leaks and tank sluicing losses, the area of the discharge was assumed to equal the area of the affected tank bottoms.	The simulated area should be related to the number of affected tanks.	Increasing the area delays the calculated cumulative breakthrough to the water table.

Table 4.8. (contd)

Assumption	Rationale	Impact
The initial soil moisture was estimated based on a steady recharge of 5 mm/yr.	5 mm/yr is estimated to be the recharge before natural vegetation was disturbed.	The senstivity of calculated cumulative breakthrough to the water table is discussed in Section 4.1.2.4. Any impacts of the initial water content are lost within a relatively short period of time.
The model was assumed to instantaneously response to changes in recharge rates.	Waste sites are generally shallow and should respond quickly to changes recharge relative to the 1000-year study period.	Changes in recharge at deeper sites will occur gradually over many years. Since decreased recharge results in decreased release from the waste form for each of the release models, when recharge rates decrease the model will underestimate the predicted release to the water table for the next few years.
Barriers were assumed to affect the entire soil profile under consideration.	Barriers are expected to be sufficiently extensive that the flow from a waste form beneath a barrier will not be influenced by the recharge rates occurring beyond the barrier.	If the barrier is small relative to the depth to the water table this assumption will not be valid. This assumption will delay the predicted discharge to the water table.
Adjacent sites were assumed to not interfere with each other.	Simulating the vadose zone transport in multiple dimensions for the entire 200 Plateau Area at the Hanford Site was not practical for the first iteration of the Composite Analysis.	Interference will generally increase the flux to the water table. This is not a conservative assumption.
The soils were represented with a total of seven main soil groups.	Inadequate data exist to characterize the soil properties beneath most sites beyond the seven main soil groups considered.	Several thin, very low permeability strata have been observed in the vadose zone beneath the 200 Area Plateau at the Hanford Site. These strata would tend to reduce the flux to the water table. Neglecting these very low permeability strata would tend to increase the predicted cumulative flux to the water table.

Table 4.8. (contd)

Assumption	Rationale	Impact
Liquid releases were assumed to occur uniformly over the period of operation.	Inadequate data exist to distribute the volume of the liquid releases and the associated inventories over time.	Many of the liquid releases had very transient behaviors. Assuming that the estimated volume of the specific site is released uniformly over the entire period of operation will generally increase the predicted cumulative flux to the water table, since a larger area would be required to handle the transient release.
The depth that K_ds change is time-invariant.	Inadequate data exist to describe the temporal variation in the depth that the K_d changes from near-field to far-field.	This assumption is conservative if the near-field K_d is less than the far-field K_d because it will underestimate the depth of the far-field early in the release. However, if the near-field K_d is greater than the far-field K_d, this assumption will underestimate the influence to the far-field K_d on early releases. Generally, K_ds increase from near-field to far-field.
Preferential flow paths were not considered in the first iteration of the Composite Analysis.	Inadequate data exist to characterize the soil properties beneath most sites beyond the seven main soil groups considered.	Preferential flow paths can significantly increase the predicted cumulative flux to the water table. This is not a conservative assumption.
A value of 0.4 m was used for dispersivity in the STOMP calculations.	STOMP was only used to estimate the travel times of unit releases from the waste form to the water table. The actual mass flux is estimated using the convolution approach discussed in Section 4.1.2.2.	Increasing the dispersivity value will result in earlier break-throughs to the water table. However, a higher dispersivity value will also result in the mass flux to be spread out over a longer time period.

The depths at which distribution coefficients change, were estimated from the maximum penetration depth of beta and gamma observed in or adjacent to facilities. These measurements mainly reflect cesium-137 and strontium-90. If measurements were available for a facility, then the measured penetration depth was used. If no measurements were available, then the depth was estimated from measurements at facilities that received the same types of waste. The assumption was made that cesium is essentially mobile to the transition depth and immobile after the transition depth is reached. However, total volume discharged was also examined, and for sites with relatively large discharge volumes, the transition depth was taken to be something less than the maximum depth of measured gamma and beta. The selection of distribution coefficients is discussed in detail in Appendix E.

Table 4.9. Summary of Key Assumptions for the Groundwater Flow Model

Assumption	Rationale	Impact
The unconfined aquifer system, overlying the basalts, can be adequately represented by nine hydrostratigraphic units.	Flow of water (and transport of radionuclides) is assumed to occur in three dimensions. Nine hydrostratigraphic units are considered adequate to represent flow in this unconfined aquifer system over a wide range of conditions. Nine units are supported by available hydrogeologic data and represent all major and areally extensive conductive and nonconductive geohydrologic units above the basalt.	Additional units would better represent local flow conditions and hydrogeology. However, data are not currently available to improve this interpretation on a sitewide basis and other uncertainties could nullify the effect of this improvement. Additionally, simulation times would be adversely affected.
Natural recharge is variable across the Hanford Site and is included as a surface condition in the flow (and transport) model.	Variability of recharge across the Hanford Site is based on the distribution of surface cover, ranging from natural shrub-steppe vegetation to gravel surfaces in some of the 200 Areas. The differences in recharge based on surface cover have been well documented for the Hanford Site (Fayer and Walters 1995).	The surface recharge affects the flow model calibration by adding water to the system. The result is a distribution of higher hydraulic conductivity than would occur without recharge. Recharge affects the transport model by diluting the contaminant plumes and driving the maximum plume concentrations below the surface nodes.
The Columbia River is treated as a constant head boundary using hydraulic heads for 1979 to represent the long-term average conditions.	Performing simulations with transient river stage boundary conditions would not be appropriate since the inland areas that are the focus of this analysis are not greatly affected by river stage variations because they damp out before they reach the 200 Areas. Additionally, how the future river stage might vary is not known, and it would be too costly computationally at the Hanford Site-wide scale of the Composite Analysis.	Including the highly variable river stage conditions in the Hanford Site-wide Composite Analysis model would not affect the long-term results.

Table 4.9. (contd)

Assumption	Rationale	Impact
Post-Hanford conditions do not include large-scale irrigation impacts.	The prospect of large-scale irrigation occurring on the Hanford Site is unlikely for the following reasons. • Public acceptance of food products grown on the Hanford Site, regardless of the actual risk associated with agricultural development is uncertain. • Sufficient water rights within the Columbia Basin for development of crops requiring large-scale irrigation on the Site are unavailable. If agriculture should develop on the Hanford Site, it is likely that the crops to be planted will use the efficient and focused irrigation methods (e.g. drip irrigation) that are used in fruit orchards or vineyards. • New technologies and advanced resource management practices will likely eliminate or significantly curtail over-irrigation of crops.	The impact of this assumption can be significant depending on the scenario that is used. Previous sitewide analyses such as the Hanford Defense Waste Environmental Impact Statement (DOE 1987) included significant agricultural irrigation scenarios, which can alter the overall flow system in the unconfined aquifer and control the direction and rate of groundwater flow and contaminant transport.

Table 4.10. Major Hydrogeologic Units Used in the Site-Wide Three-Dimensional Model

Unit Number	Hydrogeologic Unit	Lithologic Description
1	Hanford Formation	Fluvial gravels and coarse sands
2	Palouse Soils	Fine-grained sediments and eolian silts
3	Plio-Pleistocene Unit	Buried soil horizon containing caliche and basaltic gravels
4	Upper Ringold Formation	Fine-grained fluvial/lacustrine sediments
5	Middle Ringold (Unit E)	Semi-indurated coarse-grained fluvial sediments
6	Middle Ringold (Unit C)	Fine-grained sediments with some interbedded coarse-grained sediments
7	Middle Ringold (Unit B and D)	Coarse-grained sediments
8	Lower Mud Sequence (Lower Ringold and part of Basal Ringold)	Lower blue or green clay or mud sequence
9	Basal Ringold (Unit A)	Fluvial sand and gravel
10	Columbia River Basalt	Basalt

Table 4.11. Summary of Key Assumptions for the Groundwater Transport Model

Assumption	Rationale	Impact
K_ds were selected based on information documented in Appendix E.	K_ds were based on available geochemical data at the Hanford Site and by analogy to other waste forms. Best-estimate values were used in the Composite Analysis.	Some of the K_ds for specific radionuclides may be uncertain and result in different predictions than actually have occurred and will occur in the future.
A grid spacing of 375 m on a side was used for the transport simulations.	This grid spacing was sufficient to represent transport on the sitewide scale used for the Composite Analysis. This grid spacing was a compromise between resolution of predicted contaminant plumes and computational time.	The grid spacing is too coarse to adequately resolve predicted concentrations at distances less than 1 km from the contaminant sources. Away from the sources and beyond the exclusion and buffer zones, the grid spacing is adequate to represent the contaminant plumes.
The basic vertical resolution of the transport grid was 8 m. Each of the nine units was represented with as many 8-m layers as needed to represent its entire thickness. Nonconductive (e.g., mud units) were always represented by at least two transport layers while conductive units (e.g., sand-gravel units) were only represented with one transport layer if they were less than 8-m thick. Creation of excessively thick and thin transport layers to achieve total unit thickness was prevented by the layering algorithm.	The 8-m transport layers were selected based on simulations previously performed for the Effluent Treatment Facility (Cole et al. 1997).	Adding additional transport layers would improve representation of the vertical distribution of contaminants, but at the expense of computational efficiency.
The longitudinal dispersivity assumed for all contaminant transport simulations was 95 m. The transverse dispersivity was assumed to be 20 m (~20 % of the longitudinal dispersivity).	Dispersivity is not a directly measurable value and no sitewide scale estimates are available. The value selected was the smallest value that satisfies all three theoretical constraints on its value, which include grid Peclet numerical constraint, scale of uncharacterized heterogeneities constraint, and transport scale of interest constraint. A transverse dispersivity that is 1/5 of the longitudinal dispersivity is typical for transport simulations (Freeze and Cherry 1979).	Dispersivity parameters assumed for contaminant transport directly affect predicted concentrations. Lower dispersivities result in higher predicted concentrations near the source but later first arrival times; higher dispersivities result in lower predicted concentrations near the source but earlier first arrival time which can be important for radionuclides with short half-lives.

Table 4.11. (contd)

Assumption	Rationale	Impact
An effective porosity of 0.25 was assumed for calculation of the retardation factor in all contaminant transport simulations.	This value of effective porosity was based on measurements available for Hanford Site unconfined aquifer sediments. Tracer tests conducted at the Hanford Site have revealed a range of effective porosity from 0.1 to 0.25 cm^3/cm^3.	Use of the highest value of effective porosity to calculate retardation factor yields a low estimate of sorption in Hanford sediments, and is therefore biased toward a conservative (i.e., maximum) estimate of contaminant migration in groundwater.

Table 4.12. Summary of Key Assumptions for the Atmospheric Model

Assumption	Rationale	Impact
The graphite reactor cores source was the only significant contributor to dose via the atmospheric pathway.	Previous performance assessments and environmental impact statement analyses demonstrated only negligible impacts via the atmospheric pathway.	This assumption was not conservative
The entire fraction of the inventory predicted to have been released from the reactor was assumed to enter the atmosphere.	Inadequate data exist to estimate the fraction of the released inventory that will move downward through the vadose zone and the fraction that will enter the atmosphere.	Because no credit is taken for the fraction of the inventory migrating through the vadose zone, this is a conservative assumption.
Atmospheric emissions were assumed to occur uniformly over an area source of 100 m by 600 m.	The area assumed to release reflects the dimension of the likely source.	Negligible.

Table 4.13. Summary of Key Assumptions for the Exposure and Dose Model

Assumption	Rationale	Impact
The exclusive waste management area and buffer zone were assumed to remain under federal control until the lands are safe for release to the public.	Safe stewardship of land used by the DOE requires that DOE retain control of the land and groundwater inside the buffer zone until it is safe to release.	Radiological doses were not presented for the portion of the Hanford Site inside the buffer zone.
The Unit Dose Factor was used to calculate doses in the Composite Analysis.	Guidance for the completion of the Composite Analysis required the simulation of annual radiation dose.	Calculation and presentation of only the annual radiation dose is a deviation from the guidance in the HSRAM which calls for a lifetime risk assessment from both chemicals and radionuclides.
The exposure scenarios included in the Composite Analysis were recreational, industrial, residential, and agricultural.	These exposure scenarios cover the range of possible post-Hanford land uses, and formally published in the HSRAM report.	Some potential impacts may not be covered by the conditions specified in these scenario descriptions, e.g., recently defined Native American scenarios.
Radionuclide concentrations in transport media were assumed to be constant over exposure durations analyzed (e.g., annual radiation dose).	Impacts predicted for the Composite Analysis are for 1000 years. Groundwater transport is simulated using relatively short time steps, but not as short as 1 year. Therefore, the concentration applied in the exposure duration is constant for the 1-year period.	The impact is negligible. Release calculations were made on a 1-year time interval. Greater resolution of exposures would not be consistent with the prior simulation steps.
Radionuclides are assumed to reach equilibrium with soils in a time period not exceeding 50 years, and the maximum value was not varied with time in the Unit Dose Factor calculation.	Unit Dose Factors are based on constant deposition over the duration period. This simplification was also needed in order to precalculate the UDF values.	The agricultural scenario is well represented with only iodine-129, uranium-233, and uranium-235 assigned somewhat lower buildup in soils over 50 years than are predicted to occur over longer time frame. Exposures to native soils, e.g., in the recreational scenario, are underestimated when using the 50-year soil contamination buildup levels because their low leach rates cause a continuous buildup over 1000-year period.

Table 4.14. Industrial Scenario Exposure Pathways

Transport Medium	Exposure Pathway	Chemical	Radioactive
Soil (air deposition)	Ingestion	Yes	Yes
	External	No	Yes
	Dermal Contact	Yes	No
	Suspension -- Inhalation	Yes	Yes
Air	Inhalation	Yes	Yes
Groundwater	Ingestion	Yes	Yes
	Dermal Contact	Yes	No

Table 4.15. Recreational Scenario Exposure Pathways

Transport Medium	Exposure Pathway	Chemical	Radioactive
Soil (air deposition)	Ingestion	Yes	Yes
	External	No	Yes
	Dermal Contact	Yes	No
	Suspension - Inhalation	Yes	Yes
	Biota - game (deer)	Yes	Yes
Air	Inhalation	Yes	Yes
	Biota - game (deer)	Yes	Yes
Groundwater	Ingestion	Yes	Yes
	Biota - game (deer)	Yes	Yes
	Dermal Contact (bathing)	Yes	No

Table 4.16. Residential Scenario Exposure Pathways

Transport Medium	Exposure Pathway	Chemical	Radioactive
Soil (air deposition)	Ingestion	Yes	Yes
	External	No	Yes
	Dermal Contact	Yes	No
	Biota – Fruit	Yes	Yes
	Biota – Vegetables	Yes	Yes
	Suspension - Inhalation	Yes	Yes
Air	Inhalation	Yes	Yes
	Biota – Fruit	Yes	Yes
	Biota – Vegetables	Yes	Yes
Groundwater	Ingestion	Yes	Yes
	Dermal Contact (bathing)	Yes	No
	Biota – Fruit	Yes	Yes
	Biota – Vegetables	Yes	Yes

Table 4.17. Agricultural Scenario Exposure Pathways

Transport Medium	Exposure Pathway	Chemical	Radioactive
Soil (air deposition)	Ingestion	Yes	Yes
	External	No	Yes
	Dermal Contact	Yes	No
	Biota - Dairy	Yes	Yes
	Biota - Meat	Yes	Yes
	Biota - Game (deer)	Yes	Yes
	Biota - Fruit	Yes	Yes
	Biota - Vegetables	Yes	Yes
	Suspension - Inhalation	Yes	Yes
Air	Inhalation	Yes	Yes
	Biota - Dairy	Yes	Yes
	Biota - Meat	Yes	Yes
	Biota - Game (deer)	Yes	Yes
	Biota - Fruit	Yes	Yes
	Biota - Vegetables	Yes	Yes
Groundwater	Ingestion	Yes	Yes
	Dermal Contact (bathing)	Yes	No
	Biota - Dairy	Yes	Yes
	Biota - Meat	Yes	Yes
	Biota - Game (deer)	Yes	Yes
	Biota - Fruit	Yes	Yes
	Biota - Vegetables	Yes	Yes
	Inhalation indoor	Yes	Yes (Radon)

Table 4.18. Unit Dose Factors (UDFs) Used in the Composite Analysis

Radionuclide	Agricultural Scenario mrem/(pCi/L)	Residential Scenario mrem/(pCi/L)	Industrial Scenario mrem/(pCi/L)	Recreational Scenario mrem/(pCi/L)
H-3	5.69E-05	4.85E-05	1.57E-05	1.05E-06
C-14	4.09E-02	1.52E-02	5.22E-04	2.99E-05
Cl-36	1.08E-01	1.76E-02	7.58E-04	5.29E-05
Se-79	1.21E-02	6.77E-03	2.17E-03	1.28E-04
Sr-90	3.12E-01	2.53E-01	3.58E-02	2.01E-03
Tc-99	3.66E-03	1.36E-03	3.65E-04	2.10E-05
I-129	6.19E-01	2.27E-01	6.90E-02	3.95E-03
	mrem/(μg/L)	mrem/(μg/L)	mrem/(μg/L)	mrem/(μg/L)
U-total	1.86E-01	1.69E-01	5.27E-02	2.96E-03
Hazard Factor	Hazard Index/ (μ-g/L)	Hazard Index/ (μ-g/L)	Hazard Index/ (μ-g/L)	Hazard Index/ (μ-g/L)
U-Total	1.19E-02	1.08E-02	3.48E-03	1.89E-04

Table 4.19. Comparison of Unit Dose Factors Between the TWRS Low-Level Tank Waste Performance Interim Assessment and the Composite Analysis

Radionuclide	Ingestion Factor Composite Analysis (rem/pCi)	Ingestion Factor TWRS Low-Level Tank Waste IPA (rem/pCi)	Inhalation Factor Composite Analysis (rem/pCi)	Inhalation Factor TWRS Low-Level Tank Waste IPA (rem/pCi)
Carbon-14	2.09E-09	2.1E-09	2.09E-09	2.1E-09
Chlorine-36	3.03E-09	3.0E-03	2.19E-09	2.1E-09
Tritium	6.3E-11	6.3E-11	6.3E-11	6.3E-11
Iodine-129	2.67E-07	2.8E-07	1.74E-07	1.8E-07
Selenium-79	8.7E-09	8.3E-09	9.84E-09	8.9E-09
Strontium-90	1.42E-07	1.3E-07	1.3E-06	1.3E-06
Technetium-99	1.46E-09	1.3E-09	5.33E-09	7.5E-09
Uranium-234	2.83E-07	2.6E-07	1.32E-04	1.3E-04
Uranium-235	2.66E-07	2.5E-07	1.23E-04	1.2E-04
Uranium-238	2.55E-07	2.3E-07	1.18E-04	1.2E-04

IPA = Interim Performance Assessment
TWRS = Tank Waste Remediation System

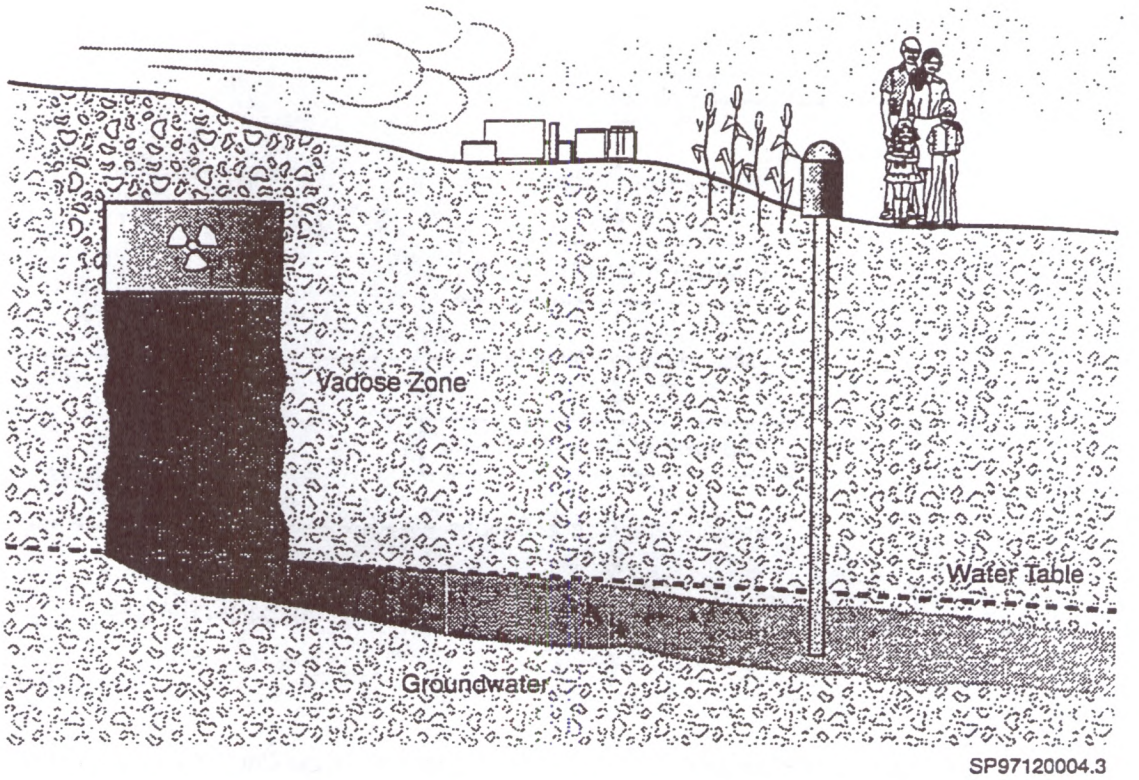


Figure 4.1. Transport and Exposure Pathways Considered in the Composite Analysis

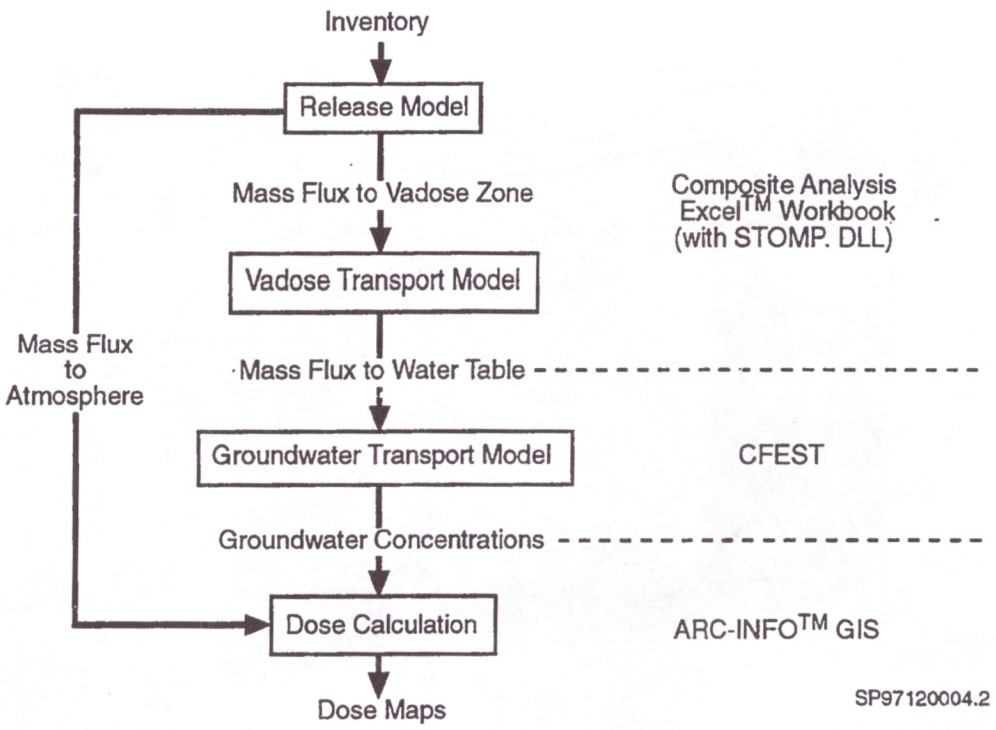


Figure 4.2. Relationship Among Software Elements in the Composite Analsysis

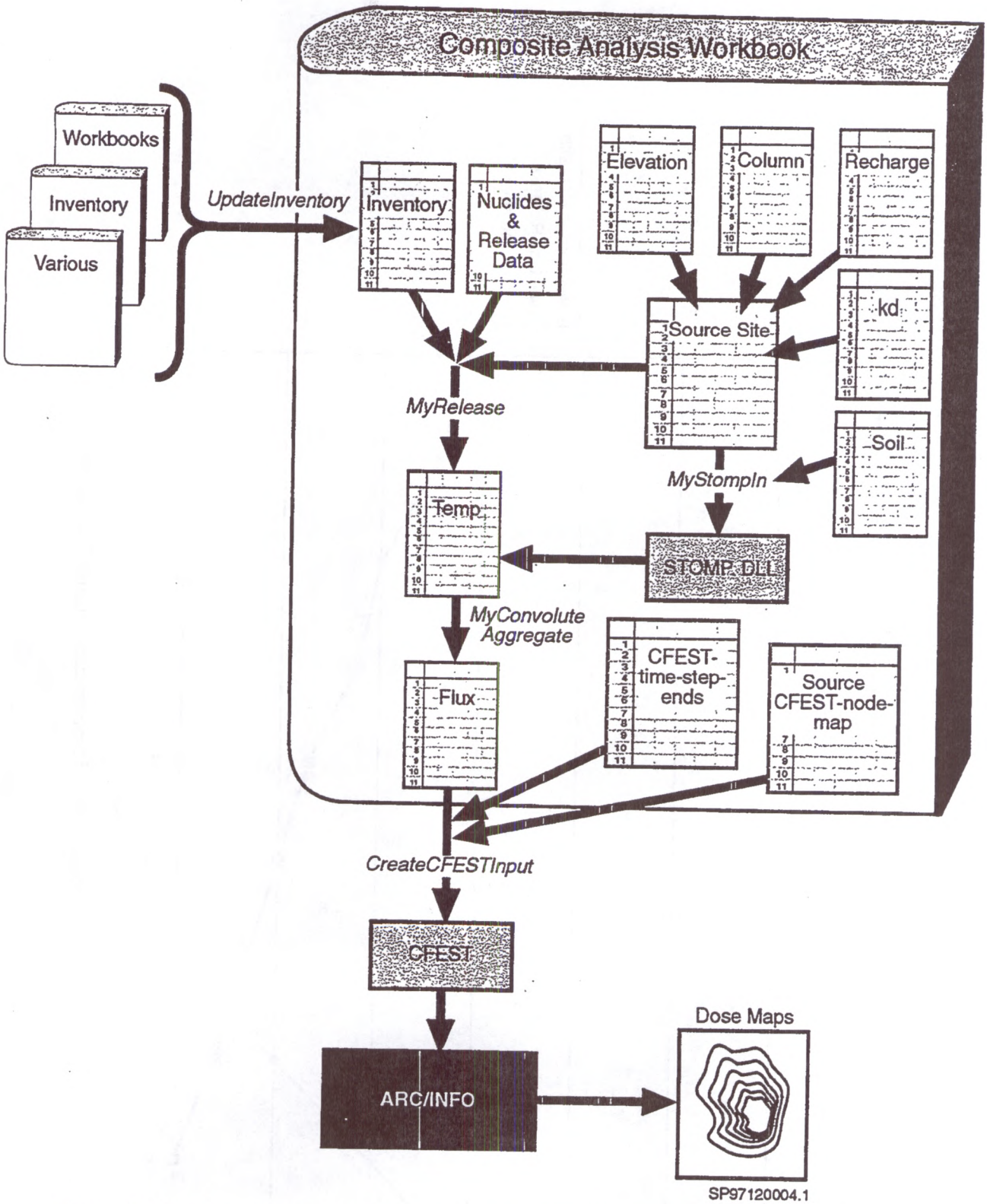


Figure 4.3. Implementation of the Release and Vadose Zone Transport Models in the Excel™ Spreadsheet

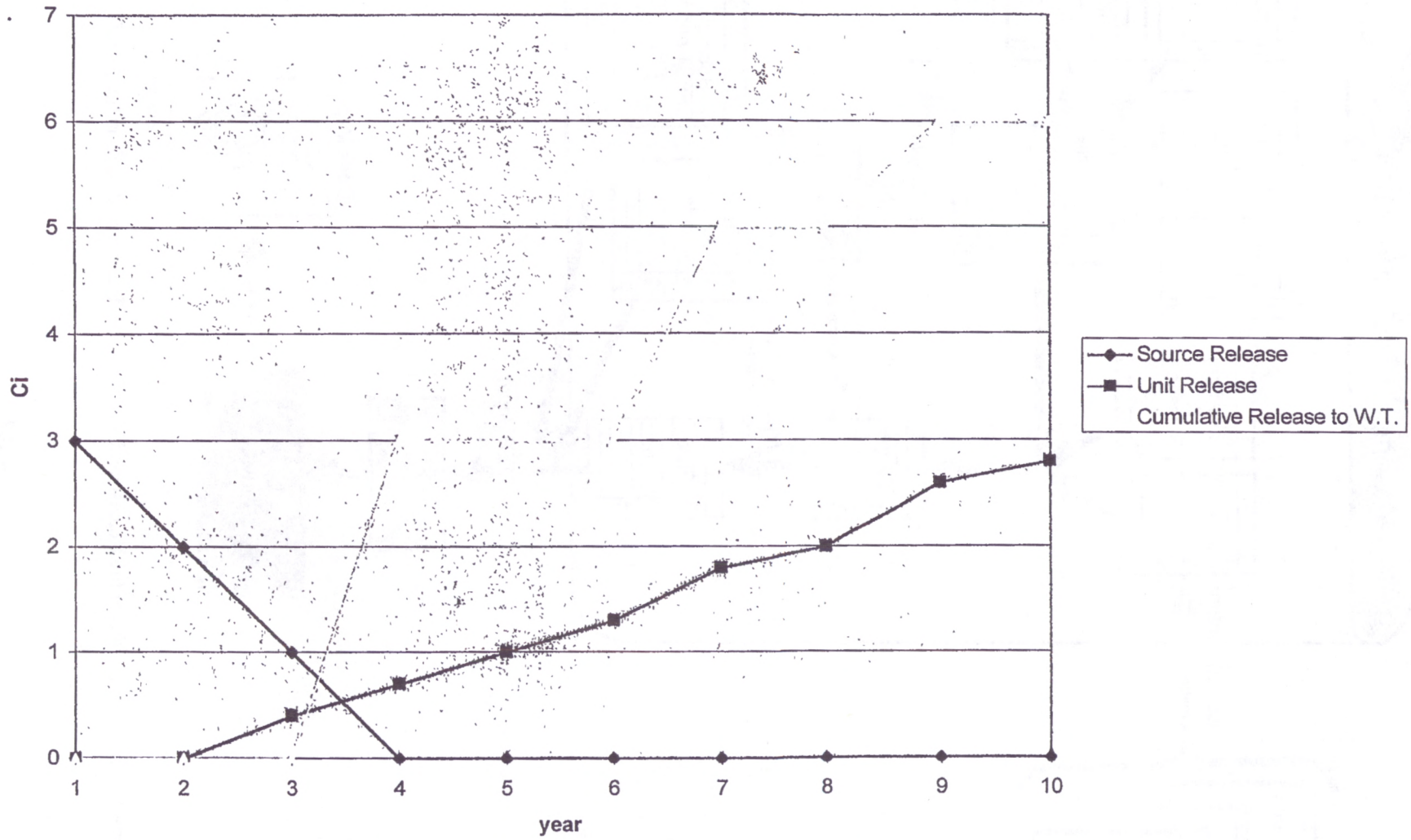


Figure 4.4. Convolution Method for Estimating Cumulative Flux to the Water Table

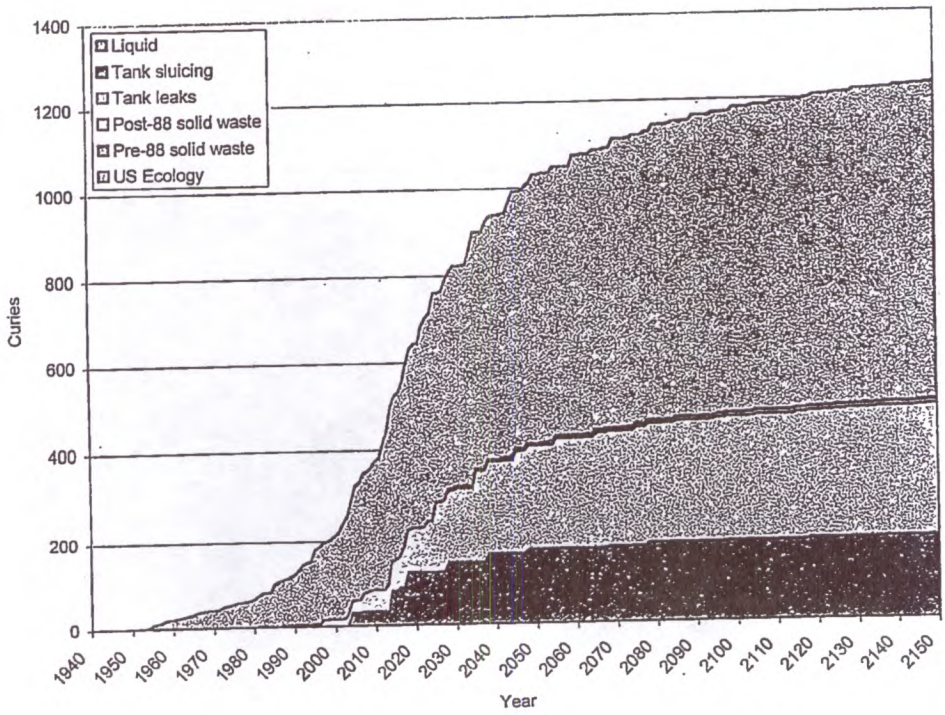


Figure 4.5a. Cumulative Release of Technetium-99 from All Sources to the Water Table from 1940 to 2150

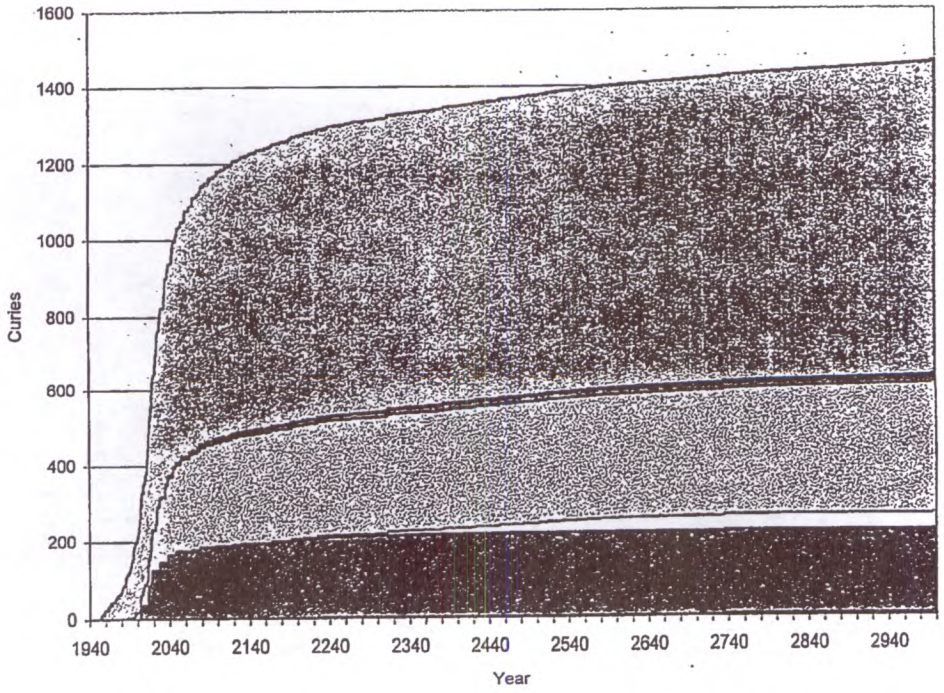


Figure 4.5b. Cumulative Release of Technetium-99 from All Sources to the Water Table from 1940 to 3000

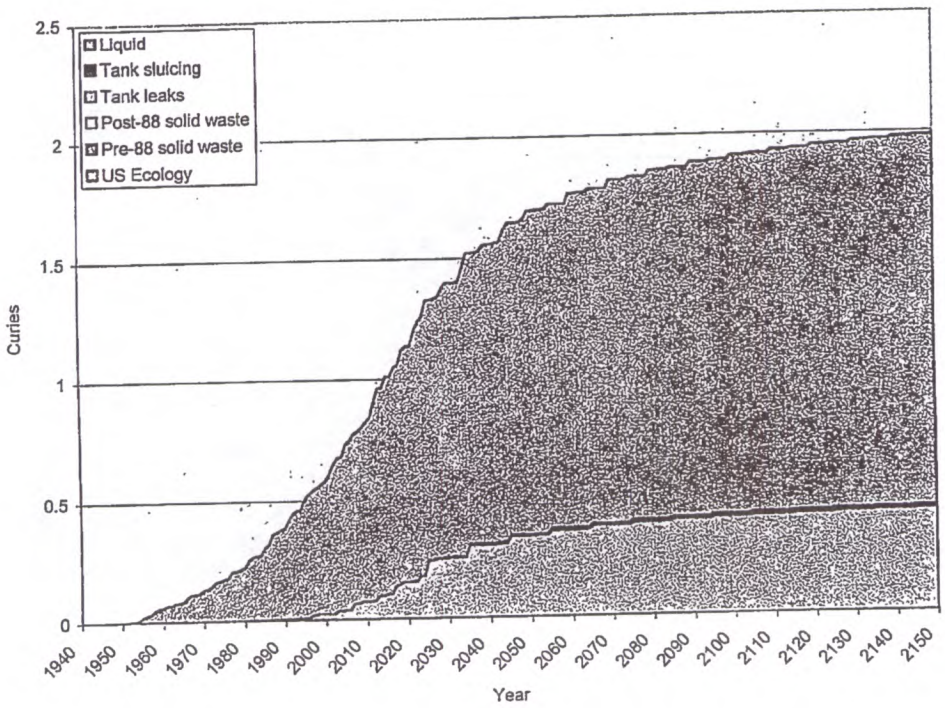


Figure 4.6a. Cumulative Release of Iodine-129 from All Sources to the Water Table from 1940 to 2150

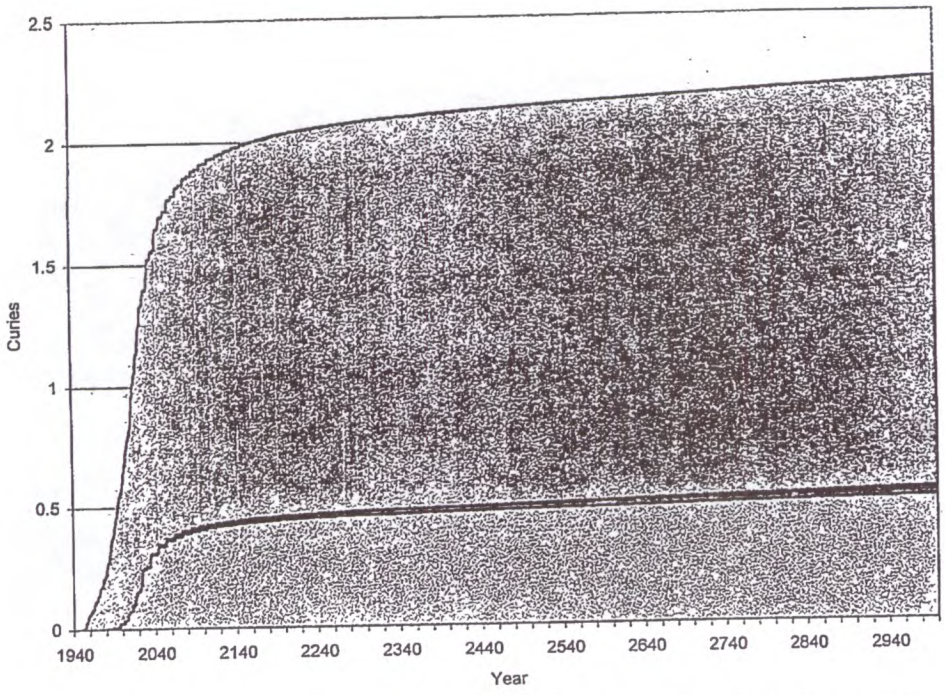


Figure 4.6b. Cumulative Release of Iodine-129 from All Sources to the Water Table from 1940 to 3000

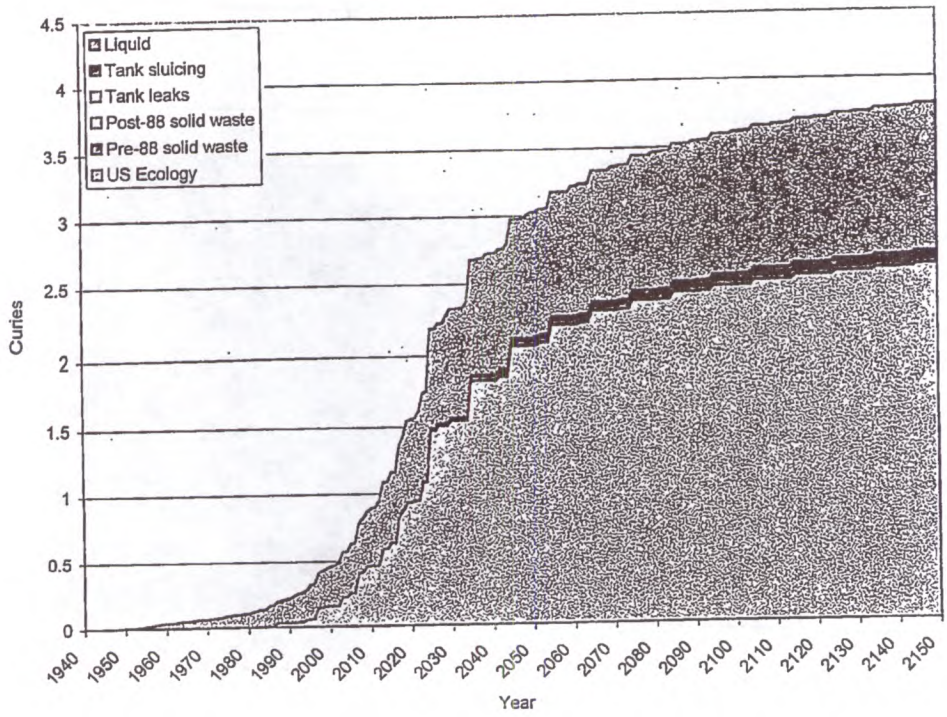


Figure 4.7a. Cumulative Release of Carbon-14 from All Sources to the Water Table from 1940 to 2150

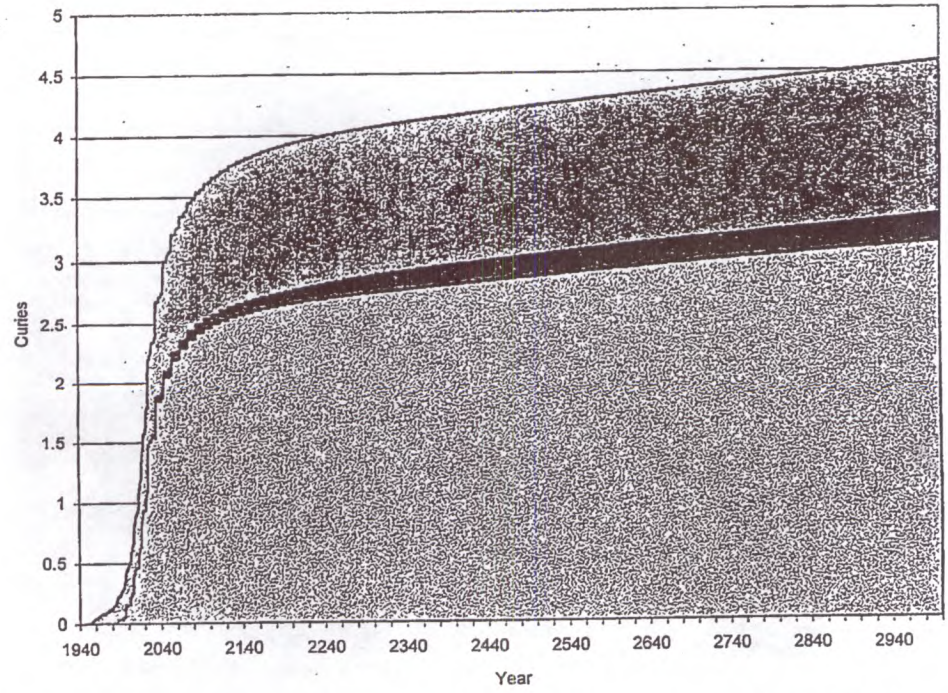


Figure 4.7b. Cumulative Release of Carbon-14 from All Sources to the Water Table from 1940 to 3000

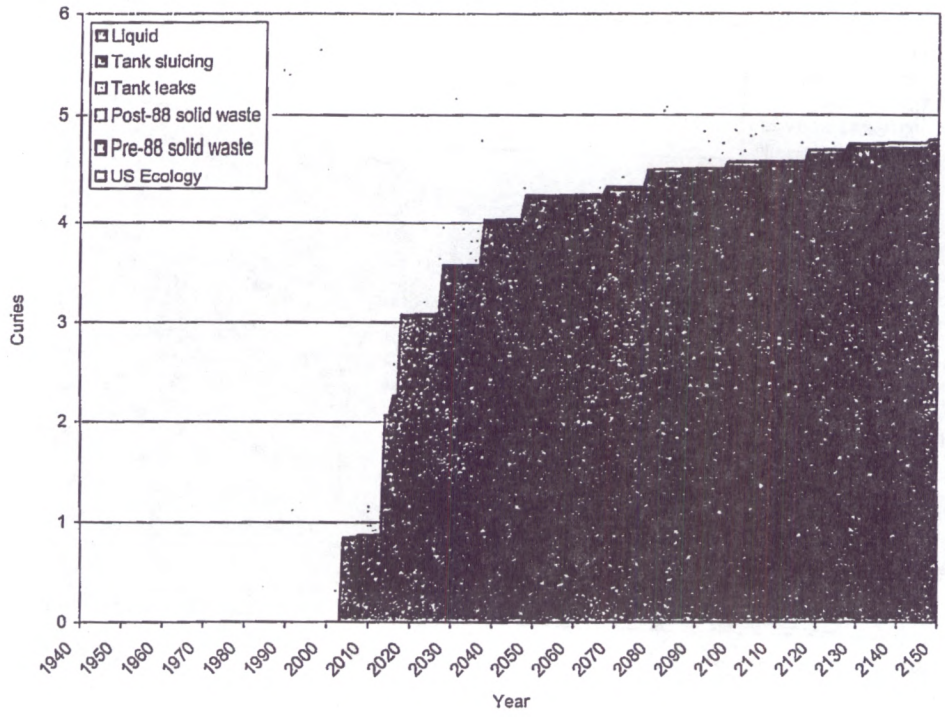


Figure 4.8a. Cumulative Release of Chlorine-36 from All Sources to the Water Table from 1940 to 2150

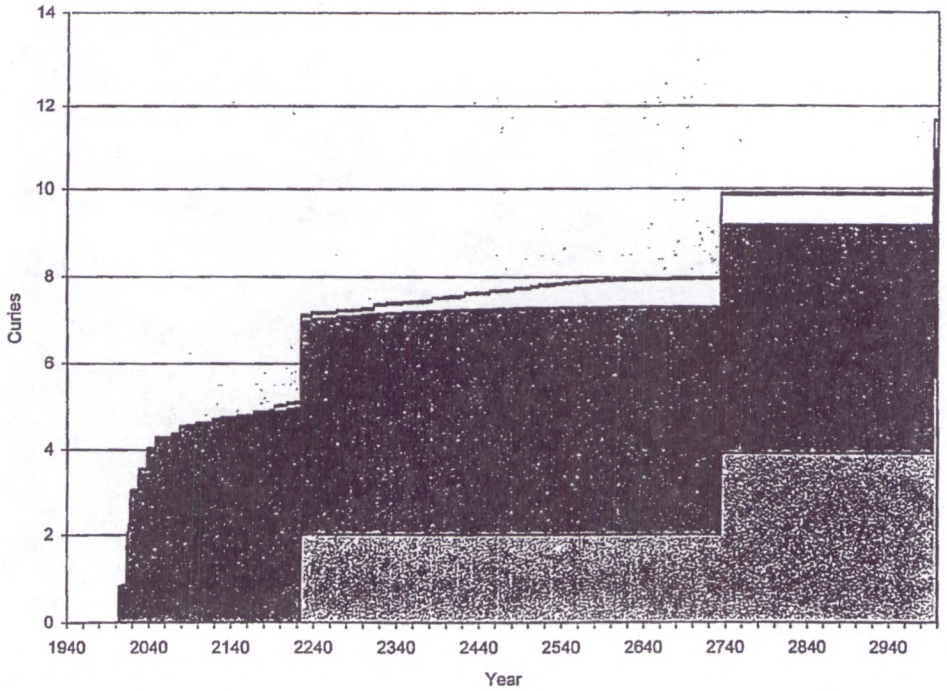


Figure 4.8b. Cumulative Release of Chlorine-36 from All Sources to the Water Table from 1940 to 3000

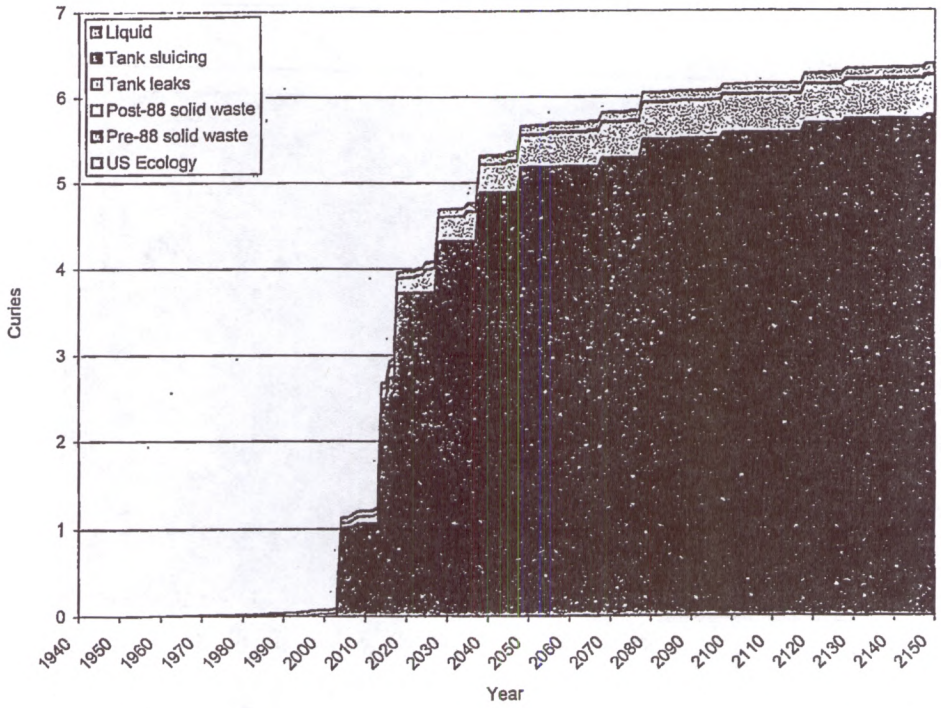


Figure 4.9a. Cumulative Release of Selenium-79 from All Sources to the Water Table from 1940 to 2150

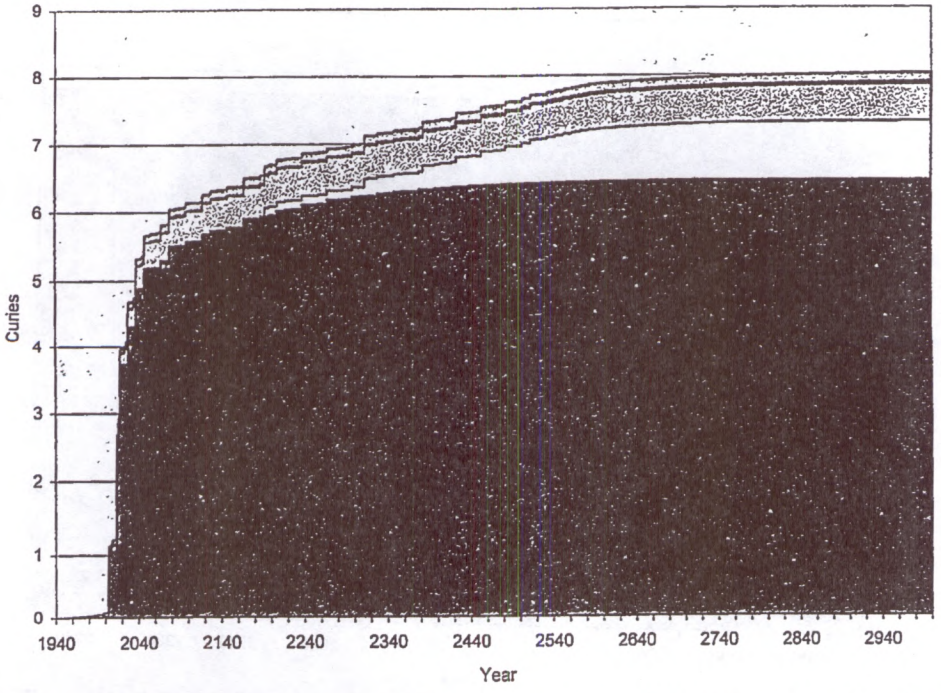


Figure 4.9b. Cumulative Release of Selenium-79 from All Sources to the Water Table from 1940 to 3000

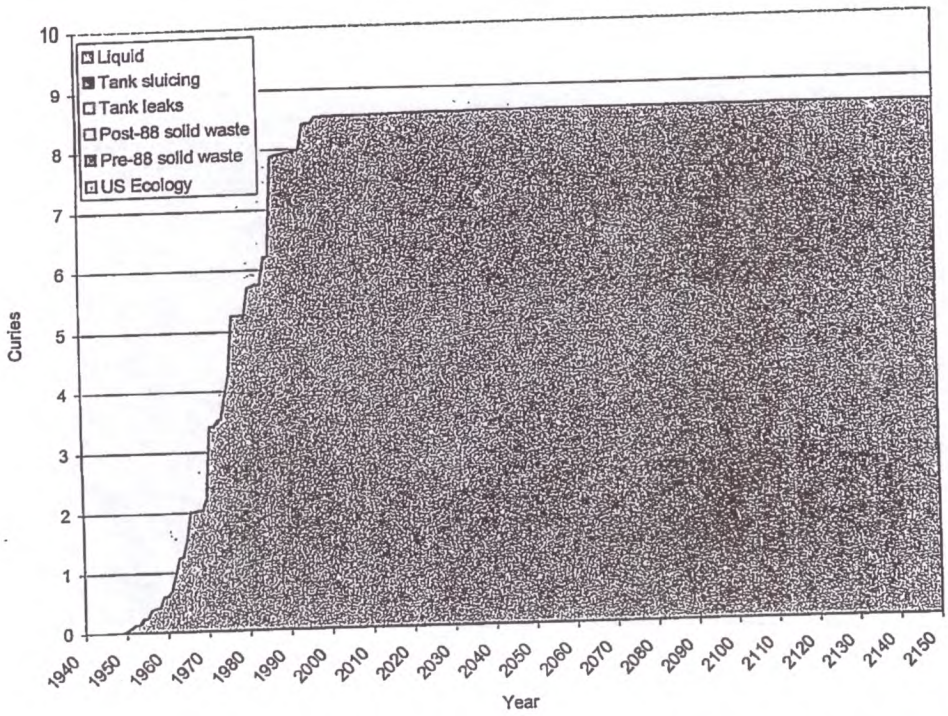


Figure 4.10a. Cumulative Release of Uranium-238 from All Sources to the Water Table from 1940 to 2150

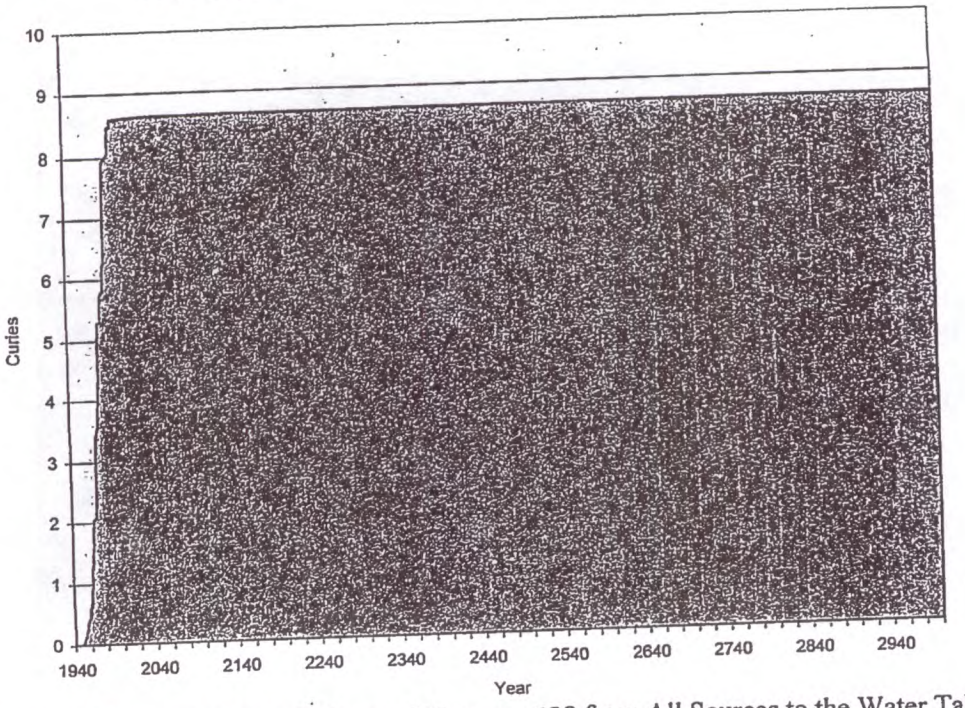


Figure 4.10b. Cumulative Release of Uranium-238 from All Sources to the Water Table from 1940 to 3000

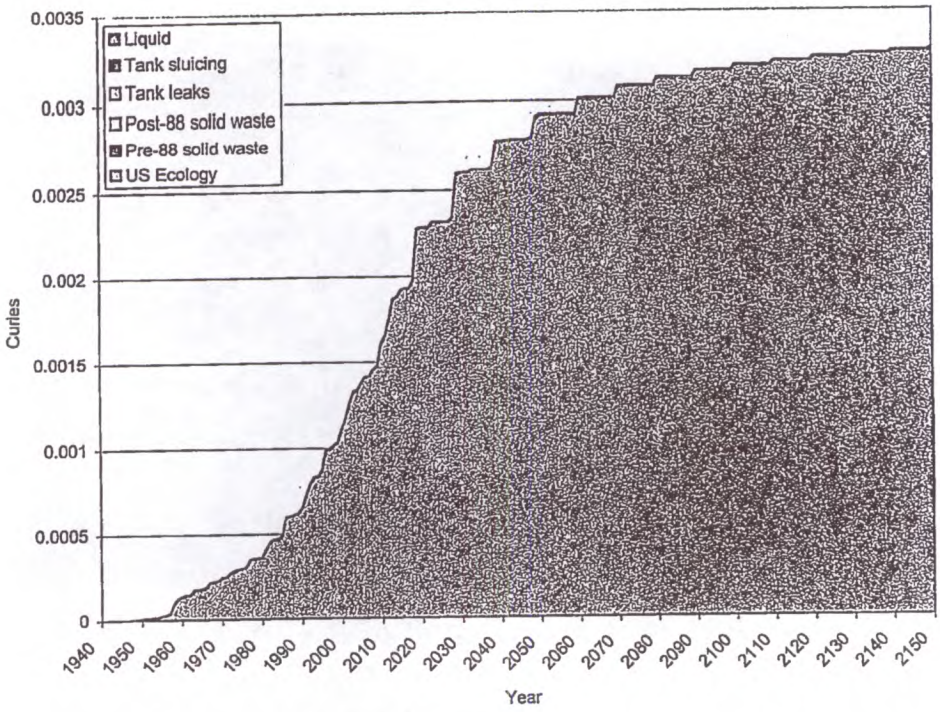


Figure 4.11a. Cumulative Release of Cobalt-60 from All Sources to the Water Table from 1940 to 2150

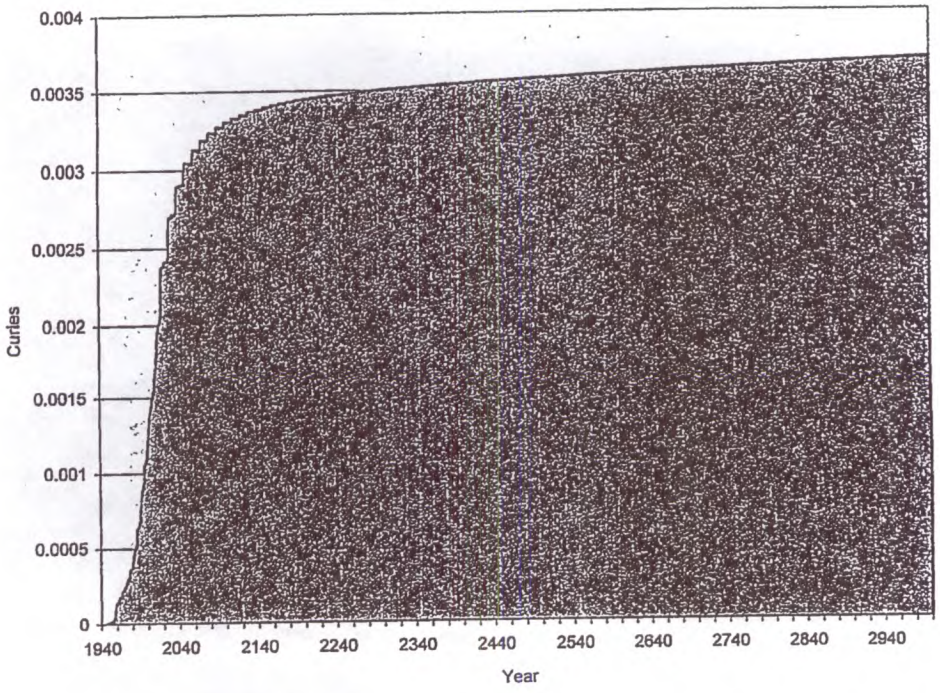


Figure 4.11b. Cumulative Release of Cobalt-60 from All Sources to the Water Table from 1940 to 3000

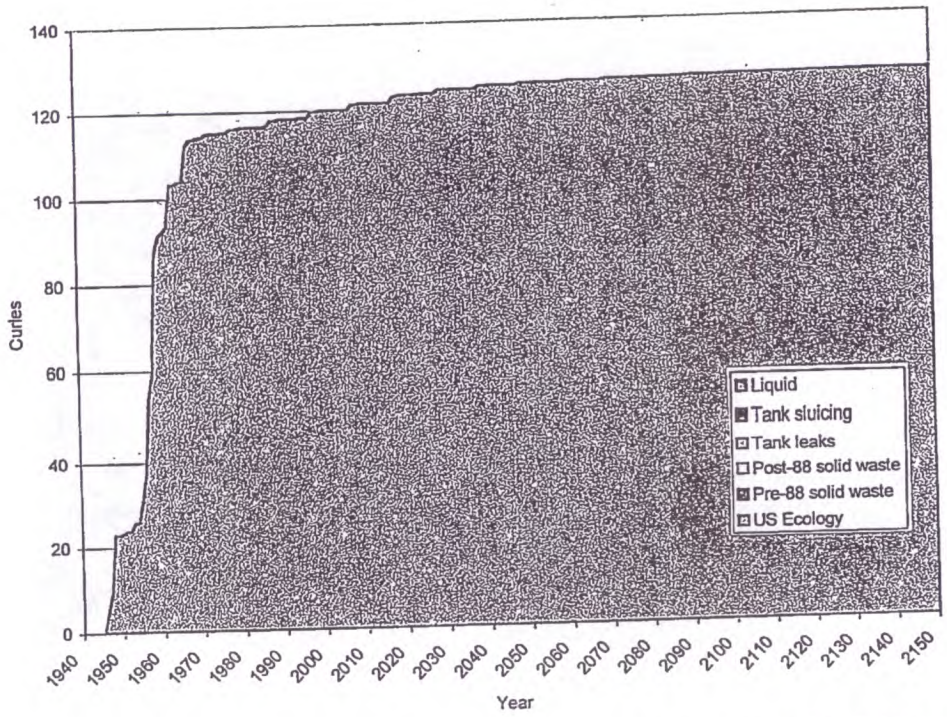


Figure 4.12a. Cumulative Release of Americium-241 from All Sources to the Water Table from 1940 to 2150

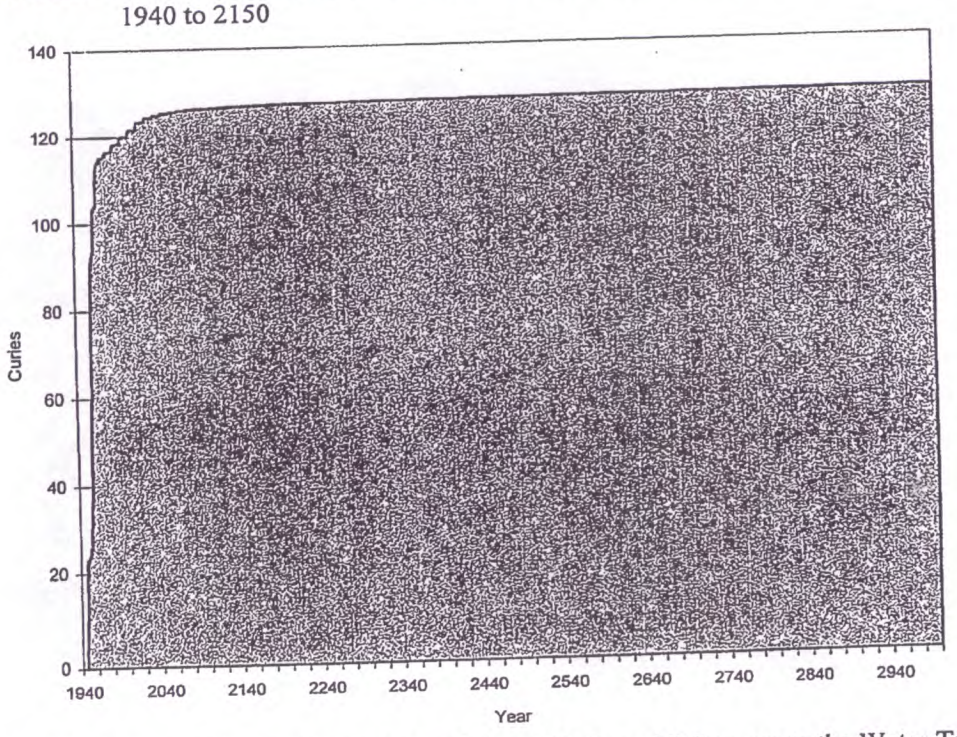


Figure 4.12b. Cumulative Release of Americium-241 from All Sources to the Water Table from 1940 to 3000

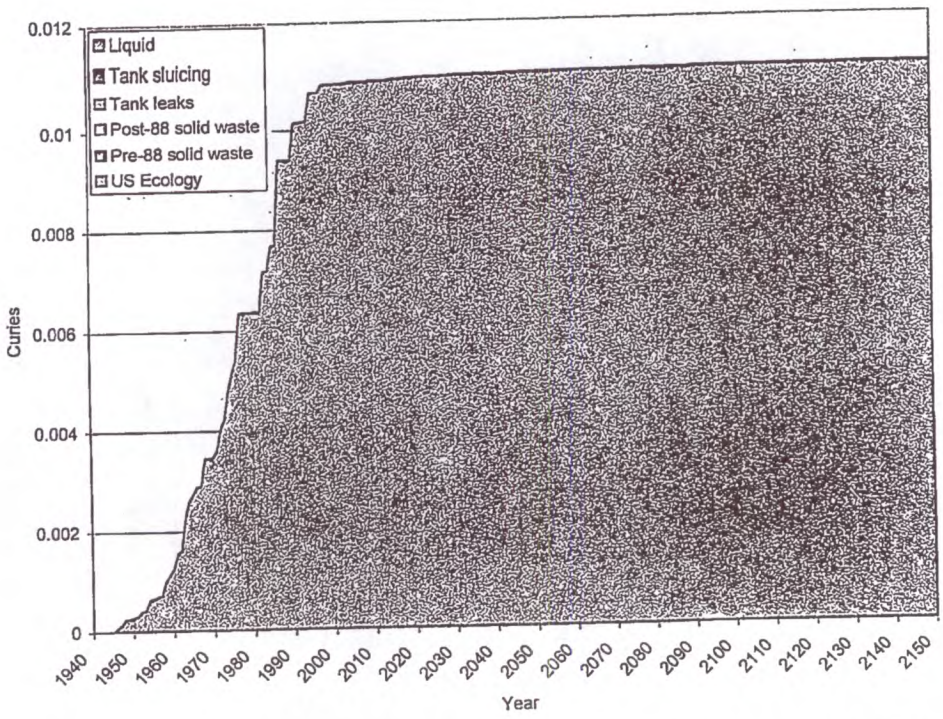


Figure 4.13a. Cumulative Release of Neptunium-237 from All Sources to the Water Table from 1940 to 2150

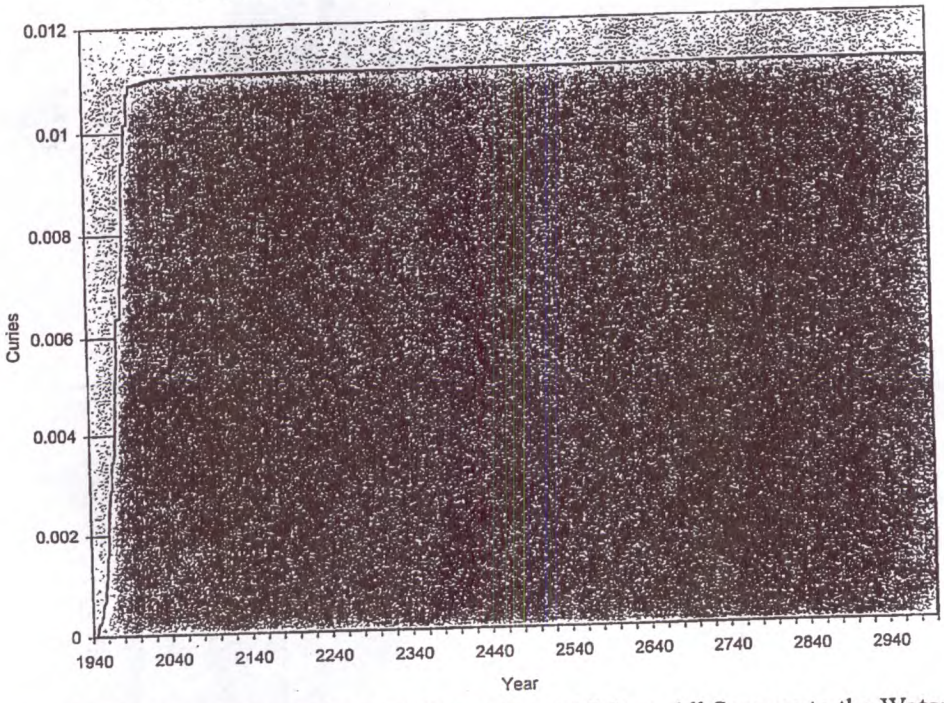


Figure 4.13b. Cumulative Release of Neptunium-237 from All Sources to the Water Table from 1940 to 3000

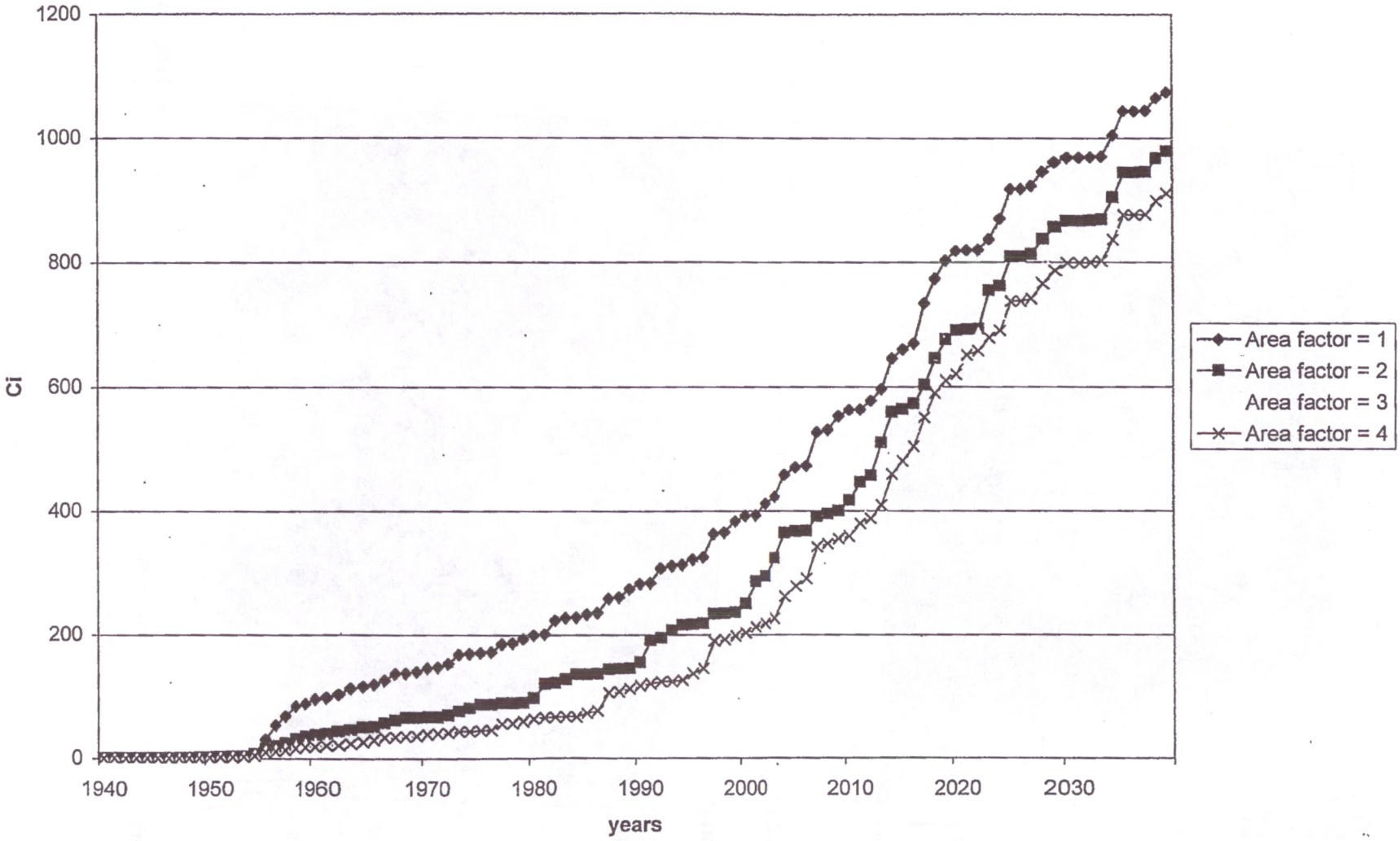


Figure 4.14a. Cumulative Release of Technetium-99 to Groundwater from Liquid Discharge Sources for Different Cross Section Area Factors from 1940 to 2040

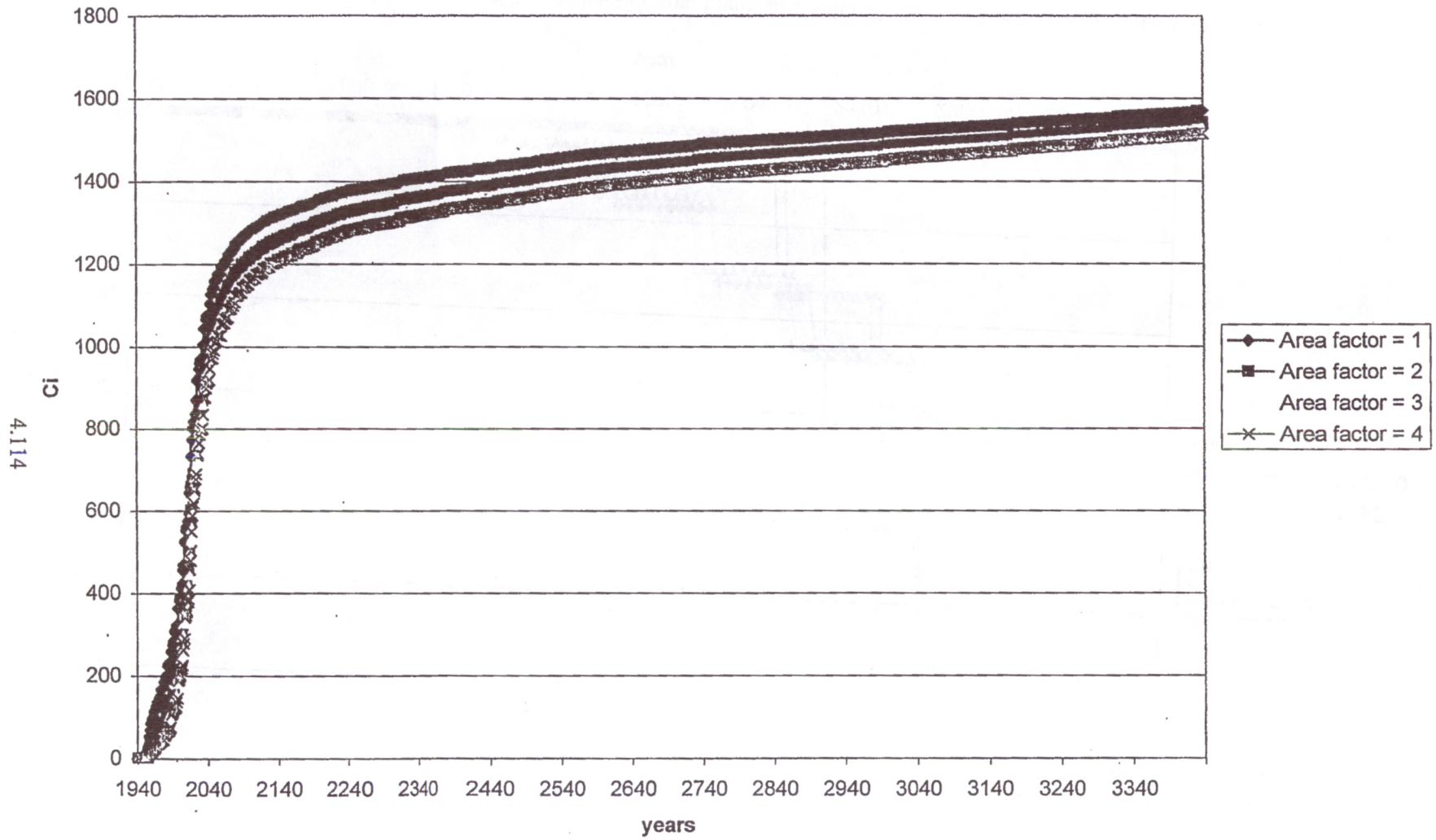


Figure 4.14b. Cumulative Release of Technetium-99 to Groundwater from Liquid Discharge Sources for Different Cross Section Area Factors from 1940 to 3440

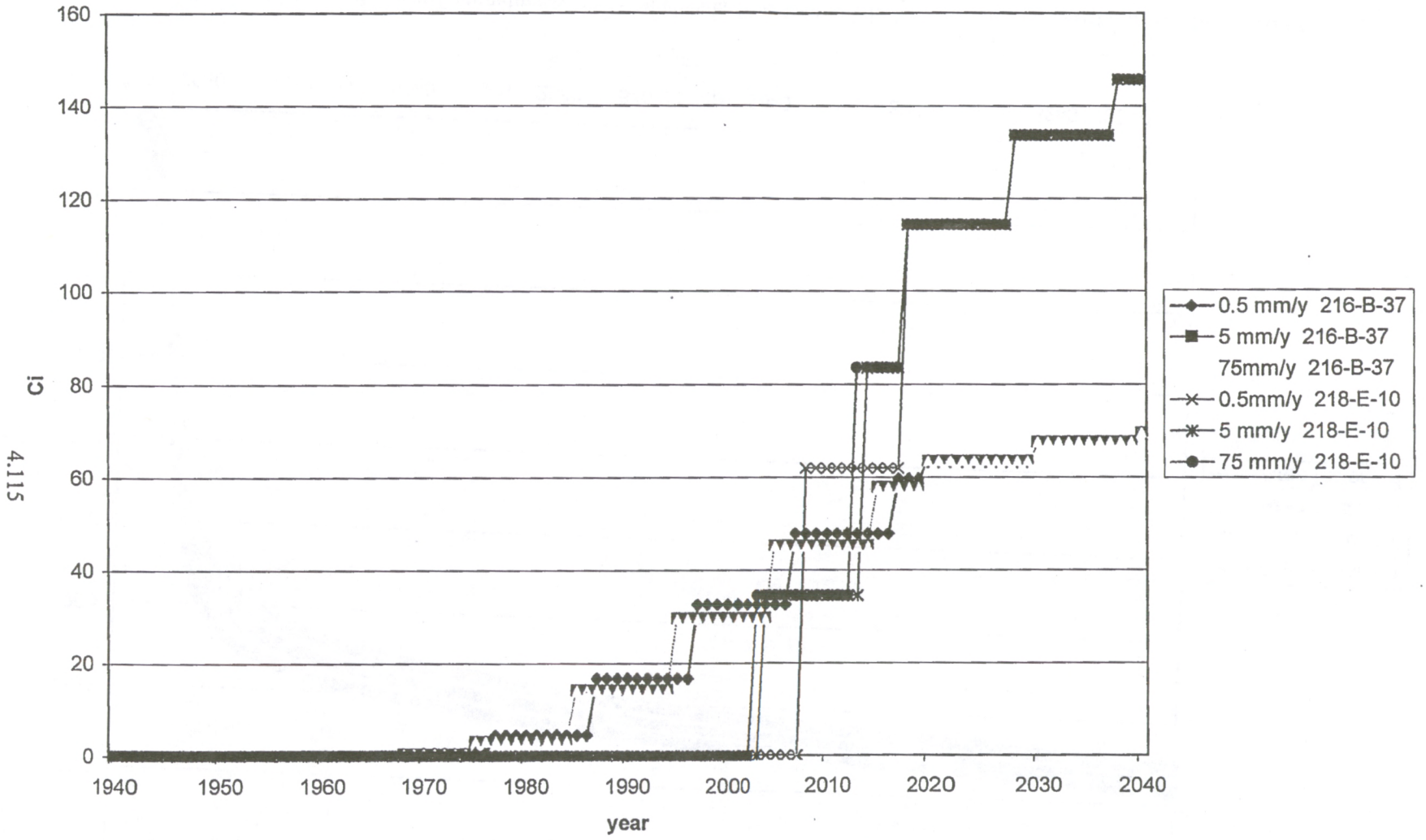


Figure 4.15. Cumulative Release of Technetium-99 from Liquid and Solid Waste Sites for Different Initial Soil Moisture Conditions

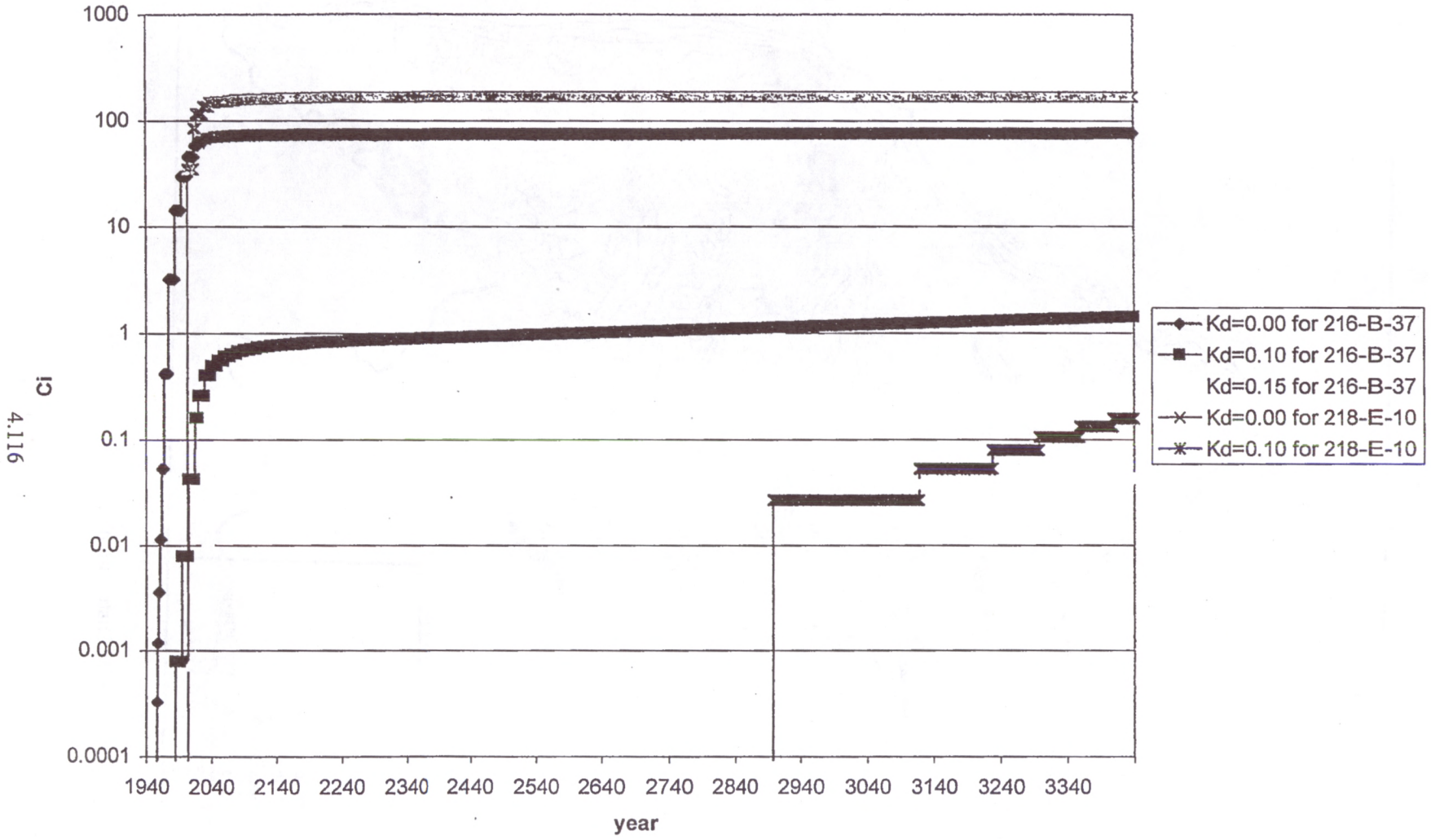
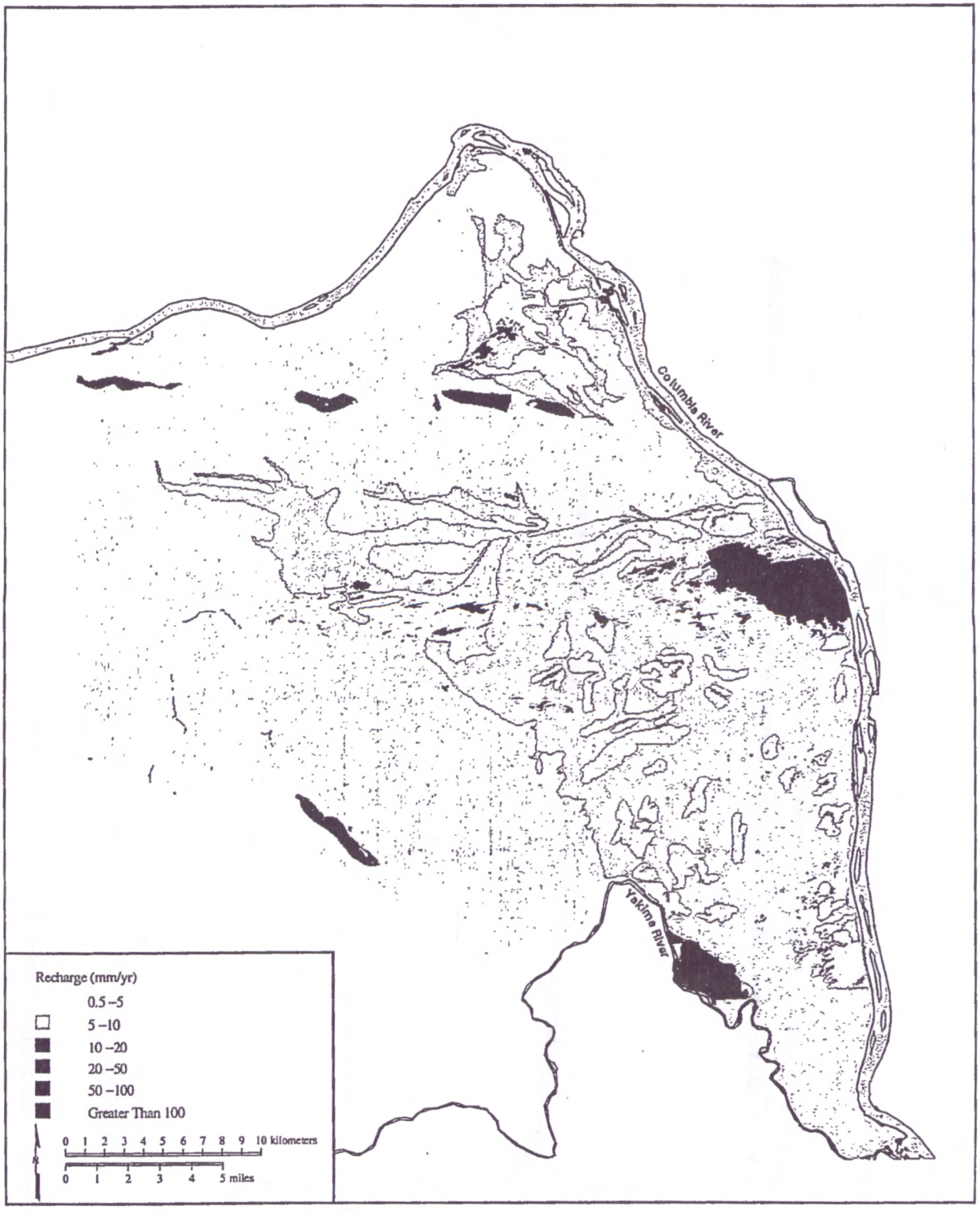
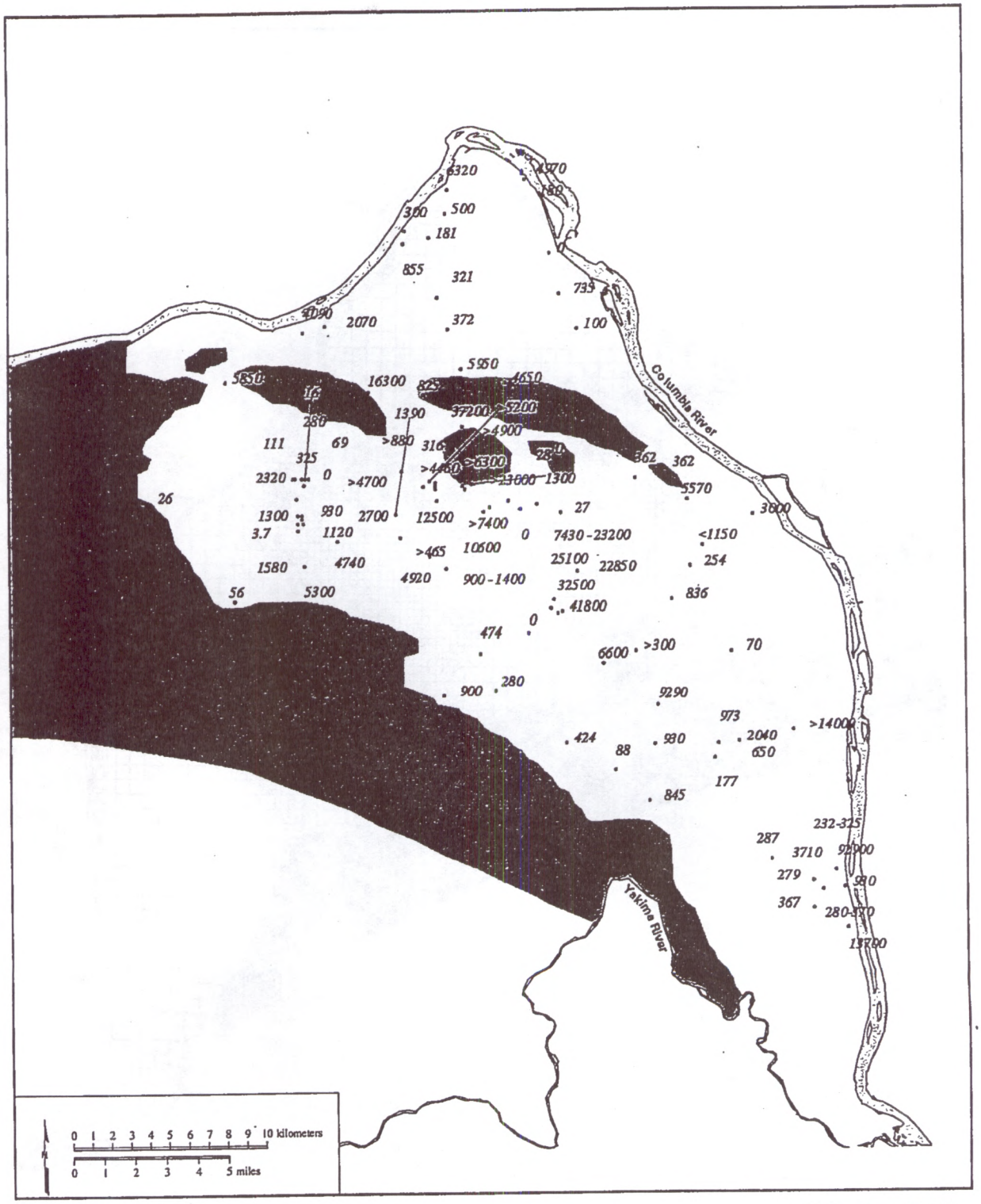


Figure 4.16. Cumulative Release of Technetium-99 from Liquid and Solid Waste Sites for Different Distribution Coefficients



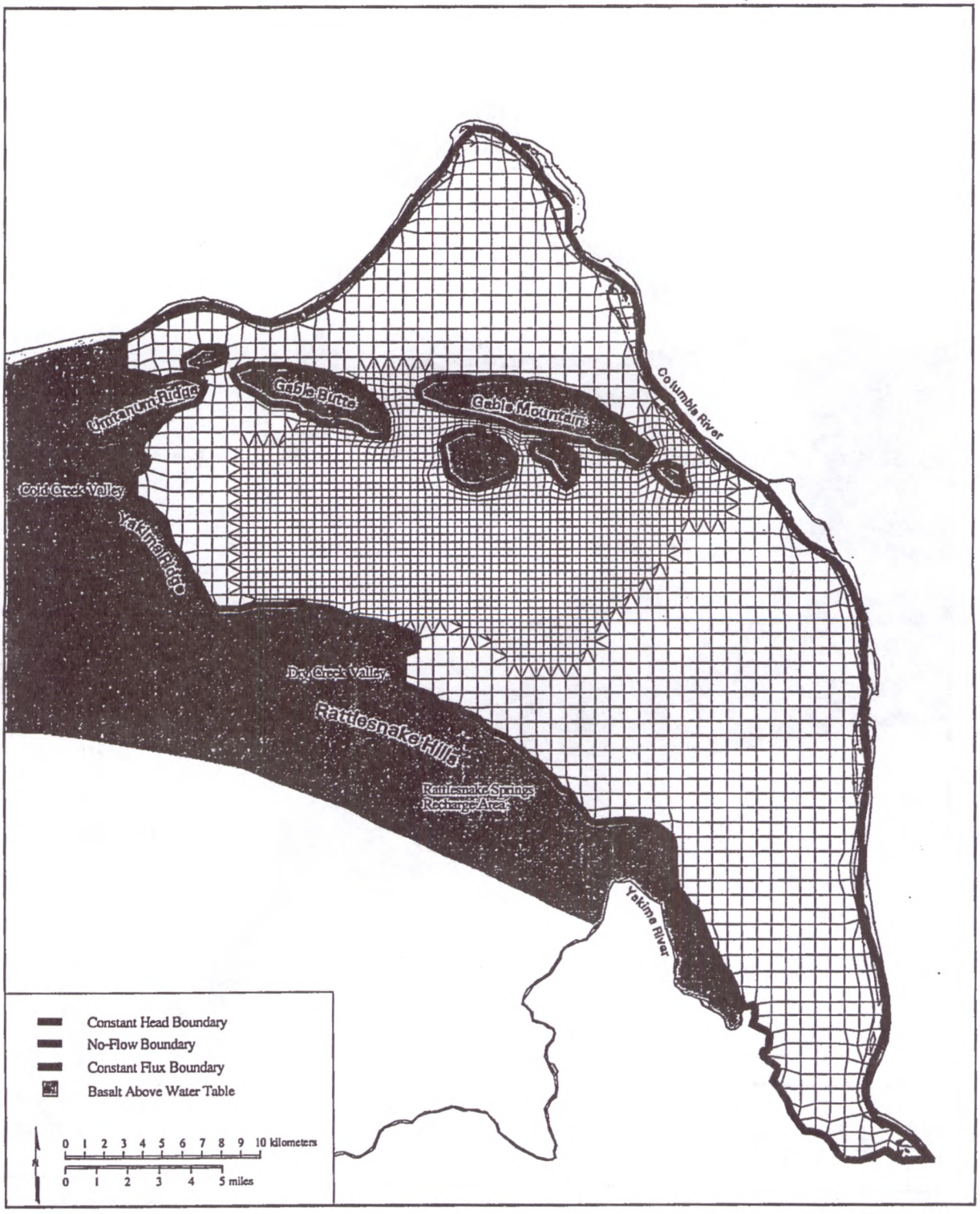
skw98001.eps January 02, 1998

Figure 4.17. Sitewide Distribution of Recharge at the Hanford Site Used in the Composite Analysis



skw98002.eps January 02, 1998

Figure 4.18. Locations of Hanford Site Wells with Aquifer Test Results Used in the Groundwater Model Calibration (Values Shown are the Interpreted Transmissivity in m^2/d)



skw98003.eps January 02, 1998

Figure 4.19. Finite Element Grid and Boundary Conditions Used in the Groundwater Model of the Unconfined Aquifer for the Composite Analysis

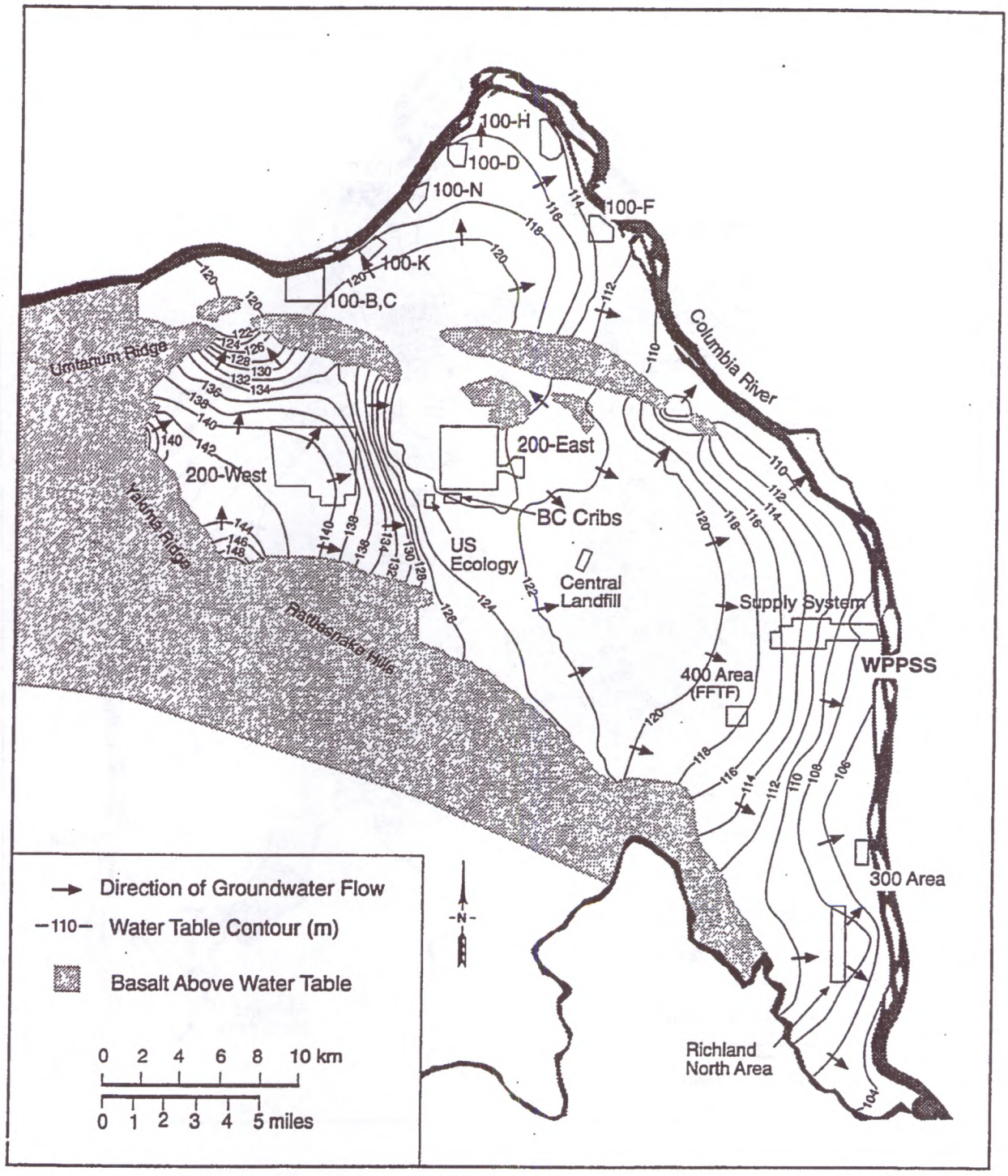


Figure 4.20. Water Table Predicted in 2000 with the Three-Dimensional Model

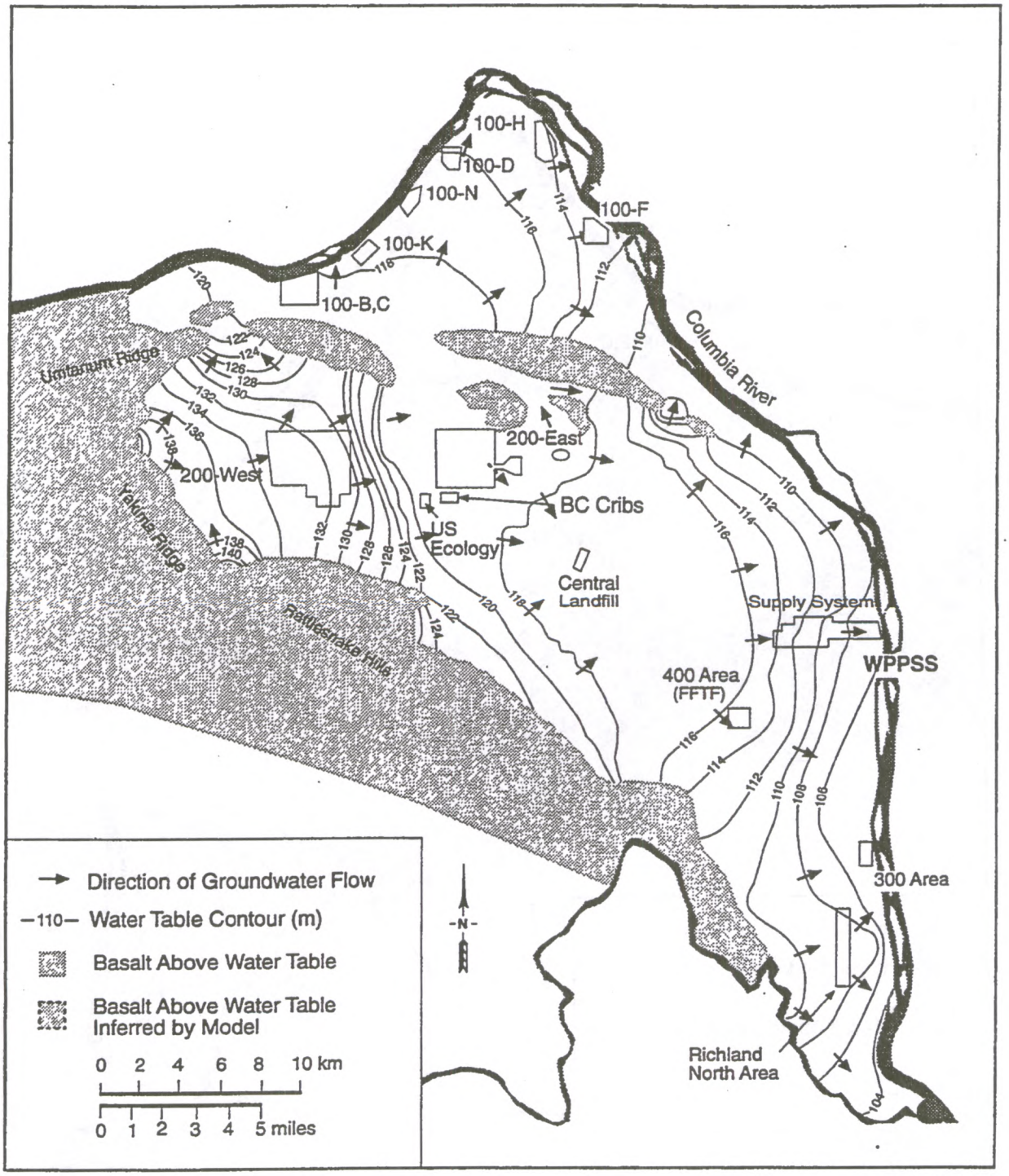


Figure 4.21. Water Table Predicted in 2100 with the Three-Dimensional Model

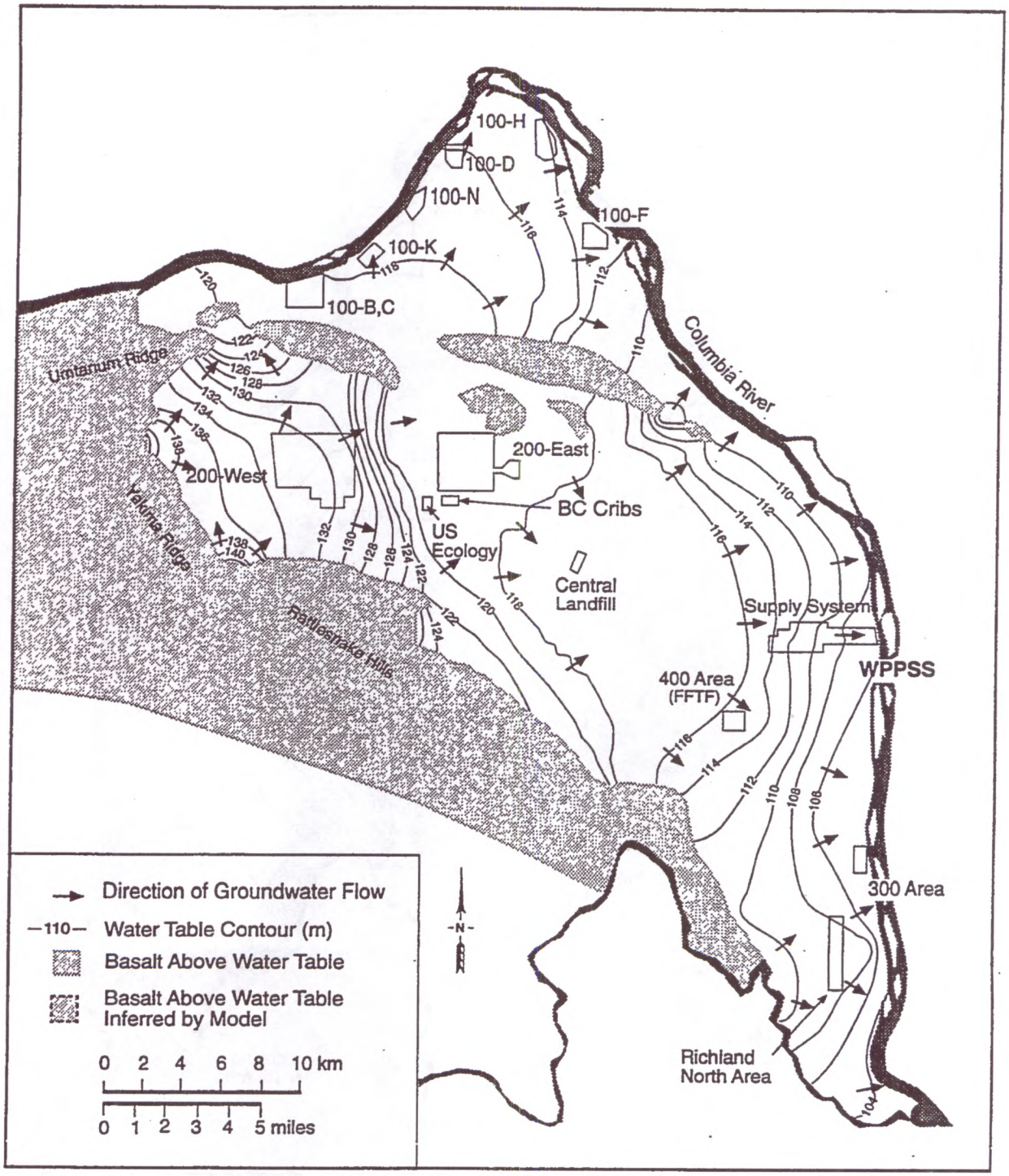


Figure 4.22. Water Table Predicted in 2200 with the Three-Dimensional Model

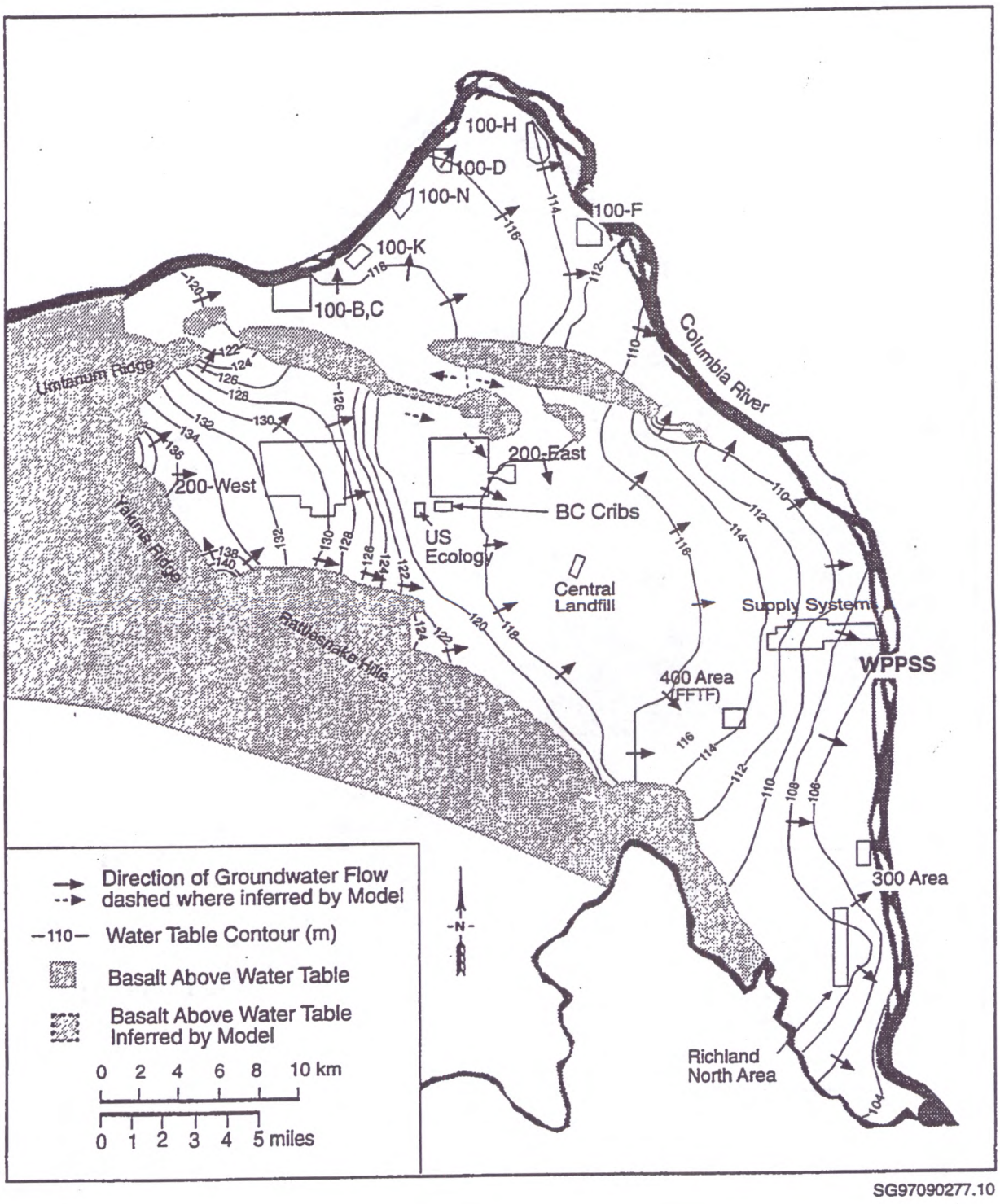
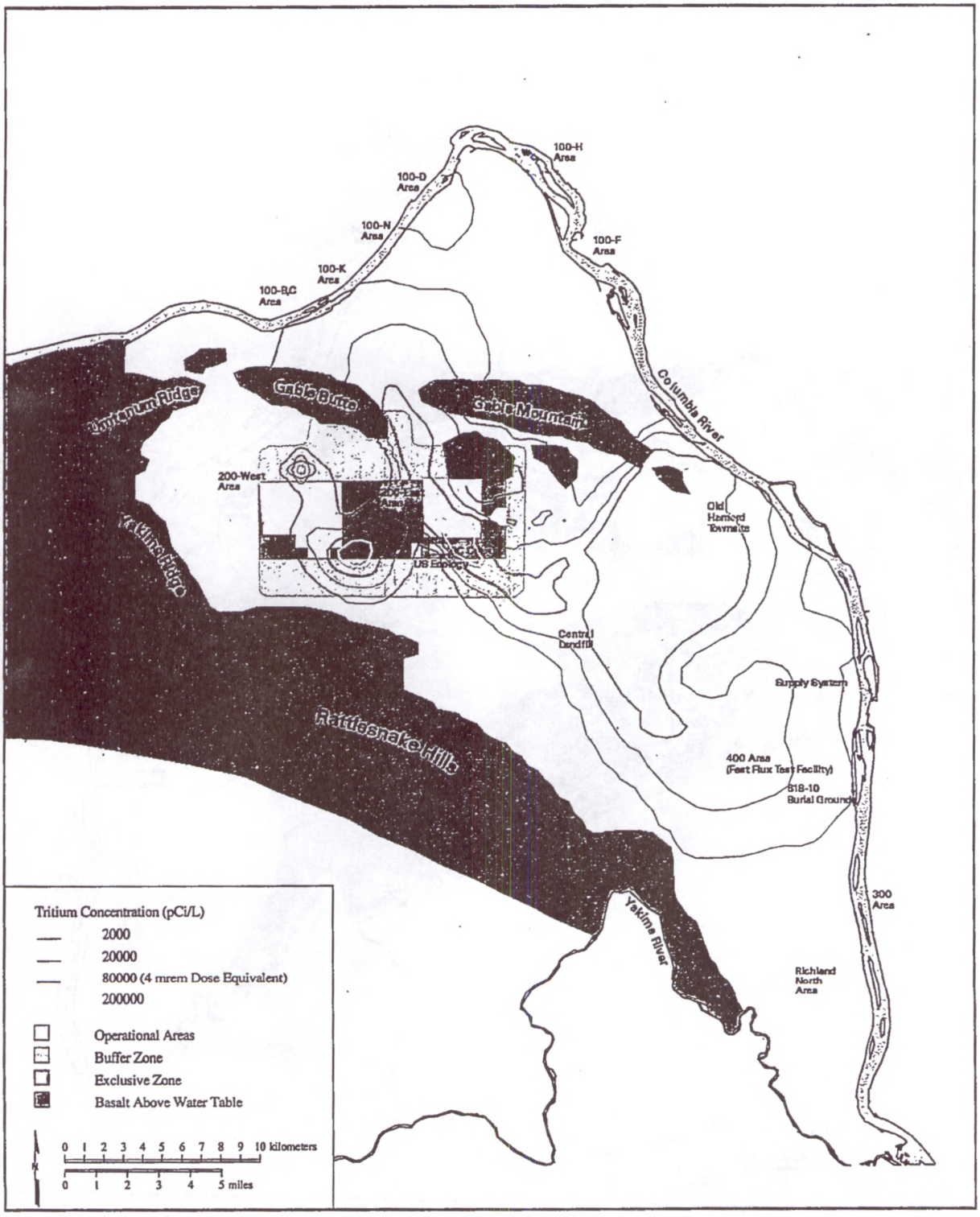
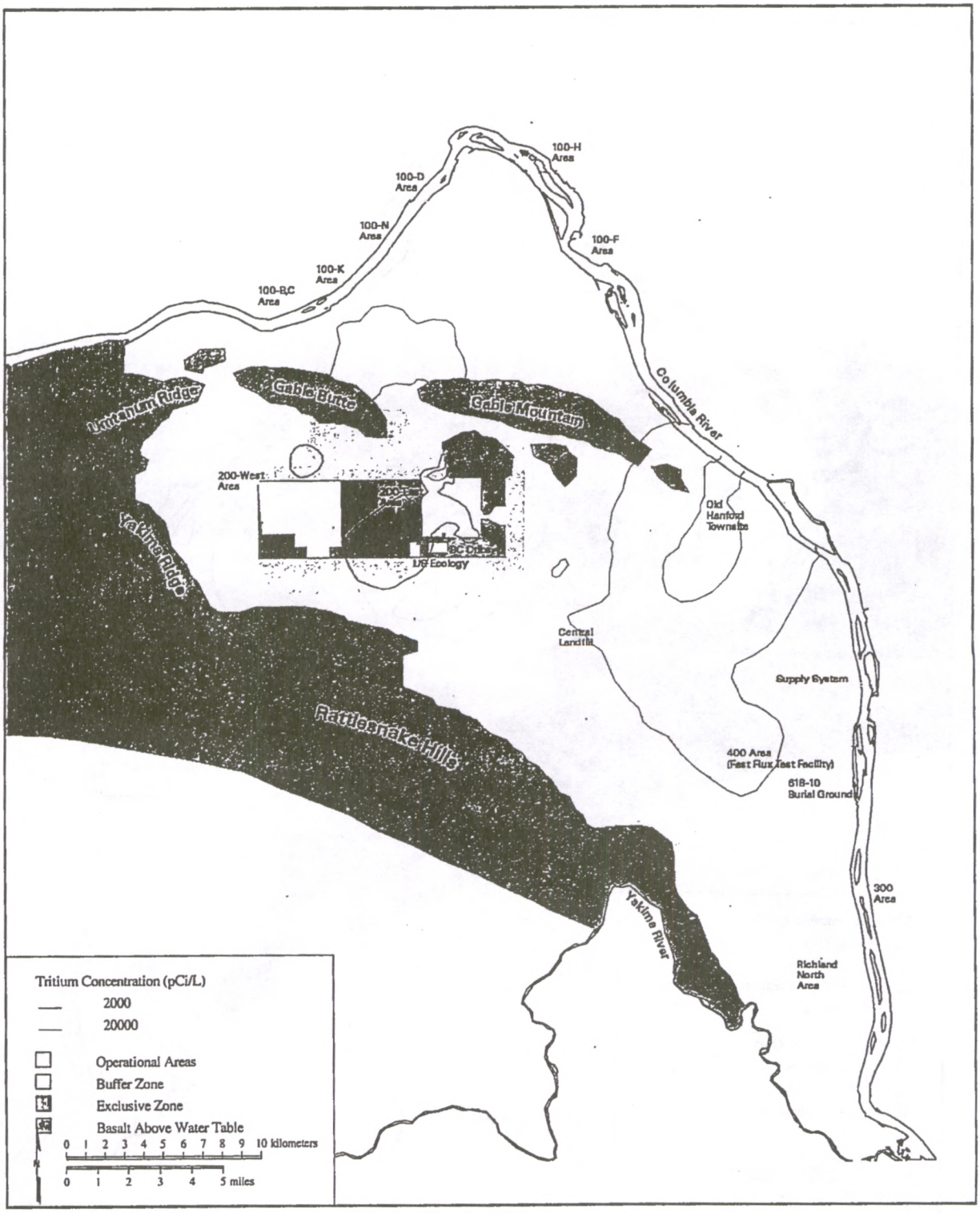


Figure 4.23. Water Table Predicted in 2350 with the Three-Dimensional Model



skw98037.eps December 23, 1997

Figure 4.24a. Predicted Distribution of Tritium in the Unconfined Aquifer from All Sources in 1997 (Time of Peak Concentration)



skw98050.eps December 23, 1997

Figure 4.24b. Predicted Distribution of Tritium in the Unconfined Aquifer from All Sources in 2050

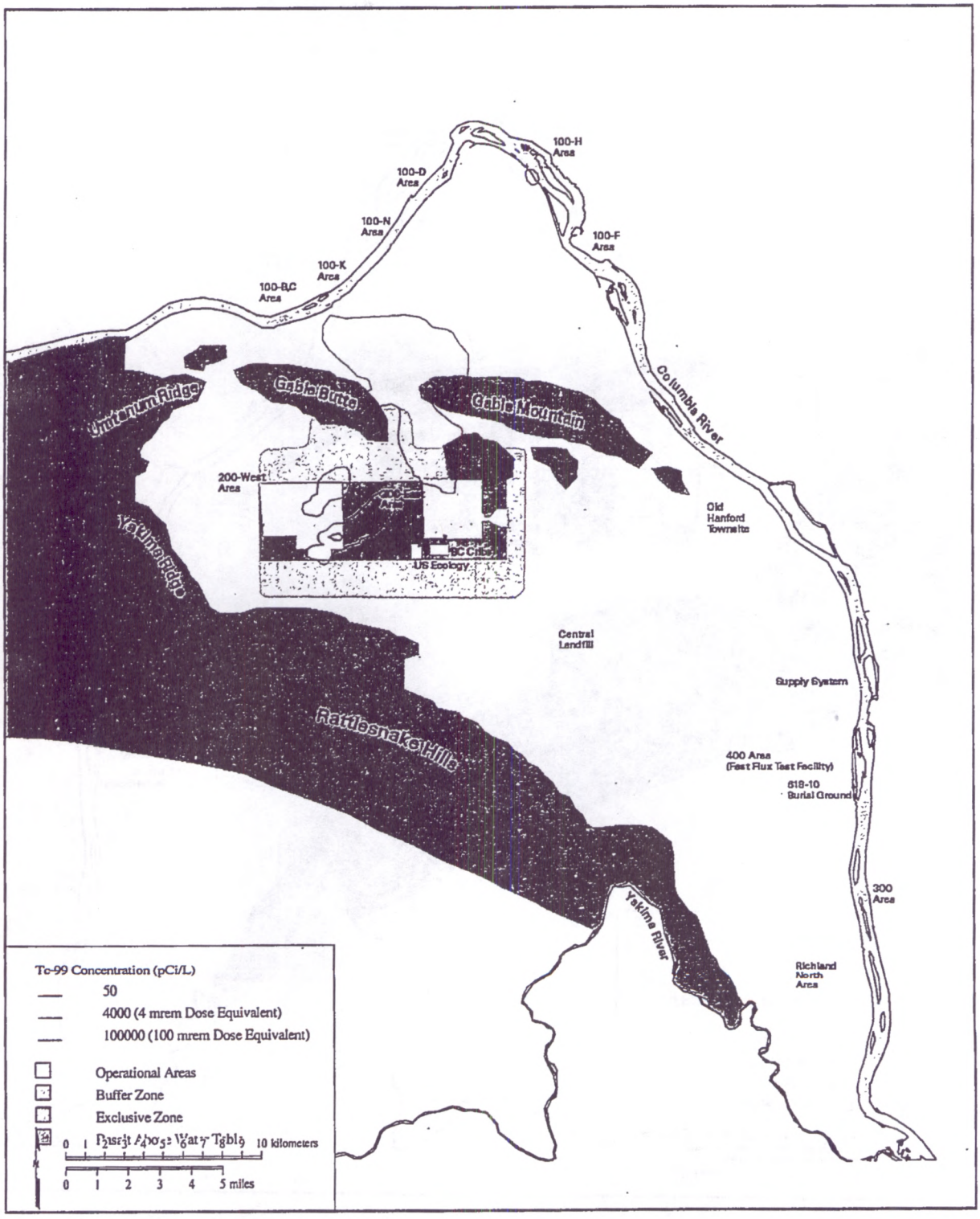
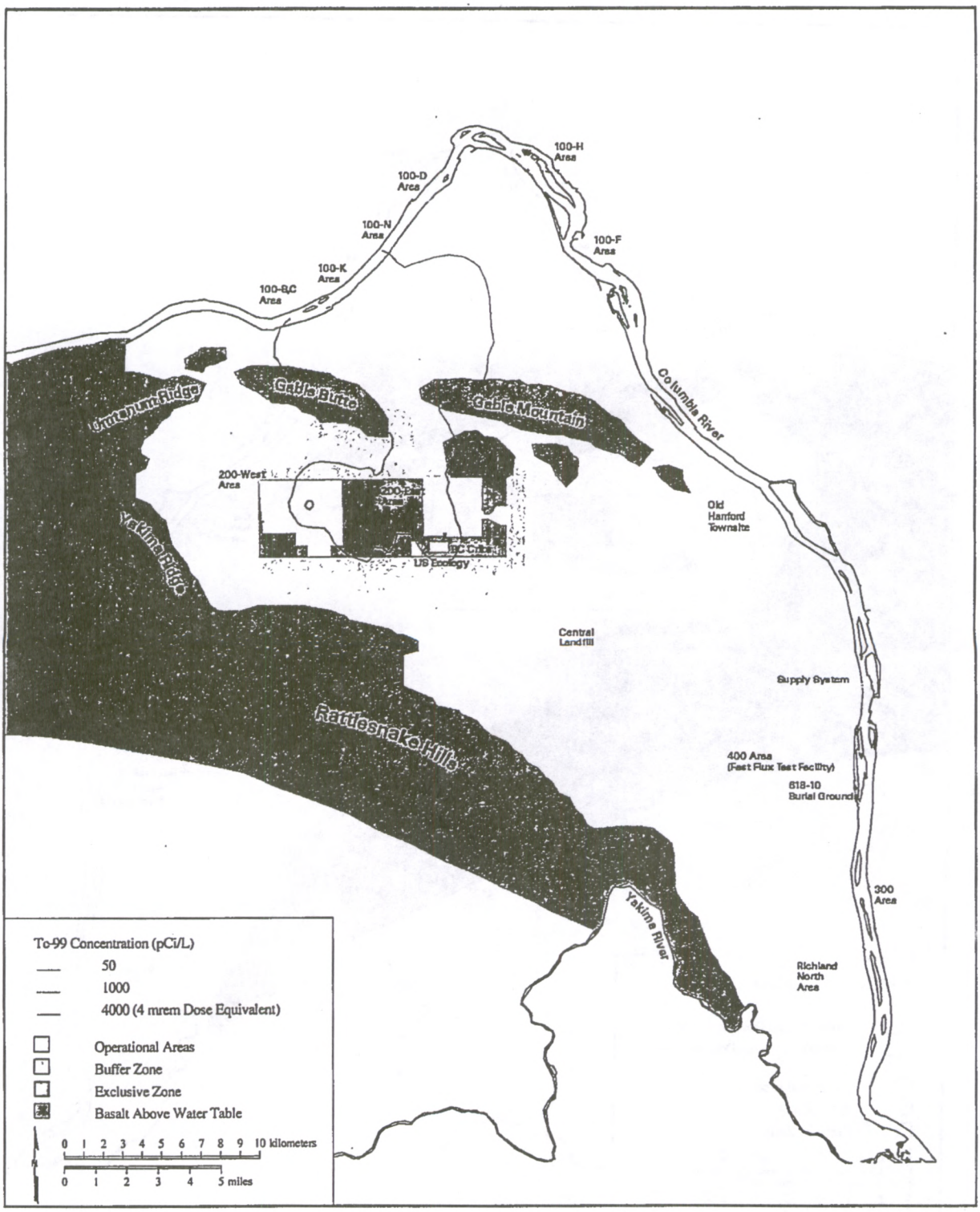
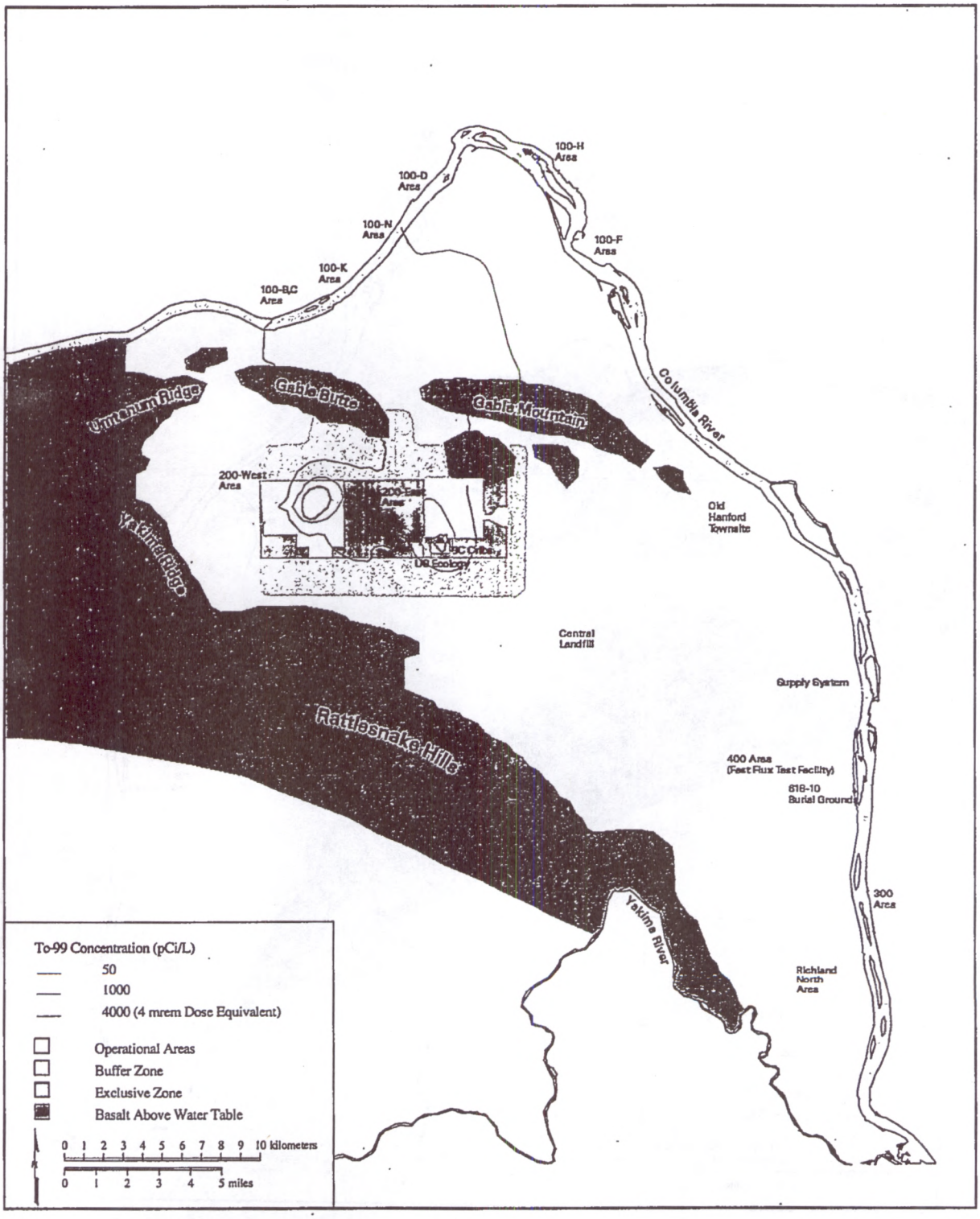


Figure 4.25a. Distribution of Technetium-99 in the Unconfined Aquifer from Existing Plumes in 1996 (Time of Peak Concentration)



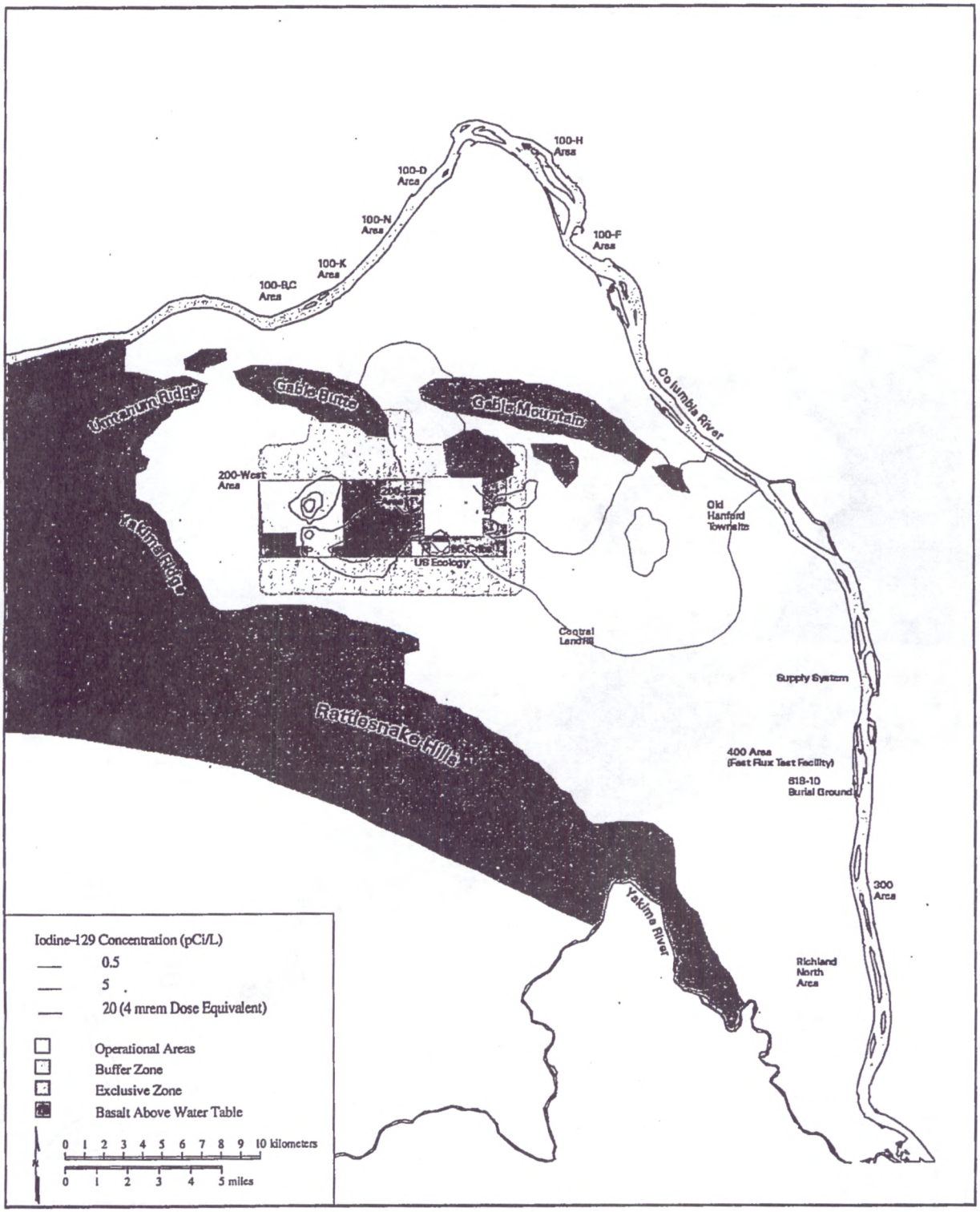
skw98055.eps December 23, 1997

Figure 4.25b. Predicted Distribution of Technetium-99 in the Unconfined Aquifer from All Sources in 2049



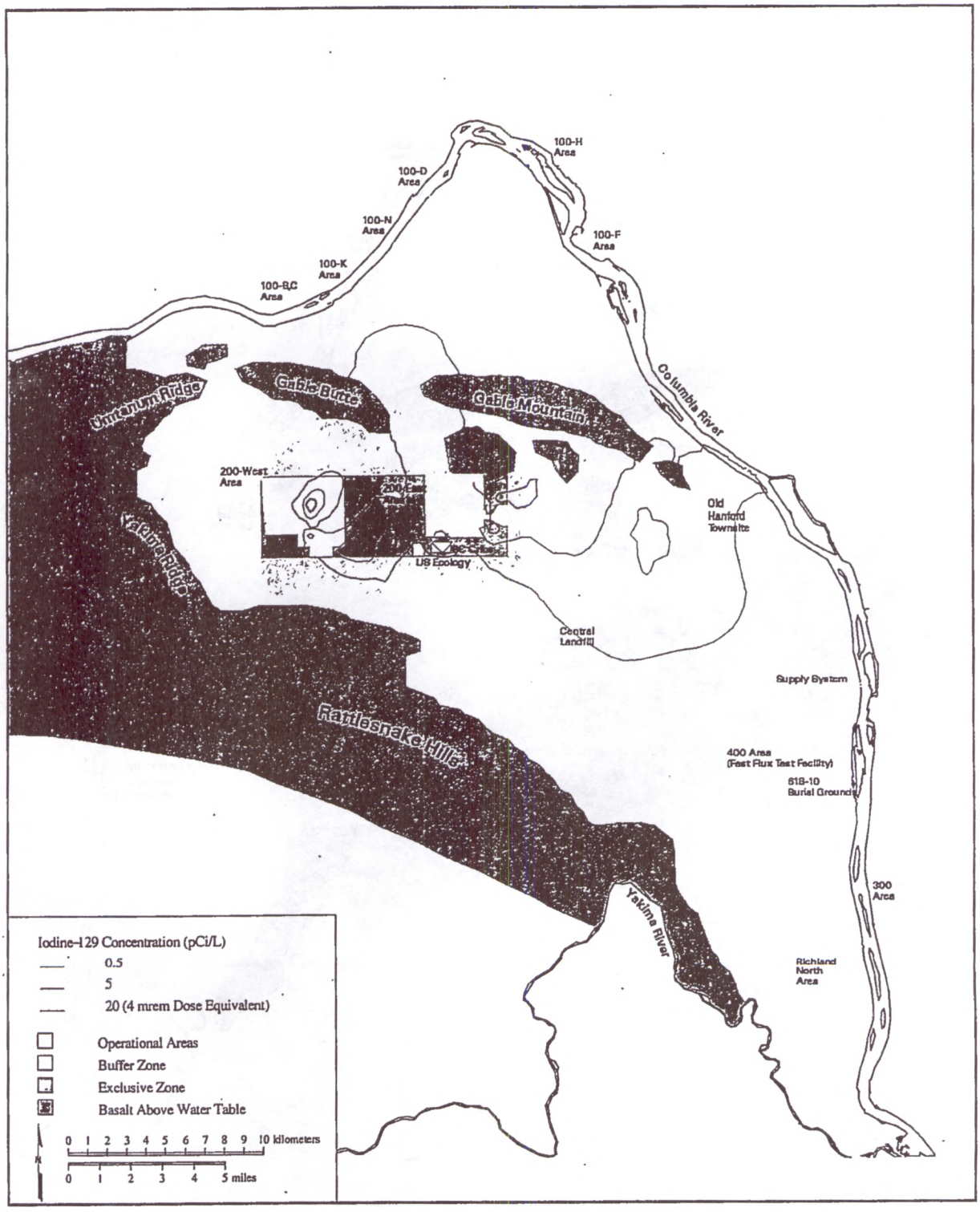
skw98058.eps December 23, 1997

Figure 4.26. Predicted Distribution of Technetium-99 in the Unconfined Aquifer from All Sources in 2036 (Time of Secondary Peak Concentration)



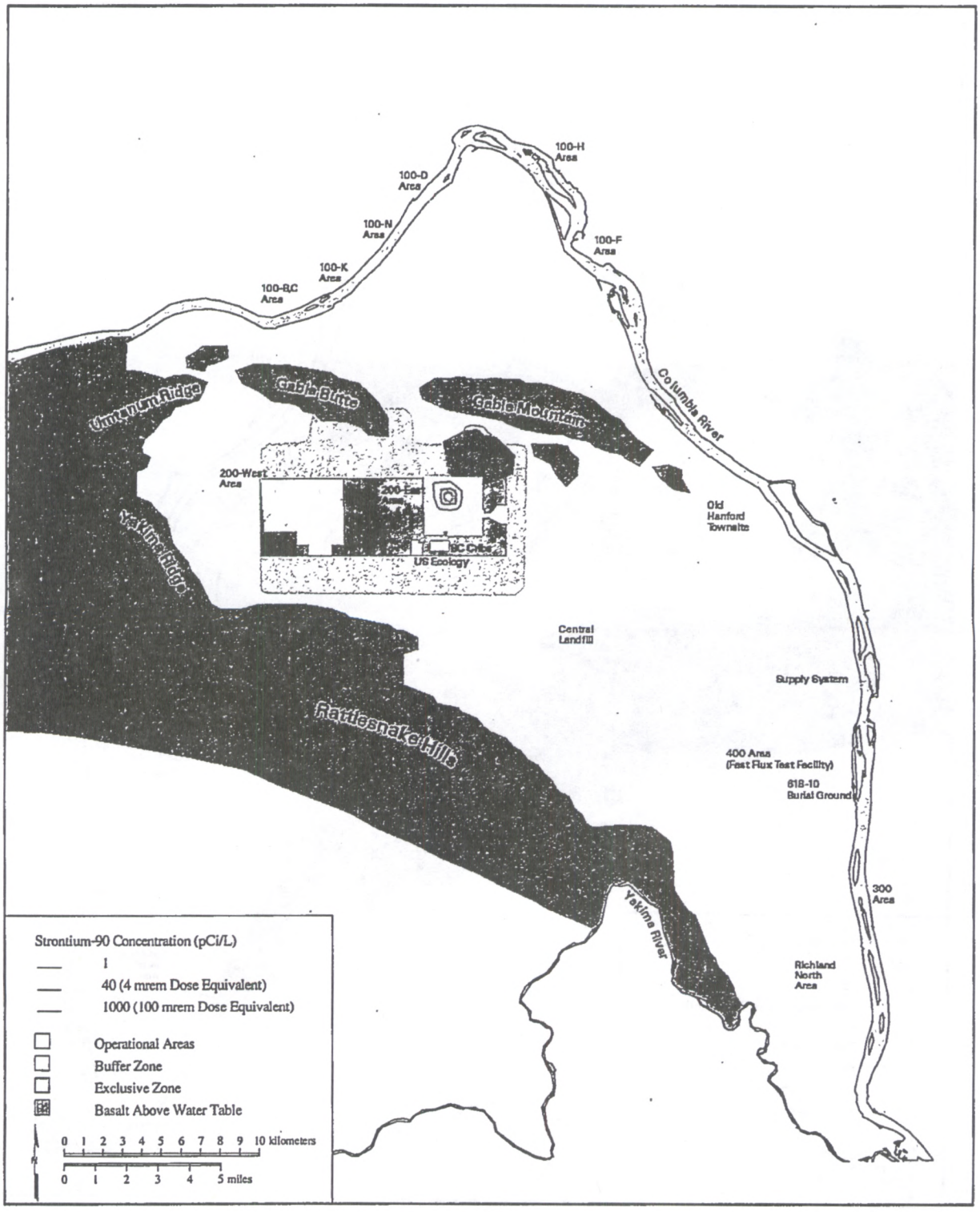
skw98059.eps December 23, 1997

Figure 4.27a. Predicted Distribution of Iodine-129 in the Unconfined Aquifer from All Sources in 2036 (Time of Peak Concentration)



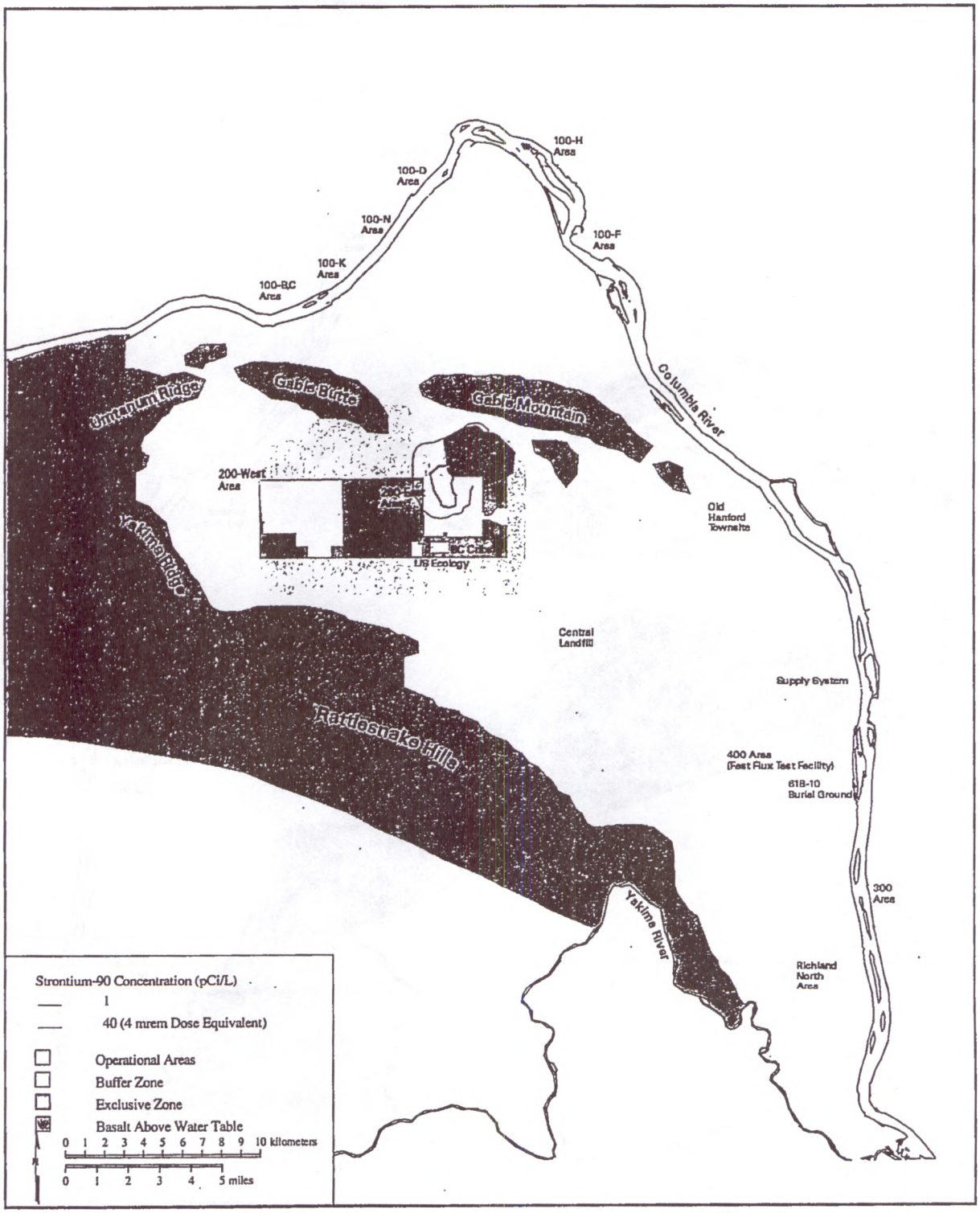
skw98056.eps March 03, 1998

Figure 4.27b. Predicted Distribution of Iodine-129 in the Unconfined Aquifer from All Sources in 2049



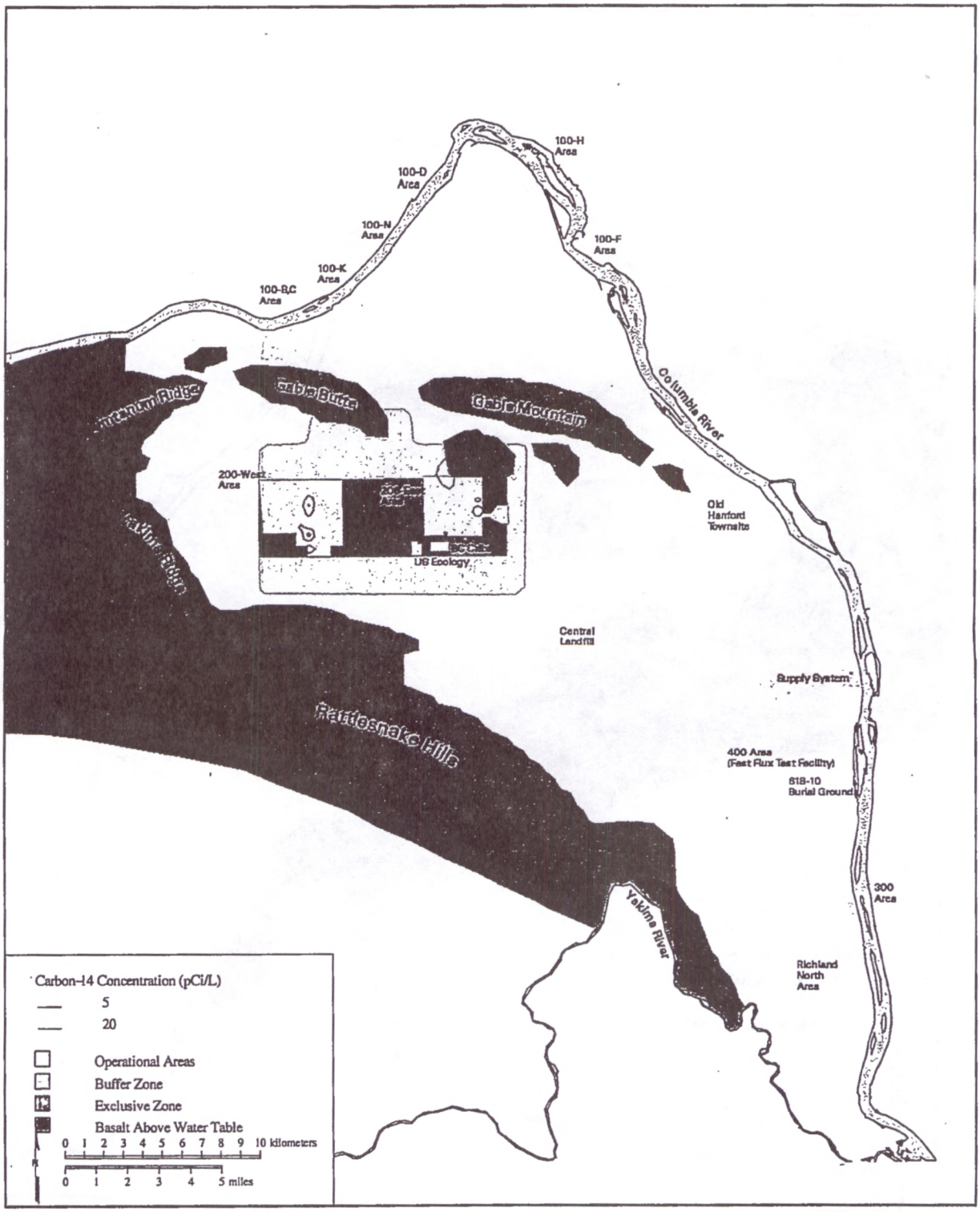
skw98041.eps December 23, 1997

Figure 4.28a. Distribution of Strontium-90 in the Unconfined Aquifer from All Sources in 1996 (Time of Peak Concentration)



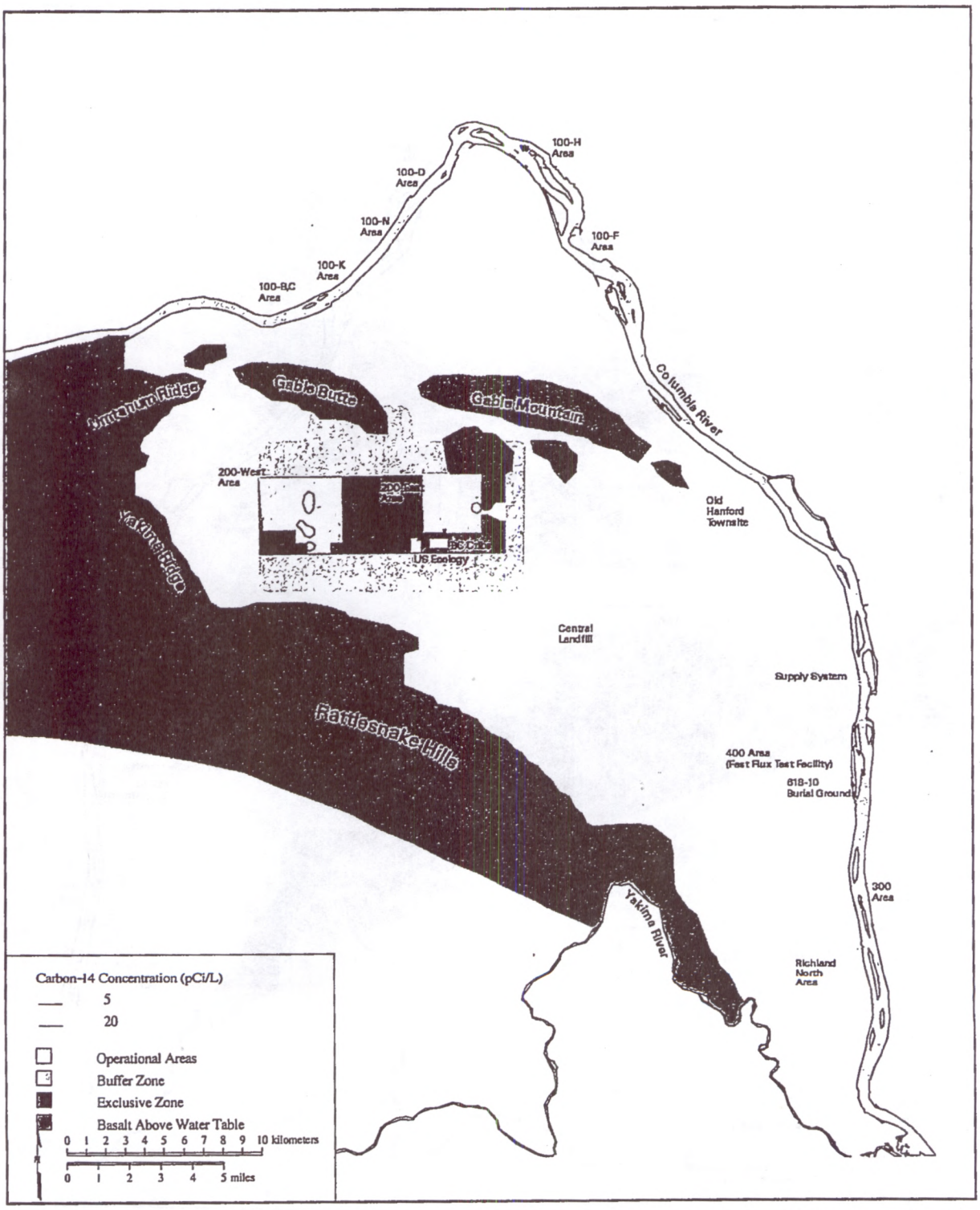
skw98054.eps December 23, 1997

Figure 4.28b. Predicted Distribution of Strontium-90 in the Unconfined Aquifer from All Sources in 2049



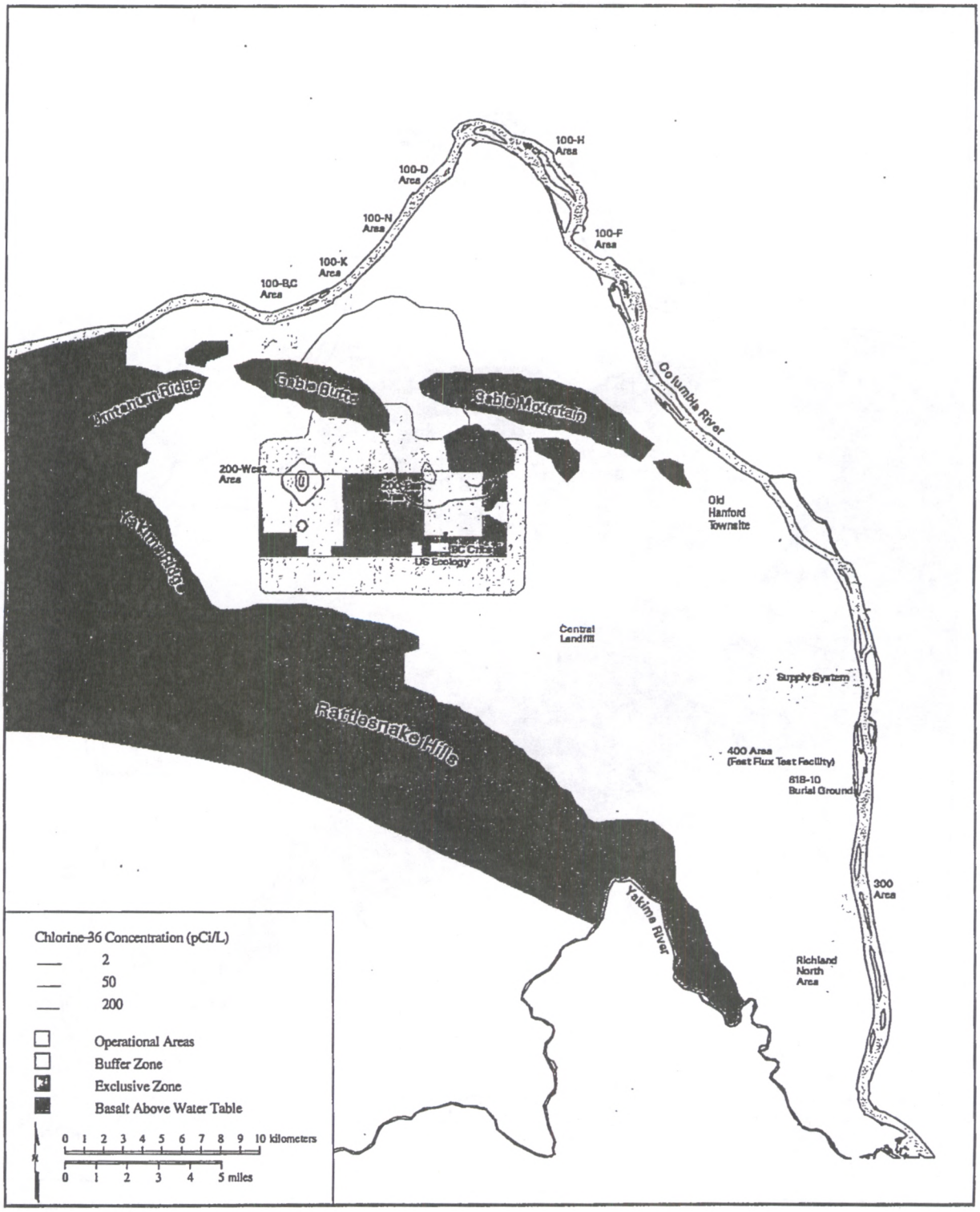
skw98038.eps December 23, 1997

Figure 4.29a. Predicted Distribution of Carbon-14 in the Unconfined Aquifer from All Sources in 2027 (Time of Peak Concentration)



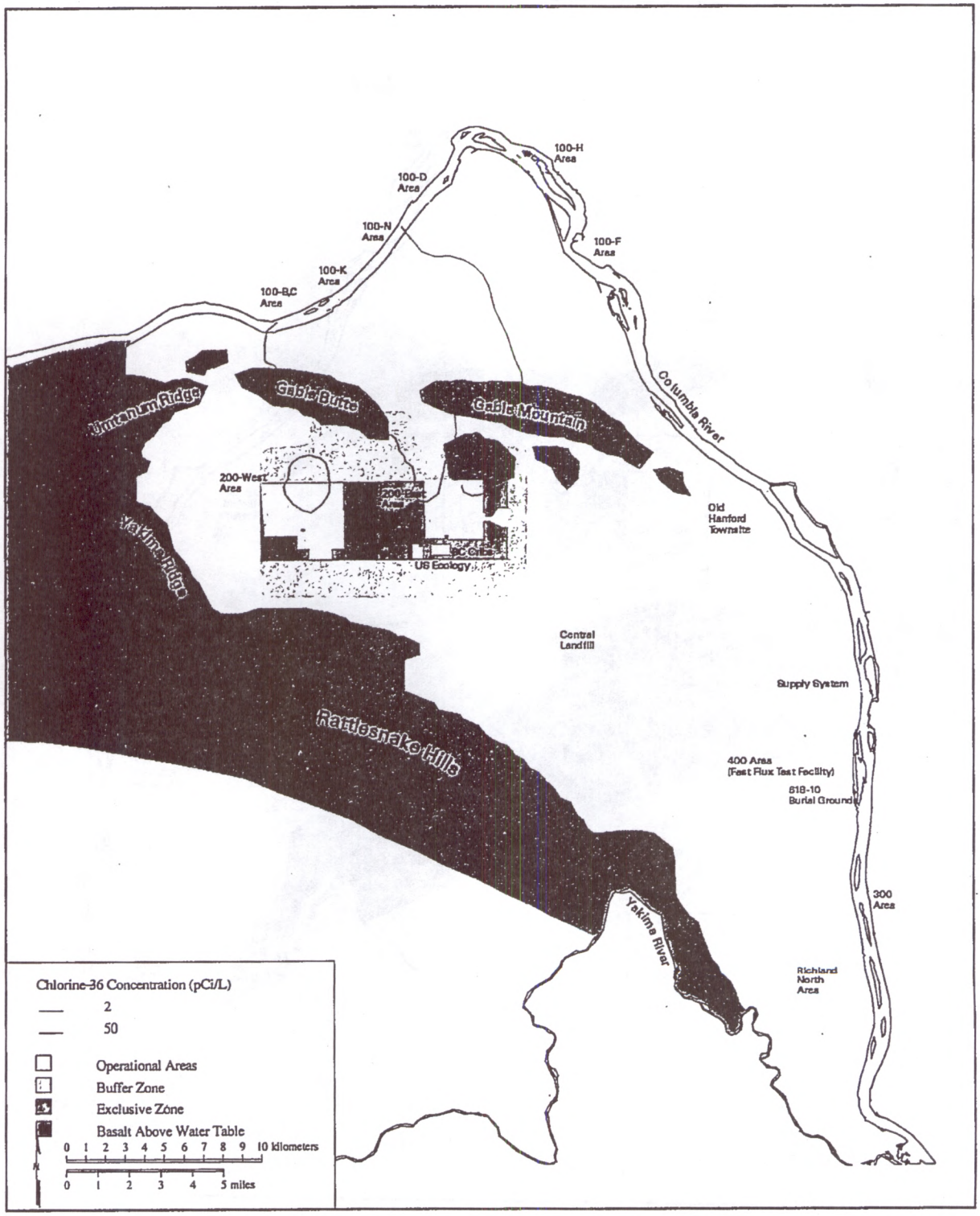
skw98051.eps December 23, 1997

Figure 4.29b. Predicted Distribution of Carbon-14 in the Unconfined Aquifer from All Sources in 2049



skw98039.eps December 23, 1997

Figure 4.30a. Predicted Distribution of Chlorine-36 in the Unconfined Aquifer from All Sources in 2019 (Time of Peak Concentration)



skw98052.eps December 23, 1997

Figure 4.30b. Predicted Distribution of Chlorine-36 in the Unconfined Aquifer from All Sources in 2049

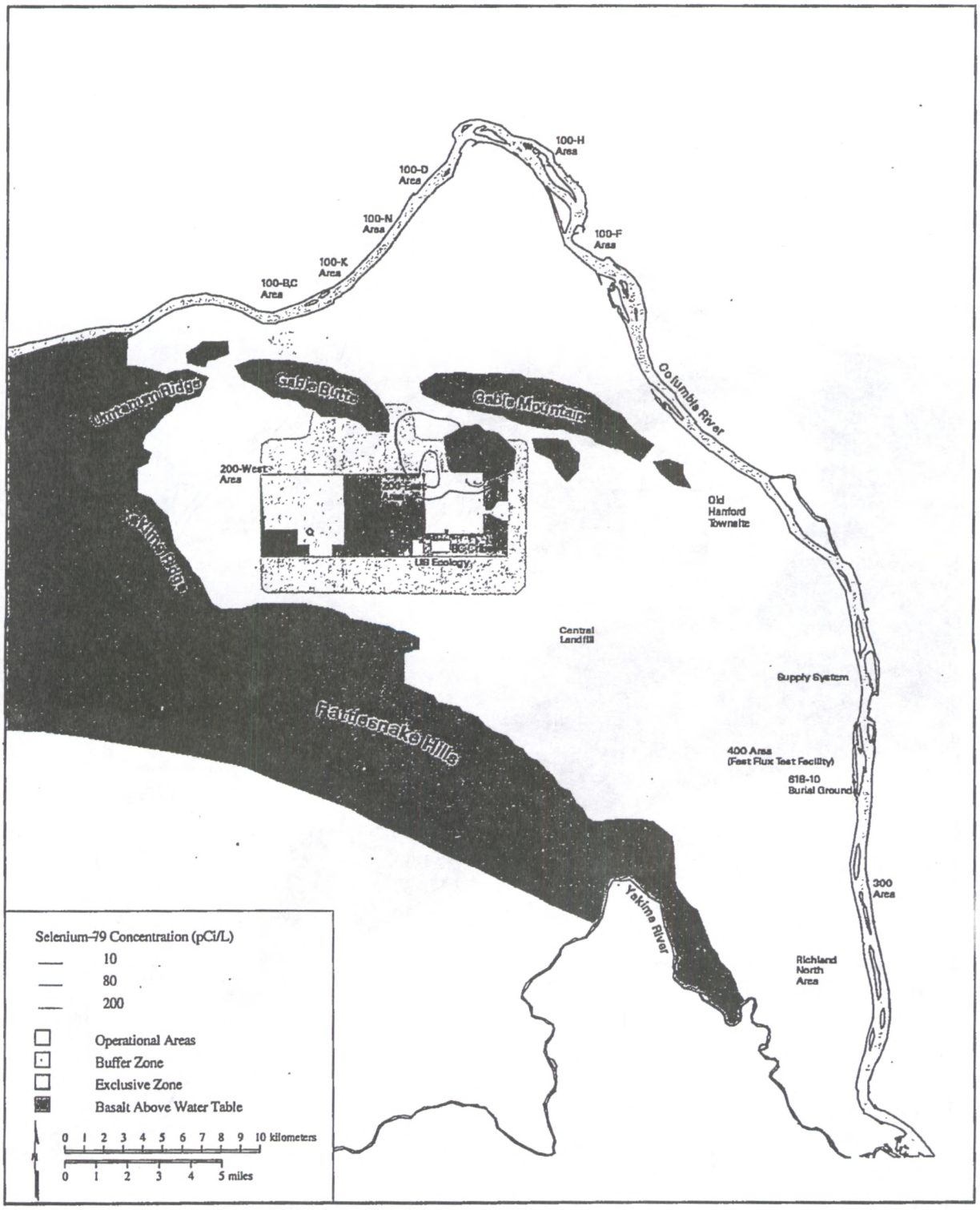


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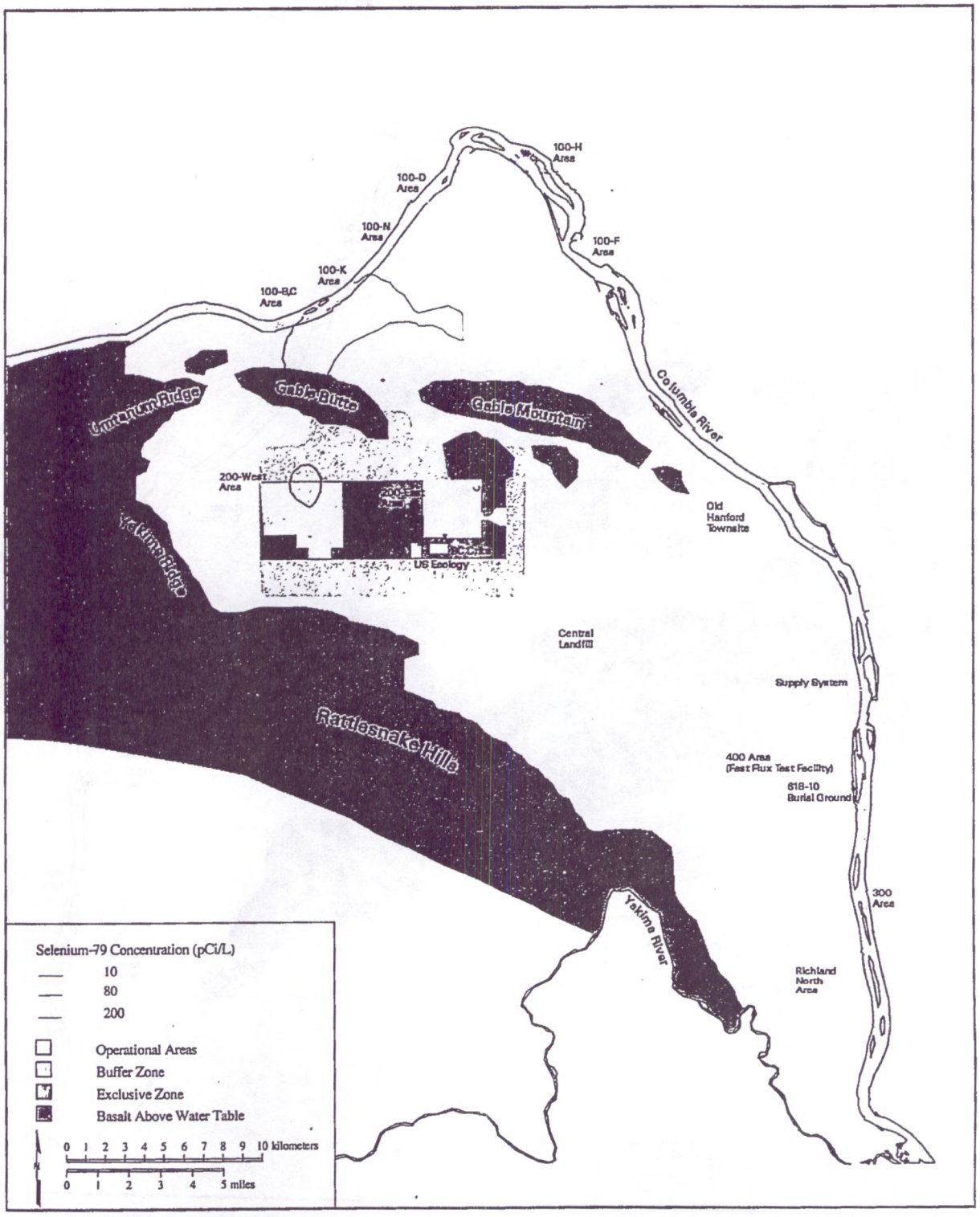
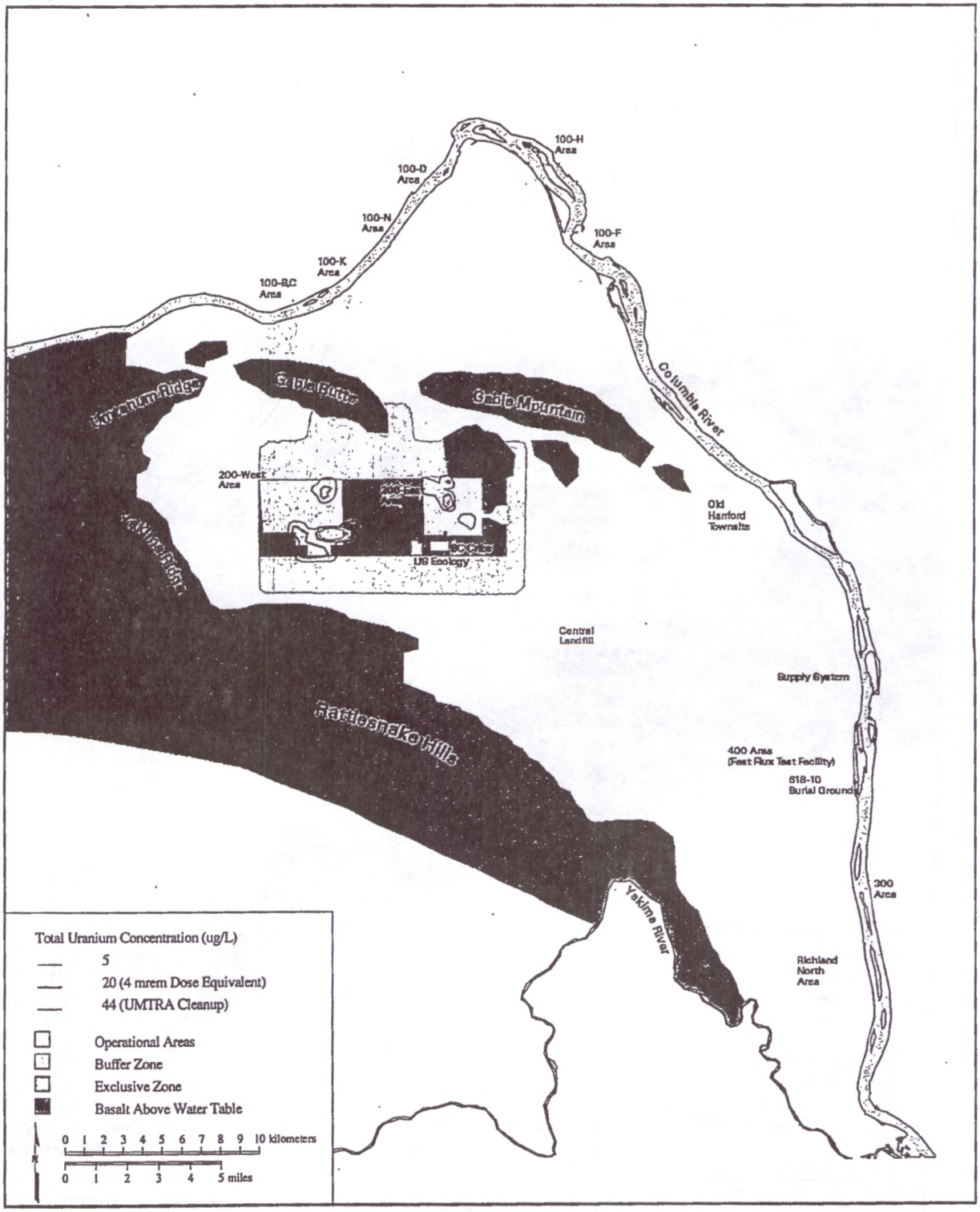
Figure 4.31a. Predicted Distribution of Selenium-79 in the Unconfined Aquifer from All Sources in 2005 (Time of Peak Concentration)



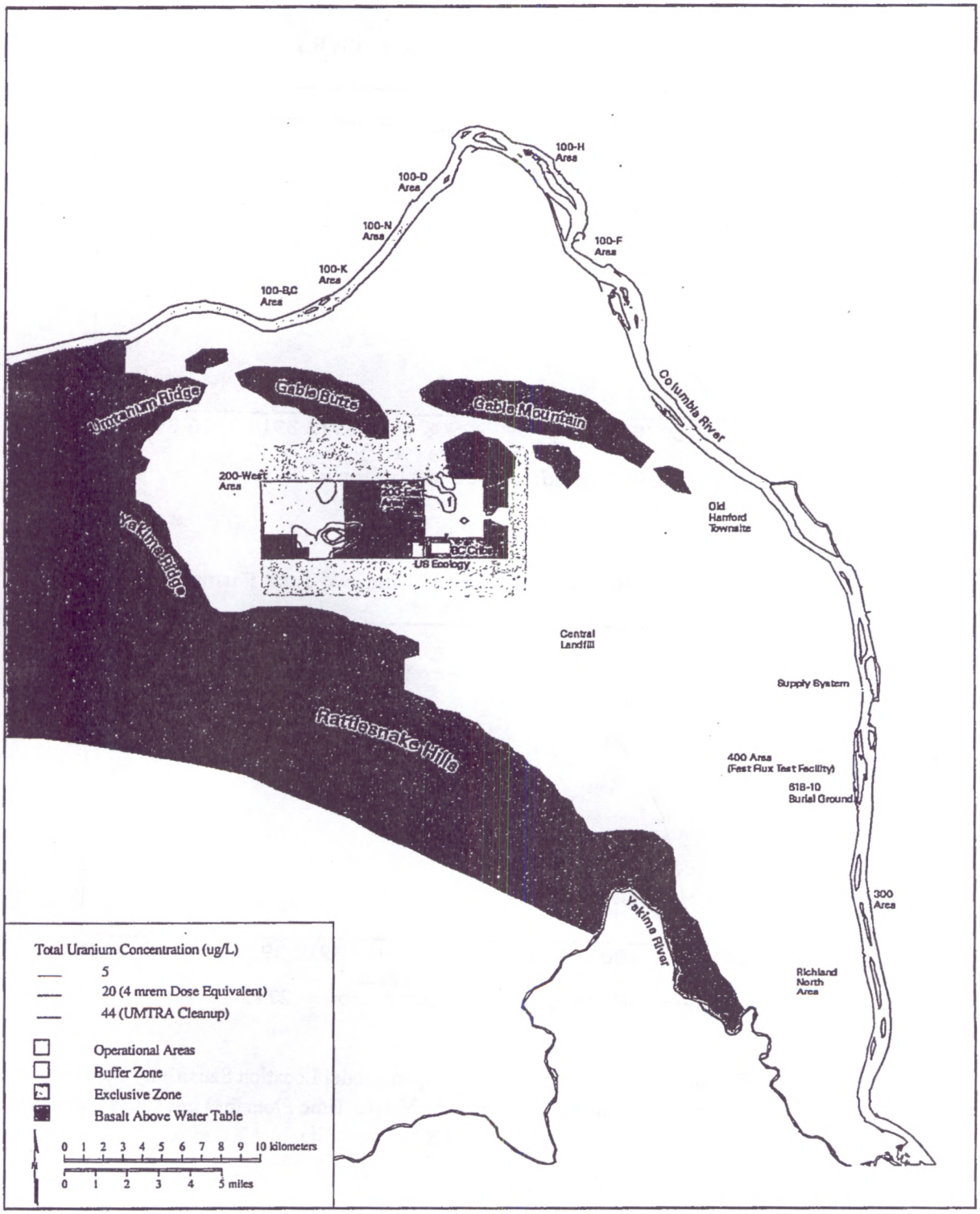
skw98053.eps December 23, 1997

Figure 4.31b. Predicted Distribution of Selenium-79 in the Unconfined Aquifer from All Sources in 2049



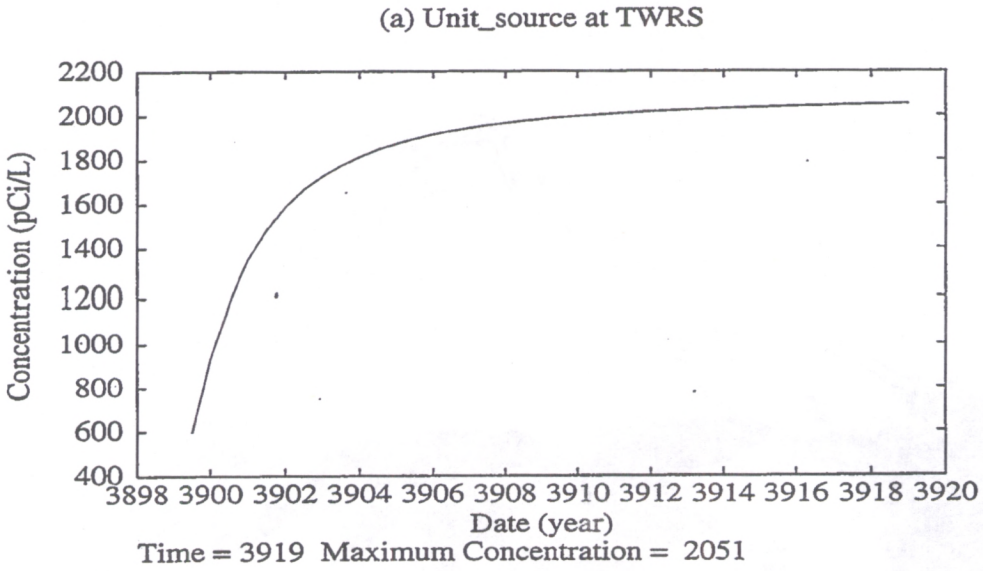
skw98048.eps December 23, 1997

Figure 4.32a. Distribution of Uranium (Total) in the Unconfined Aquifer from All Sources at 1996 (Time of Peak Concentration)



skw98057.eps December 23, 1997

Figure 4.32b. Predicted Distribution of Uranium (Total) in the Unconfined Aquifer from All Sources in 2049



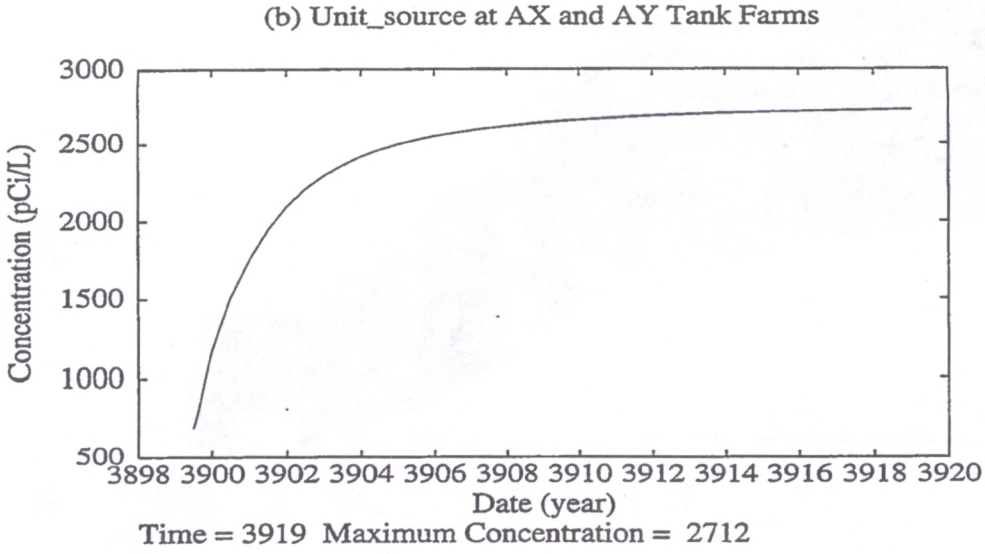
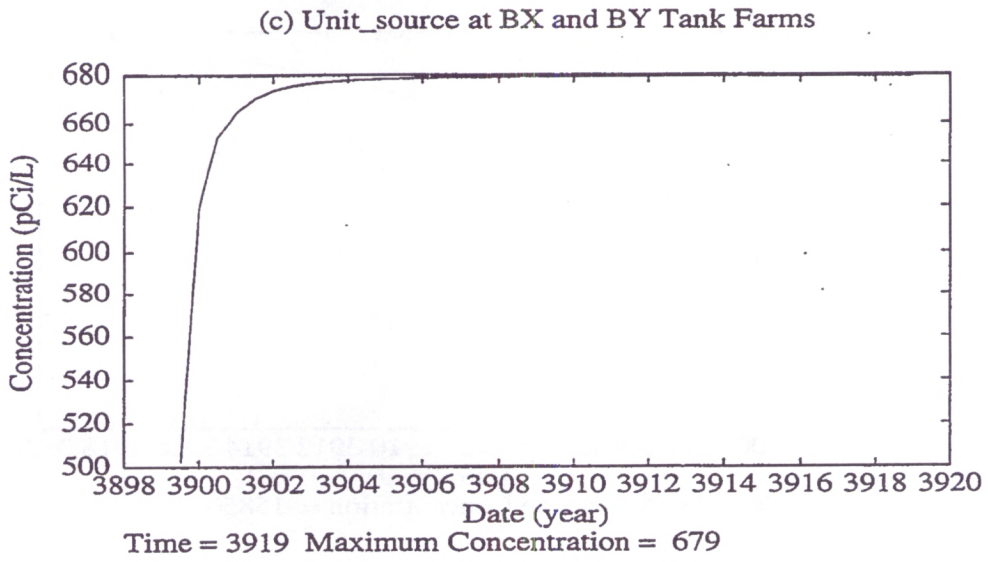


Figure 4.33. Results from a Series of Nine Transport Model Location Sensitivity Studies. Shown are Maximum Concentration Versus Time Plots for Unit Curie Sources at the Following Locations: a) TWRS Disposal Site, b) AX and AY Tank Farms.



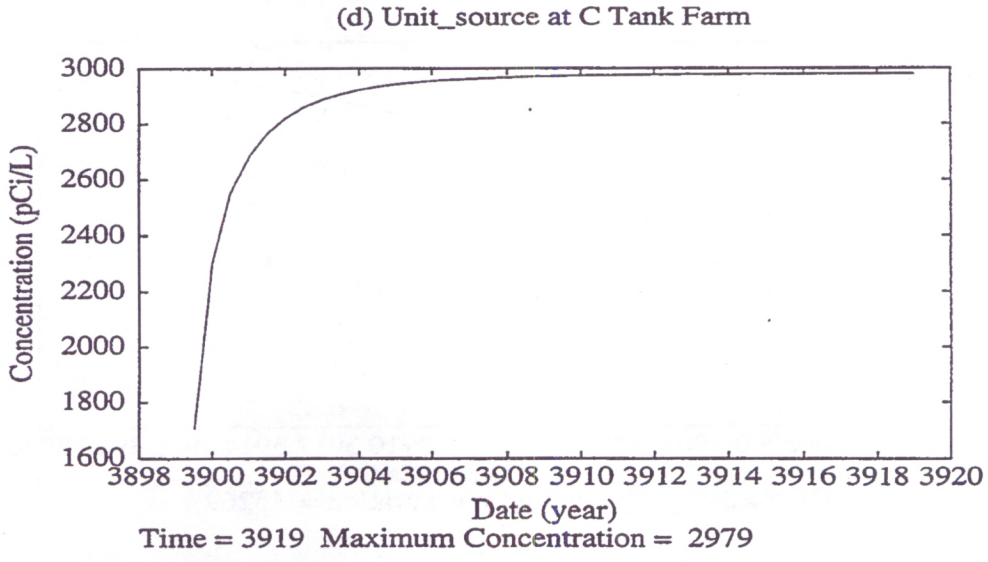
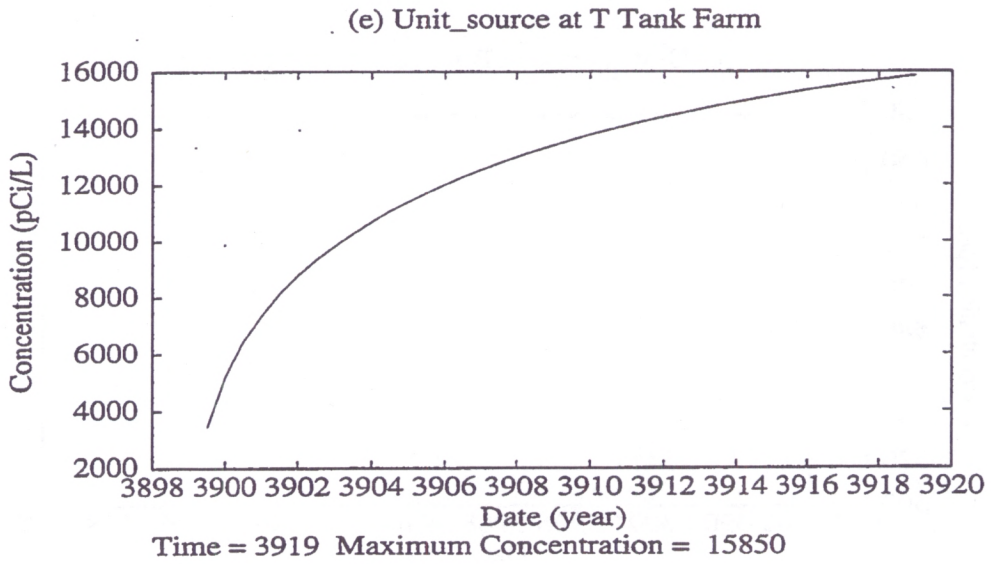


Figure 4.33. Results from a Series of Nine Transport Model Location Sensitivity Studies. Shown are Maximum Concentration Versus Time Plots for Unit Curie Sources at the Following Locations: c) BX and BY Tank Farms, d) C Tank Farm.



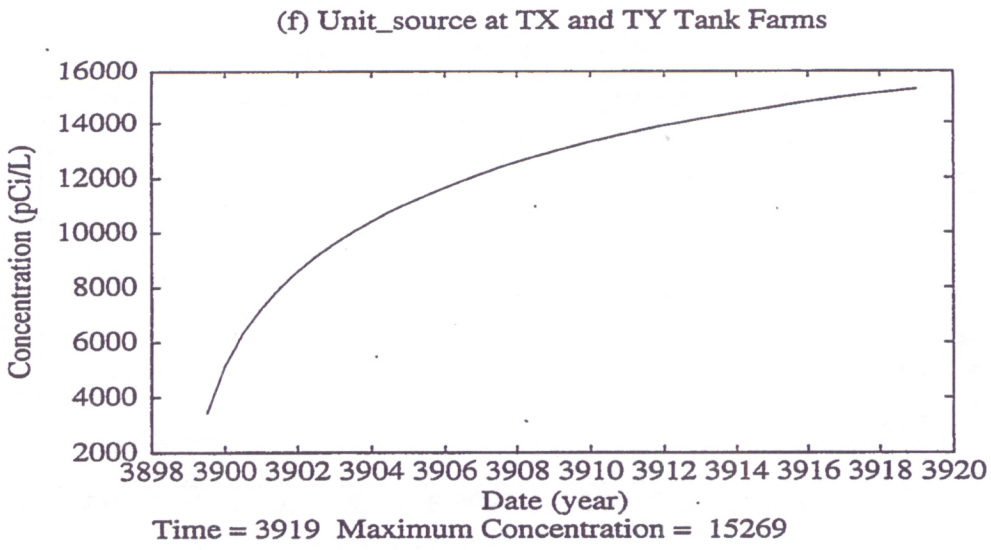


Figure 4.33. Results from a Series of Nine Transport Model Location Sensitivity Studies. Shown are Maximum Concentration Versus Time Plots for Unit Curie Sources at the Following Locations: e) T Tank Farm, f) TX and TY Tank Farms.

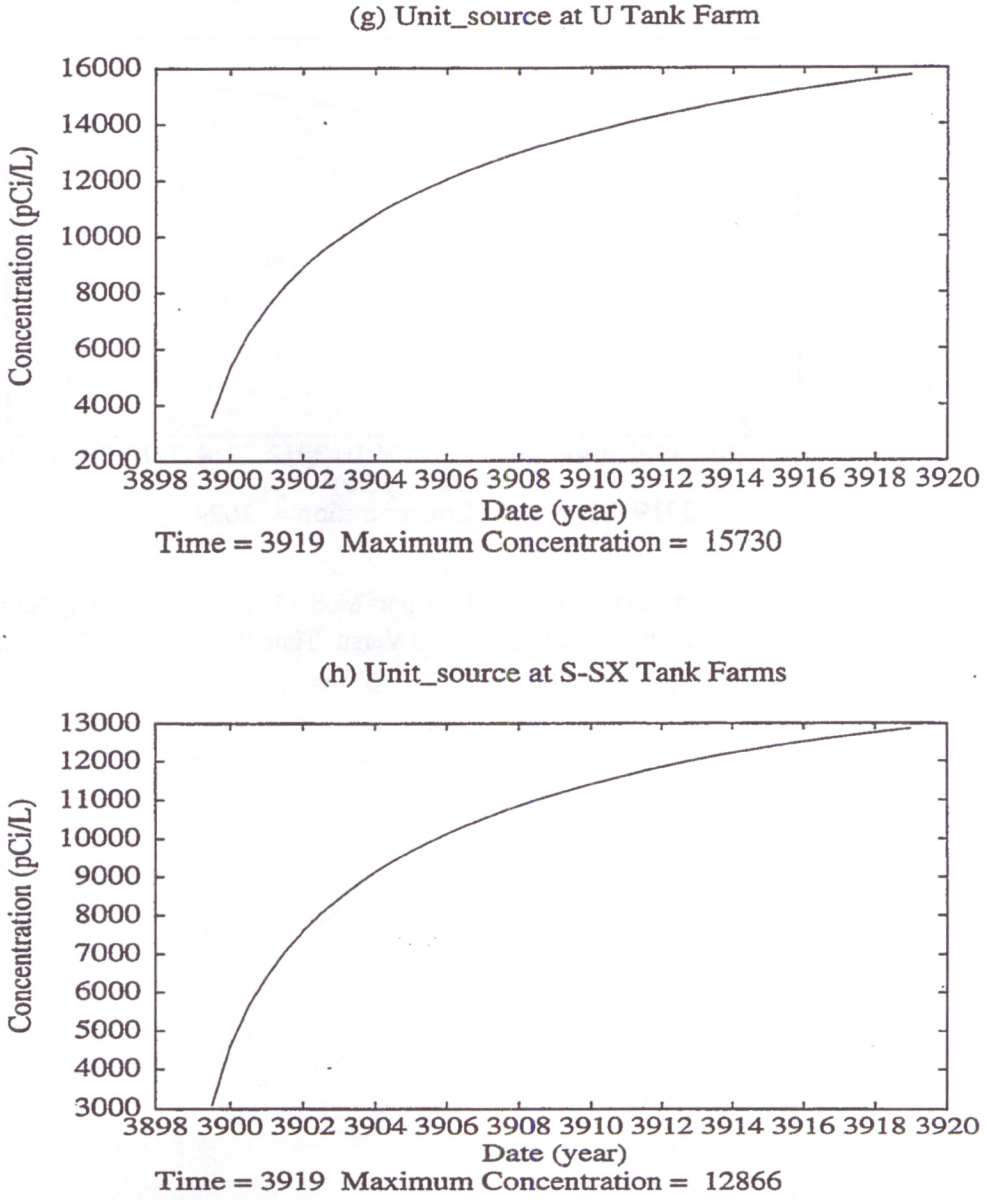


Figure 4.33. Results from a Series of Nine Transport Model Location Sensitivity Studies. Shown are Maximum Concentration Versus Time Plots for Unit Curie Sources at the Following Locations: g) U Tank Farm, h) S and SX Tank Farms.

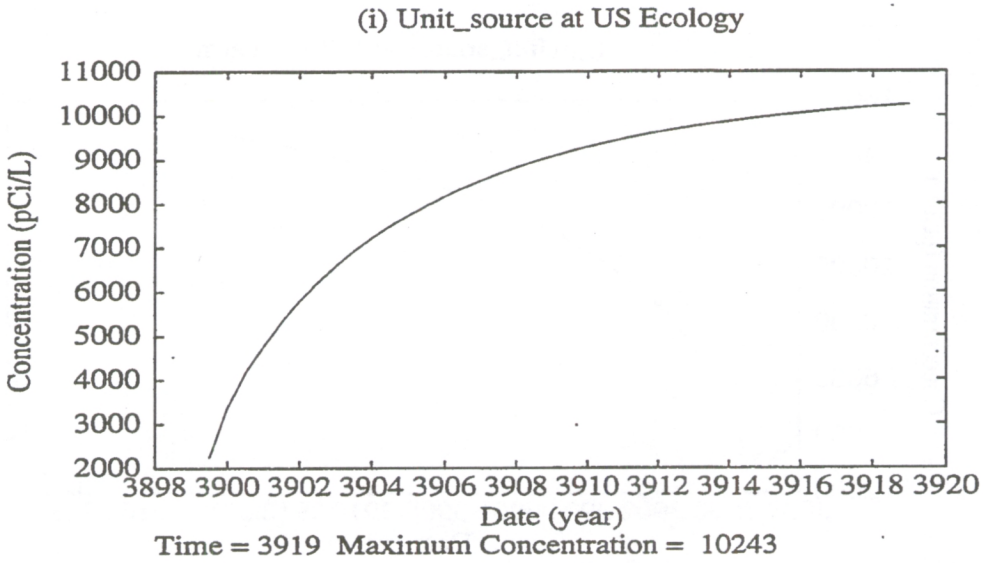


Figure 4.33. Results from a Series of Nine Transport Model Location Sensitivity Studies. Shown are Maximum Concentration Versus Time Plots for Unit Curie Sources at the Following Location: i) U.S. Ecology Site.

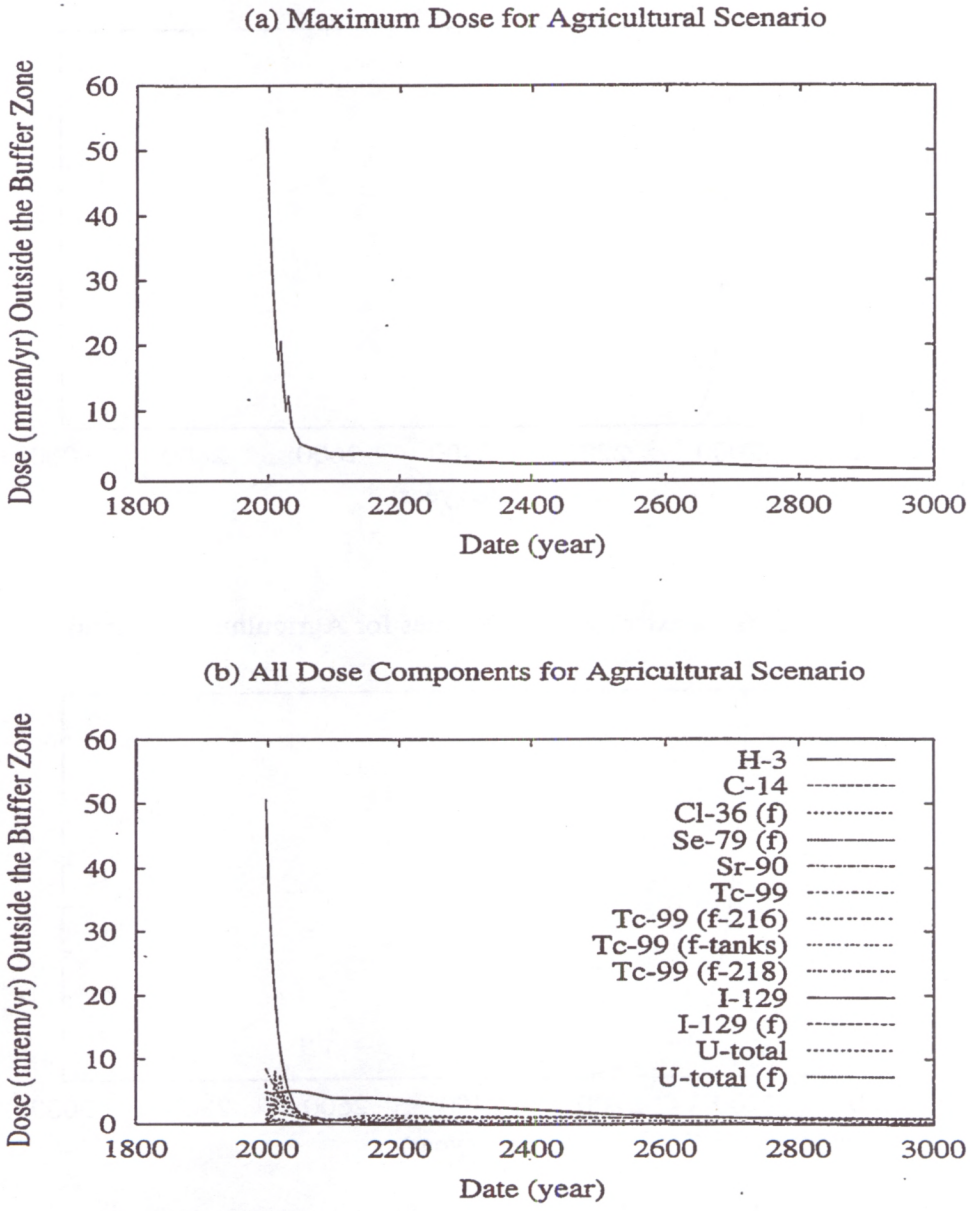
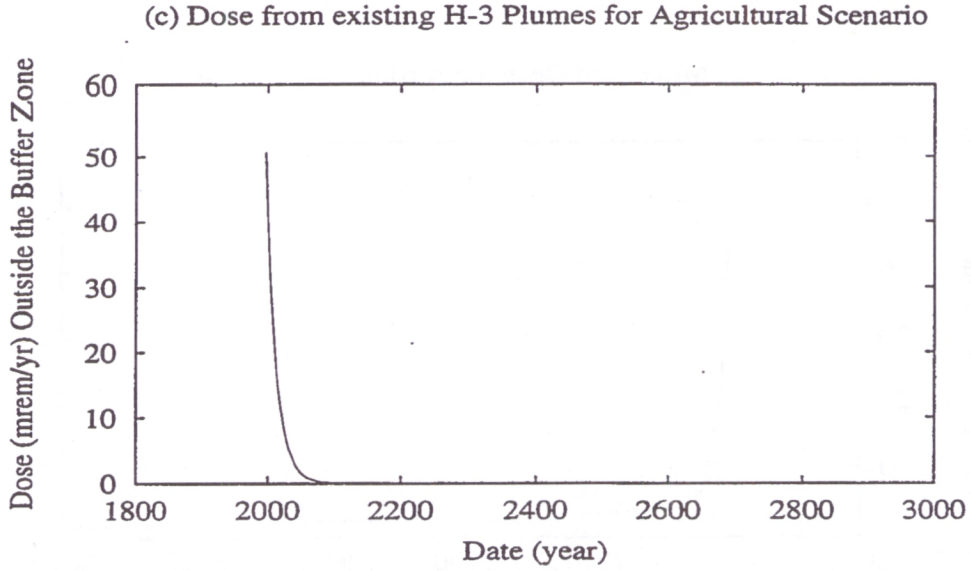


Figure 4.34. Maximum Dose Versus Time Outside the Buffer Zone for the Agricultural Scenario and the Dose the Various Radionuclides/Sources Contribute. Shown are: a) Maximum Dose, b) All Contributions on Same Scale.



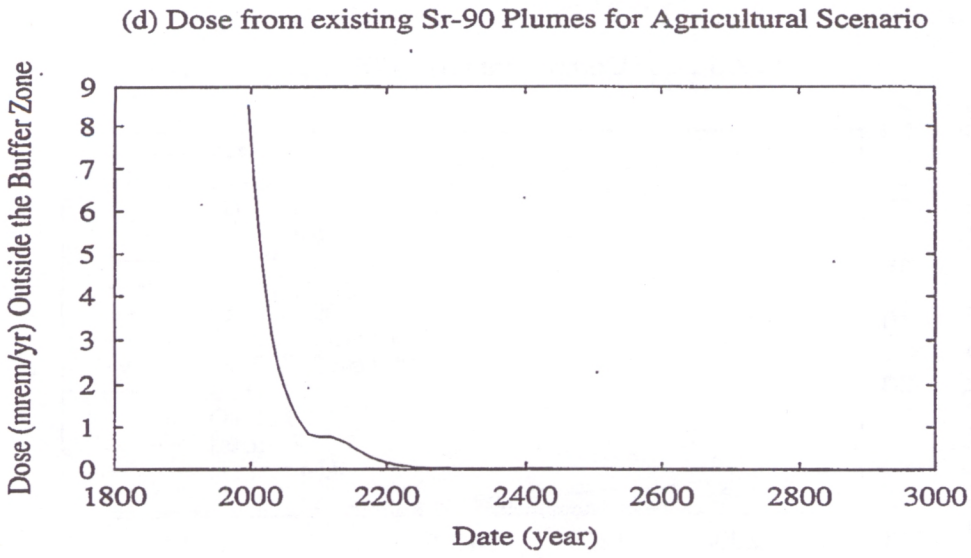
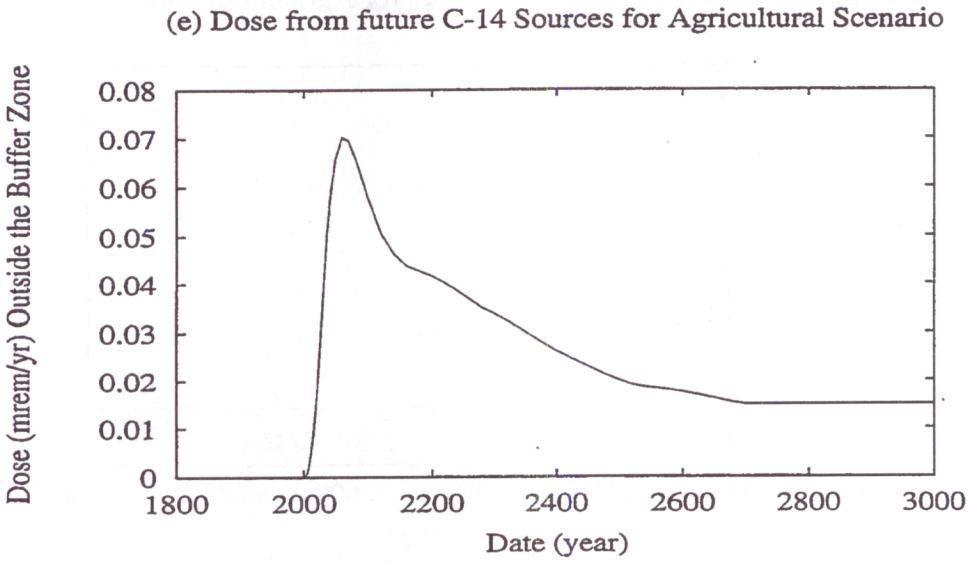


Figure 4.34. Maximum Dose Versus Time Outside the Buffer Zone for the Agricultural Scenario and the Dose the Various Radionuclides/Sources Contribute. Shown are: c) Tritium Contribution, d) Strontium-90.



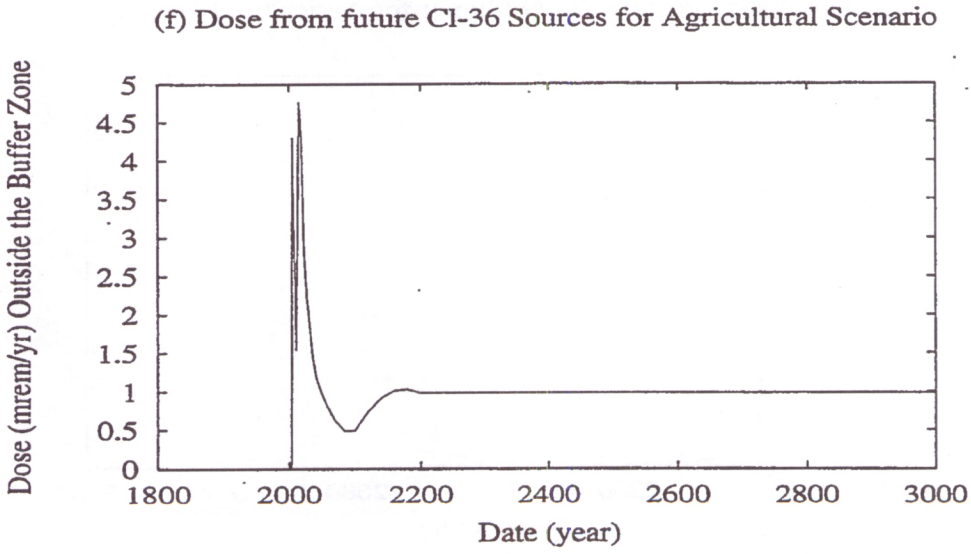
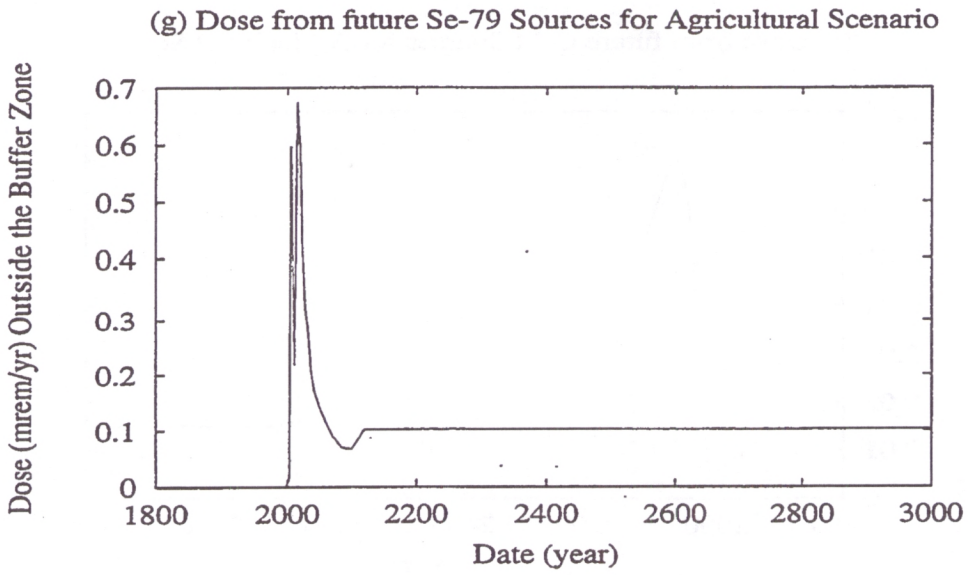


Figure 4.34. Maximum Dose Versus Time Outside the Buffer Zone for the Agricultural Scenario and the Dose the Various Radionuclides/Sources Contribute. Shown are: e) Carbon-14, f) Chlorine-36.



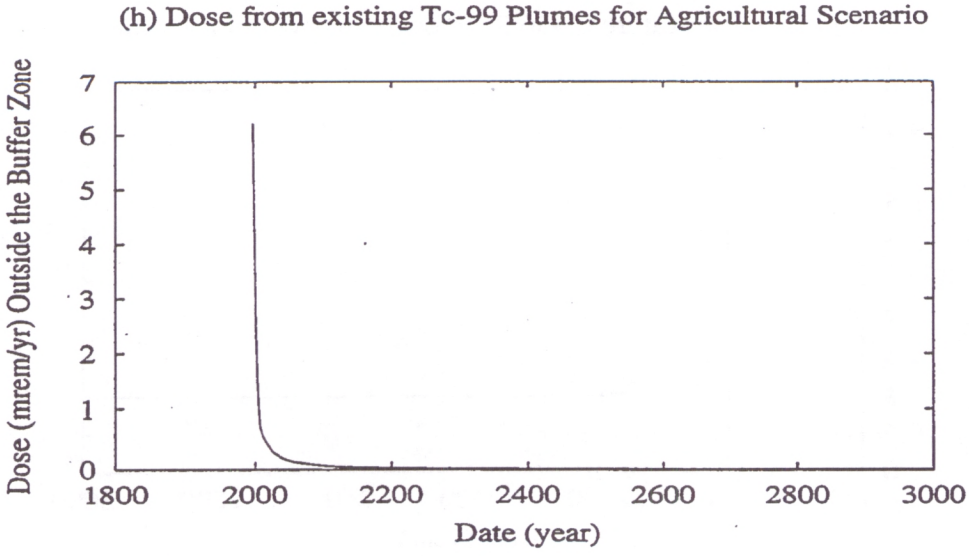
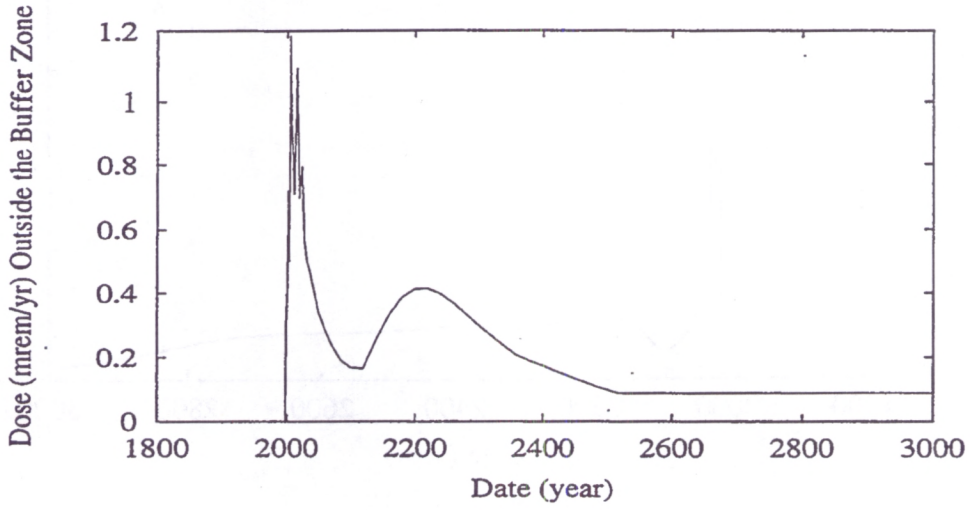


Figure 4.34. Maximum Dose Versus Time Outside the Buffer Zone for the Agricultural Scenario and the Dose the Various Radionuclides/Sources Contribute. Shown are: g) Selenium-79, h) Technetium-99 Contribution from Existing Plumes.

(i) Dose from future Tc-99 from 216 Sources for Agricultural Scenario



(j) Dose from future Tc-99 from Tank Sources for Agricultural Scenario

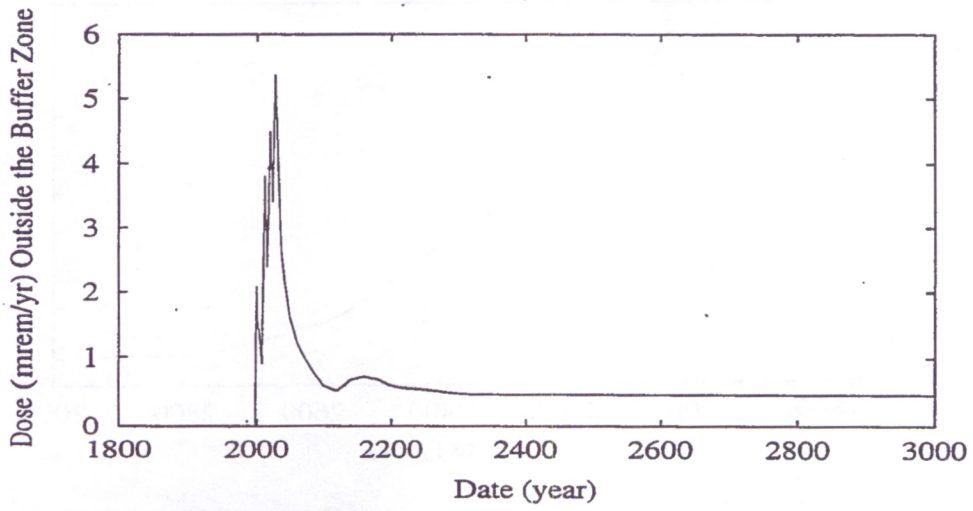
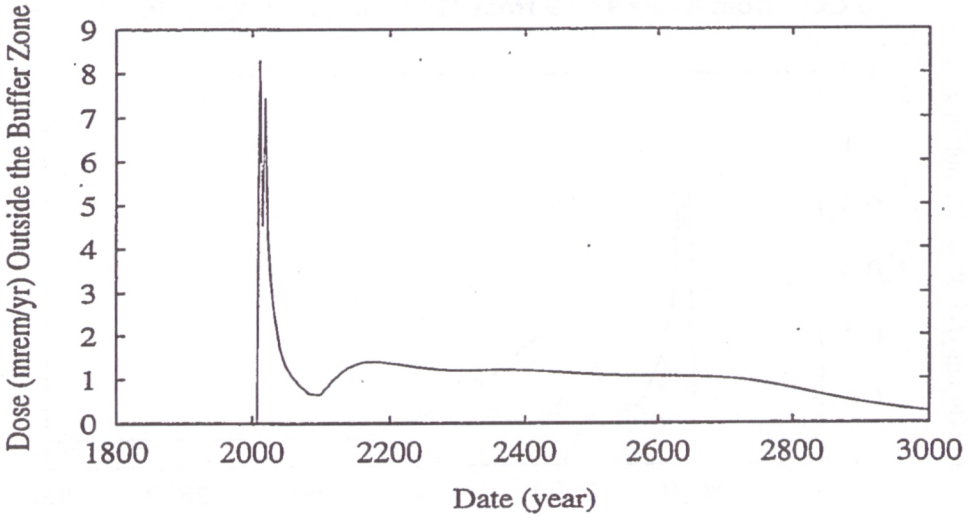


Figure 4.34. Maximum Dose Versus Time Outside the Buffer Zone for the Agricultural Scenario and the Dose the Various Radionuclides/Sources Contribute. Shown are: i) Technetium-99 Contribution from Liquid Discharges, j) Technetium-99 Contribution from Tank Sources.

(k) Dose from future Tc-99 from 218 Sources for Agricultural Scenario



(l) Dose from existing I-129 Plumes for Agricultural Scenario

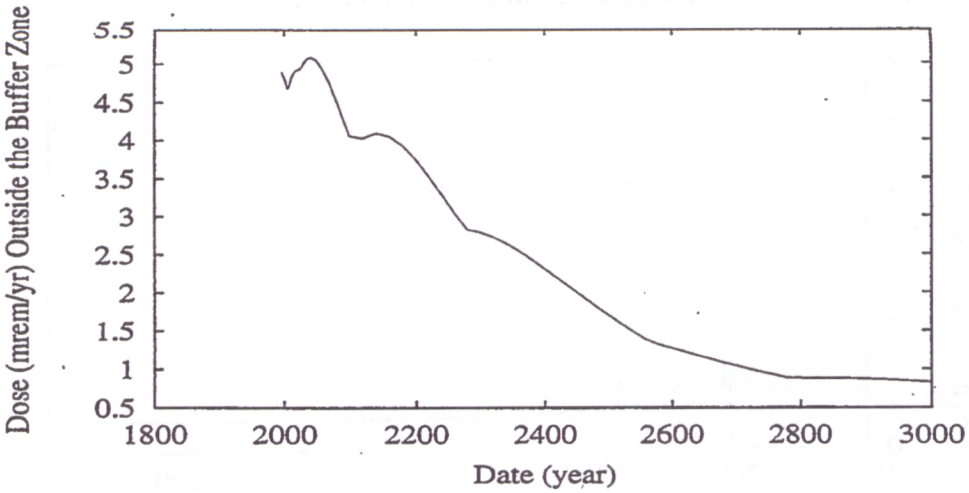


Figure 4.34. Maximum Dose Versus Time Outside the Buffer Zone for the Agricultural Scenario and the Dose the Various Radionuclides/Sources Contribute. Shown are: k) Technetium-99 Contribution from Solid Waste Sources, l) Iodine-129 Contribution from Existing Plumes.

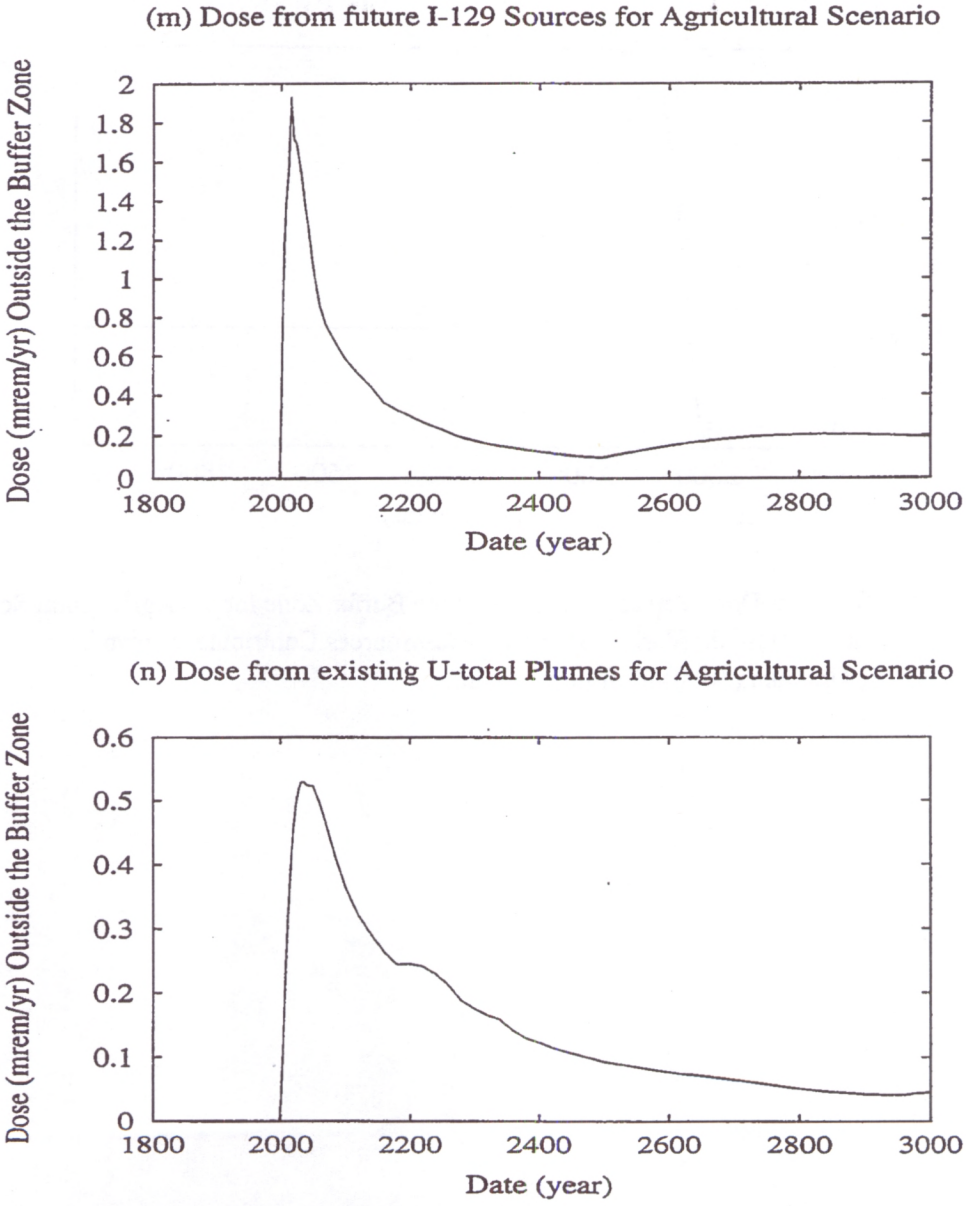


Figure 4.34. Maximum Dose Versus Time Outside the Buffer Zone for the Agricultural Scenario and the Dose the Various Radionuclides/Sources Contribute. Shown are: m) Iodine-129 Contribution from Future Sources, n) Uranium (Total) from Existing Plumes.

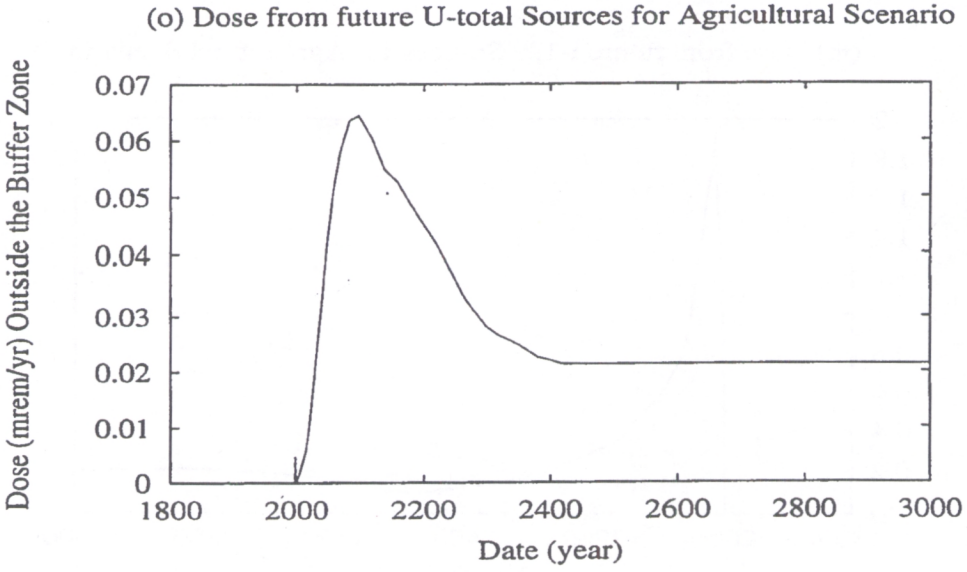


Figure 4.34. Maximum Dose Versus Time Outside the Buffer Zone for the Agricultural Scenario and the Dose the Various Radionuclides/Sources Contribute. Shown is:
o) Uranium (Total) from Future Sources.

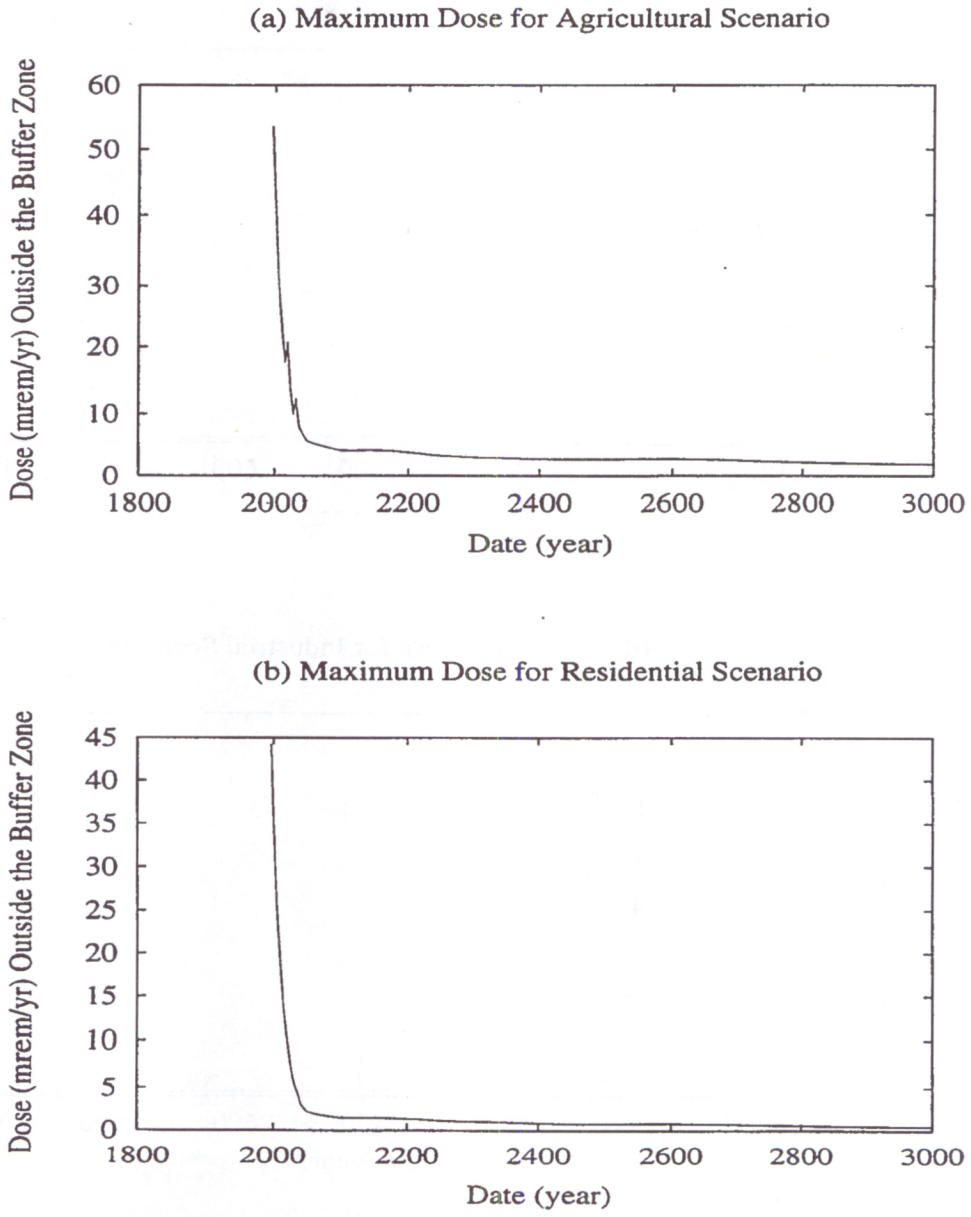


Figure 4.35. Composite Dose Outside the Buffer Zone from All Radionuclides and All Sources Modeled Versus Time for the a) Agricultural Scenario and b) Residential Scenario.

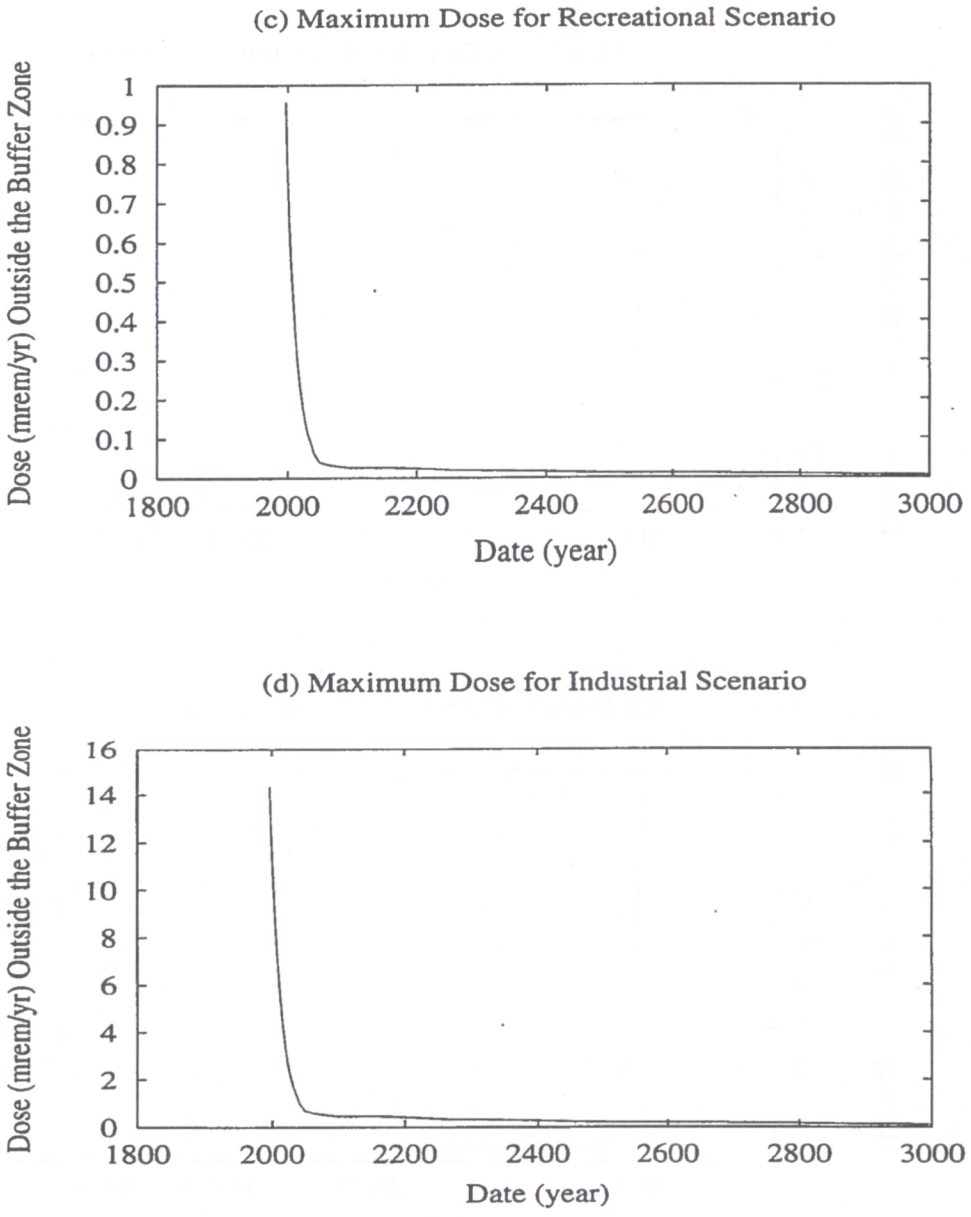


Figure 4.35. Composite Dose Outside the Buffer Zone from All Radionuclides and All Sources Modeled Versus Time for the c) Recreational Scenario and d) Industrial Exposure Scenario.

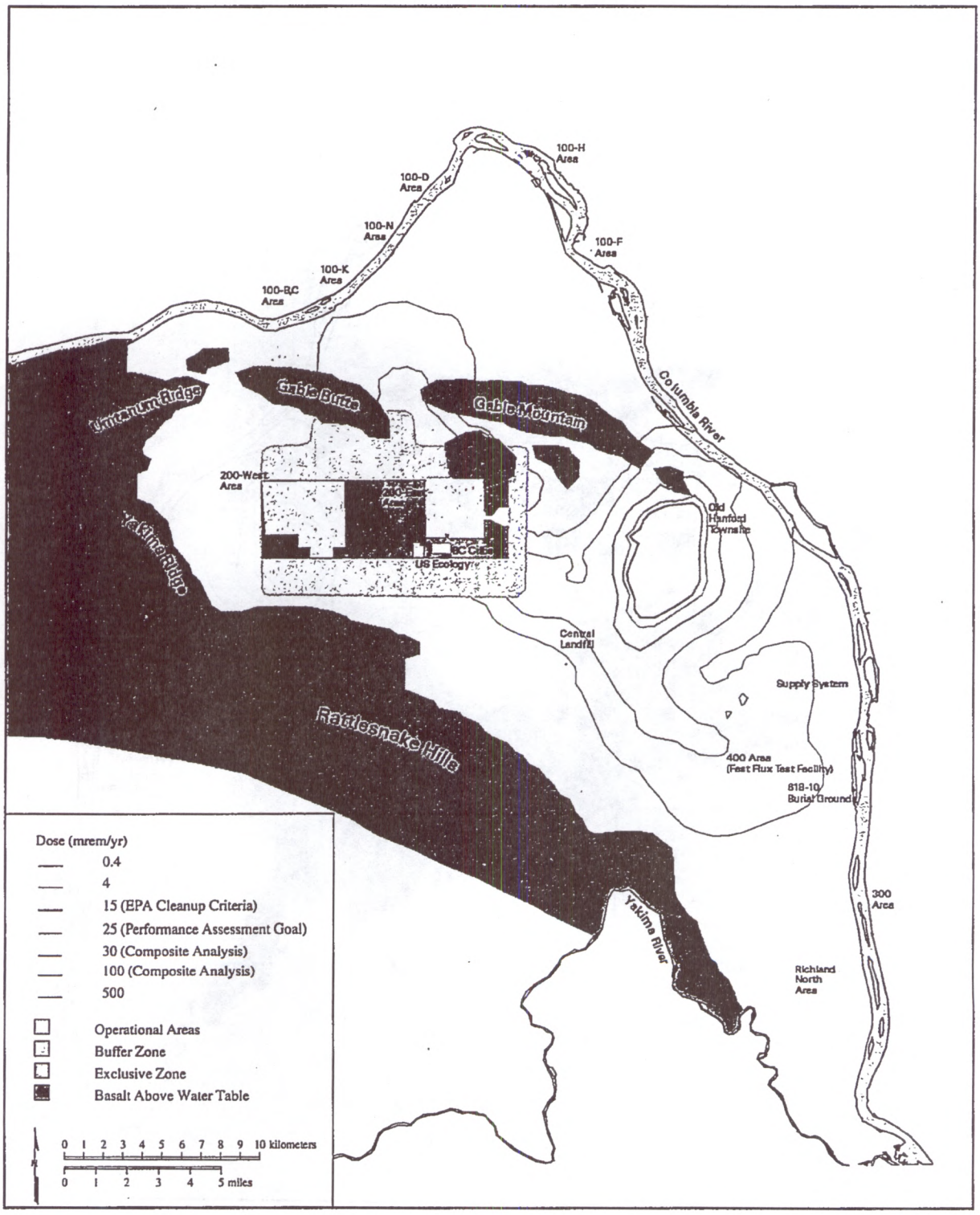


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Figure 4.36. Predicted Distribution of Composite Dose for the Agricultural Exposure Scenario in 1996

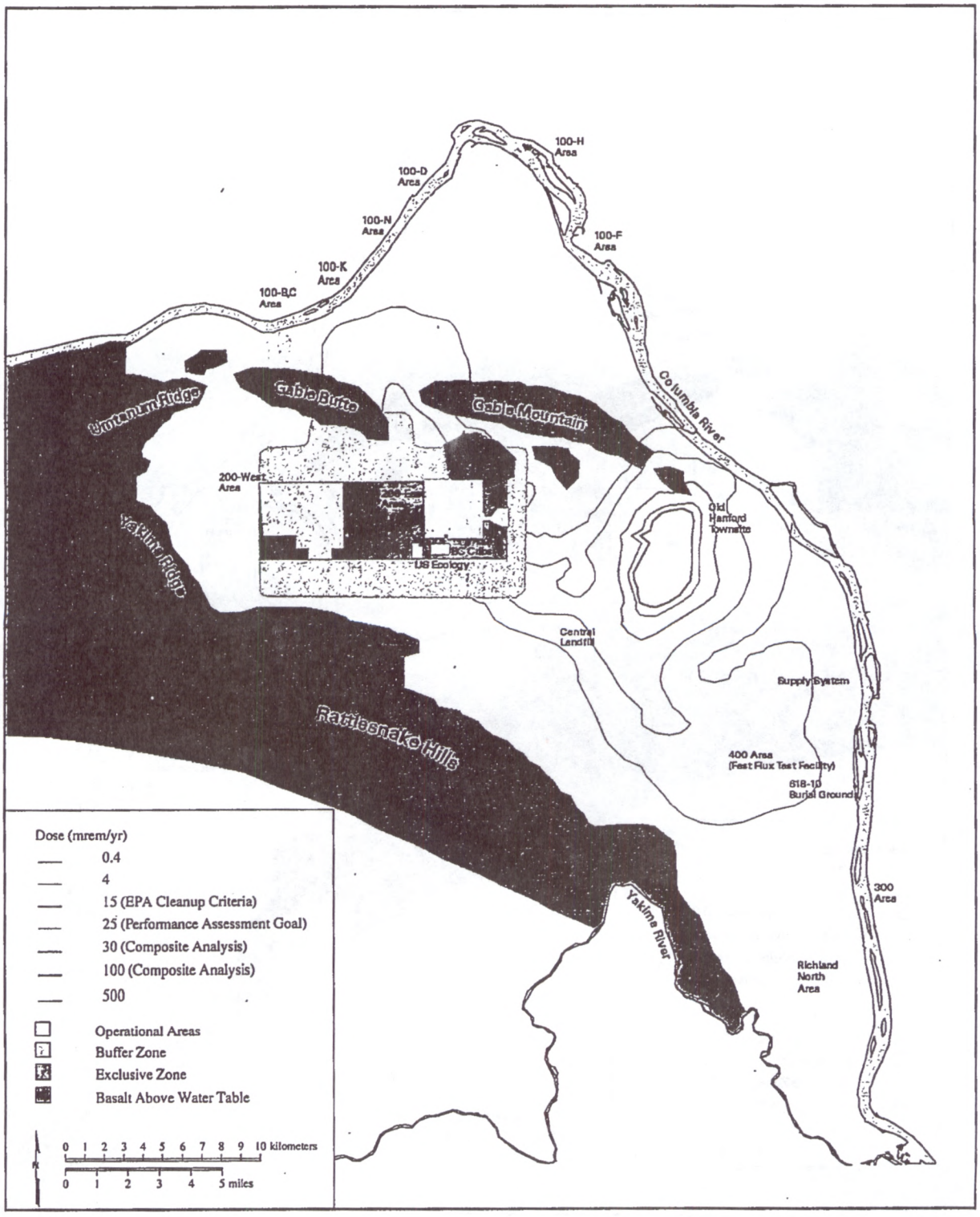

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Figure 4.37. Predicted Distribution of Composite Dose for the Residential Exposure Scenario in 1996

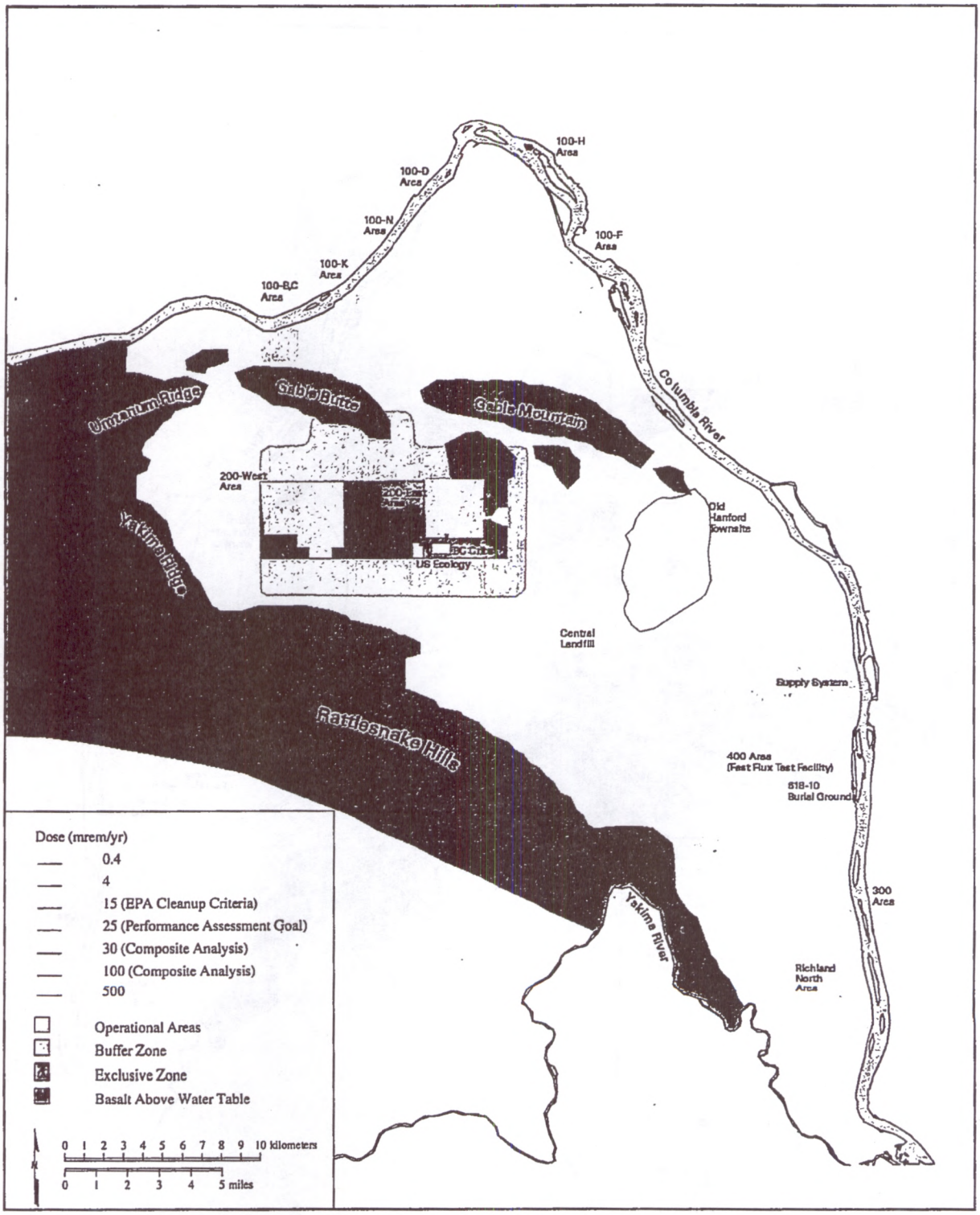

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Figure 4.38. Predicted Distribution of Composite Dose for the Recreational Exposure Scenario in 1996

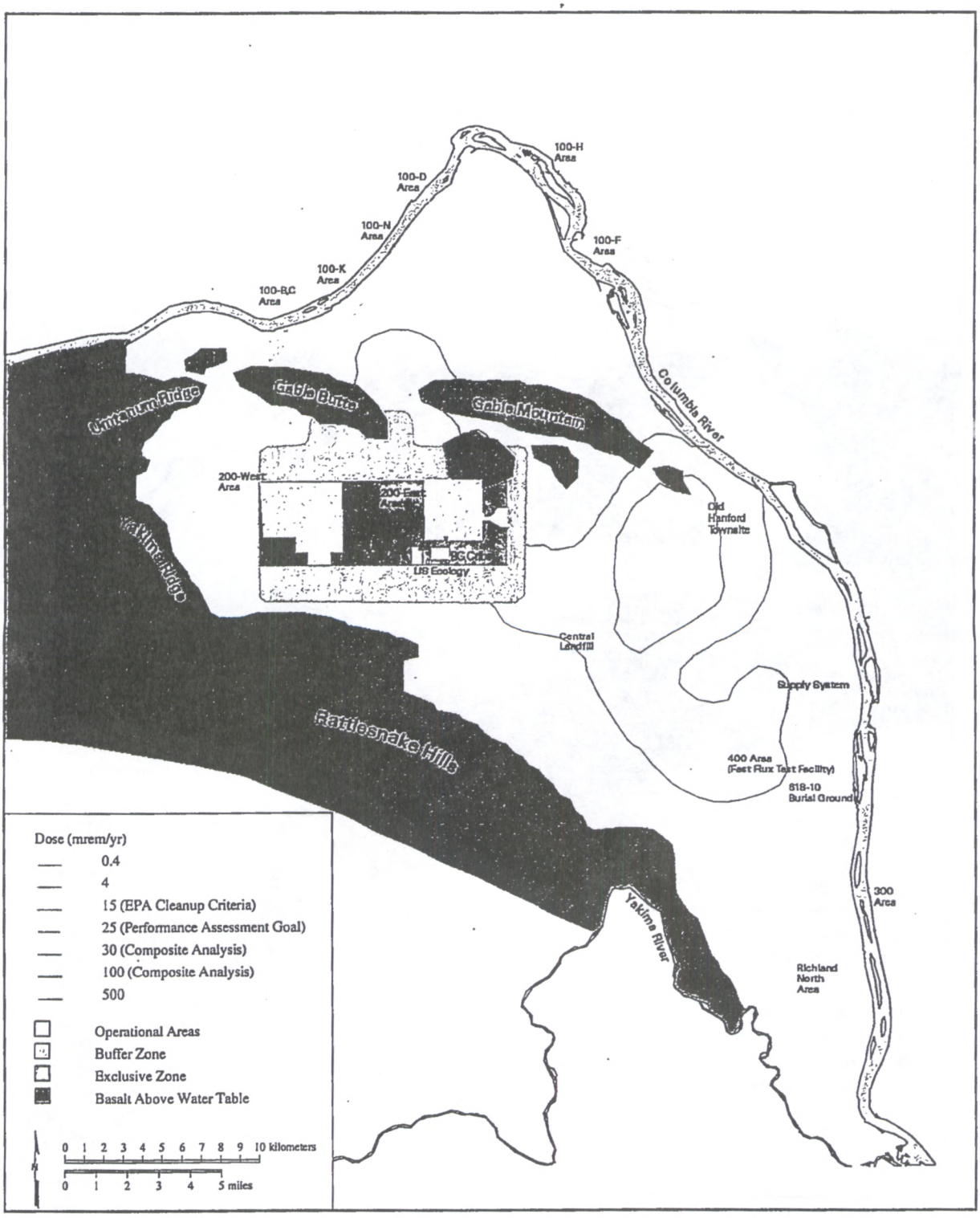

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Figure 4.39. Predicted Distribution of Composite Dose for the Industrial Exposure Scenario in 1996

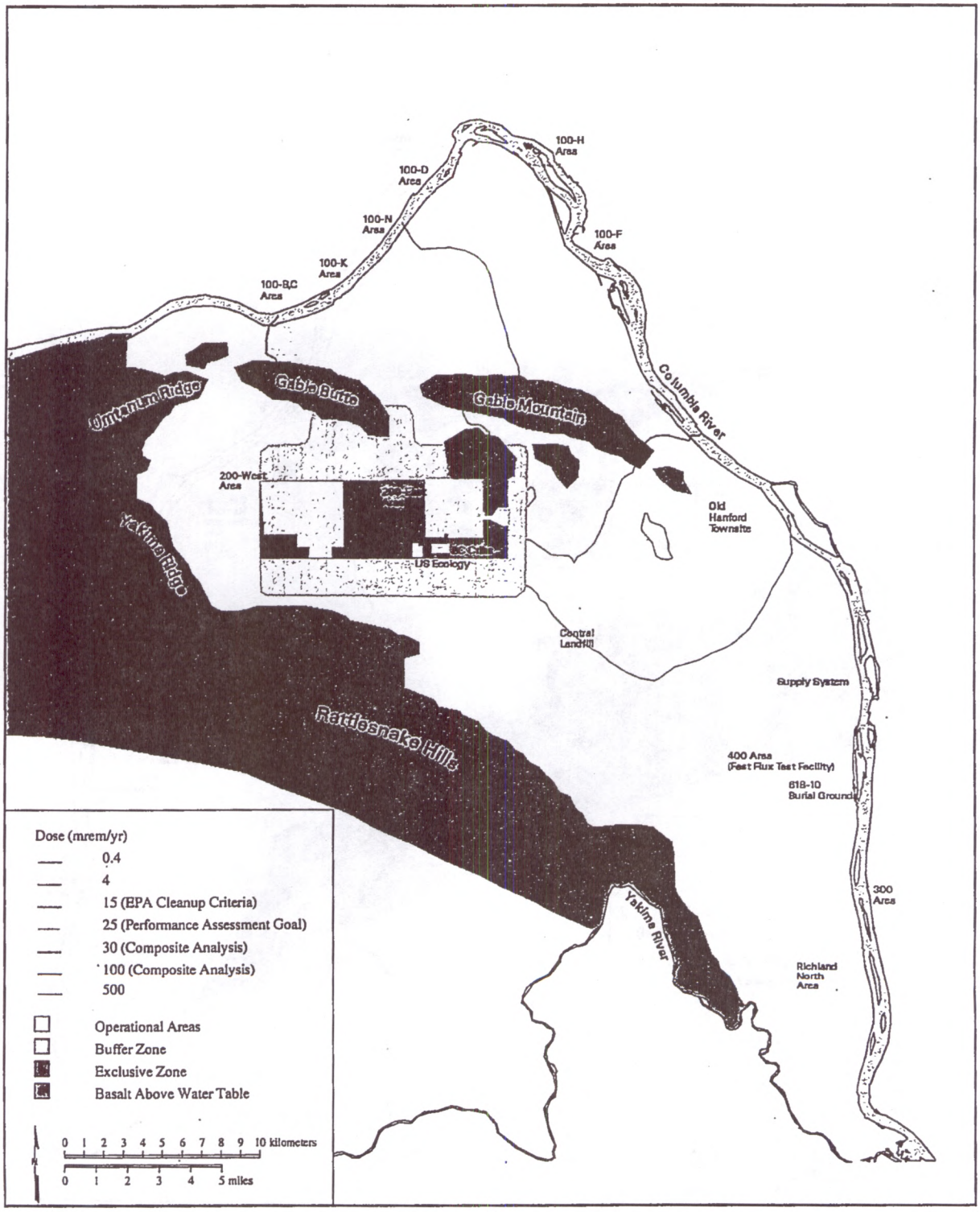

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Figure 4.40. Predicted Distribution of Composite Dose for the Agricultural Exposure Scenario in 2049

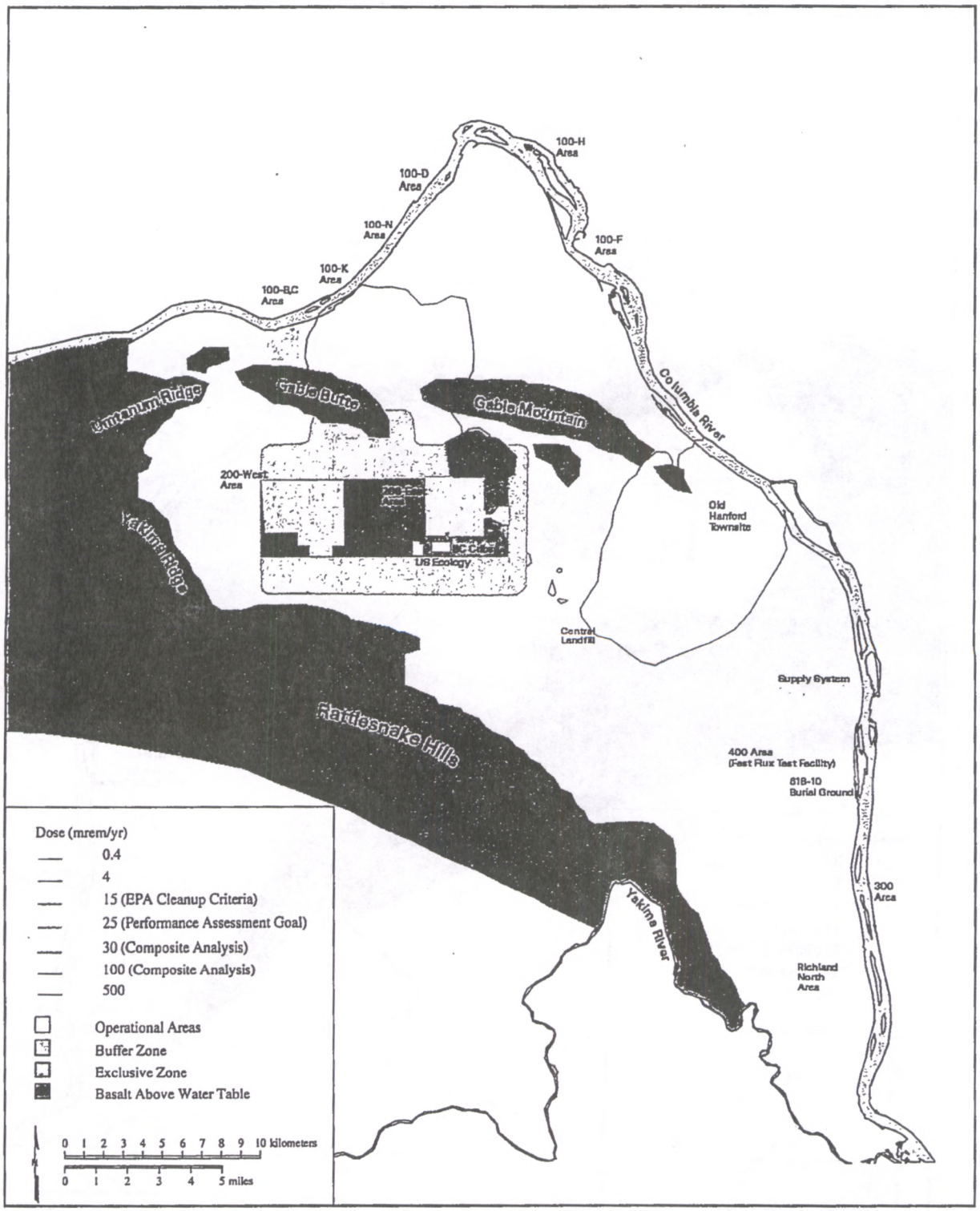

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Figure 4.41. Predicted Distribution of Composite Dose for the Residential Exposure Scenario in 2049

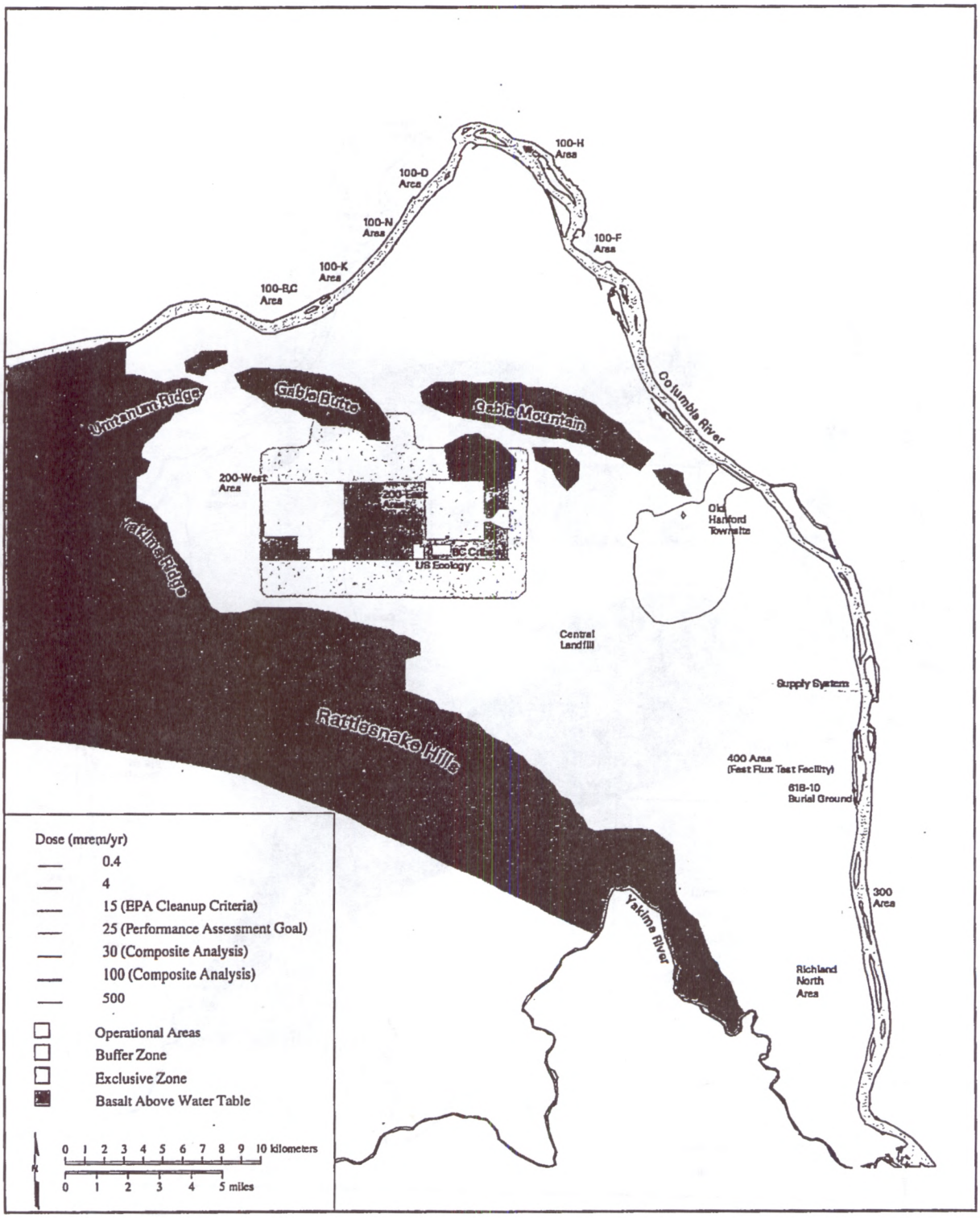

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Figure 4.42. Predicted Distribution of Composite Dose for the Industrial Exposure Scenario in 2049

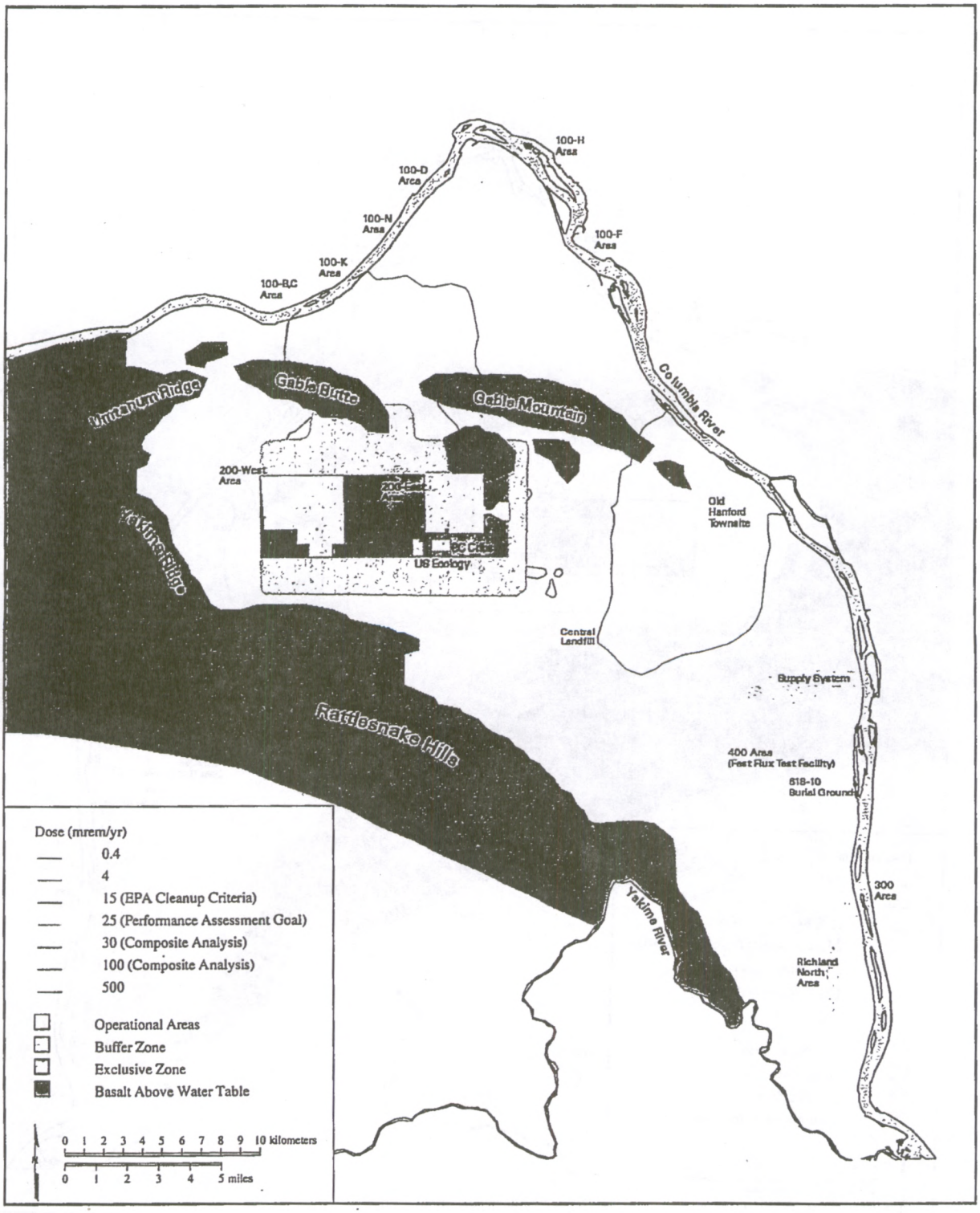

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Figure 4.43. Predicted Distribution of Composite Dose for the Agricultural Exposure Scenario in 2159

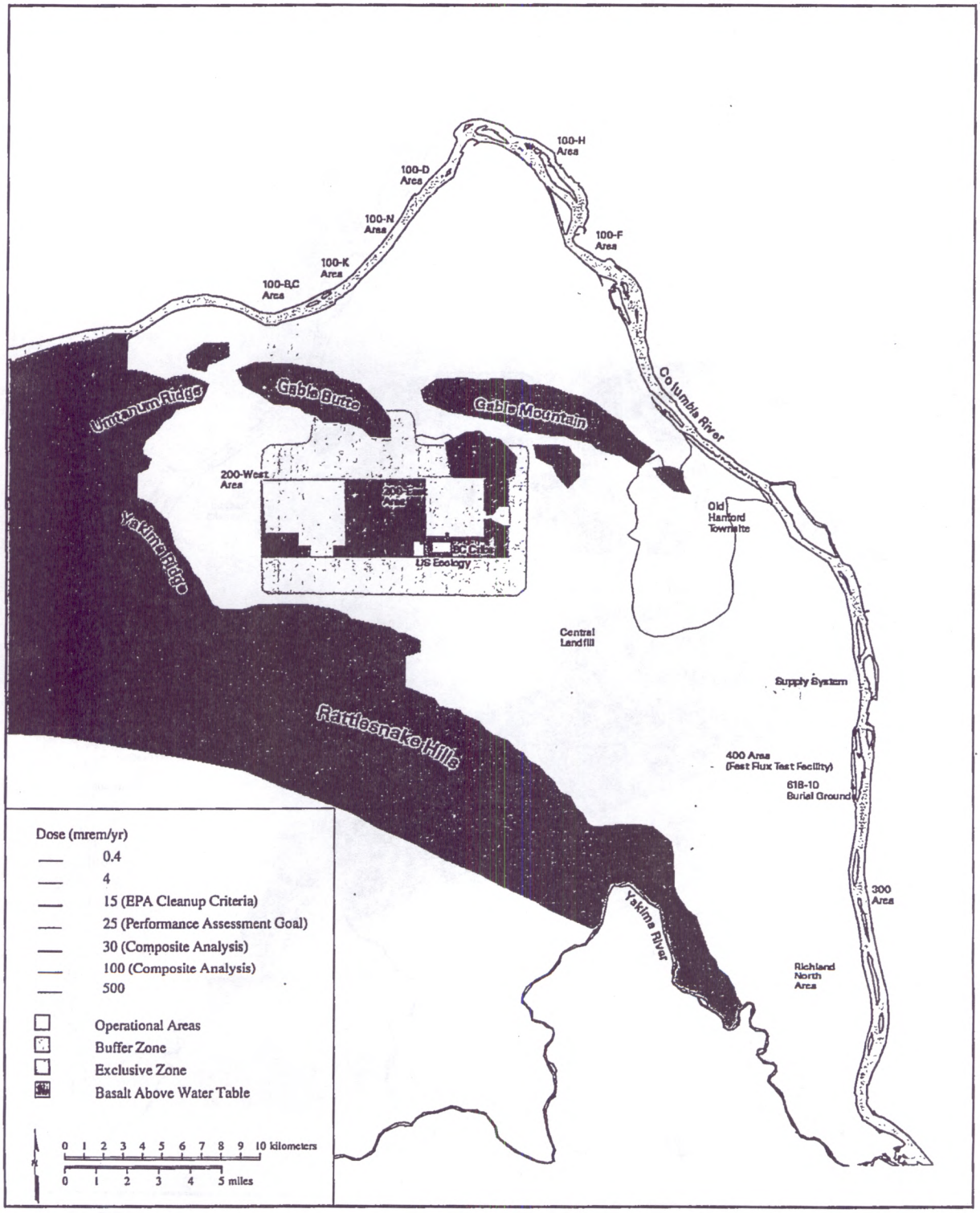

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Figure 4.44. Predicted Distribution of Composite Dose for the Residential Exposure Scenario in 2159

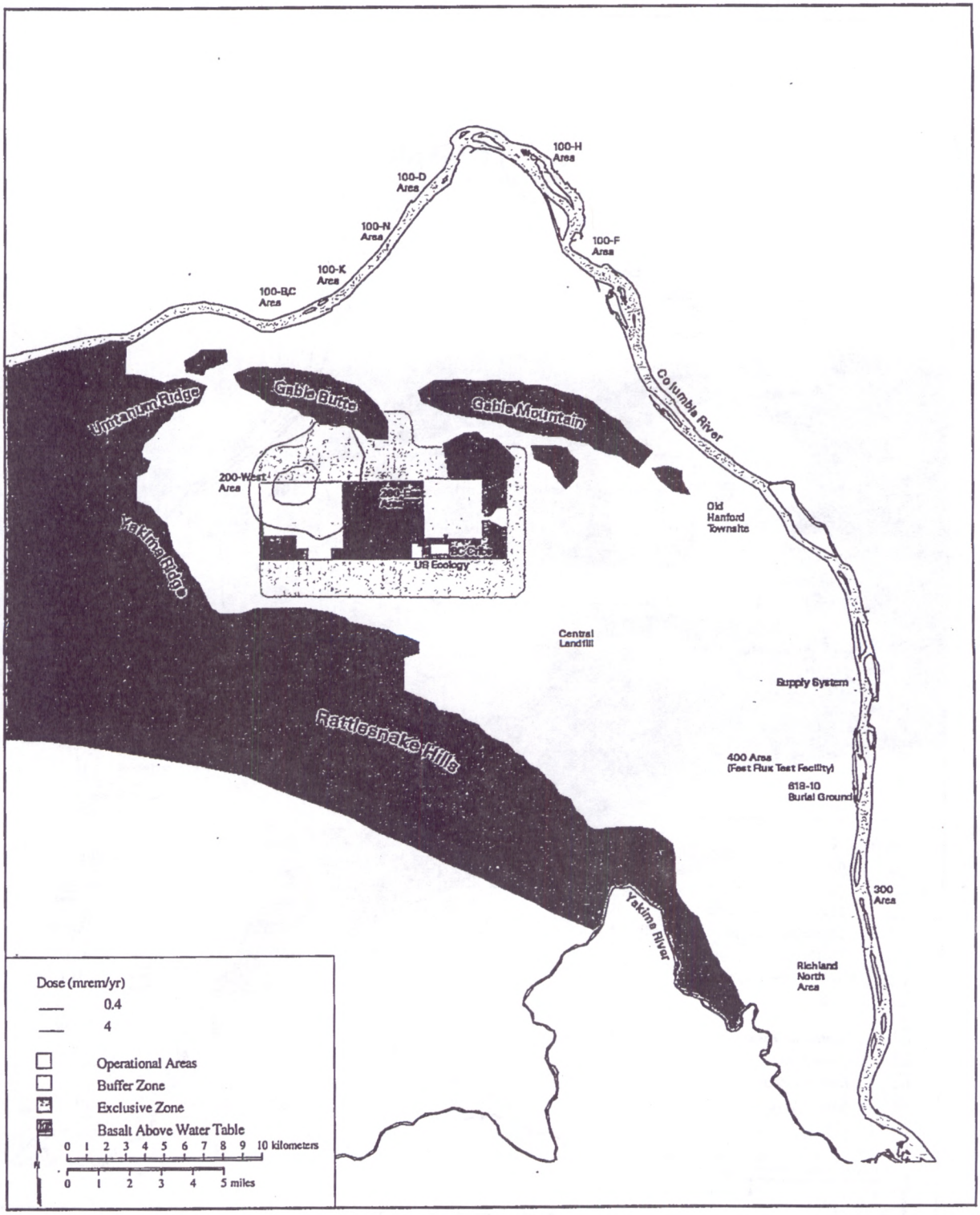

Figure 4.45. Predicted Air Pathway Dose at 2999 (Time of Peak Dose Inside the Buffer Zone)

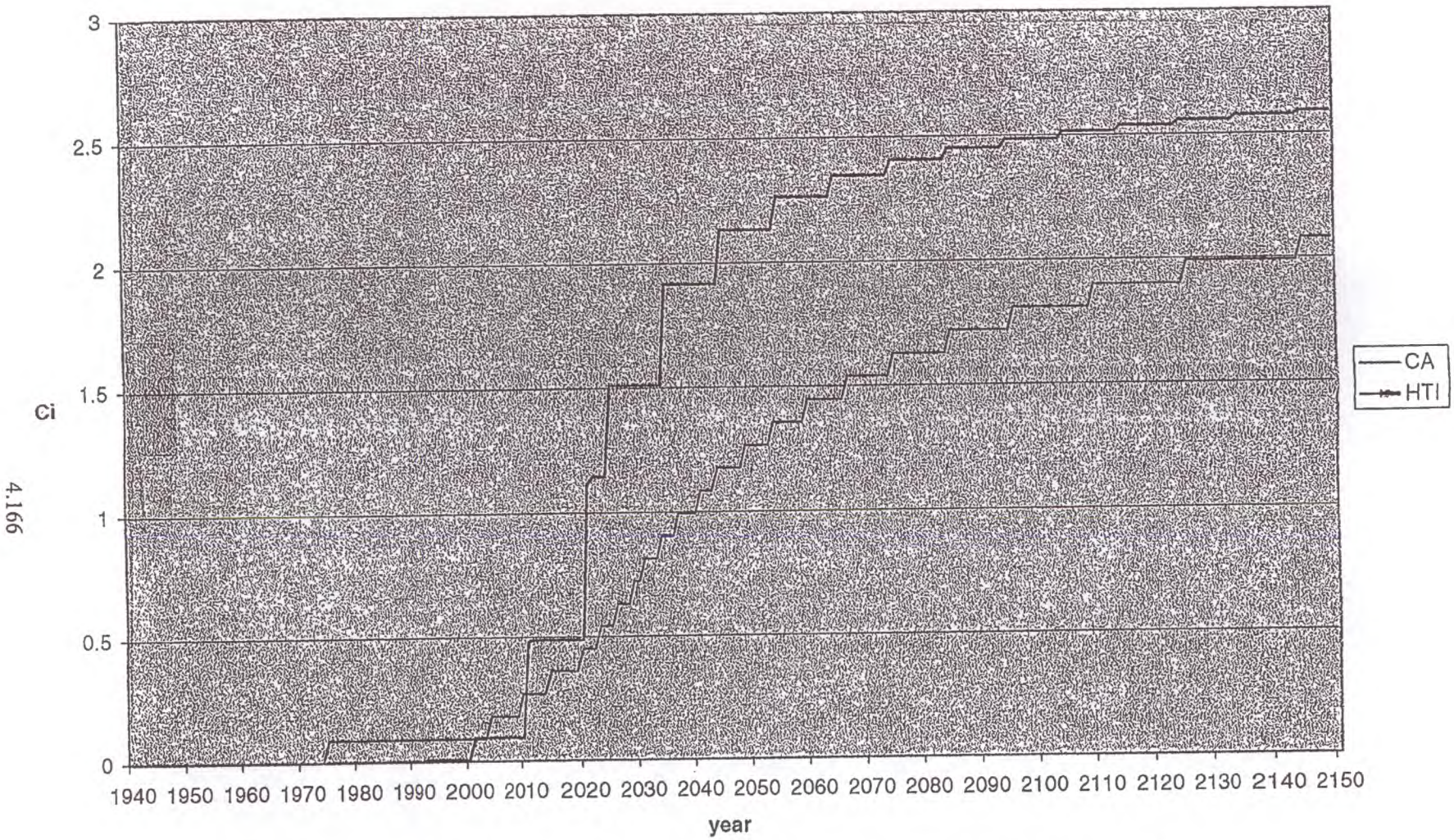

Figure 4.46. Comparison of Cumulative Release of Technetium-99 for AX Tank Farm Leaks (1940-2150)

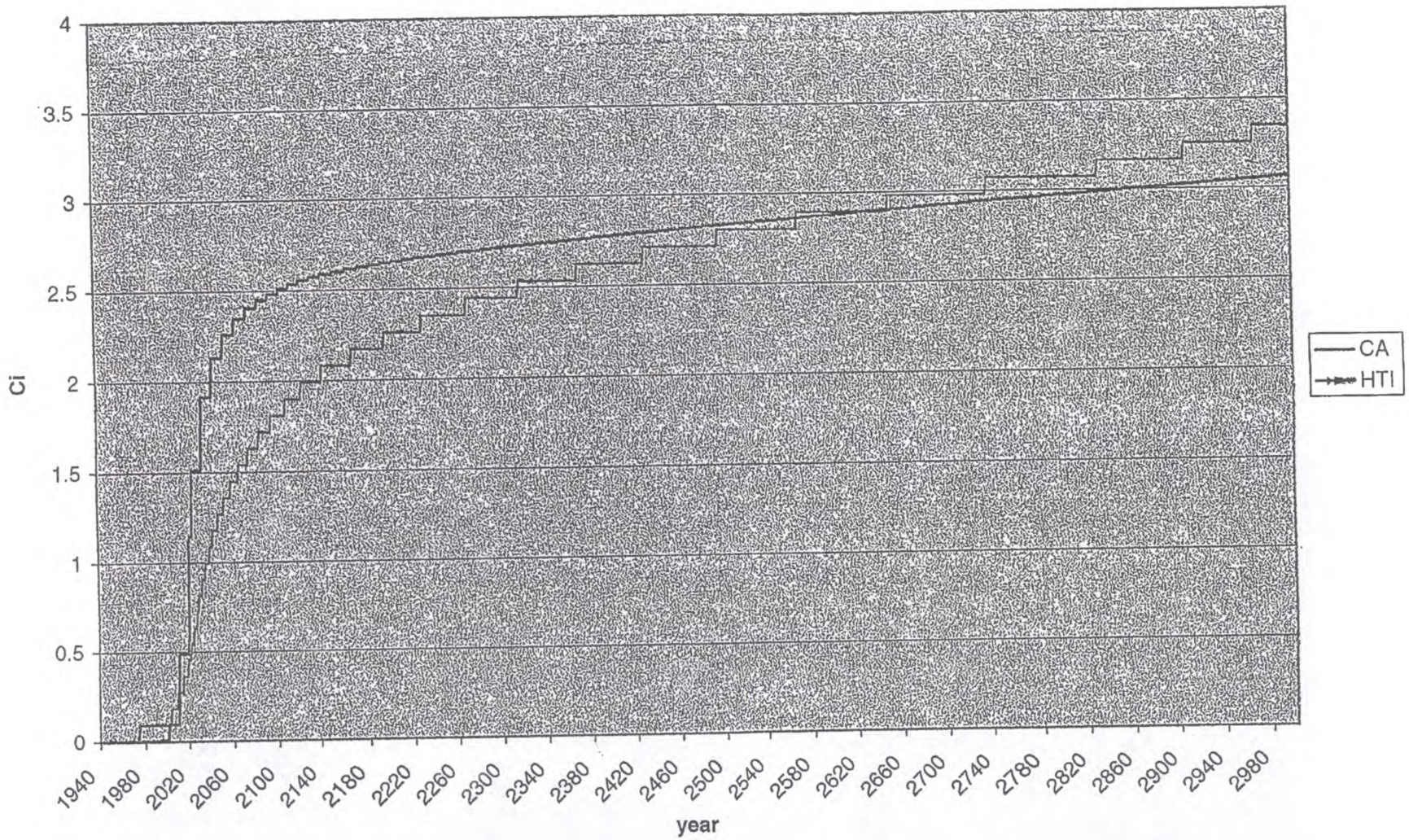

Figure 4.47. Comparison of Cumulative Release of Technetium-99 for AX Tank Farm Leaks (1940-3000)

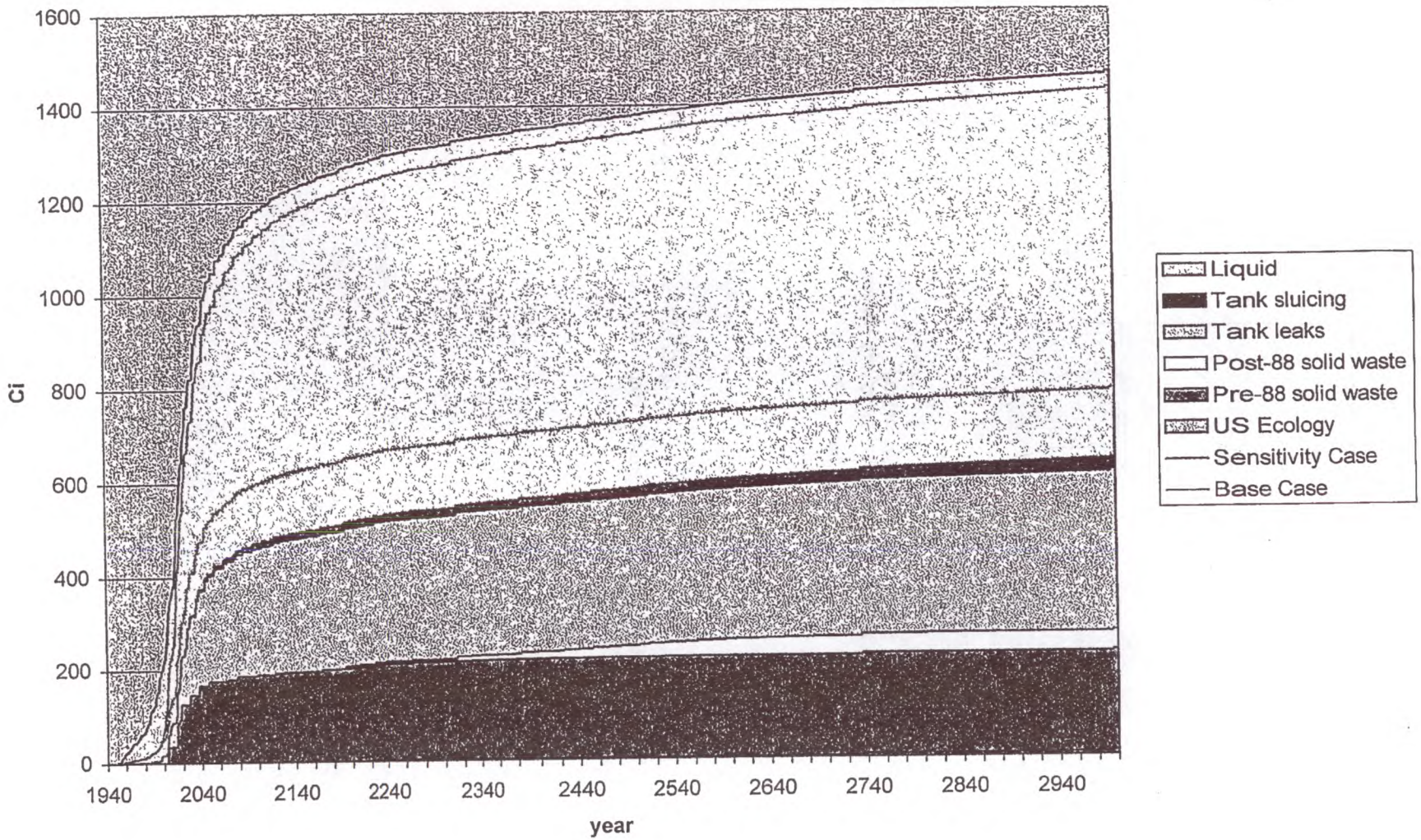

Figure 4.48. Cumulative Release of Technetium-99 from All Sources to the Water Table from 1940 to 3440 for Three Inventory/Release Scenarios for the Liquid-Discharge Sites. Shown are the Full and Scaled or Base-Case Cooney Inventory and the Sensitivity Case with Enhanced Post-1996 Release Rate.

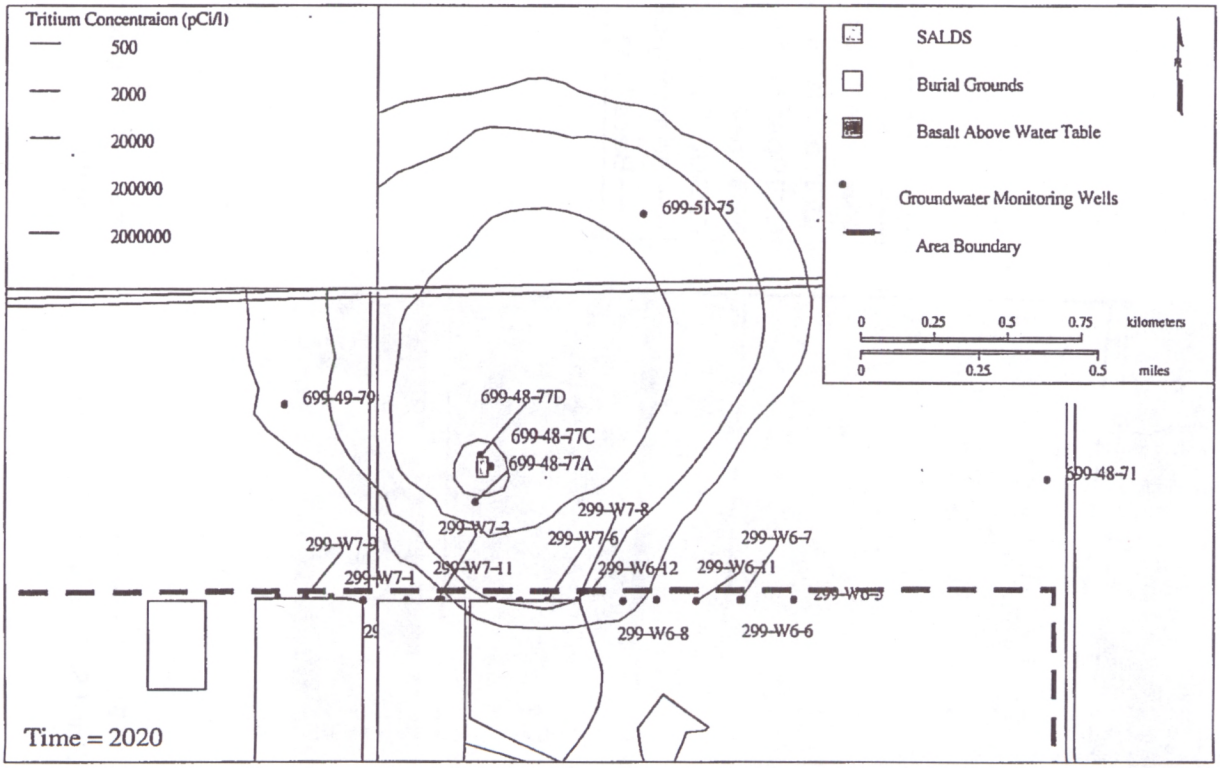

Figure 4.49a. Areal Distribution of Tritium Predicted in 2020 Near the SALDS with the High-Resolution, Three-Dimensional, Transport Model (from Barnett et al. 1997)

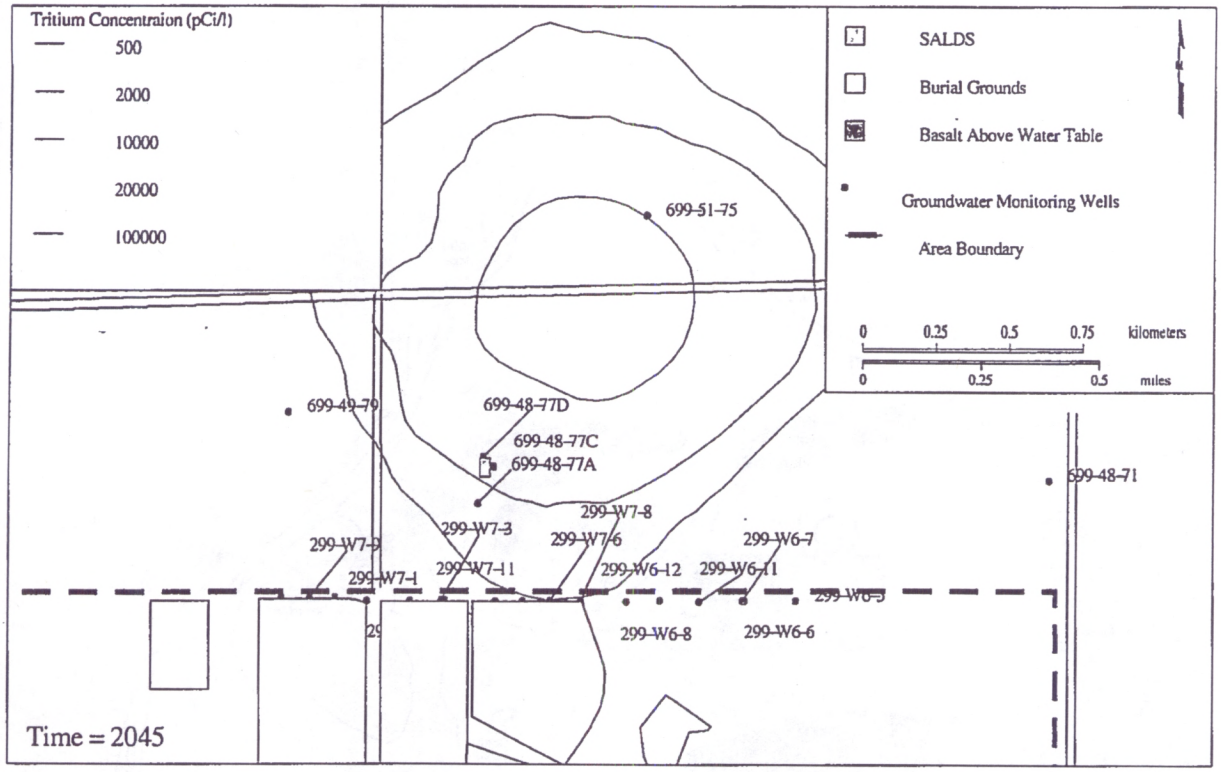

Figure 4.49b. Areal Distribution of Tritium Predicted in 2045 Near SALDS with the High-Resolution, Three-Dimensional, Transport Model (from Barnett et al. 1997)

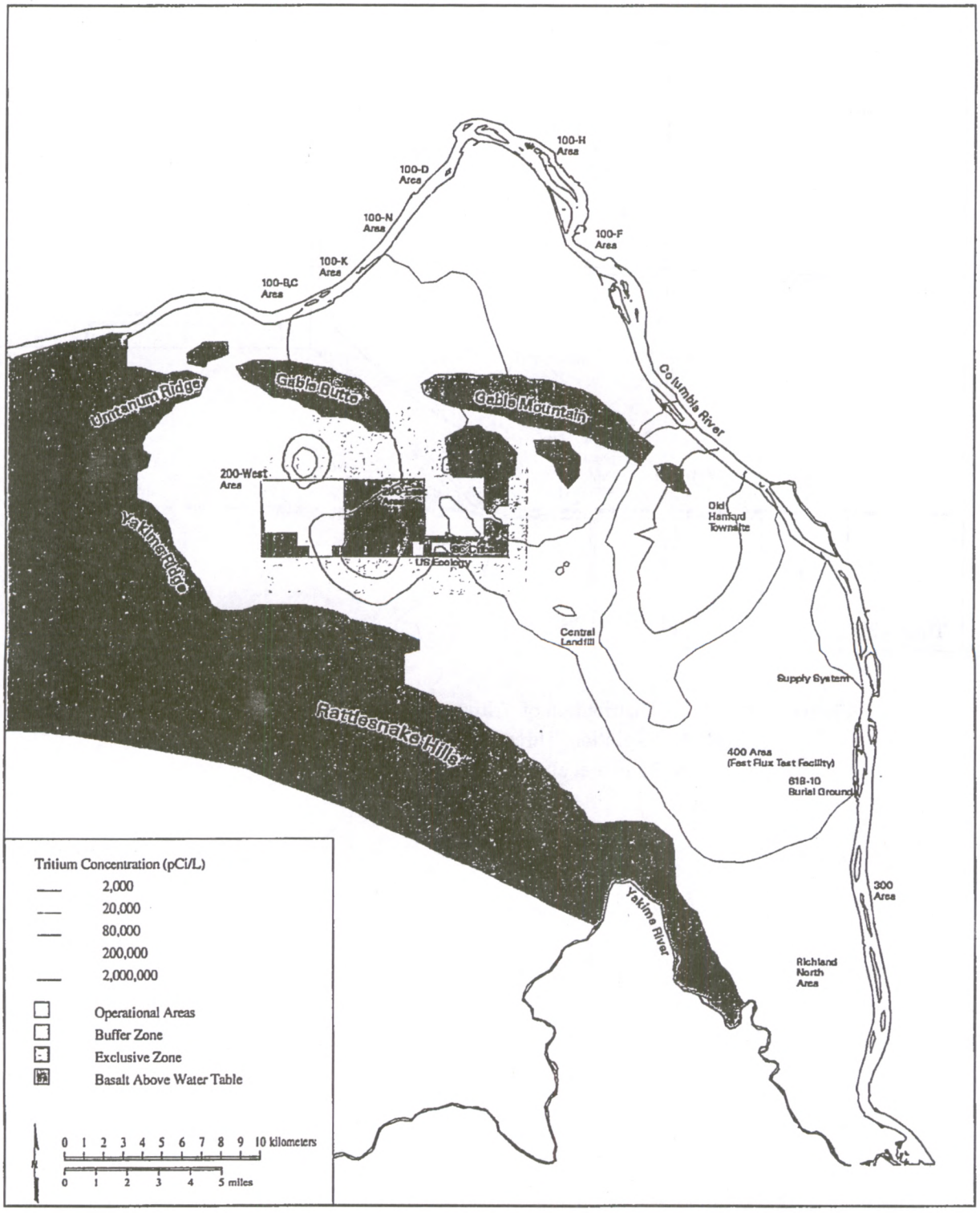

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Figure 4.50. Areal Distribution of Tritium in 2020 as Predicted with the Three-Dimensional Transport Model (from Cole et al. 1997)

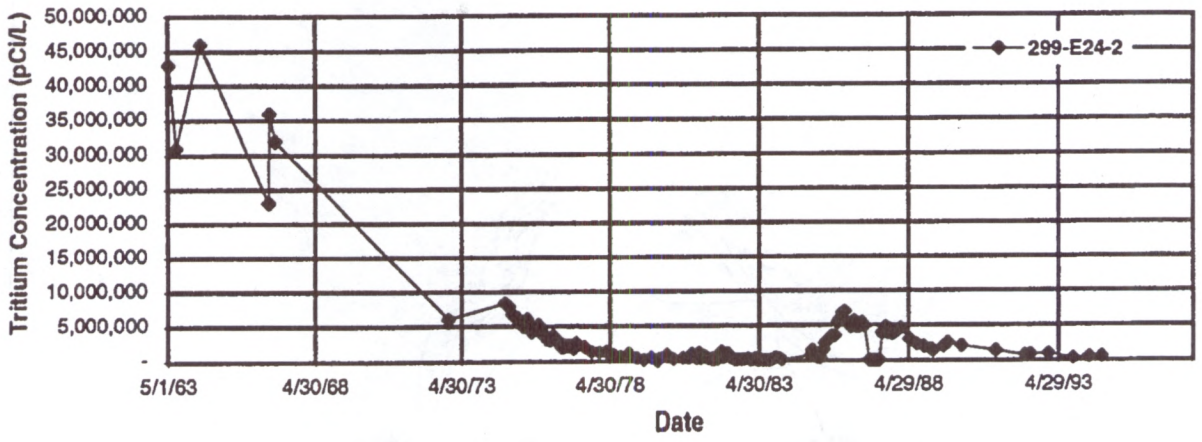

Figure 4.51. Tritium Concentration History for Observation Well 299-E24-2

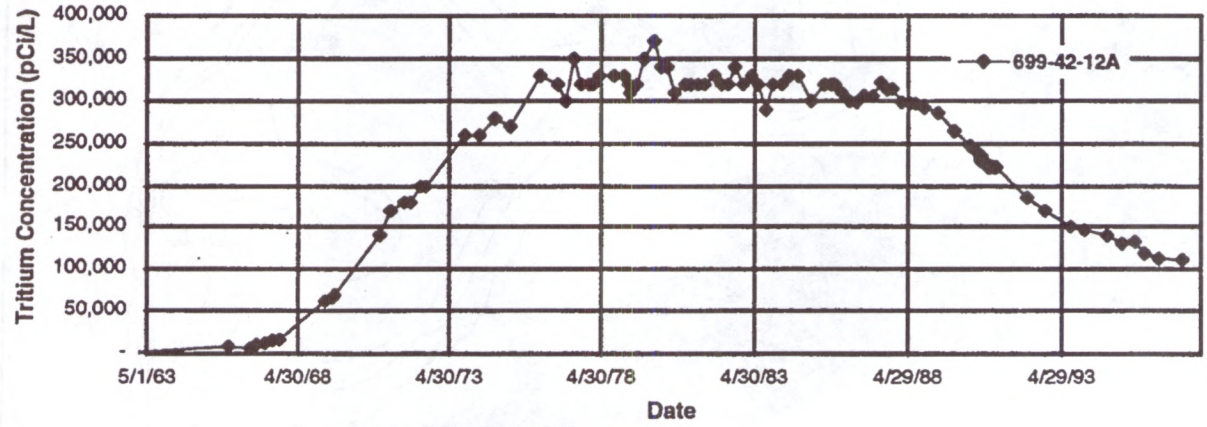

Figure 4.52. Tritium Concentration History for Observation Well 699-42-12A

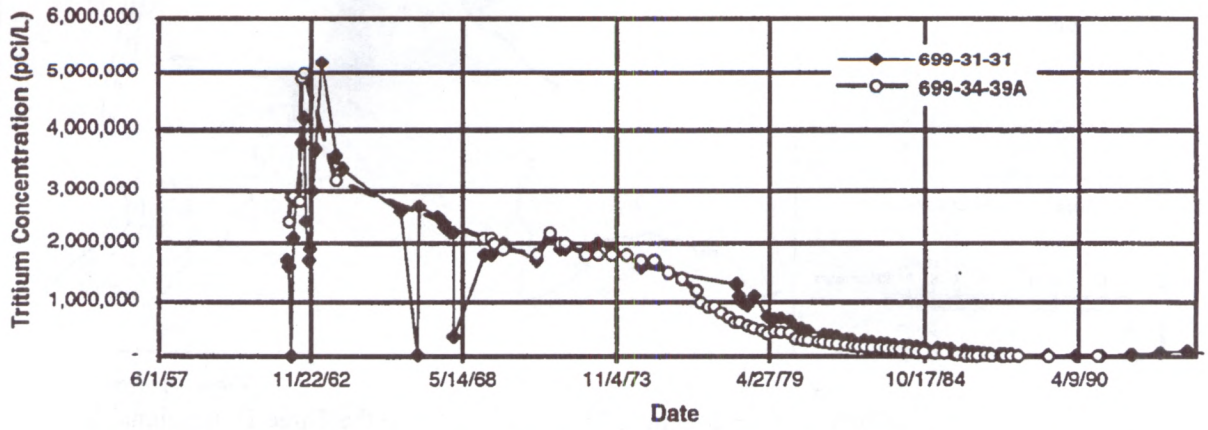

Figure 4.53. Tritium Concentration History for Observation Wells 699-31-31 and 699-34-39A

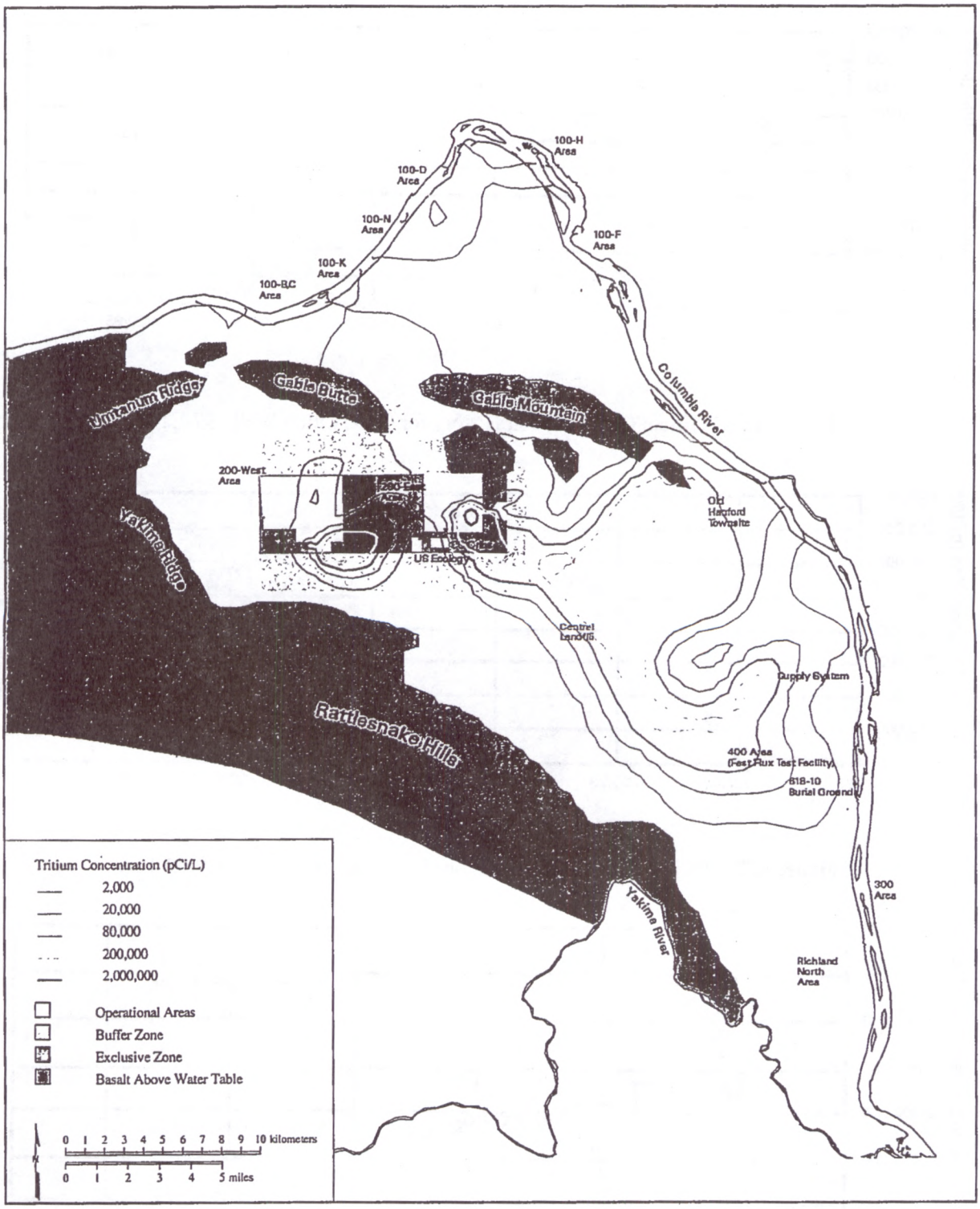

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Figure 4.54. Areal Distribution of Tritium in 1985 as Predicted with the Three-Dimensional Transport Model (from Cole et al. 1997)

5.0 Interpretation of Results

This chapter discusses the results of the Composite Analysis in comparison with the primary dose limit and the dose constraint. It includes discussions of the principle sources of uncertainty, and the implications they have for results of the base case. The results of the Composite Analysis are interpreted and salient issues are discussed. A brief qualitative ALARA (as low as reasonably achievable) assessment is presented to quantify the value to society of a detailed options analysis and ALARA assessment for alternate remediations. Finally, suggestions are made for further study in preparation for the second and subsequent iterations of the Composite Analysis.

5.1 Discussion of Results

The U.S. Department of Energy (DOE) primary dose limit of 100 mrem effective dose equivalent (EDE) in a year applies to a hypothetical future member of the public. This all-pathways dose to the maximally exposed offsite individual is calculated for 1000 years at points on the Hanford Site that a future member of the public could access. The point of access nearest the waste disposals in the future is defined by the boundary of a buffer zone designed to separate the public from the exclusive waste management area on the 200 Area Plateau (Figure 1.4). The dose constraint is defined as 30 mrem EDE in a year to the maximally exposed offsite individual for 1000 years (DOE 1996b), and is used to ensure that no single source, practice, or pathway uses an extraordinary portion of the primary dose limit. If the dose to the maximally exposed individual is above either 100 or 30 mrem in a year, an options analysis and an ALARA assessment must be performed to evaluate alternate actions the DOE could take to reduce the dose. If the dose is below 30 mrem in a year, a qualitative ALARA assessment should be performed to determine whether a quantitative ALARA analysis would be cost-beneficial.

5.1.1 Comparison with the Primary Dose Limit

To quantify potential impacts from alternate future land uses, four scenarios were used in the Hanford Site Composite Analysis to quantify dose to the hypothetical future member of the public. In order of significance with respect to the dose they yield, they are based on agricultural, residential, industrial, and recreational land use assumptions. Each of these scenarios was applied to the region of the present Hanford Site outside the buffer zone surrounding the exclusive waste management area. Maximum dose within the exclusion area and buffer zone was not compared to the dose limit.

As described in Chapter 4, a review of existing radionuclide plumes in the unconfined aquifer revealed the presence of a strontium-90 plume beneath the decommissioned Gable Mountain Pond. The observed peak concentration of strontium-90 in the vicinity of the retired pond was 1500 pCi/L in 1996 (Hartman and Dresel 1997; Figure 6.10-10). Using the unit dose factor for strontium-90 from the agricultural scenario, this concentration in groundwater converts to a dose of ~470 mrem in a year. If the site is not remediated to remove the strontium-90 in groundwater and in the overlying vadose zone, it is

recommended the exclusive waste management area be expanded to include this decommissioned pond. Furthermore, it is also recommended a buffer zone of ~1000 m be established as a region of relatively clean groundwater surrounding the existing strontium-90 plume such that monitoring can detect movement of the strontium. Strontium is highly sorbed on aquifer sediments (K_d = 20 mL/g) and its decay half-life is relatively short, 28.78 years. It is anticipated the declining water table will cause strontium in the upper sediments of the aquifer to be suspended in the vadose zone, and thereby act to further isolate the contamination. To simplify the discussion of results in the Composite Analysis, it is assumed the exclusive waste management area and buffer zone will be expanded as recommended. Hence, discussion of dose outside the buffer zone assumes the region surrounding Gable Mountain Pond is included inside the exclusive waste management area and buffer zone.

For the agriculture scenario, which exhibits the greatest dose, the maximum dose simulated from the cumulative releases is less than 6 mrem in a year during the regulatory period of 1000 years following Hanford Site closure for all lands outside the buffer zone. For this exposure scenario, the area extent of dose greater than 4 mrem in a year was projected to correspond with an area of 40 km^2 in the unconfined aquifer outside the buffer zone at the year 2050, the time of Hanford Site closure. The aquifer area outside the buffer zone associated with this level of dose is projected to vanish by 2085. Neither the primary limit, nor the dose constraint level is exceeded. During the regulatory period of 1000 years following Hanford Site closure the maximum doses simulated for the other scenarios are residential, 2.2 mrem in a year; industrial, 0.7 mrem in a year; and recreational, 0.04 mrem in a year.

This analysis has shown that in the first 1000 years after Hanford Site closure, maximum dose to an individual outside the buffer zone occurs at the time of closure and diminishes thereafter. Current groundwater contamination and its corresponding dose are a result of liquid discharges and tank leaks to the subsurface. For the post-1988 solid waste burial grounds, an initial period of relatively high recharge (75 mm/yr) was assumed to apply until a surface barrier is constructed. Under this condition, releases to groundwater of the most mobile radionuclides (e.g., selenium-79, technetium-99) were simulated to occur from the active and planned burial grounds in the next 200 years. However, sorbed radionuclides including carbon-14, iodine-129 and uranium (total) do not release in the 1000-year period. Dose at the boundary of the accessible environment (the buffer zone) resulting from releases from the post-1988 solid waste burial grounds, the Environmental Restoration Disposal Facility (ERDF), and the immobilized low-activity waste (ILAW) from Hanford Site tanks cannot be distinguished from background levels resulting from the earlier releases during the 1000-year period following Hanford Site closure.

Present calculated doses to a hypothetical onsite individual are the result of groundwater plumes originating from operational discharges that have been discontinued and early releases from accidental tank leaks. Secondary dose peaks occur in the 2020 to 2030 time frame. These secondary peaks are a result of the calculated breakthrough of radioactive contamination from accidental tank leaks, projected losses from single-shell tanks during future tank waste recovery operations, and early releases from solid waste burial grounds closed prior to September 26, 1988. At present and for several years to come, doses

calculated at points outside the exclusion area and buffer zone are dominated by tritium from past operations. By the assumed time of Hanford Site closure in 2050, doses are dominated by iodine-129 in the remnants of existing plumes.

The actual position and mobility of wastes in the vadose zone beneath liquid discharge facilities are not well known. Accordingly, analyses of liquid discharges to the aquifer are uncertain. However, existing groundwater contaminant plumes are a result of the past liquid discharges. Remnants of these wastes that remain in the vadose zone are deeper in the profile than dry wastes originally disposed in relatively shallow trenches. Many liquid wastes discharged to ground were very acidic or very basic waste streams; therefore, they may be under geochemical conditions more favorable for migration than neutralized solid waste in dry and relatively shallow disposals. Consequently, forecasts of relatively early releases from liquid discharge sites, past tank leaks, and losses during tank waste recovery operations are credible.

The analysis illustrates that, in comparison to the releases from liquid disposals or leaks, releases to the water table from the four active and planned low-level waste disposals will be delayed by hundreds or thousands of years. These disposals are essentially dry disposals. Releases from the ERDF and Tank Waste Remediation System (TWRS) ILAW disposal facilities do not release the most mobile radionuclides to the water table in the first 1000 years after Hanford Site closure. First releases of the most mobile radionuclides from the post-1988 solid waste burial grounds in 200 West and 200 East Areas appear approximately 200 years after Hanford Site closure. Minimally retarded radionuclides, including iodine-129 (0.5 mL/g) and uranium (3 mL/g), do not release to the water table from the post-1988 solid waste burial grounds in the 1000-year period following Hanford Site closure. The maximum dose from these dry disposals to the hypothetical future member of the public in the accessible environment outside the buffer zone is indistinguishable from background values within the regulatory period.

This analysis concludes that releases from the four dry disposals do not present a significant impact to the health and safety of an individual outside the buffer zone during the 1000-year regulatory period. Consequently, the impacts of these disposals do not require completion of a quantitative options analysis and an ALARA assessment.

5.1.2 The Influence of Uncertain Inventories and Contaminant Mobility

The original guidance (DOE 1996b) called for sensitivity analyses to be conducted on the issues of alternate future uses of DOE lands and alternate remediations of contaminated sites. Four land-use options were explored through the application of exposure and dose scenarios characteristic of long-term agricultural, residential, recreational, or industrial development. The analysis examined a single basic remediation alternative (i.e., the "leave undisturbed and cover with a surface barrier" alternative). Many groups view such an action as a virtual no-action alternative because wastes are not removed or further immobilized in their present setting. By this analysis, the DOE is not suggesting a preference for the alternative examined. Alternate remediations to be examined in the remedial investigations and

feasibility studies (RI/FSs) for these sites will be decided jointly by the DOE, U.S. Environmental Protection Agency (EPA), and State of Washington Department of Ecology (Ecology) agencies as part of the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) or Resource Conservation and Recovery Act (RCRA) process. Consequently, sensitivity analyses examining the impacts of alternate remediations for the variety of past-practice units including pre-1988 solid waste burial grounds, liquid discharge sites, canyon buildings, and tank farms were neither proposed nor analyzed in this first iteration of the Composite Analysis.

Aside from land use and remediation, potentially significant sources of uncertainty in this analysis lie in the assigned inventory of radionuclides and the combination of parameters assigned to influence the mobility of contaminants in the environment.

5.1.2.1 The Influence of an Uncertain Inventory

Inventory information gathered by several independent programs was used to assemble the inventory for the first iteration of the Composite Analysis. An examination of the total inventory assembled for the key radionuclides revealed a combination of issues that cannot be easily or quickly remedied, including: a) conservative estimates by individual programs, b) conservative estimates of individual radionuclides, c) no final identified disposal, d) absence of key mobile radionuclides, and e) failure to use all available Hanford Site surveillance data and process knowledge. Each of these issues is briefly discussed in this section.

The Composite Analysis is the only analysis conducted in recent years requiring an inventory compilation that applies to all the radioactive wastes that will reside at the Hanford Site after closure. The method of assembling inventory data from independent sources has proven difficult to implement. When basic records focus on major radionuclides such as cesium-137 and various isotopes of uranium and plutonium, methods of estimating the abundance of the key mobile radionuclides became central to the inventory uncertainty issue. The second iteration of the Composite Analysis would benefit greatly from the creation of an inventory that honors, or reconciles, radionuclide generation data, import data, export data, process flow sheets, and waste transaction records. The inventory should be in a form, perhaps as a model, that permits the generation of uncertainty estimates or equally likely realizations.

Conservative Estimates by Individual Programs. A conservative approach to environmental analyses is incorporated into performance assessment and risk assessment guidance and has gained acceptance. Whenever compliance to an environmental standard can be shown using a conservative analysis, there is little reason for a more accurate analysis. When a waste form can be shown to safely dispose of the entire Hanford Site-generated inventory of a radionuclide, there may be no reason for the program or project developing the waste form performance information to invest resources to better understand the true inventory. Using this logic, the TWRS program standard inventory has assigned all carbon-14, selenium-79, and iodine-129 generated in the reactors at the Hanford Site to reside in the single- and double-shell tanks. This effectively overestimates the amount of these radionuclides assigned to the tanks today, and, therefore, to the future ILAW. Similarly, the inventory estimate for the

ERDF trench was based on maximum observed contamination levels in remediation site wastes applied to the total volume of wastes to be disposed. This must result in an overestimated inventory; however, the Composite Analysis indicated leachate from this facility will not reach groundwater in the 1000-year period following Hanford Site closure. Consequently, estimated environmental performance alone will not require the development of a realistic or best-basis inventory for the ERDF trench.

Conservative Estimates of Individual Radionuclides. The quantity of selenium-79 was based on the Oak Ridge Isotope Generation and Depletion Code (ORIGEN2) simulations of the fuel irradiated in the production reactors at the Hanford Site. Among other data, those simulations relied on the decay half-life of the isotope. The half-life of this isotope was recently revised from $<6.5 \times 10^4$ years to $<6.5 \times 10^5$ years. As a result, the amount of selenium-79 generated in the fuel and introduced to the chemical separation plants will decline by a factor of up to eight. Because selenium-79 is overestimated in the current inventory, results indicating safe disposal at current inventory levels are conservative. When the total inventory of selenium-79 is revised, waste forms that now account for selenium-79 will show a decline in inventory and associated dose impacts.

No Final Disposition Identified. In some cases, the final disposition of the radionuclide inventory in waste and contained in closed facilities is not well defined. In the Composite Analysis estimates of the inventory and its location were needed. Thus, when using the calculated total inventory generated in the reactors, where that inventory will reside at the time of Hanford Site closure must be determined. Because that time is sufficiently far away, several DOE programs have not developed an understanding of their inventories and where they will finally reside. Iodine-129 is a good example. The amount of iodine-129 lost to the atmosphere and trapped in scrubbers and disposed elsewhere (e.g., solid waste burial grounds or Plutonium Uranium Extraction [PUREX] Plant tunnels) is highly uncertain, and therefore, not quantified in the standard or best-basis inventory developed by the TWRS program. Consequently, in an effort to be conservative and bound the iodine-129 issue with regard to tank waste, all iodine-129 was assigned to the tanks. Of that total, 10% is assigned to the ILAW to bound the potential dose impact of this radionuclide on ILAW performance. However, this is a conservatively high estimate of the amount of this highly volatile isotope that will be trapped in vitrified waste. The ultimate disposition of the iodine presumed in the tanks, (i.e., 66 Ci), is not well defined.

In nearly all analyses of closed facilities conducted to date (e.g., hazard assessments, waste volume estimates) the inventory data included only major radionuclides (e.g., cesium-137, strontium-90, uranium, and plutonium), or an inventory of the total fission products and total activation products. Neither of these types of inventory provides sufficient information to perform all-pathways exposure and dose analyses. This has made the simulation of some facilities and waste sites intractable for the first iteration of the Composite Analysis.

Absence of Key Mobile Radionuclides. In general, the radionuclides that have been identified as key to the estimation of maximum all-pathways dose have long decay half-lives and are relatively mobile in the subsurface environment. They are carbon-14, chlorine-36, selenium-79, technetium-99, iodine-129, and uranium (total). Chlorine-36 is included because of its known presence in the graphite reactor

cores. Its potential impact in other waste has also been studied in this analysis by incorporating a hypothetical amount in irradiated fuel and therefore, in the Hanford Site waste streams. Three of the radionuclides, carbon-14, iodine-129, and uranium (total), exhibit a small degree of sorption in the environment. With the exception of uranium, the mobile and long-lived radionuclides were not routinely measured and reported during the production period at the Hanford Site.

While it is common to find cesium-137, strontium-90, uranium (isotopic or total), and plutonium (isotopic or total) reported in inventory records for specific facilities, the others are not commonly found. The abundance of the mobile and long-lived radionuclides in irradiated fuel is estimated in the total standard inventory. However, records are incomplete with respect to their quantities discharged to the environment as gaseous atmosphere releases, to liquid discharge sites (e.g., cribs, specific retention trenches, reverse or discharge wells), or to solid waste burial grounds. Improved confidence in the quantities of the mobile radionuclides assigned to liquid discharges, tank leaks, and solid waste requires an accepted means of estimating with confidence the abundance of carbon-14, selenium-79, technetium-99, iodine-129, and perhaps chlorine-36 for gaseous, liquid, and solid waste disposals.

Failure to Use All Available Data and Process Knowledge. It is fundamentally important to use process knowledge and transfer records to estimate the timing, volume, and inventory of wastes discharged or lost to the environment. The combined 216-U-1&2 crib site is an example where a combination of process knowledge and field observations could yield an improved estimate of the original release to a liquid discharge site. A series of events culminated in the detection of a release of uranium in solution to the water table in the vicinity of this crib site in February 1985. The observed uranium plume in the groundwater was and is significant. Periodically it has been the subject of pump-and-treat programs since its discovery. A significant technetium plume appeared at the same time and occupies the same groundwater. Its source is assumed to be the same crib site. However, technetium-99 disposed to the 216-U-1&2 cribs has not been estimated, based on either the likelihood that technetium-99 followed uranium in the U Plant process that generated the waste stream, or the knowledge that a substantial quantity of technetium-99 is now in the aquifer beneath these cribs. Because the Composite Analysis was based on the assembled inventory, the analysis of liquid discharges does not predict the present technetium-99 plume beneath the 216-U-1&2 cribs. This plume was modeled as an existing plume, and results show that prior to its migration from the buffer zone, it will disperse and its dose consequences will greatly diminish. Existing databases that reveal the temporal and spatial extent of contamination in the environment are a valuable asset in the assessment of the original discharges. While these data may be incomplete, they do provide valuable clues to the presence of specific radionuclides and their relative abundance.

Case for a Single Inventory Estimate. The absence of an inventory generated with a view toward a best estimate of the final location and inventory of all wastes makes it virtually impossible to perform a meaningful study of sensitivity because too many possible realizations could be generated and improbable realizations would be admitted. In the study of an isolated facility or waste form, (i.e., as in a performance assessment), the influence of various levels of inventory can be examined. However, an assessment of uncertainty in inventory for the Composite Analysis requires alternate inventories in terms

of both location and quantity. It is the potential cumulative impact of multiple disposals at a moment in time and point in space that creates the maximum composite dose. Thus, the uncertainty in composite dose is a function of uncertainties in inventory, release, migration, and exposure. The total or global estimate of inventory, based on reactor operation, is the best information on inventory at the Hanford Site. Upper bound estimates of inventory disposed at most facilities are unknown and unknowable. Thus, inventories assigned to each facility or type of facility must be associated with the known range of inventory for each. In other words, the inventory realizations generated must be equally probable for the assessment of uncertainty to be meaningful. For example, if less of a specific isotope is in the tanks today (and will eventually be in ILAW and immobilized high-level waste), more should be assigned to the inventory of one or more of the following:

- lost in gaseous form to the atmosphere during chemical separation processes
- disposed in liquid form to the subsurface
- disposed in solid form to burial grounds
- residing in canyon building vessels or structures or filters
- residing in PUREX tunnels in process vessels.

The perturbations possible in the assignment of inventories to specific waste disposal facilities must be constrained by our knowledge of processes and field observations. Before useful sensitivity cases capturing our uncertainty in the inventory can be formulated, the internal dependencies or correlation of the inventory problem must be incorporated into a collective best-estimate model. Only then can the true significance of real uncertainties be determined through simulation of environmental consequence.

Bounding estimates of inventory may be meaningful in the Composite Analysis when examining a single facility or class of facilities. However, with few exceptions, reaching consensus on a bounding inventory for a specific facility or waste form could be difficult. Such estimates may be useful when attempting to determine the maximum potential influence of a facility outside the buffer zone. Similarly, one may wish to determine the inventory required in a facility to cause a given impact outside the buffer zone. Such analyses would be useful in evaluating the need to retain a given facility or class of facilities in the Composite Analysis. Certainly, as in any environmental assessment, if the release processes or migration pathway act to constrain the dose impact to levels well below the performance standard, then bounding inventories could be assigned to all sources to simply illustrate the ultimate safety of the waste form or physical setting. However, simulations of this type do not quantify the influence of an uncertain inventory. They provide a reasonable assurance of no impact from any reasonable inventory.

Inventory Uncertainty with Respect to Dry Disposals. Given the assumptions regarding future land use, the results obtained in the first iteration of the Composite Analysis illustrate that inventories assigned to the active and planned disposals will not yield significant releases in the 1000-year period following Hanford Site closure. The analysis also indicated that increased inventories assigned to these disposals would not yield significant releases in the 1000 years. To ensure that no significant releases from the burial grounds are possible, waste acceptance criteria and procedures (WHC 1993) were applied to screen each waste package for mobile radionuclide content (e.g. carbon-14, selenium-79,

technetium-99, iodine-129, and uranium) prior to disposal and to determine if mobile radionuclide inventories are sufficiently high to require additional isolation from the environment. Typical immobilization processes are encapsulation of waste packages in thick concrete boxes or direct grouting of the waste material. This protocol ensures that no one package can provide a substantial portion of allowable dose. Neither the ERDF nor the TWRS ILAW disposal facilities are predicted to release any radionuclides in the 1000-year regulatory period.

Thus, the inventories analyzed yield maximum dose well below the DOE dose limit and constraint levels. Additional inventories, if identified for future disposals in the burial grounds, would be immobilized prior to disposal and be determined to be safe for long-term disposal prior to acceptance. Therefore, further analysis of the radioactive waste inventory with respect of incremental dose impacts from solid waste burial grounds, the ERDF, and the TWRS ILAW would not yield additional insight and are not included in the Composite Analysis.

5.1.2.2 Uncertainty in Contaminant Mobility

A recent peer review of vadose zone contamination beneath single-shell waste tank 241-SX-109 was critical of Hanford Site knowledge of contaminant migration resulting from a tank leak (DOE 1997a). The panel found insufficient information to defend a single conceptual model of the physical path and chemical mobility of cesium-137 and other radionuclides leaked to the subsurface. Alternate conceptual models stress the potential roles of natural heterogeneity, man-made preferential flow paths, fluid density, and geochemical mobility on the migration and fate of contaminants. The TWRS Vadose Zone Program is underway to gather data to better define the present distribution and future mobility of contaminants in the vadose zone beneath tank leaks. In addition, the DOE has created a program, the Hanford Groundwater and Vadose Zone Integration Project, to coordinate the study of the vadose zone across the various environmental management and environmental restoration programs at the Hanford Site. Developing confidence in models of contaminant migration and fate for the vadose zone beneath liquid discharge sites and leaking tanks must await completion of the early stages of these programs. The second and subsequent iterations of the Composite Analysis will benefit from these programs.

Geochemical Mobility. Unlike previous sitewide analyses, the Composite Analysis of wastes within the exclusive waste management area and buffer zone distinguishes among six waste types that were discharged or leaked to the subsurface environment. Based on the waste characteristics and an assumed amount of contact with subsurface sediments, the chemical elements were assigned distribution coefficients for each of the waste types and three geologic settings; an upper vadose zone, the lower vadose zone, and the unconfined aquifer (Appendix E). Thus, in this analysis process waste streams with a high organic content and very acidic pH have been assigned a different mobility than those wastes with a low organic content and a near-neutral pH.

As described in Section 4.1.2.1.4, based on post-mortem studies of cribs, specific retention trenches, and tank leaks conducted during the late 1970s and early 1980s, the vadose zone was divided into two segments. The waste type governed the selection of the distribution coefficient in the upper segment. A

combination of waste-type and sediment interactions governed the assigned value in the lower segment. In general, although not always, the wastes are more mobile in the upper segment and less mobile in the lower segment. This conceptual model applies to the liquid discharges to the sediment profile including tank leaks and losses from tanks during recovery operations.

Wastes in the dry disposal sites, including all solid waste burial grounds, the ERDF trench, and the TWRS ILAW disposal facility, were assumed to have a low-organic content, a low-salt content, and a near-neutral pH. Accordingly, chemical elements in these wastes were assigned a single distribution coefficient that applies throughout the sediment column. Best-estimate values of distribution coefficients for carbon, iodine, and uranium were 5, 0.5, and 3 mL/g, respectively. Chlorine, selenium, and technetium were all assigned 0 mL/g and assumed to move with the water. Conservative values for carbon, iodine, and uranium are 0.5, 0.3, and 0.6 mL/g, respectively. None of the sorbed contaminants from dry disposal sites are predicted to reach the water table in the period analyzed. The more mobile chlorine, selenium, and technetium radionuclides behave identically in both cases. It is interesting to note that if carbon and uranium were simulated using their conservative values of distribution coefficients (0.5 and 0.6 mL/g, respectively), their behavior would be similar to that of iodine with its best-estimate value of distribution coefficient (0.5 mL/g). Neither would reach the reach the aquifer in 1000 years. Thus, a sensitivity analysis regarding the geochemical mobility of wastes disposed in the solid waste burial grounds, the ERDF trench and the TWRS ILAW disposal facility would not reveal significantly different results in the 1000-year regulatory period.

Hydrogeologic Mobility. There is a fundamental difference between liquid discharges (including tank leaks), and dry disposals. Liquid discharges carry the contaminants into the vadose zone beneath the liquid discharge facility. This liquid, including radioactive contamination, seeks to redistribute in the vadose zone such that it comes into equilibrium with the surrounding soil water. Continuous liquid discharges move liquid waste and associated contamination deep into the vadose zone, eventually resulting in breakthrough to the water table. Short-term and lower-quantity discharges displace the resident soil water and then are driven more slowly by natural recharge as they also migrate downward to the water table.

The hydrologic driver for dry disposals is the recharge rate. Solid waste burial grounds at the Hanford Site are typical. Once in place, they are covered by 2 m of backfill pending placement of a final surface barrier system. In the base case, a sequence of recharge rates indicative of a site covered with coarse soil and maintained free of vegetation (75 mm/yr) followed by the site covered with a surface barrier (5 mm/yr) has been examined. One important nuance of this recharge and release scenario is that wastes were leached throughout both periods, i.e., it was assumed waste containers did not present a barrier to direct and immediate leaching by pore water during the period prior to surface barrier construction. In this scenario, pre-1988 solid waste burial grounds release waste to the water table in the first decades of the next century. Post-1988 disposals exhibited releases of the most mobile contaminants (chlorine-36, selenium-79, and technetium-99) in approximately 200 years but no release of less mobile contaminants (carbon-14, iodine-129, and uranium) in 1000 years.

The comparison case for solid waste burial grounds examined the scenario where leaching of the waste did not begin until after the surface barrier with a recharge rate of 5 mm/yr was constructed over the trenches. Essentially, this case is based on the assumption that waste containers minimize or preclude direct leaching of the solid waste until the barrier is in place. This case duplicates an essential feature of the analyses presented in the performance assessments for the post-1988 solid waste burial grounds (Wood et al. 1995; Wood et al. 1996). Because of the integrity and durability of containers employed since 1984, the comparison case is believed to be a better representation of future burial ground performance. In this comparison case, post-1988 disposals exhibited mean travel times of approximately 1070 years from burial grounds in the 200 West Area, 1150 years from the 218-E-10 burial ground, and 650 years from the 218-E-12b burial ground. The majority of future solid waste is destined for 200 West Area burial grounds, and, therefore, the mean travel times on the order of 1000 years will govern the majority of future releases.

The consequences of not constructing a surface barrier and applying a higher recharge rate (e.g., 50 mm/yr) over the long-term were studied in the published performance assessments (Wood et al. 1995; Wood et al. 1996). They concluded that a surface barrier tailored to the site and waste conditions should be designed and constructed over the burial grounds.

It is unlikely that more rapid leaching of the solid waste could occur than is characterized in the base case. The comparison case performed for the Composite Analysis captures a more likely scenario; however, low-level waste (LLW) containers are not specifically designed to defeat leaching by pore water for extended periods of time. When solid waste is dry and not corrosive, it is likely the container will survive and protect the waste from leaching phenomena. Further, moisture inside waste packages is largely eliminated by waste acceptance criteria requiring free liquid to be sorbed inside the package (WHC 1993). The comparison case may be extreme in the sense of preventing any release until the low recharge rate influences both the release and its subsequent migration.

The model employed in the Composite Analysis to represent contaminant transport in the vadose zone is one-dimensional. One shortcoming of such a model is its inability to quantify the multi-dimensional aspects of the analysis. Placement of a surface barrier implies an immediate and complete change in the recharge rate that is leaching the source and driving contaminants through the vadose zone. Edge effects of a barrier are neglected. In general, the distance from the land surface to the water table beneath the exclusive waste management area is less than 100 m. The physical size of the four disposal facilities under consideration suggests their barriers will be in excess of 100 m across, and it is anticipated that the barrier will be extended well beyond the disposal facility, e.g., trench, vault. Consequently, the opportunity for edge effects, i.e., moisture moving under the barrier in the vicinity of its edge, to leach the disposed waste or accelerate its migration to the water table is less than might be envisioned. For example, only those wastes near the edge of the barrier could be exposed to greater leaching, and if the barrier is extended well beyond the edge of the disposal this is less likely to occur. Similarly, the ability of the edge-effect recharge to affect the transport pathway also will decrease with

the length of extension. Thus, assuming that the future barrier design will include sufficient edge extension beyond buried waste deposits, the base case and sensitivity cases capture the range of likely environmental responses.

Uncertain Mobility with Respect to Dry Disposals. Although dry disposals in solid waste burial grounds, the ERDF trench, and the TWRS ILAW disposal facility are a primary focus of the Composite Analysis, this first iteration of the analysis has shown the importance of liquid releases (e.g., liquid discharges and tank leaks) and their migration and fate. Dry disposals are influenced by the recharge rates through disturbed surfaces and engineered surface barriers as compared to higher rates experienced at liquid release sites. Similarly, solid wastes are subject to more favorable geochemical mobility factors (i.e., distribution coefficients) than some liquid release sites. The cases reported in the Composite Analysis capture the range of conditions most likely to govern the mobility of these wastes, and illustrate the safety of these dry disposals.

5.2 Interpretation of Composite Analysis Results

While not as detailed as either a performance assessment or CERCLA analysis of LLW sites, this Composite Analysis is a reasonable first assessment of cumulative impacts at the Hanford Site. It includes impacts from active and planned LLW disposal facilities, and other sources of radioactive contamination that could interact with these LLW disposals and affect the dose to future members of the public. This Composite Analysis provides insight into what could occur at the Hanford Site in the next 1000 years and informs the DOE of the safety of active and planned LLW disposal.

By design (DOE 1996b) and out of necessity, the Composite Analysis is less rigorous than a site-specific performance assessment or an RI/FS analysis. The 200 Area Plateau at the Hanford Site will be the final disposal location for a variety of waste forms and a considerable radionuclide inventory. A less sophisticated modeling approach and a sitewide scale were justified in this first iteration of the Composite Analysis because of the required scope of the analysis (e.g., the number and variety of sites) and the level of information readily available. Portions of the modeling effort have been less rigorous (i.e., simple zero-dimensional release models and a one-dimensional vadose zone model were employed). However, model results have been matched qualitatively with observed releases to the unconfined aquifer. A more sophisticated aquifer model than appears in previous performance assessment and RI/FS analyses has been applied.

5.2.1 Consistency with Previous Performance Assessments and with the ERDF RI/FS

The Composite Analysis is a companion document to four site- or waste-form specific studies. These studies are the performance assessments for the 200 West and 200 East Area solid waste burial grounds (Wood et al. 1995; Wood et al. 1996), the RI/FS completed for the ERDF trench (DOE 1994b), and the interim performance assessment for the ILAW now in the single- and double-shell tanks (Mann et al. 1997). The performance assessment for ILAW is scheduled for submittal in the spring of 1998.

Performance assessments for the solid waste burial grounds (Wood et al. 1995; Wood et al. 1996) show first release of the most mobile radionuclides within the 1000-year period. However, these releases occur late in the 1000-year period and they result in a projected all-pathways dose that is orders of magnitude below the standard. These published performance assessments differ from the Composite Analysis in their approach to uranium release and migration. Wood et al. (1995) and Wood et al. (1996) make a conservative assumption regarding uranium mobility and assign it a distribution coefficient of 0 (mL/g) in the subsurface sediments. They also modeled the release of most of the uranium inventory by applying a solubility controlled release model. Solubility values were assumed to be controlled by the local geochemical environment that was dominated by soil water reactions or cement-water reactions if the uranium was encapsulated in grout or disposed in concrete boxes. The net result is a much lower rate of uranium release to the subsurface followed by a more rapid migration through the vadose zone. In the first iteration of the Composite Analysis uranium was assigned a best-estimate value of 3 mL/g for distribution coefficient and does not release from the vadose zone in the 1000-year period following Hanford Site closure. The simulation of another radionuclide, iodine, using a distribution coefficient of 0.5 mL/g also revealed no release. The conservative estimate of uranium sorption is 0.6 mL/g (Appendix E). Consequently, it is not necessary to apply the more realistic but complex release model because uranium is predicted to not release from the vadose zone in the 1000-year period of regulatory concern. Note, if the Composite Analysis had shown uranium release to the aquifer from dry disposals, it would be important to apply the combined solubility and sorption model as was done in the performance assessment. Only through the application of models for both processes can the concentration and timing of the release and transport be realistically modeled.

Both the ERDF RI/FS (DOE 1994b) and the interim performance assessment for TWRS ILAW (Mann et al. 1997) call for a high-integrity surface barrier. The Hanford Protective Barrier has a design standard recharge rate of 0.5 mm/yr, and it is assumed this barrier, or a similarly effective one, will be placed over the both the ERDF trench and the TWRS ILAW facilities soon after disposals are completed. The ERDF trench is double-lined to prevent releases during disposal operations, and the surface barrier is to be applied immediately after the trench is full. Thus, no releases from the trench are anticipated prior to barrier placement. Accordingly, releases will be a result of long-term leaching at a rate defined by the recharge rate through the surface barrier. Cases in the RI/FS (DOE 1994b) that examined surface barrier and liner conditions similar to those considered in the Composite Analysis showed no release to the water table in 10,000 years. The Composite Analysis results show no releases from the ERDF to the water table in the 1500 years analyzed.

While design features of disposal facilities for ILAW have not been finalized, it is apparent that barriers to recharge will be constructed to limit infiltration soon after waste placement. The TWRS ILAW performance assessment has shown the earliest releases to the water table of the most mobile radionuclides will occur in approximately 1000 years. However, the performance assessment has employed a somewhat higher dispersivity for the vadose zone than the Composite Analysis. Consequently, the TWRS ILAW performance assessment would be expected to show an earlier release because of greater longitudinal dispersion. Because of the higher vertical resolution (i.e., finer grid)

possible with the one-dimensional model, this iteration of the Composite Analysis used a dispersivity of 80 cm and predicted no release from the TWRS ILAW facilities in the base case for the 1500-year period analyzed.

5.2.2 Other Sites in the Exclusive Waste Management Area and Buffer Zone

In addition to the active and planned disposals of LLW, this Composite Analysis examined existing plumes and future releases from pre-1988 solid waste burial grounds, liquid discharge sites, tank leaks, tank losses during recovery operations, tank residuals, and graphite cores from the production reactors. This analysis has shown a marked separation of environmental response to liquid discharges and leaks to the subsurface environment, and dry disposals of the recent past and the future. There are two episodes of groundwater contamination: the near-term contamination of the aquifer by liquid discharges, tank leaks, tank losses, and past-practice or pre-1988 burial grounds; and the long-term events associated with recent and future dry disposals.

5.2.2.1 Existing Plumes

Recent reports have examined existing plumes, their future migration, and their fate (Chiaramonte et al. 1997; Cole et al. 1997). Projections of groundwater contaminant plume migration in this analysis are based on the model described by Cole et al. (1997).

Plumes of tritium, cobalt-60, strontium-90, technetium-99, iodine-129, and uranium (total) are found in the unconfined aquifer. Because of its discharge from the aquifer and decay half-life of 12.3 years, dose from tritium in the unconfined aquifer in 2050 is less than 2 mrem in a year, and by 2100 its dose contribution has virtually vanished. Cobalt-60, with its decay half-life of 5.3 years and much lower inventory in the aquifer, will be of even less significance. Because of its strong sorption on aquifer sediments, strontium-90 is shown to remain within the exclusive waste management area and buffer zone, and does not significantly contribute to dose outside the buffer zone. As stated earlier, for the purpose of simplifying this discussion, it is assumed the exclusive waste management area has been expanded to include Gable Mountain Pond. At present, if groundwater were pumped and used as the scenarios assume, dose from tritium would dominate the maximum dose outside the buffer zone. Between now and the assumed time of Hanford Site closure, iodine-129 in the remnant of the existing plume becomes the dominant contributor to dose outside the buffer zone as the impact of tritium diminishes. Future releases from past liquid discharge sites, past tank leaks, future tank losses during tank waste recovery operations, and pre-1988 solid waste burial grounds also contribute to near-term doses. Existing plumes are evidence of environmental response to past liquid discharges and tank leaks. Only recently have tank wastes been identified as contributing to existing plumes (Johnson and Chou 1998; Hodges 1998). Thus, historical plume observations are a direct response from prior liquid discharges to the subsurface. In general, existing plumes are a result of large volume liquid discharges of process plant waste streams. They range from waste streams that were usually directed to tanks, to large volumes of cooling water carrying dilute contaminant concentrations. Sediment columns contaminated

with these discharges will continue to drain and discharge to the water table aquifer. This Composite Analysis estimated dose from these releases to be maximum now, and to decline with time.

An inconsistency exists between the technetium-99 inventory assigned to liquid discharge sites and the estimated inventory of technetium-99 in observed groundwater contaminant plumes. The release and vadose zone transport models for these discharges were driven by significant liquid discharge rates and they estimated significant inventories of technetium-99 were released to the aquifer. An estimated 930 Ci of technetium-99 were disposed to ground in these facilities (Waite 1991). The transport model routed these wastes to the groundwater rather rapidly. The model predicted a cumulative activity of 181.2 Ci of technetium-99 released to the water table from liquid discharge sites by 1996. However, based on field observations, the existing plumes of technetium-99 were estimated to contain between 15.8 and 37.6 Ci, depending on the assumed thickness of contamination in the aquifer. While the integrated mass of contaminant in the aquifer is uncertain, it is believed the unconfined aquifer does not contain the amount of technetium-99 predicted by the source release model. The inconsistency was remedied by assuming the estimated mass in the aquifer was correct. As noted in Section 4.3, the model estimate of tank leak contribution to technetium-99 plumes was ~5 Ci in 1996. Consequently, the inventory estimated as released to liquid discharge sites for the base case was scaled down to result in a release to groundwater of 32.6 Ci by 1996.

If the full 930 Ci inventory of technetium-99 were disposed, it could increase the contribution to dose from liquid discharge sites by a factor of ~6.6. However, the resulting increase in maximum dose outside the buffer zone would be less than 2 mrem in a year in 2050, and less than 3 mrem in a year during the 1000-year period following Hanford Site closure. Such increases to the overall maximum individual dose would only occur if points in space and moments in time for maximum dose from liquid discharge site releases coincided with other maximum or high contributions to dose. This sensitivity case was analyzed and yielded a maximum dose of less than 7.5 mrem in a year for exposure outside the buffer zone in 2050 from the agricultural scenario. This was the maximum dose for the 1000-year period following Hanford Site closure. The maximum dose obtained for the period after 2150 was less than 6 mrem in a year.

5.2.2.2 Liquid Discharge Sites

This analysis includes contaminant releases from the ditches, ponds, reverse (or injection) wells, cribs, and specific retention trenches located on the 200 Area Plateau. There are no known prior analyses of the large number of liquid discharge sites examined. However, several post-mortem studies of specific facilities have been conducted, and results of those field studies have been used to qualitatively fit model results to field observation.

Large discharges of cooling water were made to a variety of ditches and ponds in the 200 Areas. These discharges had a significant influence on the water table, and, therefore, on the groundwater flow direction and rate. This is revealed in the history of groundwater mounds beneath the U Pond in 200 West Area, the B Pond in 200 East Area, and the Gable Mountain Pond to the north of 200 East

Area. These mounds are now declining, but they will continue to influence the groundwater flow pattern for several decades (Chiaramonte et al. 1997; Cole et al. 1997). Other than their influence on groundwater velocities and the direction groundwater plumes have moved, the large releases to ponds have not significantly affected water quality.

Several significant discharges during the early operation period were made to reverse wells. Some of these facilities discharged into the vadose zone above the water table; some discharged directly into the water table. Plumes associated with the reverse wells, notably that associated with the 216-B-5 reverse well, exhibit significant levels of radionuclides either in the water table or deep in the vadose zone. However, radionuclides in these discharges such as strontium-90 are highly sorbed and have exhibited minimal contaminant migration in the past 50 years. Strontium-90 and other highly sorbed radionuclides (cesium-137 and plutonium-239/240) presently at these retired reverse well sites are forecast to remain inside the exclusive waste management area and buffer zone. The use of reverse wells was virtually discontinued very early in the Hanford Site operations. Reverse wells completed into the vadose zone were used after the mid-1950s at only two locations, the U Plant and the hot semi-works (Law and Lu 1982). Both of these disposals involved relatively low volumes and low inventories compared to other similar facilities. Use of the reverse wells at the U Plant and hot semi-works was discontinued by 1970 and 1988, respectively (DOE 1996a).

In comparison to ponds, smaller but still relatively large discharges were made to cribs and specific retention trenches. These discharges contained significantly greater radionuclide inventories than the cooling water discharges. Some of these discharges were tank wastes. Waite (1991) estimated these wastes contained 930 Ci of technetium-99 and 1.8 Ci of iodine-129. In the base case, the technetium-99 inventory was scaled down because of inconsistencies between estimates of technetium-99 released to the water table and existing in groundwater plumes, and only 167 Ci of technetium-99 were discharged to cribs and trenches. Some of the larger-volume discharges of these wastes were made to cribs that discharged to the aquifer. In an effort to contain discharged liquids permanently in the vadose zone, smaller-volume discharges of these wastes were made to specific retention trenches. This type of facility was designed to avoid discharges to the aquifer, however, the design was based on the assumption there was no recharge in the deep vadose zone deposits of the Hanford Site. Research and field observations have shown that under a variety of conditions this assumption is not true. Therefore, based on soil physics and contaminant transport theory, the present analysis forecasts releases from these facilities (i.e., specific retention trenches) to the water table.

The precise position and mobility today of wastes beneath liquid discharge facilities is not well known. Thus, analyses of liquid discharges to the aquifer are highly uncertain. However, existing plumes in the groundwater are a result of liquid discharges. Furthermore, remnants of these wastes that remain in the soil column are likely to be deeper in the vadose zone profile than dry wastes originally disposed in relatively shallow trenches. Many liquid waste discharges were very acidic or very basic waste streams, and therefore, they may be under geochemical conditions more favorable for migration than neutralized solid wastes in dry and relatively shallow disposals. Consequently, forecasts of

relatively early releases from liquid discharge sites are credible. The characteristic of relatively early release separates releases to the aquifer by liquid discharge sites from those of recent and future dry disposals that will release much later.

In all cases, existing plumes are the result of relatively large liquid discharges to the subsurface. These releases to the water table have occurred from ponds, reverse wells, cribs, and specific retention trenches. The Composite Analysis illustrates that, depending on the mobility of the nuclides released and the quantity of liquid discharged, inventories retained in the soil column at these sites will continue to leach into the groundwater for decades. However, those contaminants in the aquifer today are a result of the early discharge of large quantities of liquid waste or direct injection at reverse well sites. Some discharges were virtually continuous, and others were periodic. Some of these discharges were tank wastes or first derivatives of tank waste that contained significant concentrations of key radionuclides. Consequently, the resulting plumes had relatively high concentrations, and they continue to exhibit relatively high peak values today despite years of groundwater transport, radioactive decay, and dispersal.

This is illustrated by the more recent modeling of the tritium plume (Cole et al. 1997). This modeling simulated the tritium plume buildup and migration from 1979 to 2100. Effects of tritium discharges to ground prior to 1979 were accounted for through the simulation's initial conditions that were based on monitoring well measurements made in 1979. It also included both the projected future discharge of ~1000 Ci of tritium at the State-Approved Land Disposal Site (SALDS) starting in 1996, and all known past large liquid discharges to ground during the 1979 to 1996 time period. These past liquid discharges included ~30,000 Ci of tritium (1979 to 1996). The majority of the tritium disposal (~24,000 Ci) during the 1979 to 1996 time period occurred during 1984, 1985, and 1986 at 216-A-10 crib site as a result of PUREX operations. Tritium and liquid discharges to the 216-A-10 crib during this three-year period averaged ~12000 Ci per year and ~400 m^3/day respectively. These rates are orders of magnitude higher than any predicted future release rates to groundwater.

Future releases to the aquifer from the liquid discharge sites, tank leaks, tank losses, and burial grounds will occur, but with a greatly diminished driving force as compared to the past releases, because the future leaching and movement is driven by natural recharge rates, not large liquid releases. Even though more curies of specific radionuclides like technetium-99 will leach into the aquifer in the future than are present today, they will be introduced at lower rates. Since the general magnitude of groundwater flow in the aquifer flowing under these various sites will remain relatively constant through time, these lower projected release rates from the sources will create plumes with lower peak concentrations. Consequently, this analysis has shown that future doses through the time of Hanford Site closure and beyond will be dominated by the existing plumes of tritium and iodine-129. Because the total curies of tritium presently disposed to ground are far greater than for any other nuclide, the tritium in existing plumes will dominate dose estimates until it either discharges to the river or decays away. Order of magnitude estimates for tritium in the 1973 plume (ERDA 1975) indicated that it might contain as many as 35 million Ci. However, this estimate was based on the assumption that the tritium concentration was uniform over the entire thickness of the unconfined aquifer (i.e., from the water table surface to the base

of the unconfined aquifer). The recent plume modeling assessment (Cole et al. 1997) assumed the thickness of the initial 1979 tritium plume was ~25 m. As a result the total curies of tritium remaining in the plume in 1996 was estimated to be ~160,000 Ci. While this is a large number of curies, it is significantly less than the U.S. Energy Research and Development Administration estimate (ERDA 1975). As the tritium concentrations are reduced by migration to the river, dispersion, and decay, the iodine-129, which is assumed to be less mobile, then begins to dominate dose because of its very high dose conversion factor.

5.2.2.3 Past Tank Leaks, Future Tank Losses, and Tank Residuals

Data on release volumes and leak dates from Hanlon (1997) were augmented with waste concentration data gathered from tank characterization reports to provide a basis for the simulation of past tank leaks in the Composite Analysis. Release and vadose zone model parameters have been adjusted to qualitatively match the recent releases (e.g., ~5 Ci of technetium-99 from tank leaks is estimated to have reached the aquifer by 1996). The remainder of these releases is forecast to occur over the next century; however, its contribution to dose outside the buffer zone is relatively small.

Plans are now being made to recover tank wastes from the single- and double-shell tanks. Between 2003 and 2020 all of the single-shell tank wastes will be recovered. While the decision has not been made as to the methodology to be applied, the Composite Analysis assumed the sluicing method of tank waste retrieval was applied as described and analyzed in the TWRS environmental impact statement (EIS) (DOE and Ecology 1996). While the TWRS EIS analyzed consequences of an average 4000-gallon loss from each single-shell tank, the Composite Analysis uses the current estimate of 8000 gallons per tank. The model applied to tank losses is identical to that applied to tank leaks. These losses, like the tank leaks, were simulated as migrating through the vadose zone and releasing to the aquifer over the next century with most of the release coming before Hanford Site closure.

Finally, tank residuals, estimated as 1% by volume of current tank contents, are assumed to remain in the remediated and stabilized tanks. As in the TWRS EIS, the tank structure and remediation were assumed to protect the residual from leaching for 500 years. The Composite Analysis indicated this waste would not release into the unconfined aquifer during the 1500-year period analyzed.

5.2.2.4 Pre-1988 Solid Waste Burial Grounds

There are no published analyses of future waste migration from the pre-1988 burial grounds. The response of wastes disposed in these burial grounds may be much different than that of the post-1988 wastes in similar facilities. In the Composite Analysis, burial grounds without permanent surface barriers were assumed to be leached by recharge rates indicative of covers of coarse soils maintained free of vegetation (75 mm/yr). Under this assumption, some older burial grounds could experience more than 50 years of relatively high recharge and leaching. During the time of their operation, it is known that containers were less substantial and more susceptible to leaching than waste containers used today.

Consequently, this analysis indicates releases from these facilities could begin in the near future, peak in the next few decades, and continue at low rates over the next century.

5.2.2.5 Graphite Cores

The graphite cores of the production reactors are to be transported to the 200 West Area for disposal in the burial grounds (ROD 1993). These cores have unique features: they contain the greatest estimated chlorine-36 inventory on the Hanford Site, and they were identified in the first iteration of the Composite Analysis as the only waste with the potential to make a significant atmospheric pathway contribution to the all-pathways dose.

Tritium and carbon-14 inventories were analyzed for vapor-phase migration upward and lateral transport in the atmosphere. The calculated atmospheric release yielded only a minor contribution of approximately 0.4 mrem in a year to the all-pathways dose from this atmospheric release. This dose was the result of soil contamination that included the continuous buildup of contamination in surface soils over the full 1000-year period following Hanford Site closure. With regard to the timing of maximum dose, the maximum contribution from the atmospheric pathway would not superimpose on maximum doses from groundwater contamination since the latter occur at the time of Hanford Site closure. With regard to the spatial location of maximum dose, the atmospheric and groundwater pathways are also separated because the maximum contribution from the soil-atmospheric pathway occurred on the western edge of the 200 West Area. It will not superimpose on maximum groundwater contamination points to the east and south of the exclusive waste management area and buffer zone. A lower dose from the atmospheric pathway at the buffer zone boundary would result from placing the production reactor cores in the center of the western portion of the 200 West Area rather than near the boundary of the exclusive waste management area. Because of the assumed placement of a Hanford Protective Barrier or equivalent cover over the graphite cores, they are shown to not release to the groundwater aquifer during the time period analyzed in the Composite Analysis.

5.2.2.6 Chemical Separation Plants and Associated Facilities

For the first iteration of the Composite Analysis insufficient inventory and waste-form data prevented a truly credible analysis of the major facilities including the chemical separation plants or canyon buildings and their buried filters. No inventory exists for most key radionuclides, (e.g., carbon-14, selenium-79, technetium-99, and iodine-129) in these facilities. Typically, inventories for only cesium-137, strontium-90, uranium (total), and plutonium (total), are provided. The physical and chemical forms of these wastes are also poorly defined. Programs responsible for the cleanup of the facilities often describe the waste as fixed in place and immobile.

With respect to these facilities, the Composite Analysis results are preliminary. Calculations were performed to demonstrate that the massive inventories of cesium-137 and strontium-90 in B Plant and in its sand and high-efficiency particulate air (HEPA) filters would not contribute to releases to the unconfined aquifer within the next 1500 years. In the calculation, wastes in the B Plant canyon building

and HEPA filters were assumed to reside in a cementitious material, (i.e., the concrete floor and a grout matrix, respectively). The sand filter was assumed to be a simple sand deposit. Both waste deposits were assumed to be protected from infiltration by a Hanford Protective Barrier. Applying the most conservative distribution coefficients to describe adsorption, this analysis demonstrated no release of radionuclides to the unconfined aquifer over the next 1500 years.

5.2.2.7 Commercial Low-Level Radioactive Waste Disposal Facility

The commercial low-level radioactive waste disposal facility operated by US Ecology has unique features and inventory aspects. Located southwest of the 200 East Area inside the exclusive waste management area, this facility uses deep unlined trenches to dispose of commercial LLW. These trenches are excavated in a thick deposit of sand and silt. The inventory for this facility contains the second largest amount of chlorine-36 onsite, 34.4 Ci, and a significant inventory of uranium, greater than 10,000 Ci. This facility also contains 5.77 Ci of iodine-129 and 65.6 Ci of technetium-99.

In the Composite Analysis, chorine-36 is predicted to release in the near-term period prior to Hanford Site closure. However, the maximum contribution of chlorine-36 to the all-pathways dose from all sources will be less than 1 mrem in a year at the boundary of the buffer zone. With the assigned distribution coefficients of 3 and 0.5 mL/g, uranium and iodine did not release to the water table during the 1500-year period of the Composite Analysis. The analysis indicated ~ 1% of the original technetium-99 inventory will release to the water table in the 1000-year period following Hanford Site closure.

5.3 ALARA Assessment

The Composite Analysis indicates an all pathways dose well under the 30 mrem EDE in a year level that would trigger the need for a full and detailed options analysis and ALARA assessment of alternate remedial actions. A brief qualitative ALARA assessment is provided to evaluate the potential value of a more detailed analysis of alternatives.

The first iteration of the Composite Analysis has demonstrated that groundwater contamination at the Hanford Site will undergo two distinct episodes in the future. The first is more severe than the second and involves releases from numerous liquid discharge sites, tank leaks, tank losses during waste recovery operations, and past-practice solid-waste burial grounds. This first episode began with Hanford Site operations and will continue through 2050, the assumed date of Hanford Site closure. The dose prediction for the base case and the agricultural exposure scenario in 2050 is ~5.5 mrem in a year to the maximally exposed offsite individual (Figure 4.35). The dose predictions continue to decline through 2150, 100 years after Hanford Site closure, and are in the neighborhood of 4 mrem in a year to the maximally exposed offsite individual at that time. The second episode begins with releases from the post-1988 solid waste burial grounds and extends well beyond the 1500 years analyzed in this Composite Analysis. Earliest releases are predicted about 200 years from present and may not occur until very near the end of the 1000-year regulatory period. Dose predictions from the base case for the second episode,

which yielded first release in approximately 200 years, are less than 4 mrem in a year to the maximally exposed individual outside the exclusive waste management area and its buffer zone. The dose projection establishes a plateau at ~3 mrem in a year during the middle of the 1000-year period before falling to less than 2 mrem in a year at the close of the period.

The unconfined aquifer underlying the Hanford Site has a low capacity and cannot support extensive irrigated agriculture. Its recharge is limited by being in the rain shadow of the Cascade Mountains, and it is not fed by recharge from upland areas that receive substantial precipitation. The Cold Creek and Dry Creek Valleys and Rattlesnake Mountains are the origins of this aquifer. The aquifer flows to the east and north from its sources and discharges into the Columbia River. The Yakima River borders the aquifer to the south, but otherwise plays a negligible role, especially with regard to that portion of the aquifer that underlies the 200 Area Plateau. Contamination from the exclusive waste management area will enter the aquifer from above over a very small portion of the land area of the aquifer. Thus, relatively few groundwater wells placed immediately downgradient of the buffer zone boundary would be able to withdraw contaminated groundwater from the aquifer, and correspondingly few individuals would be exposed to the contamination.

A small family farm would require on the order of 1.8×10^4 m^3 of water each year. This is based on the rate of groundwater usage (150 L/m^2/month for 6 months; Appendix F) from the agricultural scenario, and a family farm of 2 hectares (Kincaid et al. 1995). Thus, if a single-family farm supported an average family of 5 individuals, the aquifer would be required to supply 3.6×10^6 m^3 of water each year to support a population of 1000 people.

Assuming that existing industrial discharges will be discontinued before Hanford Site closure, and that the groundwater system upgradient and beneath the 200 Area Plateau will come to an approximate steady state soon thereafter, one can approximate the groundwater discharge in the vicinity of the exclusive waste management area and buffer. Only a fraction of the flow of the unconfined aquifer passes beneath the 200 Area Plateau. A crude estimate of this quantity is given by the sum of groundwater entering the aquifer from the Cold Creek Valley, through the northern segment of Dry Creek Valley, and as natural recharge upgradient of the waste management area. Cole et al. (1997) estimated groundwater flux crossing the Cold Creek and northern Dry Creek boundaries as 1.05×10^6 m^3/yr and 4.41×10^5 m^3/yr, respectively. The land area upgradient of the site represents less than 25 percent of the Hanford Site. Twenty-five percent of the natural recharge to the site is ~2.12×10^6 m^3/yr (Fayer and Walters 1995). This represents an estimate of all contributions to the aquifer upgradient of the exclusive waste management area and buffer zone, and is ~3.6×10^6 m^3/yr. Not all of this water resource would pass beneath the exclusive waste management area and buffer zone, and become contaminated. Thus, it is an overestimate. This analysis suggests that approximately 200 family farms and 1000 people could be supported by the unconfined aquifer immediately downgradient of the exclusive waste management area and buffer zone.

By making the following assumptions, the potential value of a full ALARA assessment can be appraised.

- The period of interest is the 1000 years following loss of institutional control (DOE 1996b).
- The agricultural scenario yields the greatest dose and should be the basis for the long-term population impact assessment.
- A population of fewer than 1000 could consume Hanford Site groundwater and be exposed to its water quality.
- The exposure would continue for 1000 years without detection and remediation.
- A range of between $1000 and $10,000 per person-rem captures the cost to society from dose (DOE 1996b).

The dose estimate for the agricultural scenario shows a continual decline following present day maximums and is ~5.5 mrem in a year in 2050. It drops to less than 2 mrem in a year after 1000 years. During the 1000-year period, 4 mrem in a year is a reasonable yet high average value for dose from the agricultural scenario. This representative individual dose applied to 1000 people for 1000 years results in a 4000-person-rem population dose. The resulting cost to society would range between $4 million and $40 million. This cost does not justify a more detailed ALARA assessment because the cost to society of further analysis and implementation of alternatives would likely be equal to or greater than this amount.

In addition to not being justified on a cost/benefit basis, it is important to note that a more detailed ALARA assessment involving a variety of remediation options could not be performed at this time. Aside from the analyses for the four low-level waste disposals to which this Composite Analysis is a companion, the other DOE sites included in this iteration of the Composite Analysis are subject to remediation under the CERCLA and RCRA programs at the Hanford Site. Analyses of sites and their alternate remedial actions completed under these programs are being and will be conducted jointly with representatives from Ecology and the EPA. Future iterations of the Composite Analysis will involve DOE, EPA, and Ecology, and if necessary, will include evaluations of alternate remedial actions.

5.4 Suggestions for Further Study

Improved confidence in the second and subsequent iterations of the Composite Analysis will come from improvements in a number of areas. Based on the experience gained in the first iteration, the most fruitful areas for improvement are the inventory, waste handling and engineered barriers, environmental mobility and models, and inclusion of additional sources.

5.4.1 The Inventory

Much has been accomplished in the past two decades to document the inventories of radionuclides and chemicals present at the Hanford Site. Process knowledge and waste transfers have been

documented. The Track Radioactive Components (TRAC) model was developed (Jungfleisch 1980, 1983) and has been superseded by the Hanford Defined Waste (HDW) model (Agnew et al. 1997). The HDW model uses all available information; however, its development has been driven by the need to estimate the contents of single- and double-shell tanks. As mentioned above, some inventory entries are conservative estimates.

While appropriate for individual programs, conservative or bounding estimates are not as useful for the Composite Analysis. For the Composite Analysis, a conservative assessment implies a sequence of events that cause multiple plumes to arrive simultaneously at a point in space and moment in time. Performance assessments differ from the Composite Analysis because the former examine contaminants from a single source passing a single point in space. By using a conservative inventory estimate, one maximizes the dose consequence. This is not true of the Composite Analysis unless conservative estimates of inventory are used for all sources.

The inventory for the Hanford Site should be viewed in a holistic sense as a conserved quantity. Ideally, each nuclide has a known inventory for the Hanford Site based on the quantity imported or generated in the reactors. For those sites with potentially significant releases to the water table, it is important to examine the tradeoffs of inventory uncertainty. A greater inventory assigned to liquid discharge sites should correspond to a smaller inventory assigned to existing tank waste and future ILAW disposal. An estimate of the inventory emitted to the atmosphere should be accounted for in the overall inventory. That portion recovered by scrubbers and disposed at the Hanford Site should also be traced through the inventory to its final disposition. Ultimately, the most meaningful uncertainty analysis of inventory would be based on a best-estimate rather than bounding estimate.

Finally, if a full options analysis and ALARA assessment must be completed to evaluate alternate remediations, the Composite Analysis is used to identify those disposals most responsible for the dose. Alternate remediations must be proposed and studied for the wastes having the greatest impact rather than others of less significance. If bounding inventories have been used in the Composite Analysis, the analysis may need to be redone prior to proceeding with the options analysis and ALARA assessment.

Thus, the sitewide inventory assembled for the second iteration Composite Analysis should be a balanced and best estimate. The estimate should be balanced in the sense that gaseous, liquid, and solid waste inventories should be accounted for, should be consistent, and should be linked. This would enable the generation of sensitivity cases that examine the implications of a greater inventory lost to the atmosphere or sent in the liquid waste streams to cribs or tanks. The estimate needs to be centered about a best estimate that places waste where it is most likely to reside at the conclusion of Hanford Site operations. Sensitivity to inventory estimates could be analyzed as independent realizations that would be created by routing more or less waste to the atmosphere, to the liquid discharges sites, to the single- and double-shell tanks, and to the solid waste burial grounds. Reviewed and accepted methods of estimating the key mobile radionuclides of greatest importance to long-term health and safety studies should be incorporated into the inventory model.

Such an inventory should be based on the HDW (Agnew et al. 1997) or a similar model of the Hanford Site inventory and the standard or best-basis inventory of Kupfer et al. (1997). It would then be possible to examine perturbations in the inventories assigned to specific waste disposal facilities. The assignments would be conditioned on the knowledge of processes and constrained by the knowledge of waste transfers. The influence of assumptions could be traced through the inventory estimates. For example, the assumed split of iodine-129 between gaseous and liquid phase, and the assumed effectiveness of silver-nitrate saddles in removing the iodine from the gaseous phase could be traced through to their resulting inventories assigned to the atmosphere, the solid waste landfills, and the liquids stored in tanks. Such a model would remedy the present issue of full accountability for the final disposal of key radionuclides including carbon-14 and iodine-129.

5.4.2 Waste Handling and Engineered Barriers

A major finding of the first iteration Composite Analysis is the separation in time of two release episodes. The first is the result of liquid discharges and tank leaks, and the second is the result of dry disposals. Confidence in this finding relies on the waste and its protective barriers, and estimates of contaminant migration and fate in the vadose zone. To a significant extent, confidence that dry disposals since 1988 will not release to the water table for hundreds of years relies on our confidence in engineered waste forms and barriers to infiltration and leaching. The following assumptions were made.

- Any large contributions to the key mobile nuclide inventories of the solid waste burial grounds will be detected prior to acceptance of the waste, and such a waste would be placed in a high-integrity waste form (e.g., mixed with a waste form material such as grout), or placed in a high-integrity container.

- Engineered systems such as the double liner and surface barrier of the ERDF will function to specifications.

- Engineered surface barriers placed over other wastes will perform to their design standards.

- The TWRS ILAW will meet performance specifications that have been the basis of its simulation in this analysis.

Confidence in the results of this and future Composite Analyses depend on efforts that justify the assumptions regarding the waste handling protocols, waste form performance, engineered barriers, and infiltration rates.

Increased confidence in long-term aspects of contaminant release and migration implies greater confidence in the performance of surface covers and protective barriers. Covers and barriers are included in disposal facility design to control or limit a number of impacts including intrusion by plants, small mammals, and humans, and especially the infiltration of water into the waste. Not all wastes will require the same cover or barrier. Consequently a graded approach to barrier design is needed. An

understanding of the performance of the various design features or components of typical covers and protective barriers will enable DOE programs to incorporate into their designs only those cover features essential to the long-term performance of their waste. Studies should quantify the roles of surface soils, capillary interfaces between layers, climate variability, and plant dynamics in determining infiltration through the cover. Because barriers are assumed to function for decades and centuries, studies should seek to quantity the long-term durability of the components of typical covers and protective barriers. Studies should also quantify the potential for water to move laterally from the edge of the cover or barrier toward the waste form. This redistribution of water beneath the cover system may result in leaching of deep waste deposits including liquid discharge sites. Studies may show that covers have an influence over a finite depth, and that their ability to reduce infiltration rate or recharge in the deep vadose zone is mitigated by the layering of natural sediment deposits that act to spread surface infiltration laterally. Certainly, it will be important to fully understand and quantify infiltration rates applicable both before and after final covers are applied to waste sites.

5.4.3 Environmental Mobility and Models

The review of the 241-SX-109 tank leak experience (DOE 1997a) has placed previously accepted vadose zone conceptual models in question. Ongoing field studies with the purpose of developing better information on the physical extent and chemical mobility of tank wastes leaked to the subsurface are underway. This knowledge will enhance our ability to quantify the environmental response of liquids discharged to specific retention trenches and lost from tanks during waste recovery operations.

Through field study of leaks from tanks and discharges to cribs, the pathways and mobility of contaminants will become better understood. Based on the field evidence and our knowledge of the waste, alternate conceptual models of waste migration and fate in the subsurface can be posed. Conceptual models capture the physical features and physicochemical processes that produced the observed situation. They can be further studied through numerical simulation, and the alternate explanations of events can be narrowed to the few or one that best explain all of the field observations.

Issues that require resolution include explanations for the initially high mobility of some wastes and an evaluation of their ability to create or follow preferential flow paths. Another issue involves the ability of barriers to prevent leaching of the waste and ensure a slow flow and transport path to the water table (i.e., quantification of the edge effect of a surface barrier). Of particular importance to the second iteration Composite Analysis will be the development of confidence in estimates of long-term leaching and migration of wastes from dry disposals. The behavior of these dry disposal sites is difficult to study in the field because of the low release rates of dry waste. Under a dry regime, the migration of the release is less likely to find and then follow geologic formations (e.g., interfaces between coarse- and fine-grained sediments) that may represent preferential flow paths under wetter or saturated conditions. The release and migration will require special attention because they occur in a much drier regime than at the liquid disposal sites. Because of the time they have been exposed to potentially greater infiltration rates, the pre-1988 solid waste burial grounds may provide an opportunity to measure release and migration from burial grounds under less-than-optimal conditions.

Our increasing knowledge of the physical position and chemical character of radionuclides in the vadose zone beneath tank leaks and liquid discharge facilities should be incorporated into the conceptual and mathematical models. By necessity, a one-dimensional model of the vadose zone was employed to simulate the numerous waste sites within the exclusive waste management area in the first iteration Composite Analysis. The greater understanding of contaminant migration in the vadose zone that will come from the ongoing and future vadose zone studies will either lead to the creation of more comprehensive, applicable, and accepted one-dimensional models or point to the need to perform multi-dimensional simulations of specific facilities or wastes. Certainly, the decision to proceed with the development and application of more sophisticated vadose zone transport models will be based on the perceived value of their predictive capability. An evaluation of their potential value may be approached through simulations with simpler models tailored to bound the potential impacts of the unresolved processes (e.g., multiphase physics, aqueous speciation, adsorption, precipitation) and geometries (e.g., two- or three-dimensional phenomena, preferential pathways) of a more sophisticated model. Studies may also conclude that probabilistic models are required. Regardless, completion of these studies and the implementation of the next generation models will lead to greater confidence in future iterations of the Composite Analysis.

5.4.4 Inclusion of Additional Sources

Numerous liquid discharge sites and canyon facilities were not modeled in this iteration of the Composite Analysis. In the case of liquid discharge sites, this is only justified by the belief that the most significant releases have been estimated and therefore included in current inventories. However, as inventory estimates are created for liquid-release and leak sites and canyon facilities not included in the first iteration, they will be included in future iterations of the Composite Analysis. The canyon buildings, their immovable underground filter assemblies, and the PUREX tunnels are another group of sources that need to be included in future analyses. The potential impacts of cesium-137 and strontium-90 in the B Plant and in its sand and HEPA filters were included as a preliminary analysis of these major radionuclides. However, the inventory and location of the key mobile radionuclides in these structures need to be developed as the basis of a credible analysis of their potential impact.

5.4.5 Use of Data Quality Objectives

During this first iteration Composite Analysis, the concept of Data Quality Objectives (DQOs) as applied to a simulation-based analysis has been examined. Because the analysis was to use only information already at hand (DOE 1996b), the DQO process in this first iteration analysis became a data-acceptance or data-qualification process. While constrained in this iteration to not gather samples for analysis and use off-the-shelf information and capabilities, subsequent iterations will be expected to apply the DQO process. In light of the iterative character of Composite Analysis, the role of DQO in the next and subsequent iterations is important to understand.

The standard DQO process was developed in response to a need to define the quantity and quality of characterization data required for decisions at CERCLA and RCRA cleanup sites. As applied to a field

characterization problem, the standard DQO process yields the sample size and equipment quality selection criteria that will meet the needs of the decision maker. This is achieved by balancing the risk and cost of making a decision error, against the cost of increasing the number of samples and achieving greater confidence in the field characterization.

This standard DQO approach can not be directly applied to the dose or risk forecast problem. The future state of the system is unknown and unknowable. Consequently, where in the field sampling problem one could completely sample the site and know the truth, there is no ability to know the true future. In the modified DQO process, the problem is one of balancing the cost of increasing model confidence or reducing uncertainty, against the cost or consequences of making an incorrect decision based on an incorrect model forecast. The poor decision may be to cleanup a site when it is not necessary, or to leave a site unremediated that deserves cleanup. A fully consistent DQO-based program would require a probabilistic analysis that considered sources of uncertainty in the inventory, release, vadose zone migration, aquifer migration, exposure mechanism, and dose or health consequence. Furthermore, the analysis would need to consider the propagation and compounding of uncertainty throughout the sequence of calculations. It is unlikely that sufficient data or resources exist to perform this analysis.

The second iteration Composite Analysis will gather information on the range and distribution of data. This information will enable an analysis of the sensitivity of the results to the range of estimated inventory, source term release, and environmental response. Subsequent Composite Analyses may use a probabilistic methodology and address the full uncertainty analysis. Prior to undertaking either sensitivity or uncertainty analyses during the second iteration Composite Analysis, it will be important to establish a baseline confidence in all elements of the analysis including the inventory, the release models and data, and the models and data for the vadose zone, groundwater, and exposure pathways.

5.4.6 Linkage to the Hanford Groundwater Project, the 200 Area Characterization Program, and the TWRS Hanford Tanks Initiative

Field observations must play a greater role in determining the base case conditions at the Hanford Site. Existing plumes in the vadose zone and groundwater are evidence of contaminant release and mobility. In response to CERCLA and RCRA guidance and DOE Orders 5400.1 and 5400.5, a groundwater protection management plan is routinely issued (Barnett et al. 1995) for the groundwater resources at the Hanford Site. This plan describes the ongoing monitoring efforts. Before the next iteration of the Composite Analysis is completed, efforts should be made to include in this plan the work necessary to sample the aquifer for more of the key mobile radionuclides identified in the Composite Analysis including selenium-79 and chlorine-36. Similarly, efforts should be made to determine the distribution of key mobile radionuclides in the vadose zone beneath cribs, specific retention trenches, reverse wells, and tank farms.

Special efforts should be undertaken to sample groundwater and characterize the vadose zone in the vicinity of liquid discharge facilities (cribs, specific retention trenches, and reverse wells) and tank leaks

that received the largest quantities of waste having large inventories of key mobile radionuclides. Recently identified radionuclides such as selenium-79 and perhaps chlorine-36 should be added to laboratory analyses of water samples. In part, such efforts should attempt to substantiate the estimate of inventory (mass) and contaminant concentration discharged to the environment.

The sampling strategy should be designed to yield results suitable to provide an estimate of the mass of contaminant in the aquifer or vadose zone. Point samples taken at moments in time are prone to miss peak concentrations. Sampling a substantial interval of aquifer provides an integrated sample biased toward the water quality of the most conductive strata intercepted by the sample. Sampling short intervals over the depth of the vadose zone or aquifer borehole could provide valuable insight on contaminant distribution. Analysis of small intervals could identify sediment layers responsible for adsorption or precipitation phenomena in the vadose zone. Such an analysis of saturated sediments would begin to reveal the three-dimensional distribution of contaminants throughout the aquifer. Methods of estimating the contaminant mass, including a best estimate and range, are needed for comparison to model results of key mobile radionuclide discharges. Inventory estimates of key radionuclides discharged to cribs and leaked from tanks should be conditioned by our knowledge of the mass of those radionuclides found in the vadose zone and aquifer.

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Appendix A

Solid Waste Inventories

Appendix A

Solid Waste Inventories

G. A. Whyatt and C. T. Kincaid

A.1 Introduction

This appendix provides radionuclide inventory values for solid waste disposal. At the end of this appendix, Tables A.1 through A.6 present the inventories by burial ground. The data for actinides and their daughters are provided in a table separate from fission and activation products. The six waste categories are listed and described as follows:

dec96 — This inventory table includes total unsegregated and post-1970 segregated non-transuranic (non-TRU) solid waste disposal from the start of operations through December 1996. To ensure this original table is retained without revision, a new table "new_dec_96" (Table A.1) was created and used to develop the pre- and post-1988 inventories.

sept88 — This inventory table includes total unsegregated and post-1970 segregated non-TRU solid waste disposal from the start of operations through September 1988. To ensure this original table is retained without revision, a new table "new_sept_88" (Table A.2) was created and used to develop the pre- and post-1988 inventories.

New_96_88 — This inventory table (Table A.3) is the difference between the dec96 and sept88 tables and represents the inventory disposed after September 1988 to which U.S. Department of Energy (DOE) Order 5820.2A applies. This table replaced an earlier version, i.e., the "post_sept88" table, which was not based on the most current "dec96" and "sept88" tables.

Suspect_TRU — This inventory table provides estimates of radionuclides that are contained in suspect transuranic (TRU) waste that are expected to be reclassified from TRU to low-level waste (LLW) and be disposed of onsite. This inventory represents waste disposed as TRU between 1970 and 1986 using a greater than or equal to 10 nCi/g definition of TRU. It is anticipated that some of this waste will be reclassified as LLW and disposed onsite because of the current definition of TRU (i.e., greater than or equal to 100 nCi/g). A copy of this table named "new_suspect_TRU" (Table A.4) is used to develop the pre- and post-1988 inventory estimates.

New_future — This inventory table (Table A.5) provides a projection of potential future disposal based on a linear projection of disposal occurring after September 1988. The "new_96_88" table is multiplied by the factor "360 (months) / 99 (months)" to estimate the inventory disposed during the proposed 30-year operational period that began in 1988. This is the estimated post-1988 inventory that is regulated under DOE Order 5820.2a.

New_pre_1988 — Table A.6 is simply the sum of the "new_sept_88" and "new_suspect_TRU" spreadsheets and provides an estimate of the total LLW inventory disposed in the pre-1988 burial grounds. These burial grounds will be closed under the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) program.

A.2 Explanation of Basis for Solid Waste Inventories

Inventories of radionuclides disposed in each of the 200 East and 200 West solid waste burial grounds were estimated and are provided in Tables A.1 through A.6 at the end of this appendix. The inventory data are derived from the Solid Waste Information Tracking System (SWITS) database (Clark 1995). The Oak Ridge Isotope Generation and Depletion (ORIGEN2) code (Croff 1980; Wittekind 1989) was used to estimate the abundance of minor radionuclides that could potentially be present but are not be reported in the SWITS database.

Activities of cesium-137, and masses of uranium and plutonium disposed were obtained directly from the SWITS database. The following SWITS database reports were obtained on 1/4/96 to provide the inventory information:

- SWIR328D and SWIR328E: Unsegregated Waste Burial Ground Areas Waste Volumes Buried and Non-decayed Curie Content from Startup through September 30th, 1988. ("D" contains uranium and plutonium masses and "E" contains cesium-137 Ci)

- SWIR328G and SWIR328H: Post-1970 Non-transuranic Waste Burial Ground Areas Waste Volumes Buried or Stored and Non-decayed Curie Content through September 30, 1988. ("G" contains uranium and plutonium masses and "H" contains cesium-137 Ci)

The reports above were also generated for dates of startup through December 31, 1996. The inventories of uranium, plutonium, and cesium-137 disposed were totaled between the unsegregated disposal inventory and the segregated non-TRU inventory (thus excluding the TRU waste that is not expected to remain onsite). The startup through 12/31/96 (99 months) represents all the waste of interest disposed as solid waste. DOE Order 5820.2A applies to waste disposed after 9/26/88 so it was desired to

provide a separate inventory estimate for disposal after this date. The inventory of waste disposed after 9/30/88[a] was determined by subtracting the waste disposed through 9/30/88 from the total disposal through 12/31/96.

A.2.1 Suspect TRU Waste

Prior to 1970, TRU waste was not segregated before disposal. After 1970, TRU waste, which was defined as greater than or equal to 10 nCi/g, was segregated before disposal so that it could be retrieved and eventually be disposed offsite. In 1984, the definition of TRU waste was changed from >10 nCi/g to >100 nCi/g. Therefore, it is possible that some quantity of segregated TRU waste disposed between 1970 and 1984 may be reclassified as LLW and be disposed of on the Hanford Site.

It is most likely that the waste would be assayed within the disposal trench and never leave the trench in order to avoid changing the Resource Conservation and Recovery Act (RCRA) "disposed waste" classification. Thus, the waste is assumed to remain in the burial grounds where it currently resides and it is assumed that DOE Order 5820.2A would not apply to such wastes. Thus, the estimated inventory of TRU waste reclassified as LLW is added to the inventory of LLW through September 30, 1988 ("new_sept_88") to create the pre-September 1988 inventory.

Inventory values for suspect TRU waste were based on an estimate that 50% of drums and 15% of burial boxes would be reclassified as LLW and disposed of onsite. This waste volume was assumed to have a density of 200 kg/m^3 and contain 100 nCi/g of TRU radionuclides (i.e., the maximum possible value for waste that could be reclassified as LLW). The alpha activity calculated to be in the reclassified waste was then used to calculate the fraction of alpha activity in suspect TRU waste, which was presumed to be reclassified as LLW. This calculation indicates that roughly 0.3% of alpha activity in drums and 1.1% of alpha activity in boxes may be present in the reclassified waste. This fraction of the total alpha activity was then used to calculate the activity of each radionuclide in the reclassified waste by multiplying by the total estimated inventory of the suspect TRU waste.

A.2.2 Future Disposable Inventories

There is substantial uncertainty in the future solid waste disposal inventories because of the change in the mission of the Hanford Site from the production of special nuclear materials to the safe cleanup and management of the site's legacy wastes. A simple approach was used in which the inventory disposed between September 30, 1988 and December 31, 1996 was used to extrapolate for an additional 30 years of disposal assuming a constant rate of disposal. The inventory values were compared to projections made in the East and West Area Solid Waste Performance Assessments (Wood et al. 1995, 1996). In all cases, the linear extrapolation of waste disposal over 30 years exceeded the inventories assumed in the

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(a) The SWITS query was performed through September 30, 1988 before the precise effective date for DOE Order 5820.2A was determined. The 4-day discrepancy was not considered significant enough to warrant generating an additional set of reports.

solid waste performance assessments. The projected inventory values were associated with the disposal areas where waste disposal has occurred over the last 8 years, but actual future disposal may or may not occur in the same disposal areas.

A.2.3 Estimation of Non-Reported Radionuclides

While uranium, plutonium, and cesium-137 are relatively well reported within the SWITS database, a number of minor nuclides may also be present but are not consistently reported. Some of these minor radionuclides are identified as key nuclides in performance assessment calculations. In an effort to estimate inventories of minor radionuclides, ORIGEN2 was used to estimate the relative abundance of minor radionuclides compared to the major radionuclides that were reported.

ORIGEN2 calculations were made for single-pass reactor irradiations and for N Reactor irradiations to determine radionuclide concentrations in spent fuel and cladding. Impurities in the fuel and the cladding were included in the model. The quantities are based on Bergsman (1993), and are given in Table A.7. The brazing was also included in the model. It was assumed that the single-pass reactor fuel was all natural uranium as opposed to the actual situation where 25% of it was slightly enriched uranium. The average burnup of the single-pass reactor fuel was 728 MWd/MTU. It was also assumed that all of the N Reactor fuel was enriched to 0.947% uranium-235 when in fact some of it was of higher enrichment. The average burnup of the N Reactor fuel was 1045 MWd/MTU. The power density was assumed to be 10 MW/MTU for all of the fuel. For long decay times the radionuclide concentrations are insensitive to the power density. About 90% of the fuel reprocessed at Hanford was irradiated in the single-pass reactors. A weighted average between the single pass and N Reactor nuclide concentrations was used to estimate the overall average nuclide composition.

Inventories of potentially unidentified fission products in solid waste burial grounds were estimated by multiplying the undecayed cesium-137 inventory from SWITS by the ratio of the curies per kg concentration of the radionuclide of interest to the concentration of cesium-137 from the ORIGEN2 calculation at 10 years after discharge from the reactor. Estimates based on 1-year decay would be more conservative for radionuclides with half-lives less than that of cesium-137 (30 years) while estimates based on 10 years of decay prior to disposal are more conservative for radionuclides with half-lives of more than 30 years. The inventory estimates have been based on the fuel aged for 10 years after discharge from the reactor. In any instance, where the activity of a fission product increased over time beyond 1 year, the maximum activity between 1 and 3000 years was used to calculate the ratio to cesium-137 at 10 years.

The SWITS database reports provided information on uranium disposed, including both a mass of uranium that was not identified by isotope, and a quantity of uranium isotopes that were specifically identified. The ORIGEN2 results were used to divide the uranium that was not isotropically identified among the uranium isotopes and to estimate the quantity of other actinides (except plutonium) that might be present along with this uranium mass. The quantity of other actinides was estimated by multiplying the uranium mass reported in SWITS by the ratio of activity of actinide (or daughter) to uranium mass in discharged fuel at 1 year. Similar to the fission products, estimates are provided based on 10-year decay.

As in the case of fission products, the maximum actinide or daughter activity between 1 and 3000 years in the ORIGEN2 calculation was used to calculate the ratio to uranium mass.

For plutonium, the approach was similar to that used for uranium. Plutonium reported without isotopic distribution was divided into isotopes based on the relative abundances indicated in the ORIGEN2 calculation at 10 years. Quantities of plutonium that were reported in SWITS as specific isotopes of plutonium were then added to arrive at total plutonium isotopic values.

Chlorine-36 is a potentially important radionuclide that may be formed by the irradiation of chloride impurities in the fuel or cladding. No data on the chlorine-35 impurity levels within metallic uranium fuel were available and it is not known whether the impurity level was negligible. However, because of the uncertainty, a calculation was performed assuming a 1 ppm by weight impurity in the fuel. The chlorine-36 in waste was then ratioed to cesium-137 as for fission products. The purpose of including chlorine-36 in the inventory is to determine if the nuclide is potentially important to the results of the performance assessment. If the results indicate it is potentially important, then a more in-depth investigation into chlorine-36 may be justified. However, if results indicate it is many orders of magnitude below a level of concern, then additional effort may not be warranted.

The choice of using ORIGEN2-predicted ratios of nuclides in aged fuel to cesium-137 content at disposal was based on previous work by Wood et al. (1996). Their work provided a proposed breakdown of time after discharge at disposal by year in which the time after discharge at disposal varied between 1 and 10 years depending on the year of generation. For wastes disposed from 1945 through 1973, 1 year was determined to be appropriate. However, wastes disposed in the most recent years may originate from waste discharged from the reactor 10 years (or more) prior to disposal. Between 1 and 10 years after discharge, the cesium-137 inventory declines by about 20%. As a result, the inventories of long-lived fission products in some wastes estimates, based on cesium-137 content, were about 20% higher when the 10-year assumption was made. Overall, the difference between the 1- and 10- year assumption is probably small relative to the uncertainty related to using a ratio of nuclides in aged fuel to cesium-137 content that implicitly assumes that the isotopic ratios in the waste are similar to those in the discharged, irradiated fuel.

A.2.4 Specific Isotope Inventory Values Used from SWITS

The approach of estimating fission products based on a ratio of cesium-137 and estimating actinides based on a ratio to uranium is potentially in error. There are three possibilities: a high estimate, a good estimate, and a low estimate. The fundamental assumption being made when the fuel-ratio method is applied is that the abundance of fission products, cesium-137, actinides, and uranium in aged fuel is a good approximation to that in waste streams.

A high estimate of fission products or actinides will result whenever the proportion of cesium or uranium in the waste is high in comparison to the fission products or actinides. This could occur if the process creating the waste is biased toward the production of high quantities of cesium or uranium in the waste stream. Process steps in the separation of cesium or uranium from Hanford wastes could have a feed stream rich in cesium or uranium, and therefore, could yield waste streams with this characteristic.

Accordingly, the quantity of fission product or actinides in the waste stream would be overestimated by the aged-fuel-ratio method when the waste stream is rich in cesium or uranium. A low estimate of fission products and actinides will result when the opposite situation arises, i.e., the waste stream carries relatively low quantities of cesium and uranium. When a process feed stream is low in cesium or uranium because it was removed in a prior process step, waste streams could be higher in fission products and actinides than an aged-fuel ratio would indicate.

B Plant wastes reported to contain significant levels of cesium are likely to be associated with processes designed to separate cesium and produce feed streams and wastes disproportionately high in cesium. These feed streams and wastes would be disproportionately low in fission products. Accordingly, application of the aged-fuel-ratio methodology to estimate fission products would produce significant overestimates of the fission products in solid waste burial grounds. Similarly, wastes originating from the uranium extraction processes and reported to contain significant levels of uranium are likely to be disproportionately high in uranium. Feed streams and waste streams with high uranium contents may contain proportionately lower quantities of actinides. Application of the aged-fuel ratio would produce significant overestimates of the actinides in solid wastes. Certainly, examples resulting in low estimates of fission products or actinides are also possible. A detailed review of process waste steams produced during previous operations would be required to determine the true character of the error resulting from the use of the aged-fuel-ratio method to estimate fission products and actinides.

Such a detailed review was beyond the scope of the first iteration Composite Analysis. In the interim, the SWITS database was checked for curies of potentially key radionuclides (e.g., carbon-14, selenium-79, technetium-99, iodine-129, neptunium-237, and thorium-232) disposed in each solid waste burial ground. Where the SWITS database reported a larger inventory than projected, based on ratio to cesium-137 or uranium, the SWITS value was used for that burial ground. Furthermore, waste acceptance criteria are being revised to force waste generators to evaluate and report mobile radionuclides. This is being done to make the necessary inventory data available for future disposals, and eliminate the need to estimate key fission product and actinide inventories.

A.2.5 Special Inventory Items

Trench 94 of burial ground 218E-12B contains U.S. Navy ship reactor compartments. These wastes consist of activation products within corrosion-resistant metals, primarily Inconel Alloy 600. Because of the immobilized nature of the radionuclides within these activated metals, waste inventory identified in SWITS as being from offsite sources and disposed in Trench 94 of 218E-12B has been excluded from the inventory.

The SWITS inventory information for wastes from the 100 Area suggest that these wastes are activation products rather than fission products. While it is possible that the radionuclides exist in corrosion-resistant metals, this is not known for certain, as in the case of the reactor compartments. Therefore, the inventory of carbon-14 is reported separately for 100 Area wastes. Any non-100 Area wastes that were specifically identified as activated metal were included in the 100-Area carbon-14 inventory.

Trench 218W4C contains 74.8 Ci of carbon-14 and 14.62 Ci of technetium-99 which were disposed of in grouted containers. Based on performance calculations made prior to waste acceptance, this portion of the inventory is expected to be significantly less mobile than the inventory in other wastes and therefore was not included within the inventory tables. If desired, these radionuclides could be included in a separate release model and the source term added to the source term from the remainder of the burial ground.

A separate inventory prepared by Wood[(a)] was developed based on total beta-gamma measurements on waste packages. An assumed mix of fission products or activation products was then used to account for total beta-gamma not accounted for by cesium-137. In most cases this resulted in a smaller inventory than estimating using the ratio to cesium-137. However, in a few instances, a larger inventory was predicted. In these cases, the larger value predicted using the beta-gamma measurement was used in place of that predicted based on cesium-137.

Similarly, in comparing inventories to those produced by Wood,[(a)] some discrepancies that were traced to differences in SWITS database reports were noted. The reasons for the discrepancies were not determined but in each case, the larger inventory value was adopted.

A.3 Comparison to Previous Performance Assessment Inventories

Performance assessments were previously performed for the disposal of solid waste in the 200 West and 200 East Areas (Wood et al. 1995, 1996). This section provides a comparison between the currently recommended inventory to the inventory recommended in the performance assessments. This is provided for information only, since Wood no longer recommends use of the inventory values in those performance assessments. He recommends the use of future inventory estimates based on the longer period of record of disposals now available. The comparison for a few of the mobile radionuclides is provided in Tables A.8 and A.9.

Projected uranium inventories in future 200 West Area burial ground disposals are much higher in this document. These higher projected inventories for uranium are probably a direct result of recent disposals of large quantities of uranium associated with the cleanup of facilities, and the use of a simple linear assumption for forecasting future disposals. Carbon-14 values for burial grounds in both 200 East and 200 West Areas are larger in the current inventory. One possible explanation for the larger carbon-14 values is that the published performance assessments may have eliminated more waste on the basis that it was in activated metal and not available to be released. In this document, the values of the technetium-99 inventory for the 200 West Area burial grounds are also higher by an order of magnitude. Again, recent disposals and linear projections may account for the increase in estimates. In comparing the currently recommended inventory to the previous performance assessment inventories, it should be noted that waste disposal has occurred between the time of the previous performance assessment and the present, and therefore the bases for the two estimates are quite different.

(a) Electronic mail message regarding West Area Burial Ground Inventories. Sent by M. I. Wood of Waste Management Federal Services of Hanford to G. A. Whyatt on December 19, 1996.

A.4 Information on Radionuclides of Interest

This section provides information on the radionuclides within the inventory including half-life, sources, and inventories predicted by an ORIGEN2 calculation. All ORIGEN2 inventories are a weighted average of single pass and N Reactor fuels, and they represent the activity in irradiated fuel and cladding in units of curies per kg of uranium. Compositions of single-pass and N Reactor fuels and cladding used in the ORIGEN2 simulation are provided in Table A.7. These simulations and the averaging of results were performed prior to development and publication of the standard inventory (Kupfer et al. 1997; Watrous and Wootan 1997). The ORIGEN2 inventory values were used to calculate inventories for nonreported radionuclides.

A.4.1 Highest Mobility Nuclides

Tritium — $T_{1/2}$ = 12.3 years. For this nuclide, activation dominates fission by a factor of 1E+04. It is a negligible source in waste. ORIGEN2 calculation indicates the following inventories: 1 year = 7.385E-02 Ci and 10 years = 7.068E-03 Ci. Beta decays to helium-3 (stable).

Carbon-14 — $T_{1/2}$ = 5730 years. This nuclide beta decays to nitrogen-14 (stable). It is generated as both a fission and an activation product, although activation is the more important source. Activation is from various neutron interactions with nitrogen-14, nitrogen-15, oxygen-16, oxygen-17, and carbon-13. ORIGEN2 calculation indicates the following inventories: 1 year = 1.247E-04 Ci and 10 years = 1.246E-04 Ci. Sources other than the fuel and cladding may also be important and are not included in this value (e.g., production within graphite moderator).

Chlorine-36 — $T_{1/2}$ = 3.01E+05 years. This nuclide results from the activation of chlorine-35 impurity in fuel or cladding. Assuming a 1 ppm by weight contaminant level of chlorine in the fuel, the ORIGEN2 calculation indicates a 1-year inventory of 1.228E-07 Ci. 200 East Area performance assessments (Wood et al. 1996) estimated inventory only in connection with reactor compartments. The use of inventory based on 1 ppm contaminant level may allow determination if further investigation is warranted. Decays by beta to argon-36 or by positron emission to sulfur-36, both stable. Another potential source is from chloride contaminant in materials other than fuel as well.

Selenium-79 — $T_{1/2}$ = 6.5E+04 years. This nuclide is a fission product only. ORIGEN2 calculation indicates the following inventories: 1 year = 1.016E-05 Ci and 10 years = 1.015E-05. Beta decays to bromine-79 (stable).

Technetium-99 — $T_{1/2}$ = 2.13E+05 years. This nuclide is a fission product with minor activation source. ORIGEN2 calculation indicates the following inventories at 1 year: 3.434E-04 Ci fission product and 6.484E-11 Ci activation. Beta decays to ruthenium-99 (stable).

A.4.2 Moderately Mobile Nuclides

Iodine-129 — $T_{1/2}$ = 1.57E+07 years. This nuclide is a fission product with a minor activation source. ORIGEN2 calculation indicates the following inventories for 1 year: 6.774E-07 Ci fission product, and 1.149E-25 Ci activation.

Uranium-232 — $T_{1/2}$ = 68.9 years. This nuclide is a daughter of relatively short-lived parent plutonium-236 ($T_{1/2}$ = 2.85 years). ORIGEN2 calculation indicates an inventory at 1 year =3.93E-10 Ci with maximum inventory of about 4.3E-08 Ci at 10 years.

Uranium-233 — $T_{1/2}$ = 1.592E+05 years. This nuclide is an ingrowth daughter product, with long-lived neptunium-237 parent. ORIGEN2 calculation indicates an inventory of 3.93E-10 Ci at 1 year, still growing in at 1.03E-07 Ci at 3000 years. It is a potentially significant source if thorium-232 is used as fertile material [reactions are (n,γ), β^-, β^-] to produce fissile uranium-233. Thorium was used not only for targets for uranium-233 production but also for neutron shielding which could add inventories of uranium-233 depending on where the material is eventually disposed.

Uranium-234 — $T_{1/2}$ = 2.454E+05 years. This nuclide is a daughter within uranium-238 decay chain. Inventory in ORIGEN2 calculation is essentially constant at 3.49E-04 Ci from 1 through 3000 years.

Uranium-235 — $T_{1/2}$ = 7.037E+08 years. This nuclide is the primary fissile isotope in fuel, and its concentration depends on enrichment and burnup. ORIGEN2 calculation indicates a 1-year discharge at 1.425E-05 Ci. Minor source from plutonium-239 alpha decay causes an increase to 1.439E-05 Ci over 3000 years after discharge.

Uranium-236 — $T_{1/2}$ = 2.342E+07 years. This nuclide has slow ingrowth from long-lived plutonium-240 alpha decay. ORIGEN2 calculation indicates an inventory of 8.616E-06 Ci at 1 year, increasing to 9.41E-06 Ci at 3000 years.

Uranium-237 — $T_{1/2}$ = 6.75 days. This nuclide may be generated in a reactor through (n,γ), (n,γ) reactions from uranium-235 although initial inventory quickly decays because of short half-life. A more important source at 1 year is through a minor alpha decay route (0.0024%) of plutonium-241. This holds concentration up and for

	performance assessment purposes, uranium-237 should be considered as a short-lived daughter of plutonium-241. The uranium-237 beta decays to neptunium-237 and over time, the contribution to neptunium-237 is non-negligible. After 1 year ORIGEN2 calculation indicates that an inventory of 1.177E-05 Ci of uranium-237 remains.
Uranium-238	$T_{1/2}$ = 4.468E+09 years. Primary uranium isotope, which alpha decays to thorium-234 with a long decay chain. ORIGEN2 calculation indicates 1-year inventory of 3.335E-04 Ci, and 991.6g.
Uranium-239	$T_{1/2}$ = 23.5 minutes. This nuclide beta decays to neptunium-239. ORIGEN2 calculation indicates zero inventory after 1 year. This nuclide was not included in the inventory.
Uranium-240	$T_{1/2}$ = 14.1 hours. This nuclide is formed in a reactor by (n,γ), (n, γ) with uranium-238 but initial inventory quickly decays. It is produced as daughter from plutonium-244 alpha decay and should be considered as short-lived daughter of plutonium-244. Beta decays to neptunium-240, and then beta decays again ($T_{1/2}$ = 67 minutes) to plutonium-240. Uranium-240 activity at 1 year = 2.45E-15 Ci.
Neptunium-237	$T_{1/2}$ = 2.14E+06 years. This nuclide is generated by uranium-235 (n,γ)→uranium-236(n,γ)→Uranium-237(β)→neptunium-237 reactions. It is also formed by alpha decay of americium-241 and by a plutonium-241 alpha decay (0.0024%) followed by uranium-237 beta decay. Decay chain includes alpha to protactinium-233, beta to uranium-233 and then follows the uranium-233 decay chain. ORIGEN2 calculation indicates the following inventories: 1 year = 5.181E-06 Ci and 3000 year = 8.542E-06 Ci. Much of this increase after 1 year is the result of the plutonium-241 to uranium-237 to neptunium-237 decay route.

A.4.3 Moderately Immobile Nuclides

Protactinium-231	$T_{1/2}$ = 3.28E+04 years. This nuclide is a daughter of thorium-231 beta decay within the uranium-235 decay chain. Inventory slowly grows in over time. ORIGEN2 calculations indicate the following inventories: after 1 year = 3.9E-10 Ci and after 3000 years = 8.8E-07 Ci.
Radium-222	$T_{1/2}$ = 38 seconds. This nuclide is a daughter product resulting from uranium-230 alpha decay ($T_{1/2}$=20.8 days) to thorium-226 ($T_{1/2}$=30.9 minutes) which alpha decays to radium-222. Because of the short half-life of parent nuclides, this nuclide decays to negligible levels within 3 years and need not be considered in the Composite Analysis. This nuclide was not included in inventory.

Radium-223	$T_{1/2}$ = 11.435 days. This nuclide is a daughter in the uranium-235 decay chain and grows in over time. ORIGEN2 calculation indicates the following inventories: 1 year = 7.8E-12 Ci and 3000 year = 8.8E-07 Ci.
Radium-224	$T_{1/2}$ = 3.66 days. This nuclide is a daughter in the thorium-232 decay chain. ORIGEN2 calculation indicates minor initial ingrowth (8.1E-09 Ci at 1 year) with a peak at 4.0E-08 Ci at 10 years and a decline to 2.4E-12 Ci at 3000 years.
Radium-225	$T_{1/2}$ = 14.8 days. This nuclide is a daughter in the neptunium-237/uranium-233 decay chain. It grows in over time. ORIGEN2 calculation indicates the following inventories: 6.4E-14 Ci at 1 year and 1.3E-08 Ci at 3000 years.
Radium-226	$T_{1/2}$ = 1.6E+03 years. This nuclide is a daughter in the uranium-238/uranium-234 decay chain. It grows in over time. ORIGEN2 calculation indicates the following inventories: 1.0E-12 Ci at 1 year and 4.12E-6 Ci at 3000 years.
Radium-228	$T_{1/2}$ = 5.76 years. This nuclide is a thorium-232 daughter. ORIGEN2 calculation indicates ingrowth to 2.6E-17 Ci at 1 year, and 1.3E-12 Ci over 3000 years.
Ruthenium-106	$T_{1/2}$ = 367 days. This nuclide is a fission product with very minor activation source. ORIGEN2 calculation indicates the following inventories: 1 year = 5.914 Ci, and 10 year = 1.214E-02 Ci. Beta decays to short-lived daughter rhodium-106, which beta decays to palladium-106 (stable).

A.4.4 Highly Immobile Nuclides

Nickel-59	$T_{1/2}$ = 7.5E+04 years. This nuclide is an activation product. It decays by electron capture (99+%) to cobalt-59 (stable). ORIGEN2 calculation indicates inventory at 1 year = 1.841E-05 Ci.
Cobalt-60	$T_{1/2}$ = 5.271 years. This nuclide is an activation product which β^- decays to nickel-60 (stable). ORIGEN2 calculation indicates the following inventories: 1 year = 2.65E-03 Ci and 10 year = 8.111E-04 Ci.
Nickel-63	$T_{1/2}$ = 100 years. This nuclide is an activation product. It decays by beta to copper-63 (stable). ORIGEN2 calculation indicates the following inventories: 1 year = 2.272E-03 Ci and 10 years = 2.123E-03 Ci.
Strontium-90	$T_{1/2}$ = 28.8 years. This nuclide is a fission product. It beta decays to yttrium-90, then beta decays again ($T_{1/2}$=64.1 hours) to zirconium-90 (stable). ORIGEN2 calculation indicates the following inventories: 1 year = 2.039 Ci and 10 year = 1.646 Ci.

Zirconium-93	$T_{1/2}$ = 1.5E+06 years. This nuclide is primarily a fission product with minor activation source. It decays by β^- to niobium-93m then niobium-93 (stable). ORIGEN2 calculation indicates the following inventories: 1 year = 4.72E-05 Ci from fission, 1 year = 2.19E-07 Ci from activation; 10 year = 4.720E-05 Ci from fission, 10 year = 2.188E-07 Ci from activation.
Niobium-93m	$T_{1/2}$ = 13.6 years. This nuclide is a daughter of fission product with a much smaller activation source. It decays by isomeric transition to niobium-93 (stable). The source from zirconium-93 decay causes inventory to grow in over about 300 years to a level of 4.483E-05 Ci (from ORIGEN2 calculation).
Niobium-94	$T_{1/2}$ = 2.0E+4 years. This nuclide is a fission product with very minor activation source. It beta decays to molybdenum-94 (stable). ORIGEN2 calculation indicates that the inventory at 1 year = 1.55E-09 Ci.
Palladium-107	$T_{1/2}$ = 6.5E+06 years. This nuclide is a fission product with negligible activation source. It decays by β^- to (silver) Ag-107 (stable). ORIGEN2 calculation indicates that the inventory at 1 year = 1.21E-06 Ci.
(Tin) Sn-126	$T_{1/2}$ = 1E+05 years. This nuclide is a fission product with a very small activation source. It beta decays to (antimony) Sb-126, then beta decays again ($T_{1/2}$ = 12.4 days) to tellurium-126 (stable). ORIGEN2 calculation indicates that the inventory at 1 year = 1.65E-05 Ci.
Cesium-135	$T_{1/2}$ = 3.0E+06 years. This nuclide is a fission product that beta decays to barium-135 (stable). ORIGEN2 calculation indicates that the inventory at 1 year = 8.325E-06 Ci.
Cesium-137	$T_{1/2}$ = 30.17 years. This nuclide is a fission product. Beta decays through short-lived barium-137m to barium-137 (stable). ORIGEN2 calculation indicates the following inventories: 1 year = 2.379 Ci and 10 year =1.932 Ci.
Cerium-141	$T_{1/2}$ = 32.5 days. This nuclide is a fission product, which beta decays to praseodymium-141 (stable). ORIGEN2 calculation indicates that the inventories are 0.158 Ci at 1 year and approximately zero at 10 years. Because of a short half-life and lack of an ingrowth source to maintain activity, this nuclide is unlikely to be important in groundwater analysis. However, this radionuclide has been included in the inventory.
Cerium-144	$T_{1/2}$ = 284 days. This nuclide is a fission product that beta decays to praseodymium-144 ($T_{1/2}$ = 17.3 minutes) and beta decays again to neodymium-144 ($T_{1/2}$ = 2.1E+15 years). Activity decays normally after 1 year, with no source supporting activity. ORIGEN2 calculation indicates the following inventories: 1 year = 29.44 Ci and 10 years = 9.735E-03 Ci.

Samarium-147	$T_{1/2}$ = 1.06E+11 years. This nuclide grows in as a daughter of fission product promethium-147. ORIGEN2 calculation predicts a maximum inventory of 2.378E-10 Ci at about 40 years. Alpha decays to neodymium-143 (stable).
Samarium-151	$T_{1/2}$ = 90 years. This nuclide is a fission product, that decays by β^- to europium-151 (stable). ORIGEN2 calculation indicates the following inventories: 1 year = 4.435E-02 Ci and 10 years = 4.138E-02 Ci.
Europium-152	$T_{1/2}$ = 13 years. This nuclide is a fission product with no apparent decay-chain source. ORIGEN2 calculation indicates the following inventories: 1 year = 6.133E-05 Ci and 10 year = 3.877E-05 Ci.
Europium-154	$T_{1/2}$ = 8.5 years. This nuclide is a fission product with no apparent decay-chain source. ORIGEN2 calculation indicates the following inventories: 1 year = 1.392E-02 Ci and 10 year = 6.739E-03 Ci.
Europium-155	$T_{1/2}$ = 4.9 years. This nuclide is a fission product with no apparent decay-chain source. ORIGEN2 calculation indicates the following inventories: 1 year = 5.613E-02 Ci and 10 year = 1.596E-02 Ci.
(Lead) Pb-205	$T_{1/2}$ = 1.4E+07 years. This nuclide is an activation product with no apparent decay-chain source. ORIGEN2 calculation indicates that the inventory at 1 year = 8.041E-14 Ci.
Actinium-225	$T_{1/2}$ = 10 days. This nuclide is a daughter product within the neptunium-237/uranium-233 decay chain. ORIGEN2 calculation indicates that the inventory at 1 year = 6.387E-14 Ci, and grows in to 1.27E-08 Ci at 3000 years.
Actinium-227	$T_{1/2}$ = 21.773 years. This is a daughter in the uranium-235 decay chain. ORIGEN2 calculation indicates that the inventory at 1 year = 7.77E-12 Ci, and grows in to a level of 8.81E-07 Ci at 3000 years.
Thorium-227	$T_{1/2}$ = 18.718 days. This nuclide is a daughter within the uranium-235 decay chain. ORIGEN2 calculation indicates that the inventory at 1 year = 7.67E-12 Ci and grows in over time to 8.69E-07 Ci at 3000 years.
Thorium-228	$T_{1/2}$ = 1.931 years. This nuclide is a daughter within the thorium-232 decay chain. ORIGEN2 calculation indicates that the inventory grows in from 1 year = 8.1E-09 Ci to 10 year = 4.0E-08 Ci, to 3000 year = 2.4E-12 Ci.
Thorium-229	$T_{1/2}$ = 7.3E+03 years. This nuclide is a daughter within the neptunium-237 decay chain. ORIGEN2 calculation indicates that the inventory grows in from 1 year = 6.4E-14 Ci to 3000 years = 1.27E-08 Ci.

Thorium-230	$T_{1/2}$ = 8.0E+04 years. This nuclide is a daughter within the uranium-238/ uranium-234 decay chain. ORIGEN2 calculation indicates that the inventory grows in from 1 year = 3.8E-09 Ci to 3000 years = 9.32E-06 Ci.
Thorium-231	$T_{1/2}$ = 25.52 hours. This nuclide is a daughter within the uranium-235 decay chain. ORIGEN2 calculation indicates a fairly constant inventory level at 1.43E-5 Ci.
Thorium-232	$T_{1/2}$ = 1.41E+10 years. This is a natural thorium isotope. ORIGEN2 calculation indicates that it grows in from uranium-236 alpha decay from 4.72E-16 Ci at 1 year to 1.337E-12 Ci at 3000 years.
Thorium-234	$T_{1/2}$ = 24.1 days. This nuclide is a uranium-238 daughter. ORIGEN2 calculation indicates fairly steady concentration at 1 year at 3.34E-04 Ci.
Plutonium-236	$T_{1/2}$ = 2.85 years. Generation at low levels from a secondary beta decay route (9%) of neptunium-236 prevents this isotope from decaying to zero. ORIGEN2 calculation indicates inventories of 5.36E-07 Ci at 1 year, 6.00E-08 Ci at 10 years, with decay leveling off at a level of about 1.0E-12 Ci.
Plutonium-237	$T_{1/2}$ = 45.4 days. This nuclide decays through electron capture to form neptunium-237, but this is not an important source of neptunium-237. Plutonium-237 has a source from curium-241 alpha decay, which causes the isotope to be present longer than expected from its half-life, although it is still gone within 30 years. ORIGEN2 calculation indicates the following inventories: 1 year =1.19E-08 Ci and 10 years = 2.38E-30 Ci.
Plutonium-238	$T_{1/2}$ = 87.74 years. Sources from alpha decay of curium-242 and from beta decay of neptunium-238 are insignificant after 1 year and the activity decays very nearly as if there were no source. ORIGEN2 calculation indicates the following inventories: 1 year = 2.70E-03 Ci and 10 years = 2.514E-03 Ci.
Plutonium-239	$T_{1/2}$ = 2.41E+04 years. Uranium-238, which constitutes the majority of the metallic uranium fuel, undergoes an (n,γ) reaction during reactor operations to form uranium-239. The uranium-239 then undergoes two beta decays to form first neptunium-239 and then plutonium-239. Curium-243 decay is an insignificant source of plutonium-239. ORIGEN2 calculation indicates that the inventory at 1 year = 4.89E-02 Ci.
Plutonium-240	$T_{1/2}$ = 6.57E+03 years. In the reactor, plutonium-240 may be produced either directly through an (n,γ) reaction with plutonium-239 or through an (n,γ) reaction with uranium-239 followed by two beta decays. Curium-244 decay is an insignificant source of plutonium-240. ORIGEN2 calculation indicates that the inventory at 1 year = 1.05E-02 Ci.

Plutonium-241	$T_{1/2}$ = 14.4 years. During reactor operations, plutonium-241 is produced through an (n,γ) reaction with plutonium-240. Curium–245 decay is an insignificant source of plutonium-241. ORIGEN2 calculation indicates the following inventories: 1 year = 4.80E-01 Ci and 10 years = 0.3111 Ci.
Plutonium-242	$T_{1/2}$ = 3.76E+05 years. During reactor operations, plutonium–242 is produced through an (n,γ) reaction with plutonium-241. Curium-246 decay is an insignificant source of plutonium-242. ORIGEN2 calculation indicates that the inventory at 1 year = 4.64E-07 Ci.
Plutonium-243	$T_{1/2}$ = 4.956 hours. During reactor operations, plutonium–243 is produced through an (n,γ) reaction with plutonium-242. Curium-247 alpha decay also contributes to the plutonium-243 inventory. ORIGEN2 calculation indicates a fairly steady concentration at 8.6E-20 Ci.
Plutonium-244	$T_{1/2}$ = 8.1E+07 years. During reactor operations, plutonium–244 is produced through an (n,γ) reaction with plutonium-243. Curium-248 alpha decay is an insignificant source of plutonium-244. Its long half-life maintains relatively constant concentration in ORIGEN2 calculation at 2.46E-15 Ci.
Americium-241	$T_{1/2}$ = 433 years. This nuclide's source is plutonium-241 beta decay. ORIGEN2 calculation indicates that the inventory at 1 year = 8.30E-04 Ci, and grows to maximum of 1.47E-02 Ci at 100 years.
Americium-242	$T_{1/2}$ = 16.01 hours. This nuclide is the short-lived daughter of the isomeric transition of americium-242m. It beta decays to plutonium-242. ORIGEN2 calculation indicates the following inventories: 1 year = 6.91E-07 Ci and 10 year = 6.63E-07 Ci.
Americium-242m	$T_{1/2}$ = 152 years. This nuclide decays to americium-242. ORIGEN2 calculation indicates the following inventories: 1 year = 6.94E-07 Ci and 10 year = 6.67E-07 Ci.
Americium-243	$T_{1/2}$ = 7.37E+03. This nuclide has no source after 1 year. It decays by alpha to neptunium-239, which then beta decays to plutonium-239. This source of plutonium-239 is not significant. ORIGEN2 calculation indicates the following inventories: 1 year = 1.58E-07 Ci and 10 year = 1.576E-07 Ci.
Curium-242	$T_{1/2}$ = 162.8 days. This nuclide alpha decays to plutonium-238, although this source of plutonium-238 is not important after 1 year. ORIGEN2 calculation indicates the following inventories: 1 year = 3.634E-04 Ci and 10 years = 5.488E-07 Ci.

Curium-244	$T_{1/2}$ = 18.11 years. This nuclide alpha decays to plutonium-240, although this source of plutonium-240 is not important after 1 year. ORIGEN2 calculation indicates the following inventories: 1 year = 1.54E-05 Ci and 10 year = 1.092E-05 Ci.
Curium-245	$T_{1/2}$ = 8.5E+03 years. This nuclide alpha decays to plutonium-241, although this source of plutonium-241 is not important after 1 year. ORIGEN2 calculation indicates the following inventories: 1 year = 8.323E-11 Ci and 10 years = 8.317E-11 Ci.
Curium-246	$T_{1/2}$ = 4.7E+03 years. This nuclide alpha decays to plutonium-242, although it is an insignificant source of plutonium-242. ORIGEN2 calculation indicates the following inventories: 1 year = 6.925E-13 Ci and 10 year = 6.916E-13 Ci.

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Table A.1. Solid Waste Inventory for "new_dec_96" Category (page 1 of 5)

Based on Revision 1/30/96, GA Whyatt, File inv_10yr.xls																	
SWITS INVENTORY THROUGH DECEMBER 1996. Primary Source: SWITS Data Base, run date 1/4/97																	
		Non-decayed	Total	ORIGEN2 Run for Average N/single pass reactor Fuel, values are Ci per kg of U fuel taken at 10 year except where noted.													
		Post-1970	Non-segregated														
	Unsegregated	Segregated	Plus Non-TRU	H-3	C-14	C-14	Cl-36	Se-79	Tc-99	I-129	Ru-106	Ni-59	Co-60	Ni-63	Sr-90	Zr-93
	Non-decayed	Non-TRU	Segregated	7.07E-03	1.27E-04	100 Area	8.33E-06	1.02E-05	3.43E-04	6.77E-07	1.21E-02	1.84E-05	8.11E-04	2.12E-03	1.65E+00	4.72E-05	
Facility Identifier	Cs-137, Ci	Cs-137, Ci	Cs-137, Ci	Ci	Ci	Ci	Ci	Ci	Ci	Ci	Ci	Ci	Ci	Ci	Ci	Ci	
	updated(b)	updated(b)															
200 East Area																	
218e1	2.12E+00		2.12E+00	7.77E-03	1.40E-04		9.15E-06	1.12E-05	3.78E-04	7.45E-07	1.33E-02	2.02E-05	8.92E-04	2.33E-03	1.81E+00	5.19E-05	
218e10	3.60E+03	1.18E+06	1.18E+06	4.32E+03	7.78E+01	4.56E+01(a)	5.09E+00	6.20E+00	2.10E+02	4.14E-01	7.42E+03	1.12E+01	4.95E+02	1.30E+03	1.01E+06	2.88E+01	
218e12a	1.89E+01		1.89E+01	6.91E-02	1.25E-03		8.14E-05	9.93E-05	3.36E-03	6.63E-06	1.19E-01	1.80E-04	7.93E-03	2.08E-02	1.61E+01	4.62E-04	
218e12b	1.35E+02	3.09E+04	3.10E+04	1.14E+02	2.05E+00		1.34E-01	1.63E-01	5.52E+00	1.98E-02(a)	1.95E+02	2.96E-01	1.30E+01	3.41E+01	2.64E+04	7.58E-01	
218e14 (purex tunnel#1)	1.87E+03		1.87E+03	6.82E+00	1.23E-01		8.04E-03	9.80E-03	3.31E-01	6.54E-04	1.17E+01	1.78E-02	7.83E-01	2.05E+00	1.59E+03	4.56E-02	
218e15 (purex tunnel#2)	9.04E+03		9.04E+03	3.31E+01	5.96E-01		3.89E-02	4.75E-02	1.61E+00	3.17E-03	5.68E+01	8.61E-02	3.79E+00	9.93E+00	7.70E+03	2.21E-01	
218e2	5.31E+02		5.31E+02	1.94E+00	3.50E-02		2.29E-03	2.79E-03	9.44E-02	1.86E-04	3.34E+00	5.06E-03	2.23E-01	5.83E-01	4.52E+02	1.30E-02	
218e4	2.12E-01		2.12E-01	7.77E-04	1.40E-05		9.15E-07	1.12E-06	3.78E-05	7.45E-08	1.33E-03	2.02E-06	8.92E-05	2.33E-04	1.81E-01	5.19E-06	
218e5	1.59E+02		1.59E+02	5.83E-01	1.05E-02		6.86E-04	8.37E-04	2.83E-02	5.59E-05	1.00E+00	1.52E-03	6.69E-02	1.75E-01	1.36E+02	3.89E-03	
218e5a	3.51E+02		3.51E+02	1.28E+00	2.31E-02		1.51E-03	1.84E-03	6.23E-02	1.23E-04	2.20E+00	3.34E-03	1.47E-01	3.85E-01	2.99E+02	8.58E-03	
218e8	2.12E-01		2.12E-01	7.77E-04	1.40E-05		9.15E-07	1.12E-06	3.78E-05	7.45E-08	1.33E-03	2.02E-06	8.92E-05	2.33E-04	1.81E-01	5.19E-06	
222b vaults	1.27E+01		1.27E+01	4.66E-02	8.40E-04		5.49E-05	6.69E-05	2.26E-03	4.47E-06	8.01E-02	1.21E-04	5.35E-03	1.40E-02	1.09E+01	3.11E-04	
218ec9	0.00E+00	7.97E+00	7.97E+00	2.92E-02	5.26E-04		3.44E-05	4.19E-05	1.42E-03	2.80E-06	5.01E-02	7.60E-05	3.35E-03	8.76E-03	6.79E+00	1.95E-04	
C-14 subtotals					8.07E+01	4.56E+01											
total East area	1.60E+04	1.21E+06	1.22E+06	4.48E+03	1.26E+02		5.27E+00	6.43E+00	2.17E+02	4.29E-01	7.69E+03	1.17E+01	5.14E+02	1.34E+03	1.04E+06	2.99E+01	
200 West Area																	
218W1	4.25E+00		4.25E+00	1.55E-02	2.80E-04		1.83E-05	2.23E-05	7.55E-04	1.49E-06	2.67E-02	4.05E-05	1.78E-03	4.67E-03	3.62E+00	1.04E-04	
218w11	2.12E-03		2.12E-03	7.77E-06	1.40E-07		9.15E-09	1.12E-08	3.78E-07	7.45E-10	1.33E-05	2.02E-08	8.92E-07	2.33E-06	1.81E-03	5.19E-08	
218w1A	1.02E+03		1.02E+03	3.73E+00	6.73E-02		4.40E-03	5.36E-03	1.81E-01	3.58E-04	6.41E+00	9.72E-03	4.28E-01	1.12E+00	8.69E+02	2.49E-02	
218w2	1.06E+01		1.06E+01	3.89E-02	7.00E-04		4.58E-05	5.58E-05	1.89E-03	3.72E-06	6.67E-02	1.01E-04	4.46E-03	1.17E-02	9.05E+00	2.59E-04	
218w2A	4.99E+03	5.43E+02	5.54E+03	2.03E+01	3.65E-01		2.39E-02	2.91E-02	9.84E-01	1.94E-03	3.48E+01	5.28E-02	2.32E+00	6.08E+00	4.72E+03	1.35E-01	
218W3	1.91E+01		1.91E+01	6.99E-02	1.26E-03		8.24E-05	1.00E-04	3.40E-03	6.70E-06	1.20E-01	1.82E-04	8.03E-03	2.10E-02	1.63E+01	4.67E-04	
218W3A	1.66E+02	3.05E+05	3.06E+05	1.12E+03	2.01E+01	2.66E+02(a)	1.32E+00	1.61E+00	5.43E+01	1.07E-01	1.92E+03	2.91E+00	1.28E+02	3.36E+02	2.60E+05	7.46E+00	
218W3AE		4.67E+04	4.67E+04	1.78E+02	3.21E+00	4.12E+01(a)	2.10E-01	2.56E-01	9.93E+00(d)	1.45E-02(d)	3.06E+02	4.64E-01	2.05E+01	5.36E+01	4.15E+04	1.19E+00	
218W4A	7.04E+01		7.04E+01	2.58E-01	4.64E-03		3.03E-04	3.70E-04	1.16E-02(d)	2.47E-05	4.42E-01	6.71E-04	2.96E-02	7.74E-02	6.00E+01	1.72E-03	
218W4B-calsson	1.65E+03	2.13E+03	3.77E+03	1.36E+01	2.49E-01	1.25E+00(a)	1.63E-02	1.98E-02	6.71E-01	1.32E-03	2.37E+01	3.60E-02	1.58E+00	4.16E+00	3.22E+03	9.22E-02	
218W4B- non calsson	5.75E+02	7.25E+03	7.83E+03	2.88E+01	5.16E-01	4.10E+00(a)	3.37E-02	4.11E-02	1.39E+00	5.00E-01(a)	4.92E+01	7.46E-02	3.29E+00	8.60E+00	6.67E+03	1.91E-01	
218W4C		3.37E+03	3.37E+03	1.23E+01	3.61E+00(a)	1.20E+00(a)	1.45E-02	2.87E-02(d)	2.69E+00(d)	3.45E-03(d)	2.12E+01	3.21E-02	1.42E+00	3.71E+00	2.87E+03	8.24E-02	
218W5		3.18E+03	3.18E+03	1.16E+01	4.44E+00(a)	1.10E+00(a)	1.37E-02	1.67E-02	7.10E-01(d)	3.33E-02(d)	2.00E+01	3.03E-02	1.34E+00	3.50E+00	2.71E+03	7.77E-02	
218w9	2.12E-03		2.12E-03	7.77E-06	1.40E-07		9.15E-09	1.12E-08	3.78E-07	7.45E-10	1.33E-05	2.02E-08	8.92E-07	2.33E-06	1.81E-03	5.19E-08	
				0.00E+00	0.00E+00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
221T, T Plant				0.00E+00	0.00E+00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
222s vaults	8.56E+01		8.56E+01	3.13E-01	5.64E-03		3.69E-04	4.50E-04	1.52E-02	3.00E-05	5.38E-01	8.16E-04	3.59E-02	9.41E-02	7.29E+01	2.09E-03	
222T vaults	1.64E+01		1.64E+01	5.98E-02	1.08E-03		7.05E-05	8.59E-05	2.91E-03	5.73E-06	1.03E-01	1.56E-04	6.86E-03	1.80E-02	1.39E+01	3.99E-04	
241T Facility			0.00E+00	0.00E+00	0.00E+00		0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	
TRUSAF		1.31E-02	1.31E-02	4.80E-05	8.66E-07		5.66E-08	6.90E-08	2.33E-06	4.60E-09	8.25E-05	1.25E-07	5.51E-06	1.44E-05	1.12E-02	3.21E-07	
Z plant, PFP		1.21E-03	1.21E-03	4.44E-06	8.00E-08		5.23E-09	6.37E-09	2.16E-07	4.25E-10	7.62E-06	1.16E-08	5.09E-07	1.33E-06	1.03E-03	2.96E-08	
cen. wst cmplx buildings, numerous		2.15E+02	2.15E+02	7.88E-01	5.74E-01(a)		9.29E-04	1.13E-03	3.69E+00(a)	1.83E-03(a)	1.35E+00	2.05E-03	9.05E-02	2.37E-01	1.84E+02	5.26E-03	
2706T		2.40E+01	2.40E+01	8.77E-02	1.58E-03		1.03E-04	1.26E-04	2.34E-02(a)	1.84E-03(a)	1.51E-01	2.28E-04	1.01E-02	2.63E-02	2.04E+01	5.85E-04	
Alk Metal Waste Storage units 1,2,3,4		7.43E-02	7.43E-02	2.72E-04	4.90E-06		3.20E-07	3.90E-07	1.32E-05	2.61E-08	4.67E-04	7.08E-07	3.12E-05	8.17E-05	6.33E-02	1.82E-06	
Flammable storage units 1 through 20		6.67E-01	6.67E-01	3.17E-03	7.38E-02(a)		3.73E-06	4.55E-06	1.32E-02(a)	1.00E-02(a)	5.45E-03	8.26E-06	3.64E-04	9.52E-04	7.38E-01	2.12E-05	
C-14 subtotals					3.35E+01	3.35E+02											
west area total	8.61E+03	3.71E+05	3.79E+05	1.39E+03	3.69E+02		1.64E+00	2.00E+00	7.46E+01	6.76E-01	2.38E+03	3.62E+00	1.59E+02	4.17E+02	3.23E+05	9.27E+00	
Notes:																	
218E-12B inventories of C-14, Tc-99, I-129 and Se-79 neglect offsite sources to exclude reactor compartments from inventory.																	
218W4C excludes 74.8 Ci of C-14 and and 14.62 Ci Tc-99 which is immobilized in grouted containers. The C-14 was shown in SWITS report, Tc-99 was not.																	
(a) Values indicate entries where SWITS reported inventory exceeds projected inventory based on Cs-137 activity and results of ORIGEN2 run for single pass and N reactors																	
(b) Indicate SWITS data input columns.																	
(c) Indicate that the inventory projection is based on the maximum ingrowth amount predicted by ORIGEN2 between 1 and 3000 years.																	
(d) Entries identify instances where the inventory calculated by M. I. Wood (from an electronic mail message sent by M. I. Wood of Waste Management Federal Services of Hanford to G. A. Whyatt on December 19, 1996, Subject - Waste Area Burial Ground Inventories) based on total beta exceeded that based on ORIGEN2 run. The larger of the two estimates was used.																	

Table B.1. (page 7 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Cesium-137, Units (Ci)	Strontium-90, Units (Ci)	Ruthenium-106, Units (Ci)	Total Plutonium, Units (g, unless otherwise stated)
216-T-11	Trench	216-T-11 Trench	Decon Waste	TP-4				
216-T-12	Trench	216-T-12 Trench	Cooling Water	TP-3	4.34	2.05	1.38E-10	1
216-T-13	Trench	216-T-13 Trench	Decon Waste	TP-2				
216-T-14	Trench	216-T-14 Trench	Tank Farm Waste	TP-3	204	2.46	2.07E-10	0.88
216-T-15	Trench	216-T-15 Trench	Tank Farm Waste	TP-3	450	8.62	1.66E-10	0.94
216-T-16	Trench	216-T-16 Trench	Tank Farm Waste	TP-3	227	3.28	1.79E-10	0.65
216-T-17	Trench	216-T-17 Trench	Tank Farm Waste	TP-3	162	1.23	1.38E-10	0.53
216-T-18	Cribs	216-T-18 Crib	Tank Farm Waste	TP-2	24.2	2.8	1.38E-09	1800
216-T-19	Cribs	216-T-19TF Crib and Tile Field	Process Waste	TP-2	17.5	27.8	0.00000603	14.4
216-T-2	Reverse Well	216-T-2 Reverse Well	Lab Waste	TP-4				
216-T-20	Trench	216-T-20 Trench	Process Waste	TP-2	0.44	0.388	7.44E-12	
216-T-21	Trench	216-T-21 Trench	Tank Farm Waste	TP-1	174	3.28	8.56E-10	1
216-T-22	Trench	216-T-22 Trench	Tank Farm Waste	TP-1	803	20.9	4.14E-10	2
216-T-23	Trench	216-T-23 Trench	Tank Farm Waste	TP-1	577	16.82	3.59E-10	1
216-T-24	Trench	216-T-24 Trench	Tank Farm Waste	TP-1	617	16.4	4.42E-10	2
216-T-25	Trench	216-T-25 Trench	Process Waste	TP-1	3860	1.64	1.38E-09	1
216-T-26	Cribs	216-T-26 Crib	Tank Farm Waste	TP-2	75.6	282	8.02E-08	59
216-T-27	Cribs	216-T-27 Crib	Lab Waste	TP-2	55.9	75.3	0.0000409	13
216-T-28	Cribs	216-T-28 Crib	Decon Waste	TP-2	193	106	0.0000196	70
216-T-29	Cribs	216-T-29 Crib	Miscellaneous Drainage	TP-4				
216-T-3	Reverse Well	216-T-3 Reverse Well	Process Waste	TP-4	21.3	18.6	5.22E-12	3350
216-T-31	French Drain	216-T-31 French Drain	Miscellaneous Drainage	TP-2				
216-T-32	Cribs	216-T-32 Crib	Process Waste	TP-1	9.71	10.9	4.44E-11	3200
216-T-33	Cribs	216-T-33 Crib	Decon Waste	TP-4	0.267	0.256	6.86E-08	5
216-T-34	Cribs	216-T-34 Crib	Lab Waste	TP-4	157	178	0.00000598	107
216-T-35	Cribs	216-T-35 Crib	Lab Waste	TP-4	11.7	11.4	0.0000144	66.2
216-T-36	Cribs	216-T-36 Crib	Steam Condensate	TP-1	3.79	4.36	0.00000524	2.48
216-T-4-1D	Ditches	216-T-4-1D Ditch	Cooling Water	TP-3				
216-T-4-2	Ditches	216-T-4-2 Ditch	Steam Condensate	TP-3				
216-T-4A	Ponds	216-T-4A Pond	Cooling Water	TP-3				
216-T-4B	Ponds	216-T-4B Pond	Cooling Water	TP-3	6.23	3.37	0.000000867	3.71
216-T-5	Trench	216-T-5 Trench	Tank Farm Waste	TP-1	31.1	0.42	8.25E-10	180
216-T-6	Cribs	216-T-6 Crib	Process Waste	TP-3	110	124	6.07E-11	390

Table B.1. (page 8 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Cesium-137, Units (Ci)	Strontium-90, Units (Ci)	Ruthenium-106, Units (Ci)	Total Plutonium, Units (g, unless otherwise stated)
216-T-7	Cribs	216-T-7TF Crib and Tile Field	Tank Farm Waste	TP-1	21.2	24	2.02E-09	130
216-T-8	Cribs	216-T-8 Crib	Lab Waste	TP-4	0.0401	0.376	6.63E-12	5
216-T-9	Trench	216-T-9 Trench	Decon Waste	TP-4				
216-U-1&2	Cribs	216-U-1 & 216-U-2	Process Condensate	UP-2	4.36	2.11	0.0000006	42.6
216-U-10	Ponds	216-U-10	Cooling Water	UP-2	11	11	0.0000278	8000
216-U-11	Ditches	216-U-11	Cooling Water	UP-2				
216-U-12	Cribs	216-U-12	Process Condensate	UP-2	0.0566	55.9	0.00000218	1
216-U-13	Trench	216-U-13 (same as UN-200-W- 125)	Decon Waste	UP-2	0.0444	0.042		0.1
216-U-14	Ditches	216-U-14	Cooling Water	UP-2				
216-U-15	Trench	216-U-15	Process Waste	UP-2	0.0465	0.0442		0.1
216-U-16	Cribs	216-U-16	Process Condensate	UP-2	0.0165	0.0092		
216-U-17	Cribs	216-U-17	Process Condensate	UP-2				
216-U-21		216-U-21			85.5	21.8	0.00000139	2.08
216-U-3	French Drain	216-U-3	Miscellaneous Drainage	UP-2	0.434	0.041		0.1
216-U-4	Reverse Well	216-U-4	Lab Waste	UP-2				
216-U-4A	French Drain	216-U-4A	Miscellaneous Drainage	UP-2	0.185	0.0159	0.00000012	0.009
216-U-4B	French Drain	216-U-4B	Miscellaneous Drainage	UP-2	0.197	0.00165		0.054
216-U-5	Trench	216-U-5 & 216-U-6	Process Waste	UP-2	0.0207	0.0195		0.05
216-U-7	French Drain	216-U-7	Miscellaneous Drainage	UP-2				
216-U-8	Cribs	216-U-8	Process Condensate	UP-2	0.0455	0.0431	0.00000001	370
216-U-9	Ditches	216-U-9	Cooling Water	RO-1				
216-W-LWC	Cribs	216-W-LWC Crib	Chemical Sewer	SS-2				
216-Z-1&2	Cribs	216-Z-1 & 216-Z-2 Cribs	Process Waste	ZP-2	0.04	0.037	1.6E-11	7000
216-Z-10	Reverse Well	216-Z-10 Reverse Well	Process Waste	ZP-2				50
216-Z-11	Ditches	216-Z-11	Cooling Water	UP-2				
216-Z-12	Cribs	216-Z-12 Crib	Process Waste	ZP-2	0.053	0.051	0.00000093	25000
216-Z-13	French Drain	216-Z-13 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-14	French Drain	216-Z-14 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-15	French Drain	216-Z-15 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-16	Cribs	216-Z-16 Crib	Lab Waste	ZP-2				72
216-Z-17	Ditches	216-Z-17 Trench	Lab Waste	ZP-2				50
216-Z-18	Cribs	216-Z-18 Crib	Process Waste	ZP-2				23000
216-Z-19	Ditches	216-Z-19	Cooling Water	UP-2				

Table B.1. (page 9 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Cesium-137, Units (Ci)	Strontium-90, Units (Ci)	Ruthenium-106, Units (Ci)	Total Plutonium, Units (g, unless otherwise stated)
216-Z-1A	Cribs	216-Z-1A Tile Field	Process Waste	ZP-2	0.16	0.15	0.0000052	57000
216-Z-1D	Ditches	216-Z-1D	Cooling Water	UP-2				
216-Z-20	Cribs	216-Z-20	Cooling Water	UP-2	0.0864	0.083	0.000107	0.148
216-Z-21	Retention Basin	216-Z-21 Seepage Basin	Cooling Water	ZP-2				
216-Z-3	Cribs	216-Z-3 Crib	Process Waste	ZP-2	0.048	0.045	0.000000006	5700
216-Z-4	Trench	216-Z-4 Trench	Process Waste	ZP-2	0.035	0.033	2.7E-14	2
216-Z-5	Cribs	216-Z-5 Crib	Process Waste	ZP-2	3.6	1.7	5.2E-12	340
216-Z-6	Cribs	216-Z-6 Crib	Process Waste	ZP-2	0.035	0.033	2.7E-14	5
216-Z-7	Cribs	216-Z-7 Crib	Lab Waste	ZP-2	200	200	0.0000051	2000
216-Z-8	Cribs	216-Z-8 French Drain	Process Waste	ZP-2				2
216-Z-9	Cribs	216-Z-9 Trench	Process Waste	ZP-2	0.052	0.049	0.000000019	48000
218-C-9	Burial Site	218-C-9 Burial Ground	LLW - SOLID	SO-1	8.1		0.0000054	0.0001
218-E-1	Burial Site	218-E-1	LLW - SOLID	PO-2	0.8166	0.7165	7.69E-12	900
218-E-10	Burial Site	218-E-10 Burial Ground	LLW - SOLID	BP-10	931000	768000	0.771	4900
218-E-12A	Burial Site	218-E-12A	LLW - SOLID	PO-6	10.99	9.056	0.000001222	8930
218-E-12B	Burial Site	218-E-12B	LLW - SOLID	PO-6				
218-E-13	Burial Site	218-E-13		PO-2				
218-E-2	Burial Site	218-E-2 Burial Ground	LLW - SOLID	BP-10	213	187	0	800
218-E-2A	Burial Site	218-E-2A Burial Ground	LLW - SOLID	BP-10				
218-E-4	Burial Site	218-E-4 Burial Ground	LLW - SOLID	BP-10	940	0.0833	0	10
218-E-5	Burial Site	218-E-5 Burial Ground	LLW - SOLID	BP-10	70.7	62.7	0	620
218-E-5A	Burial Site	218-E-5A Burial Ground	LLW - SOLID	BP-10	165	147	0	1380
218-E-6	Burial Site	218-E-6 Burial Ground	Debris	BP-6				
218-E-7	Burial Site	218-E-7 Burial Ground	Lab Waste	BP-6	4.96	4.36	0	1
218-E-8	Burial Site	218-E-8	TRU Solid Waste	PO-6	0.1017	0.09058	1.177E-10	20
218-E-9	Burial Site	218-E-9 Burial Ground	LLW - SOLID	BP-10				
218-W-1	Burial Site	218-W-1 Burial Ground	TRU Solid Waste	ZP-3	1.63	1.44	8.83E-12	94000
218-W-11	Burial Site	218-W-11 Burial Ground	LLW - SOLID	ZP-3	0.002	0.0009	1.6E-09	
218-W-1A	Burial Site	218-W-1A Burial Ground	LLW - SOLID	ZP-3	359	359	5.23E-09	2000
218-W-2	Burial Site	218-W-2 Burial Ground	TRU Solid Waste	ZP-3	4.86	4.1	5.72E-10	126000
218-W-2A	Burial Site	218-W-2A Burial Ground	LLW - SOLID	ZP-3	2766	2467	0.0025	
218-W-3	Burial Site	218-W-3 Burial Ground	TRU Solid Waste	ZP-3	9.15	8.15	1.31E-08	68000
218-W-3A	Burial Site	218-W-3A Burial Ground	TRU Solid Waste	ZP-3	302000	101000	12.7	29300

Table B.1. (page 10 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Cesium-137, Units (Ci)	Strontium-90, Units (Ci)	Ruthenium-106, Units (Ci)	Total Plutonium, Units (g, unless otherwise stated)
218-W-3AE	Burial Site	218-W-3AE Burial Ground	LLW - SOLID	ZP-3	14300	4240	0.0268	122
218-W-4A	Burial Site	218-W-4A Burial Ground	TRU Solid Waste	ZP-3	39.3	35.4	0.00000842	35400
218-W-4B	Burial Site	218-W-4B Caissons	TRU Solid Waste	ZP-3	12340	11000	216	7290
218-W-4B	Burial Site	218-W-4B Trenches	TRU Solid Waste	ZP-3	6410	89700	390	48800
218-W-4C	Burial Site	218-W-4C Burial Ground	TRU Solid Waste	ZP-3	165000	111000	927	383000
218-W-5	Burial Site	218-W-5 Burial Ground	TRU Solid Waste	ZP-3	1500	1350	1.58	154
218-W-6	Burial Site	218-W-6 Burial Ground	LLW - SOLID	ZP-3				
218-W-7	Burial Site	218-W-7	LLW - SOLID	RO-3	39.24	34.84	2.295E-08	0.7
218-W-8	Burial Site	218-W-8 Burial Ground	Lab Waste	TP-4	6.403	5.625	3.607E-11	0.3
218-W-9	Burial Site	218-W-9	LLW - SOLID	RO-2	0.000921	0.000815	5.766E-14	
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
232-Z	Building	232-Z Incinerator		ZP-2				
240-S-151	Diversion Box	240-S-151	LLW - SOLID	RO-3				
240-S-152	Diversion Box	240-S-152	Tank Farm Waste	RO-3				
240-S-302	Tanks	240-S-302	Lab Waste	RO-3				
241-A-151	Diversion Box	241-A-151	Tank Farm Waste	PO-2				
241-A-152	Diversion Box	241-A-152	Process Waste	PO-3				
241-A-153	Diversion Box	241-A-153	Process Waste	PO-3				
241-A-302A	Tanks	241-A-302A		PO-2				
241-A-302B	Tanks	241-A-302B		PO-5				
241-A-350	Tanks	241-A-350	Process Waste	PO-3				
241-A-417	Tanks	241-A-417	Process Waste	PO-3				
241-A-A	Diversion Box	241-A-A	Process Waste	PO-3				
241-A-B	Diversion Box	241-A-B	Process Waste	PO-3				
241-AN-A	Diversion Box	241-AN-A	Process Waste	PO-3				
241-AN-B	Diversion Box	241-AN-B	Process Waste	PO-3				

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Cesium-137, Units (Ci)	Strontium-90, Units (Ci)	Ruthenium-106, Units (Ci)	Total Plutonium, Units (g, unless otherwise stated)
241-AP VP	Valve Pit	241-AP	Process Waste	PO-3				
241-AR-151	Diversion Box	241-AR-151	Process Waste	PO-3				
241-AW-A	Diversion Box	241-AW-A	Process Waste	PO-3				
241-AW-B	Diversion Box	241-AW-B	Process Waste	PO-3				
241-AX-151	Diversion Box	241-AX-151	Process Waste	PO-3				
241-AX-152DS	Tanks	241-AX-152DS	Process Waste	PO-3				
241-AX-155	Diversion Box	241-AX-155	Tank Farm Waste	PO-3				
241-AX-501	Valve Pit	241-AX-501		PO-3				
241-AX-A	Diversion Box	241-AX-A		PO-3				
241-AX-B	Diversion Box	241-AX-B		PO-3				
241-AY-151	Diversion Box	241-AY-151	Process Waste	PO-3				
241-AY-152	Diversion Box	241-AY-152	Process Waste	PO-3				
241-AZ-151DS	Diversion Box	241-AZ-151DS		PO-3				
241-AZ-152	Diversion Box	241-AZ-152		PO-3				
241-C-151	Diversion Box	241-C-151		PO-3				
241-C-152	Diversion Box	241-C-152		PO-3				
241-C-153	Diversion Box	241-C-153		PO-3				
241-C-154	Diversion Box	241-C-154 Diversion Box	Process Waste	SO-1				
241-C-252	Diversion Box	241-C-252		PO-3				
241-C-301C	Tanks	241-C-301C		PO-3				
241-CR-151	Diversion Box	241-CR-151		PO-3				
241-CR-152	Diversion Box	241-CR-152		PO-3				
241-CR-153	Diversion Box	241-CR-153		PO-3				
241-CX-TK-70	Tanks	241-CX-70 Storage Tank	Tank Farm Waste	SO-1				
241-CX-TK-71	Tanks	241-CX-71 Storage Tank	Process Condensate	SO-1	0.0496	93		
241-CX-TK-72	Tanks	241-CX-72 Storage Tank	Process Waste	SO-1	15000	0.0000028		200
241-ER-153	Diversion Box	241-ER-153		PO-3				
241-S-151	Diversion Box	241-S-151	LLW - SOLID	RO-2				
241-S-152	Diversion Box	241-S-152	Tank Farm Waste	RO-4				
241-S-302A	Tanks	241-S-302A	Lab Waste	RO-2				
241-S-302B	Tanks	241-S-302B	LLW - SOLID	RO-4				
241-S-A	Diversion Box	241-S-A	Tank Farm Waste	RO-4				
241-S-B	Diversion Box	241-S-B	Tank Farm Waste	RO-4				

Table B.1. (page 12 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Cesium-137, Units (Ci)	Strontium-90, Units (Ci)	Ruthenium-106, Units (Ci)	Total Plutonium, Units (g, unless otherwise stated)
241-S-C	Diversion Box	241-S-C	Tank Farm Waste	RO-4				
241-S-D	Diversion Box	241-S-D	Tank Farm Waste	RO-4				
241-SX-151	Diversion Box	241-SX-151	Tank Farm Waste	RO-4				
241-SX-152	Diversion Box	241-SX-152	Tank Farm Waste	RO-4				
241-SX-302	Tanks	241-SX-302		RO-2				
241-SX-A	Diversion Box	241-SX-A		RO-4				
241-SX-B	Diversion Box	241-SX-B		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-T-151	Diversion Box	241-T-151 Diversion Box	Tank Farm Waste	TP-6				
241-T-152	Diversion Box	241-T-152 Diversion Box	Tank Farm Waste	TP-6				
241-T-153	Diversion Box	241-T-153 Diversion Box	Tank Farm Waste	TP-6				
241-T-252	Diversion Box	241-T-252 Diversion Box	Tank Farm Waste	TP-6				
241-T-301	Tanks	241-T-301 Catch Tank	Tank Farm Waste	TP-6				
241-T-302	Tanks	241-T-302 Catch Tank	Tank Farm Waste	TP-6				
241-T-361	Tanks	241-T-361 Settling Tank	Process Waste	TP-4				15500 Ci
241-TR-152	Diversion Box	241-TR-152 Diversion Box	Tank Farm Waste	TP-6				
241-TR-153	Diversion Box	241-TR-153 Diversion Box	Tank Farm Waste	TP-6				
241-TX-152	Diversion Box	241-TX-152 Diversion Box	Tank Farm Waste	TP-2				
241-TX-153	Diversion Box	241-TX-153 Diversion Box	Tank Farm Waste	TP-5				
241-TX-154	Diversion Box	241-TX-154 Diversion Box	Tank Farm Waste	TP-4				
241-TX-155	Diversion Box	241-TX-155 Diversion Box	Tank Farm Waste	TP-2				
241-TX-302A	Tanks	241-TX-302A Catch Tank	Tank Farm Waste	TP-5				
241-TX-302B	Tanks	241-TX-302B Catch Tank	Tank Farm Waste	TP-2				
241-TX-302C	Tanks	241-TX-302C Catch Tank	Tank Farm Waste	TP-4				
241-TXR-151	Diversion Box	241-TXR-151 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-152	Diversion Box	241-TXR-152 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-153	Diversion Box	241-TXR-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-153	Diversion Box	241-TY-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-302A	Tanks	241-TY-302A Catch Tank	Tank Farm Waste	TP-5				
241-TY-302B	Tanks	241-TY-302B Catch Tank	Tank Farm Waste	TP-5				

Table B.1. (page 13 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Cesium-137, Units (Ci)	Strontium-90, Units (Ci)	Ruthenium-106, Units (Ci)	Total Plutonium, Units (g, unless otherwise stated)
241-Z-361	Tanks	241-Z-361 Settling Tank	Process Waste	ZP-2				75000
241-Z-TK-8	Tanks	216-Z-8 Settling Tank	Process Waste	ZP-2				1600
241-Z-TK-D5	Tanks	241-Z Treatment Tank	Process Waste	ZP-2				
242-T-151	Diversion Box	242-T-151 Diversion Box	Process Condensate	TP-5				
244-A RT	Tanks	244-A	Process Waste	PO-3				
244-AR VAULT	Vault	244-AR	Process Waste	PO-3				
244-CR VAULT	Vault	244-CR	Process Waste	PO-3				
244-S RT	Tanks	244-S Receiver Tank		RO-2				
244-TX RT	Tanks	244-TX Receiving Tank		TP-5				
244-TXR	Vault	244-TXR Vault	Tank Farm Waste	TP-5				
2607-E5	Septic System	2607-E-5 Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-E6	Septic System	2607-E6	Sanitary Waste	PO-2				
2607-E7A	Septic System	2607-E-7A Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-EA	Septic System	2607-EA	Sanitary Waste	PO-2				
2607-EC	Septic System	2607-EC	Sanitary Waste	PO-5				
2607-ED	Septic System	2607-ED	Sanitary Waste	PO-3				
2607-EE	Septic System	2607-EL	Sanitary Waste	PO-2				
2607-EG	Septic System	2607-EG	Sanitary Waste	PO-3				
2607-EJ	Septic System	2607-EJ	Sanitary Waste	PO-3				
2607-N	Septic System	2607-N Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-P	Septic System	2607-P Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-R	Septic System	2607-R Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-W1	Septic System	2607-W1 Septic Tank	Sanitary Waste	SS-2				
2607-W2	Septic System	2607-W2 Septic Tank	Sanitary Waste	SS-2				
2607-W3	Septic System	2607-W3 Septic Tank	Sanitary Waste	TP-4				
2607-W4	Septic System	2607-W4 Septic Tank	Sanitary Waste	TP-4				
2607-W6	Septic System	2607-W6	Sanitary Waste	RO-3				
2607-W8	Septic System	2607-W-8 Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WA	Septic System	2607-WA Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WB		2607-WB Septic Tank and Drain Field						
2607-WT	Septic System	2607-WT Septic Tank	Sanitary Waste	TP-5				
2607-WTX	Septic System	2607-WTX Septic Tank	Sanitary Waste	TP-5				
2607-WZ	Septic System	2607-WZ	Sanitary Waste	RO-1				
2607-Z	Septic System	2607-Z Septic Tank and Drain Field	Sanitary Waste	ZP-2				

Table B.1. (page 14 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Cesium-137, Units (Ci)	Strontium-90, Units (Ci)	Ruthenium-106, Units (Ci)	Total Plutonium, Units (g, unless otherwise stated)
2607-Z-1		2607-Z-1 Septic Tank and Drain Field						
2704-C-WS-1	French Drain	2704-C-WS-1, 2704-C French Drain, Gatehouse French Drain	Miscellaneous Drainage	SO-1				
2904-S-160	Diversion Box	2904-S-160	Cooling Water	RO-1				
2904-S-170	Diversion Box	2904-S-170	Process Waste	RO-1				
2904-S-171	Diversion Box	2904-S-171	Cooling Water	RO-1				
291-C	Building	291-C Ventilation System	Process Condensate	SO-1				
299-E24-111	Reverse Well	299-E24-111		PO-2				
HSVP	Diversion Box	Semi-Works Valve Pit	Process Waste	SO-1				
UPR-200-E-141		UN-200-E-141	Solution Storage (1)	SO-1				
UPR-200-E-36		UN-200-E-36	Process & Decon Wastes (1)	SO-1				
UPR-200-E-37		UN-200-E-37	Process & Decon Wastes (1)	SO-1				
UPR-200-E-98		UN-200-E-98	Process & Decon Wastes (3)	SO-1				
UPR-200-W-160		UPR-200-W-160 Unplanned Release	Tank Farm Waste	TP-4	17	16	3.46E-10	1
Z PLANT BP	Burial Site	Z Plant Burn Pit	Debris	ZP-3				

Table B.2. Environmental Restoration Waste Site Inventories for Plutonium-238, -239, -240, and -241 (page 1 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
		212-N to 216-N-1 Pipeline						
		212-P Hazardous Waste Staging Area						
		212-P to 216-N-4 Pipeline						
		212-P Transformer Oil Tank						
		212-R to 216-N-6 Pipeline						
		241-C Waste Line Unplanned Release No. 1						
		241-C Waste Line Unplanned Release No. 2						
		241-Z Diversion Box No. 1						
		241-Z Diversion Box No. 2						
		Sanitary Crib						
200-E BP	Burial Site	200-E Burning Pit	Debris	PO-6				
200-E PAP	Burial Site	200-E Ash Pit		SS-1				
200-E PD	Ditches	200 East Powerhouse Ditch	Cooling Water	SO-1				
200-E-4	French Drain	Critical Mass Laboratory Dry Well North	Miscellaneous Drainage	SO-1				
200-N-3	Burial Site	Ballast Pits	Debris	NO-1				
200-W ADB	Burial Site	200-W Ash Disposal Basin	Ash	SS-2				
200-W ADS	Burial Site	200-W Ash Pit Demolition Site	N/A	SS-2				
200-W BP	Burial Site	200-W Burning Pit	Debris	SS-2				
200-W PAP	Burial Site	200-W Powerhouse Ash Pit	Ash	SS-2				
200-W PP	Ponds	200-W Powerhouse Pond	Cooling Water	TP-2				
201-C	Building	201-C Process Building	Process Condensate	SO-1	3.7	4.9		
207-A		207-A						
207-B	Retention Basin	207-Bb/ Retention Basin	Cooling Water	BP-8				
207-S	Retention Basin	207-S	Cooling Water	RO-2				
207-SL	Retention Basin	207-SL	Lab Waste	RO-3				
207-T	Retention Basin	207-T Retention Basin	Cooling Water	TP-3				
207-Z	Retention Basin	207-Z Retention Basin	Steam Condensate	ZP-2				
209-E-WS-1	French Drain	Critical Mass Laboratory Dry Well East	Miscellaneous Drainage	SO-1				
209-E-WS-2	French Drain	Critical Mass Laboratory Dry Well South	Miscellaneous Drainage	SO-1				
209-E-WS-3	Diversion Box	Critical Mass Laboratory Valve Pit	Process Waste	SO-1				
2101-M POND	Ponds	2101-M Pond	Lab Waste	SS-1				
216-A-1	Cribs	216-A-1	Process Waste	PO-5		0.00571	0.00154	
216-A-10	Cribs	216-A-10	Process Condensate	PO-2	0.329	3.49		42.3

Table B.2. (page 2 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
216-A-11	French Drain	216-A-11	Miscellaneous Drainage	PO-2				
216-A-12	French Drain	216-A-12	Miscellaneous Drainage	PO-2				
216-A-13	French Drain	216-A-13	Miscellaneous Drainage	PO-2				
216-A-14	French Drain	216-A-14	Miscellaneous Drainage	PO-2				
216-A-15	French Drain	216-A-15	Process Condensate	PO-2				
216-A-16	French Drain	216-A-16	Chemical Sewer	PO-5				
216-A-17	French Drain	216-A-17	Chemical Sewer	PO-5				
216-A-18	Trench	216-A-18	Process Waste	PO-5		0.00571	0.00154	
216-A-19	Trench	216-A-19	Process Waste	PO-5		0.00571	0.00154	
216-A-2	Cribs	216-A-2	Process Waste	PO-2		7.42	2	
216-A-20	Trench	216-A-20	Process Waste	PO-5		0.00571	0.00154	
216-A-21	Cribs	216-A-21	Lab Waste	PO-2		8.56	2.31	
216-A-22	French Drain	216-A-22	Miscellaneous Drainage	PO-2				
216-A-23A	French Drain	216-A-23A	Process Condensate	PO-5				
216-A-23B	French Drain	216-A-23B	Process Condensate	PO-5				
216-A-24	Cribs	216-A-24	Process Condensate	PO-5		0.289	0.0779	
216-A-25	Ponds	216-A-25 Pond	Cooling Water	IU-6				
216-A-26	French Drain	216-A-26	Miscellaneous Drainage	PO-2				
216-A-26A	French Drain	216-A-26A	Miscellaneous Drainage	PO-2				
216-A-27	Cribs	216-A-27	Miscellaneous Drainage	PO-2		5.51	1.49	
216-A-28	Cribs	216-A-28	Process Condensate	PO-2				
216-A-29	Ditches	216-A-29	Chemical Sewer	BP-11				
216-A-3	Cribs	216-A-3	Process Waste	PO-2				
216-A-30	Cribs	216-A-30	Steam Condensate	PO-4		0.0751		
216-A-31	Cribs	216-A-31	Process Waste	PO-2		0.514	0.139	
216-A-32	Cribs	216-A-32	Miscellaneous Drainage	PO-2				
216-A-33	French Drain	216-A-33	Miscellaneous Drainage	PO-2				
216-A-34	Cribs	216-A-34	Process Condensate	PO-5				
216-A-35	French Drain	216-A-35	Miscellaneous Drainage	PO-2				
216-A-36A	Cribs	216-A-36A	Process Waste	PO-2		4.57	1.23	
216-A-36B	Cribs	216-A-36B	Process Waste	PO-2		0.0569		0.558
216-A-37-1	Cribs	216-A-37-1	Process Condensate	PO-4		0.000201		
216-A-37-2	Cribs	216-A-37-2	Steam Condensate	PO-4		373		

Table B.2. (page 3 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
216-A-38-1	Cribs	216-A-38-1	N/A	PO-2				
216-A-39	Ditches	216-A-39	Miscellaneous Drainage	PO-3				
216-A-4	Cribs	216-A-4	Lab Waste	PO-2		7.99	2.16	
216-A-40	Retention Basin	216-A-40	Steam Condensate	PO-2				
216-A-41	Cribs	216-A-41	Miscellaneous Drainage	PO-2				
216-A-42	Retention Basin	216-A-42	Cooling Water	PO-4				
216-A-45	Cribs	216-A-45	Process Condensate	PO-2	0.00613	0.0556		0.658
216-A-5	Cribs	216-A-5	Process Condensate	PO-2		3.71	1	
216-A-524	Diversion Box	216-A-524		PO-5				
216-A-6	Cribs	216-A-6	Steam Condensate	PO-4		2.09	0.548	
216-A-7	Cribs	216-A-7	Process Waste	PO-5		0.0571	0.0154	
216-A-8	Cribs	216-A-8	Process Condensate	PO-5				
216-A-9	Cribs	216-A-9	Cooling Water	PO-2		0.02851	0.0077	
216-B-10A	Cribs	216-B-10A Crib	Lab Waste	BP-6	0	0.56		0
216-B-10B	Cribs	216-B-10B Crib	Lab Waste	BP-6	0	0		0
216-B-11A&B	Reverse Well	216-B-11A&B Reverse Wells	Process Condensate	BP-4	0	0.228		0
216-B-12	Cribs	216-B-12 Crib	Process Condensate	BP-9	0	21.4		0
216-B-13	French Drain	216-B-13 French Drain	Miscellaneous Drainage	BP-6				
216-B-14	Cribs	216-B-14 Crib	Scavenged Waste	BP-2	0	1.43		0
216-B-15	Cribs	216-B-15 Crib	Scavenged Waste	BP-2	0	0.285		0
216-B-16	Cribs	216-B-16 Crib	Scavenged Waste	BP-2	0	0.571		0
216-B-17	Cribs	216-B-17 Crib	Scavenged Waste	BP-2	0	0.571		0
216-B-18	Cribs	216-B-18 Crib	Scavenged Waste	BP-2	0	0.571		0
216-B-19	Cribs	216-B-19 Crib	Scavenged Waste	BP-2	0	0.571		0
216-B-2-1	Ditches	216-B-2-1 Ditchb/	Cooling Water	BP-11	0.0026	0.799		
216-B-2-2	Ditches	216-B-2-2 Ditchc/	Cooling Water	BP-11		0.0024		0
216-B-2-3	Ditches	216-B-2-3 Ditch	Cooling Water	BP-11				
216-B-20	Trench	216-B-20 Trench	Scavenged Waste	BP-2	0	0.0742		0
216-B-21	Trench	216-B-21 Trench	Scavenged Waste	BP-2	0	0.58		0
216-B-22	Trench	216-B-22 Trench	Scavenged Waste	BP-2	0	0.148		0
216-B-23	Trench	216-B-23 Trench	Scavenged Waste	BP-2	0	0.102		0
216-B-24	Trench	216-B-24 Trench	Scavenged Waste	BP-2	0	0.44		0
216-B-25	Trench	216-B-25 Trench	Scavenged Waste	BP-2	0	0.114		0

Table B.2. (page 4 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
216-B-26	Trench	216-B-26 Trench	Scavenged Waste	BP-2	0	0.143		0
216-B-27	Trench	216-B-27 Trench	Scavenged Waste	BP-2	0	0.04		0
216-B-28	Trench	216-B-28 Trench	Scavenged Waste	BP-2	0	0.32		0
216-B-29	Trench	216-B-29 Trench	Scavenged Waste	BP-2	0	0.0628		0
216-B-3	Ponds	216-B-3 Ponde/	Cooling Water	BP-11	0.0026	0.799		
216-B-3-1	Ditches	216-B-3-1 Ditchb/	Cooling Water	BP-11				
216-B-3-2	Ditches	216-B-3-2 Ditchc/	Cooling Water	BP-11				
216-B-3-3	Ditches	216-B-3-3 Ditch	Cooling Water	BP-11				
216-B-30	Trench	216-B-30 Trench	Scavenged Waste	BP-2	0	0.12		0
216-B-32	Trench	216-B-32 Trench	Scavenged Waste	BP-2	0	0.148		0
216-B-33	Trench	216-B-33 Trench	Scavenged Waste	BP-2	0	0.674		0
216-B-34	Trench	216-B-34 Trench	Scavenged Waste	BP-2	0	0.325		0
216-B-35	Trench	216-B-35 Trench	Tank Farm Waste	BP-3	0	0.0685		0
216-B-36	Trench	216-B-36 Trench	Tank Farm Waste	BP-3	0	0.0457		0
216-B-37	Trench	216-B-37 Trench	Process Waste	BP-3	0	0.114		0
216-B-38	Trench	216-B-38 Trench	Tank Farm Waste	BP-3	0	0.0685		0
216-B-39	Trench	216-B-39 Trench	Tank Farm Waste	BP-3	0	0.0826		0
216-B-3A	Ponds	216-B-3A Pond	Cooling Water	BP-11				
216-B-3B	Ponds	216-B-3B Pond	Cooling Water	BP-11				
216-B-3C	Ponds	216-B-3C Pond	Cooling Water	BP-11				
216-B-4	Reverse Well	216-B-4 Reverse Well	Miscellaneous Drainage	BP-6	0	0		0
216-B-40	Trench	216-B-40 Trench	Tank Farm Waste	BP-3	0	0.0571		0
216-B-41	Trench	216-B-41 Trench	Tank Farm Waste	BP-3	0	0.0171		0
216-B-42	Trench	216-B-42 Trench	Scavenged Waste	BP-3	0	0.0571		0
216-B-43	Cribs	216-B-43 Crib	Scavenged Waste	BP-1	0	0.0285		0
216-B-44	Cribs	216-B-44 Crib	Scavenged Waste	BP-1	0	0.856		0
216-B-45	Cribs	216-B-45 Crib	Scavenged Waste	BP-1	0	0.571		0
216-B-46	Cribs	216-B-46 Crib	Scavenged Waste	BP-1	0	1010		0
216-B-47	Cribs	216-B-47 Crib	Scavenged Waste	BP-1	0	0.285		0
216-B-48	Cribs	216-B-48 Crib	Scavenged Waste	BP-1	0	0.285		0
216-B-49	Cribs	216-B-49 Crib	Scavenged Waste	BP-1	0	0.856		0
216-B-5	Reverse Well	216-B-5 Reverse Well	Process Waste	BP-6	0	244		0
216-B-50	Cribs	216-B-50 Crib	Process Condensate	BP-1	0	0.0136		0

Table B.2. (page 5 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
216-B-51	French Drain	216-B-51 French Drain	Miscellaneous Drainage	BP-4				
216-B-52	Trench	216-B-52 Trench	Scavenged Waste	BP-2	0	1.08		0
216-B-53A	Trench	216-B-53A Trench	Lab Waste	BP-2	0	5.71		0
216-B-53B	Trench	216-B-53B Trench	Lab Waste	BP-2	0	0.285		0
216-B-54	Trench	216-B-54 Trench	Lab Waste	BP-2	0	0.285		0
216-B-55	Cribs	216-B-55 Crib	Steam Condensate	BP-9	0	0.0000038		0
216-B-56	Cribs	216-B-56 Crib	N/A	BP-6				
216-B-57	Cribs	216-B-57 Crib	Process Condensate	BP-1	0	0.0106		0
216-B-58	Trench	216-B-58 Trench	Lab Waste	BP-2	0	0.393		0
216-B-59	Retention Basin	216-B-59 Basin	Cooling Water	BP-6				
216-B-6	Reverse Well	216-B-6 Reverse Well	Lab Waste	BP-6	0	0		0
216-B-60	Cribs	216-B-60 Crib	Decon Waste	BP-6				
216-B-61	Cribs	216-B-61 Crib	N/A	BP-1				
216-B-62	Cribs	216-B-62 Crib	Process Condensate	BP-9		0.0023		
216-B-63	Ditches	216-B-63 Trench	Chemical Sewer	BP-11		0.0108		
216-B-64	Retention Basin	216-B-64 Basin	N/A	BP-9				
216-B-7A&B	Cribs	216-B-7A&B Crib	Process Waste	BP-4	0	246		0
216-B-8	Cribs	216-B-8TF Crib	Process Waste	BP-4	0	1.7		0
216-B-9	Cribs	216-B-9TF Crib	Process Waste	BP-6	0	9.94		0
216-C-1	Cribs	216-C-1 Crib	Process Condensate	SO-1		0.4579	0.123	
216-C-10	Cribs	216-C-10 Crib	Process Condensate	SO-1				
216-C-2	Reverse Well	216-C-2 Reverse Well	Miscellaneous Drainage	SO-1				
216-C-3	Cribs	216-C-3 Crib	Process Waste	SO-1				
216-C-4	Cribs	216-C-4 Crib	Process Waste	SO-1				
216-C-5	Cribs	216-C-5 Crib	Process Waste	SO-1				
216-C-6	Cribs	216-C-6 Crib	Process Condensate	SO-1				
216-C-7	Cribs	216-C-7 Crib	Process Waste	SO-1				
216-C-8	French Drain	216-C-8	Process Waste	PO-3				
216-C-9	Ponds	216-C-9 Pond	Cooling Water	SO-1				
216-E-28	Ponds	216-E-28 Pond	N/A	BP-11				
216-N-1	Ponds	216-N-1 Pond	Cooling Water	NO-1				
216-N-2	Trench	216-N-2 Trench	Cooling Water	NO-1				
216-N-3	Trench	216-N-3 Trench	Cooling Water	NO-1				

Table B.2. (page 6 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
216-N-4	Ponds	216-N-4 Pond	Cooling Water	NO-1		0.0571	0.0154	
216-N-5	Trench	216-N-5 Trench	Cooling Water	NO-1				
216-N-6	Ponds	216-N-6 Pond	Cooling Water	NO-1		0.0571	0.0132	
216-N-7	Trench	216-N-7 Trench	Cooling Water	NO-1				
216-N-8	Ponds	216-N-8 Pond		IU-6				
216-S-1&2	Cribs	216-S-1 & 2	Process Condensate	RO-2				
216-S-10D	Ditches	216-S-10D	Chemical Sewer	RO-1		0.00468		
216-S-10P	Ponds	216-S-10P	Chemical Sewer	RO-1				
216-S-11	Ponds	216-S-11	Chemical Sewer	RO-1				
216-S-12	Trench	216-S-12	Miscellaneous Drainage	RO-3				
216-S-13	Cribs	216-S-13	Process Waste	RO-2				
216-S-14	Trench	216-S-14	Process Waste	RO-3				
216-S-15	Ponds	216-S-15	Cooling Water	RO-2				
216-S-16D	Ditches	216-S-16D	Cooling Water	RO-1				
216-S-16P	Ponds	216-S-16P	Cooling Water	RO-1				
216-S-17	Ponds	216-S-17	Cooling Water	RO-1				
216-S-172	Diversion Box	216-S-172	Cooling Water	RO-1				
216-S-18	Trench	216-S-18	Debris	RO-2				
216-S-19	Ponds	216-S-19	Lab Waste	RO-1				
216-S-20	Cribs	216-S-20	Lab Waste	RO-3				
216-S-22	Cribs	216-S-22	Process Waste	RO-3				
216-S-23	Cribs	216-S-23	Process Condensate	RO-2				
216-S-25	Cribs	216-S-25	Steam Condensate	RO-1				
216-S-26	Cribs	216-S-26	Lab Waste	RO-3		0.000172		
216-S-3	French Drain	216-S-3	Process Condensate	RO-2				
216-S-4	French Drain	216-S-4	Process Condensate	UP-2				
216-S-5	Cribs	216-S-5	Cooling Water	RO-1				
216-S-6	Cribs	216-S-6	Cooling Water	RO-1				
216-S-7	Cribs	216-S-7	Process Condensate	RO-2				
216-S-8	Trench	216-S-8	Process Waste	RO-2				
216-S-9	Cribs	216-S-9	Process Condensate	RO-2				
216-T-1	Ditches	216-T-1 Ditch	Cooling Water	TP-4				
216-T-10	Trench	216-T-10 Trench	Decon Waste	TP-4				

Table B.2. (page 7 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
216-T-11	Trench	216-T-11 Trench	Decon Waste	TP-4				
216-T-12	Trench	216-T-12 Trench	Cooling Water	TP-3		0.0571	0.0154	
216-T-13	Trench	216-T-13 Trench	Decon Waste	TP-2				
216-T-14	Trench	216-T-14 Trench	Tank Farm Waste	TP-3		0.0502	0.135	
216-T-15	Trench	216-T-15 Trench	Tank Farm Waste	TP-3		0.0537	0.0145	
216-T-16	Trench	216-T-16 Trench	Tank Farm Waste	TP-3		0.0372	0.101	
216-T-17	Trench	216-T-17 Trench	Tank Farm Waste	TP-3		0.303	0.00816	
216-T-18	Cribs	216-T-18 Crib	Tank Farm Waste	TP-2		103	27.7	
216-T-19	Cribs	216-T-19TF Crib and Tile Field	Process Waste	TP-2				
216-T-2	Reverse Well	216-T-2 Reverse Well	Lab Waste	TP-4				
216-T-20	Trench	216-T-20 Trench	Process Waste	TP-2				
216-T-21	Trench	216-T-21 Trench	Tank Farm Waste	TP-1		0.571	0.154	
216-T-22	Trench	216-T-22 Trench	Tank Farm Waste	TP-1		0.114	0.308	
216-T-23	Trench	216-T-23 Trench	Tank Farm Waste	TP-1		0.0571	0.0154	
216-T-24	Trench	216-T-24 Trench	Tank Farm Waste	TP-1		0.114	0.0308	
216-T-25	Trench	216-T-25 Trench	Process Waste	TP-1		0.571	0.154	
216-T-26	Cribs	216-T-26 Crib	Tank Farm Waste	TP-2		3.37	0.908	
216-T-27	Cribs	216-T-27 Crib	Lab Waste	TP-2		0.742	0.2	
216-T-28	Cribs	216-T-28 Crib	Decon Waste	TP-2		4	1.08	
216-T-29	Cribs	216-T-29 Crib	Miscellaneous Drainage	TP-4				
216-T-3	Reverse Well	216-T-3 Reverse Well	Process Waste	TP-4		191	51.5	
216-T-31	French Drain	216-T-31 French Drain	Miscellaneous Drainage	TP-2				
216-T-32	Cribs	216-T-32 Crib	Process Waste	TP-1		1.83	49.3	
216-T-33	Cribs	216-T-33 Crib	Decon Waste	TP-4		0.285	0.077	
216-T-34	Cribs	216-T-34 Crib	Lab Waste	TP-4		6.11	1.65	
216-T-35	Cribs	216-T-35 Crib	Lab Waste	TP-4		3.78	1.02	
216-T-36	Cribs	216-T-36 Crib	Steam Condensate	TP-1		0.142	0.0381	
216-T-4-1D	Ditches	216-T-4-1D Ditch	Cooling Water	TP-3				
216-T-4-2	Ditches	216-T-4-2 Ditch	Steam Condensate	TP-3				
216-T-4A	Ponds	216-T-4A Pond	Cooling Water	TP-3				
216-T-4B	Ponds	216-T-4B Pond	Cooling Water	TP-3				
216-T-5	Trench	216-T-5 Trench	Tank Farm Waste	TP-1		10.3	2.77	
216-T-6	Cribs	216-T-6 Crib	Process Waste	TP-3		22.3	6.01	

Table B.2. (page 8 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
216-T-7	Cribs	216-T-7TF Crib and Tile Field	Tank Farm Waste	TP-1		7.42	2	
216-T-8	Cribs	216-T-8 Crib	Lab Waste	TP-4		0.285	0.077	
216-T-9	Trench	216-T-9 Trench	Decon Waste	TP-4				
216-U-1&2	Cribs	216-U-1 & 216-U-2	Process Condensate	UP-2		2.43	0.656	
216-U-10	Ponds	216-U-10	Cooling Water	UP-2		0.768		
216-U-11	Ditches	216-U-11	Cooling Water	UP-2				
216-U-12	Cribs	216-U-12	Process Condensate	UP-2		0.0123		
216-U-13	Trench	216-U-13 (same as UN-200-W- 125)	Decon Waste	UP-2		0.00571	0.00154	
216-U-14	Ditches	216-U-14	Cooling Water	UP-2				
216-U-15	Trench	216-U-15	Process Waste	UP-2		0.00571	0.00154	
216-U-16	Cribs	216-U-16	Process Condensate	UP-2		0.0902		
216-U-17	Cribs	216-U-17	Process Condensate	UP-2		0.0000296		
216-U-21		216-U-21				0.119	0.032	
216-U-3	French Drain	216-U-3	Miscellaneous Drainage	UP-2		0.00571	0.00154	
216-U-4	Reverse Well	216-U-4	Lab Waste	UP-2				
216-U-4A	French Drain	216-U-4A	Miscellaneous Drainage	UP-2		0.00051	0.00013	
216-U-4B	French Drain	216-U-4B	Miscellaneous Drainage	UP-2		0.00308	0.00083	
216-U-5	Trench	216-U-5 & 216-U-6	Process Waste	UP-2		0.00285	0.00077	
216-U-7	French Drain	216-U-7	Miscellaneous Drainage	UP-2				
216-U-8	Cribs	216-U-8	Process Condensate	UP-2		21.8	5.7	
216-U-9	Ditches	216-U-9	Cooling Water	RO-1				
216-W-LWC	Cribs	216-W-LWC Crib	Chemical Sewer	SS-2				
216-Z-1&2	Cribs	216-Z-1 & 216-Z-2 Cribs	Process Waste	ZP-2		2680	992	
216-Z-10	Reverse Well	216-Z-10 Reverse Well	Process Waste	ZP-2	0.14	2.85	0.77	2
216-Z-11	Ditches	216-Z-11	Cooling Water	UP-2		137	37	
216-Z-12	Cribs	216-Z-12 Crib	Process Waste	ZP-2		1430	386	
216-Z-13	French Drain	216-Z-13 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-14	French Drain	216-Z-14 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-15	French Drain	216-Z-15 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-16	Cribs	216-Z-16 Crib	Lab Waste	ZP-2		4.09	1.1	
216-Z-17	Ditches	216-Z-17 Trench	Lab Waste	ZP-2		2.87	0.225	
216-Z-18	Cribs	216-Z-18 Crib	Process Waste	ZP-2		1310	353	
216-Z-19	Ditches	216-Z-19	Cooling Water	UP-2				

Table B.2. (page 9 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
216-Z-1A	Cribs	216-Z-1A Tile Field	Process Waste	ZP-2		137	37	
216-Z-1D	Ditches	216-Z-1D	Cooling Water	UP-2		137	37	
216-Z-20	Cribs	216-Z-20	Cooling Water	UP-2	0.0153	2.03		2.51
216-Z-21	Retention Basin	216-Z-21 Seepage Basin	Cooling Water	ZP-2				
216-Z-3	Cribs	216-Z-3 Crib	Process Waste	ZP-2		325	87.8	
216-Z-4	Trench	216-Z-4 Trench	Process Waste	ZP-2				
216-Z-5	Cribs	216-Z-5 Crib	Process Waste	ZP-2		19.4	5.24	
216-Z-6	Cribs	216-Z-6 Crib	Process Waste	ZP-2		0.28	0.077	
216-Z-7	Cribs	216-Z-7 Crib	Lab Waste	ZP-2		114	30.8	
216-Z-8	Cribs	216-Z-8 French Drain	Process Waste	ZP-2	0.13	2.76	0.745	
216-Z-9	Cribs	216-Z-9 Trench	Process Waste	ZP-2		2190	590	
218-C-9	Burial Site	218-C-9 Burial Ground	LLW - SOLID	SO-1				
218-E-1	Burial Site	218-E-1	LLW - SOLID	PO-2		51.4	13.9	
218-E-10	Burial Site	218-E-10 Burial Ground	LLW - SOLID	BP-10				
218-E-12A	Burial Site	218-E-12A	LLW - SOLID	PO-6		510	138	
218-E-12B	Burial Site	218-E-12B	LLW - SOLID	PO-6				
218-E-13	Burial Site	218-E-13		PO-2				
218-E-2	Burial Site	218-E-2 Burial Ground	LLW - SOLID	BP-10				
218-E-2A	Burial Site	218-E-2A Burial Ground	LLW - SOLID	BP-10				
218-E-4	Burial Site	218-E-4 Burial Ground	LLW - SOLID	BP-10				
218-E-5	Burial Site	218-E-5 Burial Ground	LLW - SOLID	BP-10				
218-E-5A	Burial Site	218-E-5A Burial Ground	LLW - SOLID	BP-10				
218-E-6	Burial Site	218-E-6 Burial Ground	Debris	BP-6				
218-E-7	Burial Site	218-E-7 Burial Ground	Lab Waste	BP-6				
218-E-8	Burial Site	218-E-8	TRU Solid Waste	PO-6		1.14	0.308	
218-E-9	Burial Site	218-E-9 Burial Ground	LLW - SOLID	BP-10				
218-W-1	Burial Site	218-W-1 Burial Ground	TRU Solid Waste	ZP-3		5370	1450	
218-W-11	Burial Site	218-W-11 Burial Ground	LLW - SOLID	ZP-3				
218-W-1A	Burial Site	218-W-1A Burial Ground	LLW - SOLID	ZP-3		114	30.8	
218-W-2	Burial Site	218-W-2 Burial Ground	TRU Solid Waste	ZP-3		7190	1940	
218-W-2A	Burial Site	218-W-2A Burial Ground	LLW - SOLID	ZP-3				
218-W-3	Burial Site	218-W-3 Burial Ground	TRU Solid Waste	ZP-3		3880	1050	
218-W-3A	Burial Site	218-W-3A Burial Ground	TRU Solid Waste	ZP-3				

Table B.2. (page 10 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
218-W-3AE	Burial Site	218-W-3AE Burial Ground	LLW - SOLID	ZP-3				
218-W-4A	Burial Site	218-W-4A Burial Ground	TRU Solid Waste	ZP-3				
218-W-4B	Burial Site	218-W-4B Caissons	TRU Solid Waste	ZP-3				
218-W-4B	Burial Site	218-W-4B Trenches	TRU Solid Waste	ZP-3				
218-W-4C	Burial Site	218-W-4C Burial Ground	TRU Solid Waste	ZP-3				
218-W-5	Burial Site	218-W-5 Burial Ground	TRU Solid Waste	ZP-3				
218-W-6	Burial Site	218-W-6 Burial Ground	LLW - SOLID	ZP-3				
218-W-7	Burial Site	218-W-7	LLW - SOLID	RO-3				
218-W-8	Burial Site	218-W-8 Burial Ground	Lab Waste	TP-4		0.171	0.00462	
218-W-9	Burial Site	218-W-9	LLW - SOLID	RO-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
232-Z	Building	232-Z Incinerator		ZP-2				
240-S-151	Diversion Box	240-S-151	LLW - SOLID	RO-3				
240-S-152	Diversion Box	240-S-152	Tank Farm Waste	RO-3				
240-S-302	Tanks	240-S-302	Lab Waste	RO-3				
241-A-151	Diversion Box	241-A-151	Tank Farm Waste	PO-2				
241-A-152	Diversion Box	241-A-152	Process Waste	PO-3				
241-A-153	Diversion Box	241-A-153	Process Waste	PO-3				
241-A-302A	Tanks	241-A-302A		PO-2				
241-A-302B	Tanks	241-A-302B		PO-5				
241-A-350	Tanks	241-A-350	Process Waste	PO-3				
241-A-417	Tanks	241-A-417	Process Waste	PO-3				
241-A-A	Diversion Box	241-A-A	Process Waste	PO-3				
241-A-B	Diversion Box	241-A-B	Process Waste	PO-3				
241-AN-A	Diversion Box	241-AN-A	Process Waste	PO-3				
241-AN-B	Diversion Box	241-AN-B	Process Waste	PO-3				

Table B.2. (page 11 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
241-AP VP	Valve Pit	241-AP	Process Waste	PO-3				
241-AR-151	Diversion Box	241-AR-151	Process Waste	PO-3				
241-AW-A	Diversion Box	241-AW-A	Process Waste	PO-3				
241-AW-B	Diversion Box	241-AW-B	Process Waste	PO-3				
241-AX-151	Diversion Box	241-AX-151	Process Waste	PO-3				
241-AX-152DS	Tanks	241-AX-152DS	Process Waste	PO-3				
241-AX-155	Diversion Box	241-AX-155	Tank Farm Waste	PO-3				
241-AX-501	Valve Pit	241-AX-501		PO-3				
241-AX-A	Diversion Box	241-AX-A		PO-3				
241-AX-B	Diversion Box	241-AX-B		PO-3				
241-AY-151	Diversion Box	241-AY-151	Process Waste	PO-3				
241-AY-152	Diversion Box	241-AY-152	Process Waste	PO-3				
241-AZ-151DS	Diversion Box	241-AZ-151DS		PO-3				
241-AZ-152	Diversion Box	241-AZ-152		PO-3				
241-C-151	Diversion Box	241-C-151		PO-3				
241-C-152	Diversion Box	241-C-152		PO-3				
241-C-153	Diversion Box	241-C-153		PO-3				
241-C-154	Diversion Box	241-C-154 Diversion Box	Process Waste	SO-1				
241-C-252	Diversion Box	241-C-252		PO-3				
241-C-301C	Tanks	241-C-301C		PO-3				
241-CR-151	Diversion Box	241-CR-151		PO-3				
241-CR-152	Diversion Box	241-CR-152		PO-3				
241-CR-153	Diversion Box	241-CR-153		PO-3				
241-CX-TK-70	Tanks	241-CX-70 Storage Tank	Tank Farm Waste	SO-1				
241-CX-TK-71	Tanks	241-CX-71 Storage Tank	Process Condensate	SO-1		0.4579	0.123	
241-CX-TK-72	Tanks	241-CX-72 Storage Tank	Process Waste	SO-1				
241-ER-153	Diversion Box	241-ER-153		PO-3				
241-S-151	Diversion Box	241-S-151	LLW - SOLID	RO-2				
241-S-152	Diversion Box	241-S-152	Tank Farm Waste	RO-4				
241-S-302A	Tanks	241-S-302A	Lab Waste	RO-2				
241-S-302B	Tanks	241-S-302B	LLW - SOLID	RO-4				
241-S-A	Diversion Box	241-S-A	Tank Farm Waste	RO-4				
241-S-B	Diversion Box	241-S-B	Tank Farm Waste	RO-4				

Table B.2. (page 12 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
241-S-C	Diversion Box	241-S-C	Tank Farm Waste	RO-4				
241-S-D	Diversion Box	241-S-D	Tank Farm Waste	RO-4				
241-SX-151	Diversion Box	241-SX-151	Tank Farm Waste	RO-4				
241-SX-152	Diversion Box	241-SX-152	Tank Farm Waste	RO-4				
241-SX-302	Tanks	241-SX-302		RO-2				
241-SX-A	Diversion Box	241-SX-A		RO-4				
241-SX-B	Diversion Box	241-SX-B		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-T-151	Diversion Box	241-T-151 Diversion Box	Tank Farm Waste	TP-6				
241-T-152	Diversion Box	241-T-152 Diversion Box	Tank Farm Waste	TP-6				
241-T-153	Diversion Box	241-T-153 Diversion Box	Tank Farm Waste	TP-6				
241-T-252	Diversion Box	241-T-252 Diversion Box	Tank Farm Waste	TP-6				
241-T-301	Tanks	241-T-301 Catch Tank	Tank Farm Waste	TP-6				
241-T-302	Tanks	241-T-302 Catch Tank	Tank Farm Waste	TP-6				
241-T-361	Tanks	241-T-361 Settling Tank	Process Waste	TP-4				
241-TR-152	Diversion Box	241-TR-152 Diversion Box	Tank Farm Waste	TP-6				
241-TR-153	Diversion Box	241-TR-153 Diversion Box	Tank Farm Waste	TP-6				
241-TX-152	Diversion Box	241-TX-152 Diversion Box	Tank Farm Waste	TP-2				
241-TX-153	Diversion Box	241-TX-153 Diversion Box	Tank Farm Waste	TP-5				
241-TX-154	Diversion Box	241-TX-154 Diversion Box	Tank Farm Waste	TP-4				
241-TX-155	Diversion Box	241-TX-155 Diversion Box	Tank Farm Waste	TP-2				
241-TX-302A	Tanks	241-TX-302A Catch Tank	Tank Farm Waste	TP-5				
241-TX-302B	Tanks	241-TX-302B Catch Tank	Tank Farm Waste	TP-2				
241-TX-302C	Tanks	241-TX-302C Catch Tank	Tank Farm Waste	TP-4				
241-TXR-151	Diversion Box	241-TXR-151 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-152	Diversion Box	241-TXR-152 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-153	Diversion Box	241-TXR-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-153	Diversion Box	241-TY-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-302A	Tanks	241-TY-302A Catch Tank	Tank Farm Waste	TP-5				
241-TY-302B	Tanks	241-TY-302B Catch Tank	Tank Farm Waste	TP-5				

Table B.2. (page 13 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
241-Z-361	Tanks	241-Z-361 Settling Tank	Process Waste	ZP-2				
241-Z-TK-8	Tanks	216-Z-8 Settling Tank	Process Waste	ZP-2				
241-Z-TK-D5	Tanks	241-Z Treatment Tank	Process Waste	ZP-2				
242-T-151	Diversion Box	242-T-151 Diversion Box	Process Condensate	TP-5				
244-A RT	Tanks	244-A	Process Waste	PO-3				
244-AR VAULT	Vault	244-AR	Process Waste	PO-3				
244-CR VAULT	Vault	244-CR	Process Waste	PO-3				
244-S RT	Tanks	244-S Receiver Tank		RO-2				
244-TX RT	Tanks	244-TX Receiving Tank		TP-5				
244-TXR	Vault	244-TXR Vault	Tank Farm Waste	TP-5				
2607-E5	Septic System	2607-E-5 Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-E6	Septic System	2607-E6	Sanitary Waste	PO-2				
2607-E7A	Septic System	2607-E-7A Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-EA	Septic System	2607-EA	Sanitary Waste	PO-2				
2607-EC	Septic System	2607-EC	Sanitary Waste	PO-5				
2607-ED	Septic System	2607-ED	Sanitary Waste	PO-3				
2607-EE	Septic System	2607-EL	Sanitary Waste	PO-2				
2607-EG	Septic System	2607-EG	Sanitary Waste	PO-3				
2607-EJ	Septic System	2607-EJ	Sanitary Waste	PO-3				
2607-N	Septic System	2607-N Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-P	Septic System	2607-P Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-R	Septic System	2607-R Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-W1	Septic System	2607-W1 Septic Tank	Sanitary Waste	SS-2				
2607-W2	Septic System	2607-W2 Septic Tank	Sanitary Waste	SS-2				
2607-W3	Septic System	2607-W3 Septic Tank	Sanitary Waste	TP-4				
2607-W4	Septic System	2607-W4 Septic Tank	Sanitary Waste	TP-4				
2607-W6	Septic System	2607-W6	Sanitary Waste	RO-3				
2607-W8	Septic System	2607-W-8 Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WA	Septic System	2607-WA Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WB		2607-WB Septic Tank and Drain Field						
2607-WT	Septic System	2607-WT Septic Tank	Sanitary Waste	TP-5				
2607-WTX	Septic System	2607-WTX Septic Tank	Sanitary Waste	TP-5				
2607-WZ	Septic System	2607-WZ	Sanitary Waste	RO-1				

Table B.2. (page 14 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-238, Units (Ci)	Plutonium-239, Units (Ci)	Plutonium-240, Units (Ci)	Plutonium-241, Units (Ci)
2607-Z	Septic System	2607-Z Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-Z-1		2607-Z-1 Septic Tank and Drain Field						
2704-C-WS-1	French Drain	2704-C-WS-1, 2704-C French Drain, Gatehouse French Drain	Miscellaneous Drainage	SO-1				
2904-S-160	Diversion Box	2904-S-160	Cooling Water	RO-1				
2904-S-170	Diversion Box	2904-S-170	Process Waste	RO-1				
2904-S-171	Diversion Box	2904-S-171	Cooling Water	RO-1				
291-C	Building	291-C Ventilation System	Process Condensate	SO-1				
299-E24-111	Reverse Well	299-E24-111		PO-2				
HSVP	Diversion Box	Semi-Works Valve Pit	Process Waste	SO-1				
UPR-200-E-141		UN-200-E-141	Solution Storage (1)	SO-1				
UPR-200-E-36		UN-200-E-36	Process & Decon Wastes (1)	SO-1				
UPR-200-E-37		UN-200-E-37	Process & Decon Wastes (1)	SO-1				
UPR-200-E-98		UN-200-E-98	Process & Decon Wastes (3)	SO-1				
UPR-200-W-160		UPR-200-W-160 Unplanned Release	Tank Farm Waste	TP-4				
Z PLANT BP	Burial Site	Z Plant Burn Pit	Debris	ZP-3				

Table B.3. Environmental Waste Site Inventories for Plutonium-242, Total Uranium, Gross Uranium, and Uranium-235 (page 1 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
		212-N to 216-N-1 Pipeline						
		212-P Hazardous Waste Staging Area						
		212-P to 216-N-4 Pipeline						
		212-P Transformer Oil Tank						
		212-R to 216-N-6 Pipeline						
		241-C Waste Line Unplanned Release No. 1						
		241-C Waste Line Unplanned Release No. 2						
		241-Z Diversion Box No. 1						
		241-Z Diversion Box No. 2						
		Sanitary Crib						
200-E BP	Burial Site	200-E Burning Pit	Debris	PO-6				
200-E PAP	Burial Site	200-E Ash Pit		SS-1				
200-E PD	Ditches	200 East Powerhouse Ditch	Cooling Water	SO-1				
200-E-4	French Drain	Critical Mass Laboratory Dry Well North	Miscellaneous Drainage	SO-1				
200-N-3	Burial Site	Ballast Pits	Debris	NO-1				
200-W ADB	Burial Site	200-W Ash Disposal Basin	Ash	SS-2				
200-W ADS	Burial Site	200-W Ash Pit Demolition Site	N/A	SS-2				
200-W BP	Burial Site	200-W Burning Pit	Debris	SS-2				
200-W PAP	Burial Site	200-W Powerhouse Ash Pit	Ash	SS-2				
200-W PP	Ponds	200-W Powerhouse Pond	Cooling Water	TP-2				
201-C	Building	201-C Process Building	Process Condensate	SO-1				
207-A		207-A						
207-B	Retention Basin	207-Bb/ Retention Basin	Cooling Water	BP-8				
207-S	Retention Basin	207-S	Cooling Water	RO-2				
207-SL	Retention Basin	207-SL	Lab Waste	RO-3				
207-T	Retention Basin	207-T Retention Basin	Cooling Water	TP-3				
207-Z	Retention Basin	207-Z Retention Basin	Steam Condensate	ZP-2				
209-E-WS-1	French Drain	Critical Mass Laboratory Dry Well East	Miscellaneous Drainage	SO-1				
209-E-WS-2	French Drain	Critical Mass Laboratory Dry Well South	Miscellaneous Drainage	SO-1				
209-E-WS-3	Diversion Box	Critical Mass Laboratory Valve Pit	Process Waste	SO-1				
2101-M POND	Ponds	2101-M Pond	Lab Waste	SS-1				
216-A-1	Cribs	216-A-1	Process Waste	PO-5		0.0514		
216-A-10	Cribs	216-A-10	Process Condensate	PO-2		0.081		

Table B.3. (page 2 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
216-A-11	French Drain	216-A-11	Miscellaneous Drainage	PO-2				
216-A-12	French Drain	216-A-12	Miscellaneous Drainage	PO-2				
216-A-13	French Drain	216-A-13	Miscellaneous Drainage	PO-2				
216-A-14	French Drain	216-A-14	Miscellaneous Drainage	PO-2				
216-A-15	French Drain	216-A-15	Process Condensate	PO-2				
216-A-16	French Drain	216-A-16	Chemical Sewer	PO-5				
216-A-17	French Drain	216-A-17	Chemical Sewer	PO-5				
216-A-18	Trench	216-A-18	Process Waste	PO-5		0.469		
216-A-19	Trench	216-A-19	Process Waste	PO-5		13		
216-A-2	Cribs	216-A-2	Process Waste	PO-2		0.026		
216-A-20	Trench	216-A-20	Process Waste	PO-5		0.135		
216-A-21	Cribs	216-A-21	Lab Waste	PO-2		0.065		
216-A-22	French Drain	216-A-22	Miscellaneous Drainage	PO-2				
216-A-23A	French Drain	216-A-23A	Process Condensate	PO-5				
216-A-23B	French Drain	216-A-23B	Process Condensate	PO-5				
216-A-24	Cribs	216-A-24	Process Condensate	PO-5		0.0167		
216-A-25	Ponds	216-A-25 Pond	Cooling Water	IU-6		4.24		
216-A-26	French Drain	216-A-26	Miscellaneous Drainage	PO-2				
216-A-26A	French Drain	216-A-26A	Miscellaneous Drainage	PO-2				
216-A-27	Cribs	216-A-27	Miscellaneous Drainage	PO-2		0.0227		
216-A-28	Cribs	216-A-28	Process Condensate	PO-2		0.211		
216-A-29	Ditches	216-A-29	Chemical Sewer	BP-11				
216-A-3	Cribs	216-A-3	Process Waste	PO-2		0.559		
216-A-30	Cribs	216-A-30	Steam Condensate	PO-4		0.1		
216-A-31	Cribs	216-A-31	Process Waste	PO-2		0.00683		
216-A-32	Cribs	216-A-32	Miscellaneous Drainage	PO-2				
216-A-33	French Drain	216-A-33	Miscellaneous Drainage	PO-2				
216-A-34	Cribs	216-A-34	Process Condensate	PO-5				
216-A-35	French Drain	216-A-35	Miscellaneous Drainage	PO-2				
216-A-36A	Cribs	216-A-36A	Process Waste	PO-2		0.0484		
216-A-36B	Cribs	216-A-36B	Process Waste	PO-2		0.0398		
216-A-37-1	Cribs	216-A-37-1	Process Condensate	PO-4		0.0109		
216-A-37-2	Cribs	216-A-37-2	Steam Condensate	PO-4		0.0172		

Table B.3. (page 3 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
216-A-38-1	Cribs	216-A-38-1	N/A	PO-2				
216-A-39	Ditches	216-A-39	Miscellaneous Drainage	PO-3				
216-A-4	Cribs	216-A-4	Lab Waste	PO-2		0.133		
216-A-40	Retention Basin	216-A-40	Steam Condensate	PO-2				
216-A-41	Cribs	216-A-41	Miscellaneous Drainage	PO-2				
216-A-42	Retention Basin	216-A-42	Cooling Water	PO-4				
216-A-45	Cribs	216-A-45	Process Condensate	PO-2		0.00225		
216-A-5	Cribs	216-A-5	Process Condensate	PO-2		0.0877		
216-A-524	Diversion Box	216-A-524		PO-5				
216-A-6	Cribs	216-A-6	Steam Condensate	PO-4		0.055		
216-A-7	Cribs	216-A-7	Process Waste	PO-5		0.00227		
216-A-8	Cribs	216-A-8	Process Condensate	PO-5		0.123		
216-A-9	Cribs	216-A-9	Cooling Water	PO-2		0.0000757		
216-B-10A	Cribs	216-B-10A Crib	Lab Waste	BP-6		0.00302		0
216-B-10B	Cribs	216-B-10B Crib	Lab Waste	BP-6		0.000000249		0
216-B-11A&B	Reverse Well	216-B-11A&B Reverse Wells	Process Condensate	BP-4		0.00454		0
216-B-12	Cribs	216-B-12 Crib	Process Condensate	BP-9		6.96		0
216-B-13	French Drain	216-B-13 French Drain	Miscellaneous Drainage	BP-6				
216-B-14	Cribs	216-B-14 Crib	Scavenged Waste	BP-2		0.0726		0
216-B-15	Cribs	216-B-15 Crib	Scavenged Waste	BP-2		0.0348		0
216-B-16	Cribs	216-B-16 Crib	Scavenged Waste	BP-2		0.107		0
216-B-17	Cribs	216-B-17 Crib	Scavenged Waste	BP-2		0.118		0
216-B-18	Cribs	216-B-18 Crib	Scavenged Waste	BP-2		0.0786		0
216-B-19	Cribs	216-B-19 Crib	Scavenged Waste	BP-2		0.0605		0
216-B-2-1	Ditches	216-B-2-1 Ditchb/	Cooling Water	BP-11		2.1		
216-B-2-2	Ditches	216-B-2-2 Ditchc/	Cooling Water	BP-11		0.0000157		0
216-B-2-3	Ditches	216-B-2-3 Ditch	Cooling Water	BP-11				
216-B-20	Trench	216-B-20 Trench	Scavenged Waste	BP-2		0.117		0
216-B-21	Trench	216-B-21 Trench	Scavenged Waste	BP-2		0.225		0
216-B-22	Trench	216-B-22 Trench	Scavenged Waste	BP-2		0.139		0
216-B-23	Trench	216-B-23 Trench	Scavenged Waste	BP-2		0.052		0
216-B-24	Trench	216-B-24 Trench	Scavenged Waste	BP-2		0.082		0
216-B-25	Trench	216-B-25 Trench	Scavenged Waste	BP-2		0.0051		0

Table B.3. (page 4 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
216-B-26	Trench	216-B-26 Trench	Scavenged Waste	BP-2		0.196		0
216-B-27	Trench	216-B-27 Trench	Scavenged Waste	BP-2		0.114		0
216-B-28	Trench	216-B-28 Trench	Scavenged Waste	BP-2		0.1		0
216-B-29	Trench	216-B-29 Trench	Scavenged Waste	BP-2		0.115		0
216-B-3	Ponds	216-B-3 Ponds/	Cooling Water	BP-11		2.1		
216-B-3-1	Ditches	216-B-3-1 Ditchb/	Cooling Water	BP-11				
216-B-3-2	Ditches	216-B-3-2 Ditchc/.	Cooling Water	BP-11				
216-B-3-3	Ditches	216-B-3-3 Ditch	Cooling Water	BP-11				
216-B-30	Trench	216-B-30 Trench	Scavenged Waste	BP-2		0.0293		0
216-B-32	Trench	216-B-32 Trench	Scavenged Waste	BP-2		0.00367		0
216-B-33	Trench	216-B-33 Trench	Scavenged Waste	BP-2		0.00667		0
216-B-34	Trench	216-B-34 Trench	Scavenged Waste	BP-2		0.0283		0
216-B-35	Trench	216-B-35 Trench	Tank Farm Waste	BP-3		0.00557		0
216-B-36	Trench	216-B-36 Trench	Tank Farm Waste	BP-3		0.00532		0
216-B-37	Trench	216-B-37 Trench	Process Waste	BP-3		0.00121		0
216-B-38	Trench	216-B-38 Trench	Tank Farm Waste	BP-3		0.0141		0
216-B-39	Trench	216-B-39 Trench	Tank Farm Waste	BP-3		0.00193		0
216-B-3A	Ponds	216-B-3A Pond	Cooling Water	BP-11				
216-B-3B	Ponds	216-B-3B Pond	Cooling Water	BP-11				
216-B-3C	Ponds	216-B-3C Pond	Cooling Water	BP-11				
216-B-4	Reverse Well	216-B-4 Reverse Well	Miscellaneous Drainage	BP-6		0		0
216-B-40	Trench	216-B-40 Trench	Tank Farm Waste	BP-3		0.017		0
216-B-41	Trench	216-B-41 Trench	Tank Farm Waste	BP-3		0.0025		0
216-B-42	Trench	216-B-42 Trench	Scavenged Waste	BP-3		0.227		0
216-B-43	Cribs	216-B-43 Crib	Scavenged Waste	BP-1		0.00454		0
216-B-44	Cribs	216-B-44 Crib	Scavenged Waste	BP-1		0.000756		0
216-B-45	Cribs	216-B-45 Crib	Scavenged Waste	BP-1		0.00227		0
216-B-46	Cribs	216-B-46 Crib	Scavenged Waste	BP-1		0.0635		0
216-B-47	Cribs	216-B-47 Crib	Scavenged Waste	BP-1		0.00227		0
216-B-48	Cribs	216-B-48 Crib	Scavenged Waste	BP-1		0.000757		0
216-B-49	Cribs	216-B-49 Crib	Scavenged Waste	BP-1		0.106		0
216-B-5	Reverse Well	216-B-5 Reverse Well	Process Waste	BP-6		0		0
216-B-50	Cribs	216-B-50 Crib	Process Condensate	BP-1		0.000095		0

Table B.3. (page 5 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
216-B-51	French Drain	216-B-51 French Drain	Miscellaneous Drainage	BP-4				
216-B-52	Trench	216-B-52 Trench	Scavenged Waste	BP-2		0.00998		0
216-B-53A	Trench	216-B-53A Trench	Lab Waste	BP-2		0.00756		0
216-B-53B	Trench	216-B-53B Trench	Lab Waste	BP-2		0.00302		0
216-B-54	Trench	216-B-54 Trench	Lab Waste	BP-2		0.00302		0
216-B-55	Cribs	216-B-55 Crib	Steam Condensate	BP-9		0.0268		0
216-B-56	Cribs	216-B-56 Crib	N/A	BP-6				
216-B-57	Cribs	216-B-57 Crib	Process Condensate	BP-1		0.000297		0
216-B-58	Trench	216-B-58 Trench	Lab Waste	BP-2		0.00304		0
216-B-59	Retention Basin	216-B-59 Basin	Cooling Water	BP-6				
216-B-6	Reverse Well	216-B-6 Reverse Well	Lab Waste	BP-6		0		0
216-B-60	Cribs	216-B-60 Crib	Decon Waste	BP-6				
216-B-61	Cribs	216-B-61 Crib	N/A	BP-1				
216-B-62	Cribs	216-B-62 Crib	Process Condensate	BP-9		0.01		
216-B-63	Ditches	216-B-63 Trench	Chemical Sewer	BP-11		0.15		
216-B-64	Retention Basin	216-B-64 Basin	N/A	BP-9				
216-B-7A&B	Cribs	216-B-7A&B Crib	Process Waste	BP-4		0.0606		0
216-B-8	Cribs	216-B-8TF Crib	Process Waste	BP-4		0.0151		0
216-B-9	Cribs	216-B-9TF Crib	Process Waste	BP-6		0.0151		0
216-C-1	Cribs	216-C-1 Crib	Process Condensate	SO-1				
216-C-10	Cribs	216-C-10 Crib	Process Condensate	SO-1				
216-C-2	Reverse Well	216-C-2 Reverse Well	Miscellaneous Drainage	SO-1				
216-C-3	Cribs	216-C-3 Crib	Process Waste	SO-1				
216-C-4	Cribs	216-C-4 Crib	Process Waste	SO-1				
216-C-5	Cribs	216-C-5 Crib	Process Waste	SO-1				
216-C-6	Cribs	216-C-6 Crib	Process Condensate	SO-1				
216-C-7	Cribs	216-C-7 Crib	Process Waste	SO-1				
216-C-8	French Drain	216-C-8	Process Waste	PO-3				
216-C-9	Ponds	216-C-9 Pond	Cooling Water	SO-1				
216-E-28	Ponds	216-E-28 Pond	N/A	BP-11				
216-N-1	Ponds	216-N-1 Pond	Cooling Water	NO-1				
216-N-2	Trench	216-N-2 Trench	Cooling Water	NO-1				
216-N-3	Trench	216-N-3 Trench	Cooling Water	NO-1				

Table B.3. (page 6 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
216-N-4	Ponds	216-N-4 Pond	Cooling Water	NO-1		4500 g	0.00151	
216-N-5	Trench	216-N-5 Trench	Cooling Water	NO-1				
216-N-6	Ponds	216-N-6 Pond	Cooling Water	NO-1		4500 g	0.00151	
216-N-7	Trench	216-N-7 Trench	Cooling Water	NO-1				
216-N-8	Ponds	216-N-8 Pond		IU-6				
216-S-1&2	Cribs	216-S-1 & 2	Process Condensate	RO-2		0.756		
216-S-10D	Ditches	216-S-10D	Chemical Sewer	RO-1		0.0671		
216-S-10P	Ponds	216-S-10P	Chemical Sewer	RO-1				
216-S-11	Ponds	216-S-11	Chemical Sewer	RO-1		0.00685		
216-S-12	Trench	216-S-12	Miscellaneous Drainage	RO-3		0.00166		
216-S-13	Cribs	216-S-13	Process Waste	RO-2		0.0303		
216-S-14	Trench	216-S-14	Process Waste	RO-3				
216-S-15	Ponds	216-S-15	Cooling Water	RO-2				
216-S-16D	Ditches	216-S-16D	Cooling Water	RO-1				
216-S-16P	Ponds	216-S-16P	Cooling Water	RO-1		1.05		
216-S-17	Ponds	216-S-17	Cooling Water	RO-1		0.0453		
216-S-172	Diversion Box	216-S-172	Cooling Water	RO-1				
216-S-18	Trench	216-S-18	Debris	RO-2				
216-S-19	Ponds	216-S-19	Lab Waste	RO-1		0.0518		
216-S-20	Cribs	216-S-20	Lab Waste	RO-3		0.0125		
216-S-22	Cribs	216-S-22	Process Waste	RO-3		0.000015		
216-S-23	Cribs	216-S-23	Process Condensate	RO-2		0.000129		
216-S-25	Cribs	216-S-25	Steam Condensate	RO-1		0.0555		
216-S-26	Cribs	216-S-26	Lab Waste	RO-3				
216-S-3	French Drain	216-S-3	Process Condensate	RO-2		0.000127		
216-S-4	French Drain	216-S-4	Process Condensate	UP-2				
216-S-5	Cribs	216-S-5	Cooling Water	RO-1		0.0907		
216-S-6	Cribs	216-S-6	Cooling Water	RO-1		0.0906		
216-S-7	Cribs	216-S-7	Process Condensate	RO-2		0.862		
216-S-8	Trench	216-S-8	Process Waste	RO-2		0.065		
216-S-9	Cribs	216-S-9	Process Condensate	RO-2		0.0113		
216-T-1	Ditches	216-T-1 Ditch	Cooling Water	TP-4				
216-T-10	Trench	216-T-10 Trench	Decon Waste	TP-4				

Table B.3. (page 7 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
216-T-11	Trench	216-T-11 Trench	Decon Waste	TP-4				
216-T-12	Trench	216-T-12 Trench	Cooling Water	TP-3				
216-T-13	Trench	216-T-13 Trench	Decon Waste	TP-2				
216-T-14	Trench	216-T-14 Trench	Tank Farm Waste	TP-3				
216-T-15	Trench	216-T-15 Trench	Tank Farm Waste	TP-3				
216-T-16	Trench	216-T-16 Trench	Tank Farm Waste	TP-3				
216-T-17	Trench	216-T-17 Trench	Tank Farm Waste	TP-3				
216-T-18	Cribs	216-T-18 Crib	Tank Farm Waste	TP-2				
216-T-19	Cribs	216-T-19TF Crib and Tile Field	Process Waste	TP-2				
216-T-2	Reverse Well	216-T-2 Reverse Well	Lab Waste	TP-4				
216-T-20	Trench	216-T-20 Trench	Process Waste	TP-2				
216-T-21	Tronch	216-T-21 Trench	Tank Farm Waste	TP-1				
216-T-22	Trench	216-T-22 Trench	Tank Farm Waste	TP-1				
216-T-23	Trench	216-T-23 Trench	Tank Farm Waste	TP-1				
216-T-24	Trench	216-T-24 Trench	Tank Farm Waste	TP-1				
216-T-25	Trench	216-T-25 Trench	Process Waste	TP-1				
216-T-26	Cribs	216-T-26 Crib	Tank Farm Waste	TP-2				
216-T-27	Cribs	216-T-27 Crib	Lab Waste	TP-2				
216-T-28	Cribs	216-T-28 Crib	Decon Waste	TP-2				
216-T-29	Cribs	216-T-29 Crib	Miscellaneous Drainage	TP-4				
216-T-3	Reverse Well	216-T-3 Reverse Well	Process Waste	TP-4				
216-T-31	French Drain	216-T-31 French Drain	Miscellaneous Drainage	TP-2				
216-T-32	Cribs	216-T-32 Crib	Process Waste	TP-1				
216-T-33	Cribs	216-T-33 Crib	Decon Waste	TP-4				
216-T-34	Cribs	216-T-34 Crib	Lab Waste	TP-4				
216-T-35	Cribs	216-T-35 Crib	Lab Waste	TP-4				
216-T-36	Cribs	216-T-36 Crib	Steam Condensate	TP-1				
216-T-4-1D	Ditches	216-T-4-1D Ditch	Cooling Water	TP-3				
216-T-4-2	Ditches	216-T-4-2 Ditch	Steam Condensate	TP-3				
216-T-4A	Ponds	216-T-4A Pond	Cooling Water	TP-3				
216-T-4B	Ponds	216-T-4B Pond	Cooling Water	TP-3				
216-T-5	Trench	216-T-5 Trench	Tank Farm Waste	TP-1				
216-T-6	Cribs	216-T-6 Crib	Process Waste	TP-3				

Table B.3. (page 8 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
216-T-7	Cribs	216-T-7TF Crib and Tile Field	Tank Farm Waste	TP-1				
216-T-8	Cribs	216-T-8 Crib	Lab Waste	TP-4				
216-T-9	Trench	216-T-9 Trench	Decon Waste	TP-4				
216-U-1&2	Cribs	216-U-1 & 216-U-2	Process Condensate	UP-2		0.702		
216-U-10	Ponds	216-U-10	Cooling Water	UP-2		1.88		
216-U-11	Ditches	216-U-11	Cooling Water	UP-2				
216-U-12	Cribs	216-U-12	Process Condensate	UP-2		0.677		
216-U-13	Trench	216-U-13 (same as UN-200-W- 125)	Decon Waste	UP-2				
216-U-14	Ditches	216-U-14	Cooling Water	UP-2				
216-U-15	Trench	216-U-15	Process Waste	UP-2				
216-U-16	Cribs	216-U-16	Process Condensate	UP-2		0.00592		
216-U-17	Cribs	216-U-17	Process Condensate	UP-2		0.000478		
216-U-21		216-U-21						
216-U-3	French Drain	216-U-3	Miscellaneous Drainage	UP-2		0.00606		
216-U-4	Reverse Well	216-U-4	Lab Waste	UP-2				
216-U-4A	French Drain	216-U-4A	Miscellaneous Drainage	UP-2		0.00297		
216-U-4B	French Drain	216-U-4B	Miscellaneous Drainage	UP-2				
216-U-5	Trench	216-U-5 & 216-U-6	Process Waste	UP-2				
216-U-7	French Drain	216-U-7	Miscellaneous Drainage	UP-2				
216-U-8	Cribs	216-U-8	Process Condensate	UP-2		8.04		
216-U-9	Ditches	216-U-9	Cooling Water	RO-1				
216-W-LWC	Cribs	216-W-LWC Crib	Chemical Sewer	SS-2				
216-Z-1&2	Cribs	216-Z-1 & 216-Z-2 Cribs	Process Waste	ZP-2				
216-Z-10	Reverse Well	216-Z-10 Reverse Well	Process Waste	ZP-2	0.00004			
216-Z-11	Ditches	216-Z-11	Cooling Water	UP-2				
216-Z-12	Cribs	216-Z-12 Crib	Process Waste	ZP-2				
216-Z-13	French Drain	216-Z-13 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-14	French Drain	216-Z-14 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-15	French Drain	216-Z-15 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-16	Cribs	216-Z-16 Crib	Lab Waste	ZP-2				
216-Z-17	Ditches	216-Z-17 Trench	Lab Waste	ZP-2				
216-Z-18	Cribs	216-Z-18 Crib	Process Waste	ZP-2				
216-Z-19	Ditches	216-Z-19	Cooling Water	UP-2				

Table B.3. (page 9 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
216-Z-1A	Cribs	216-Z-1A Tile Field	Process Waste	ZP-2				
216-Z-1D	Ditches	216-Z-1D	Cooling Water	UP-2				
216-Z-20	Cribs	216-Z-20	Cooling Water	UP-2				
216-Z-21	Retention Basin	216-Z-21 Seepage Basin	Cooling Water	ZP-2				
216-Z-3	Cribs	216-Z-3 Crib	Process Waste	ZP-2				
216-Z-4	Trench	216-Z-4 Trench	Process Waste	ZP-2				
216-Z-5	Cribs	216-Z-5 Crib	Process Waste	ZP-2				
216-Z-6	Cribs	216-Z-6 Crib	Process Waste	ZP-2				
216-Z-7	Cribs	216-Z-7 Crib	Lab Waste	ZP-2				
216-Z-8	Cribs	216-Z-8 French Drain	Process Waste	ZP-2				
216-Z-9	Cribs	216-Z-9 Trench	Process Waste	ZP-2				
218-C-9	Burial Site	218-C-9 Burial Ground	LLW - SOLID	SO-1				
218-E-1	Burial Site	218-E-1	LLW - SOLID	PO-2		400000		
218-E-10	Burial Site	218-E-10 Burial Ground	LLW - SOLID	BP-10		800000		
218-E-12A	Burial Site	218-E-12A	LLW - SOLID	PO-6		990000		
218-E-12B	Burial Site	218-E-12B	LLW - SOLID	PO-6				
218-E-13	Burial Site	218-E-13		PO-2				
218-E-2	Burial Site	218-E-2 Burial Ground	LLW - SOLID	BP-10				
218-E-2A	Burial Site	218-E-2A Burial Ground	LLW - SOLID	BP-10				
218-E-4	Burial Site	218-E-4 Burial Ground	LLW - SOLID	BP-10				
218-E-5	Burial Site	218-E-5 Burial Ground	LLW - SOLID	BP-10				
218-E-5A	Burial Site	218-E-5A Burial Ground	LLW - SOLID	BP-10				
218-E-6	Burial Site	218-E-6 Burial Ground	Debris	BP-6				
218-E-7	Burial Site	218-E-7 Burial Ground	Lab Waste	BP-6				
218-E-8	Burial Site	218-E-8	TRU Solid Waste	PO-6		2000		
218-E-9	Burial Site	218-E-9 Burial Ground	LLW - SOLID	BP-10				
218-W-1	Burial Site	218-W-1 Burial Ground	TRU Solid Waste	ZP-3				
218-W-11	Burial Site	218-W-11 Burial Ground	LLW - SOLID	ZP-3				
218-W-1A	Burial Site	218-W-1A Burial Ground	LLW - SOLID	ZP-3				
218-W-2	Burial Site	218-W-2 Burial Ground	TRU Solid Waste	ZP-3				
218-W-2A	Burial Site	218-W-2A Burial Ground	LLW - SOLID	ZP-3				
218-W-3	Burial Site	218-W-3 Burial Ground	TRU Solid Waste	ZP-3				
218-W-3A	Burial Site	218-W-3A Burial Ground	TRU Solid Waste	ZP-3				

Table B.3. (page 10 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
218-W-3AE	Burial Site	218-W-3AE Burial Ground	LLW - SOLID	ZP-3				
218-W-4A	Burial Site	218-W-4A Burial Ground	TRU Solid Waste	ZP-3				
218-W-4B	Burial Site	218-W-4B Caissons	TRU Solid Waste	ZP-3				
218-W-4B	Burial Site	218-W-4B Trenches	TRU Solid Waste	ZP-3				
218-W-4C	Burial Site	218-W-4C Burial Ground	TRU Solid Waste	ZP-3				
218-W-5	Burial Site	218-W-5 Burial Ground	TRU Solid Waste	ZP-3				
218-W-6	Burial Site	218-W-6 Burial Ground	LLW - SOLID	ZP-3				
218-W-7	Burial Site	218-W-7	LLW - SOLID	RO-3		700 g		
218-W-8	Burial Site	218-W-8 Burial Ground	Lab Waste	TP-4				
218-W-9	Burial Site	218-W-9	LLW - SOLID	RO-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
232-Z	Building	232-Z Incinerator		ZP-2				
240-S-151	Diversion Box	240-S-151	LLW - SOLID	RO-3				
240-S-152	Diversion Box	240-S-152	Tank Farm Waste	RO-3				
240-S-302	Tanks	240-S-302	Lab Waste	RO-3				
241-A-151	Diversion Box	241-A-151	Tank Farm Waste	PO-2				
241-A-152	Diversion Box	241-A-152	Process Waste	PO-3				
241-A-153	Diversion Box	241-A-153	Process Waste	PO-3				
241-A-302A	Tanks	241-A-302A		PO-2				
241-A-302B	Tanks	241-A-302B		PO-5				
241-A-350	Tanks	241-A-350	Process Waste	PO-3				
241-A-417	Tanks	241-A-417	Process Waste	PO-3				
241-A-A	Diversion Box	241-A-A	Process Waste	PO-3				
241-A-B	Diversion Box	241-A-B	Process Waste	PO-3				
241-AN-A	Diversion Box	241-AN-A	Process Waste	PO-3				
241-AN-B	Diversion Box	241-AN-B	Process Waste	PO-3				

Table B.3. (page 11 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
241-AP VP	Valve Pit	241-AP	Process Waste	PO-3				
241-AR-151	Diversion Box	241-AR-151	Process Waste	PO-3				
241-AW-A	Diversion Box	241-AW-A	Process Waste	PO-3				
241-AW-B	Diversion Box	241-AW-B	Process Waste	PO-3				
241-AX-151	Diversion Box	241-AX-151	Process Waste	PO-3				
241-AX-152DS	Tanks	241-AX-152DS	Process Waste	PO-3				
241-AX-155	Diversion Box	241-AX-155	Tank Farm Waste	PO-3				
241-AX-501	Valve Pit	241-AX-501		PO-3				
241-AX-A	Diversion Box	241-AX-A		PO-3				
241-AX-B	Diversion Box	241-AX-B		PO-3				
241-AY-151	Diversion Box	241-AY-151	Process Waste	PO-3				
241-AY-152	Diversion Box	241-AY-152	Process Waste	PO-3				
241-AZ-151DS	Diversion Box	241-AZ-151DS		PO-3				
241-AZ-152	Diversion Box	241-AZ-152		PO-3				
241-C-151	Diversion Box	241-C-151		PO-3				
241-C-152	Diversion Box	241-C-152		PO-3				
241-C-153	Diversion Box	241-C-153		PO-3				
241-C-154	Diversion Box	241-C-154 Diversion Box	Process Waste	SO-1				
241-C-252	Diversion Box	241-C-252		PO-3				
241-C-301C	Tanks	241-C-301C		PO-3				
241-CR-151	Diversion Box	241-CR-151		PO-3				
241-CR-152	Diversion Box	241-CR-152		PO-3				
241-CR-153	Diversion Box	241-CR-153		PO-3				
241-CX-TK-70	Tanks	241-CX-70 Storage Tank	Tank Farm Waste	SO-1				
241-CX-TK-71	Tanks	241-CX-71 Storage Tank	Process Condensate	SO-1				
241-CX-TK-72	Tanks	241-CX-72 Storage Tank	Process Waste	SO-1				
241-ER-153	Diversion Box	241-ER-153		PO-3				
241-S-151	Diversion Box	241-S-151	LLW - SOLID	RO-2				
241-S-152	Diversion Box	241-S-152	Tank Farm Waste	RO-4				
241-S-302A	Tanks	241-S-302A	Lab Waste	RO-2				
241-S-302B	Tanks	241-S-302B	LLW - SOLID	RO-4				
241-S-A	Diversion Box	241-S-A	Tank Farm Waste	RO-4				
241-S-B	Diversion Box	241-S-B	Tank Farm Waste	RO-4				

Table B.3. (page 12 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
241-S-C	Diversion Box	241-S-C	Tank Farm Waste	RO-4				
241-S-D	Diversion Box	241-S-D	Tank Farm Waste	RO-4				
241-SX-151	Diversion Box	241-SX-151	Tank Farm Waste	RO-4				
241-SX-152	Diversion Box	241-SX-152	Tank Farm Waste	RO-4				
241-SX-302	Tanks	241-SX-302		RO-2				
241-SX-A	Diversion Box	241-SX-A		RO-4				
241-SX-B	Diversion Box	241-SX-B		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-T-151	Diversion Box	241-T-151 Diversion Box	Tank Farm Waste	TP-6				
241-T-152	Diversion Box	241-T-152 Diversion Box	Tank Farm Waste	TP-6				
241-T-153	Diversion Box	241-T-153 Diversion Box	Tank Farm Waste	TP-6				
241-T-252	Diversion Box	241-T-252 Diversion Box	Tank Farm Waste	TP-6				
241-T-301	Tanks	241-T-301 Catch Tank	Tank Farm Waste	TP-6				
241-T-302	Tanks	241-T-302 Catch Tank	Tank Farm Waste	TP-6				
241-T-361	Tanks	241-T-361 Settling Tank	Process Waste	TP-4				
241-TR-152	Diversion Box	241-TR-152 Diversion Box	Tank Farm Waste	TP-6				
241-TR-153	Diversion Box	241-TR-153 Diversion Box	Tank Farm Waste	TP-6				
241-TX-152	Diversion Box	241-TX-152 Diversion Box	Tank Farm Waste	TP-2				
241-TX-153	Diversion Box	241-TX-153 Diversion Box	Tank Farm Waste	TP-5				
241-TX-154	Diversion Box	241-TX-154 Diversion Box	Tank Farm Waste	TP-4				
241-TX-155	Diversion Box	241-TX-155 Diversion Box	Tank Farm Waste	TP-2				
241-TX-302A	Tanks	241-TX-302A Catch Tank	Tank Farm Waste	TP-5				
241-TX-302B	Tanks	241-TX-302B Catch Tank	Tank Farm Waste	TP-2				
241-TX-302C	Tanks	241-TX-302C Catch Tank	Tank Farm Waste	TP-4				
241-TXR-151	Diversion Box	241-TXR-151 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-152	Diversion Box	241-TXR-152 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-153	Diversion Box	241-TXR-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-153	Diversion Box	241-TY-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-302A	Tanks	241-TY-302A Catch Tank	Tank Farm Waste	TP-5				
241-TY-302B	Tanks	241-TY-302B Catch Tank	Tank Farm Waste	TP-5				

Table B.3. (page 13 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
241-Z-361	Tanks	241-Z-361 Settling Tank	Process Waste	ZP-2				
241-Z-TK-8	Tanks	216-Z-8 Settling Tank	Process Waste	ZP-2				
241-Z-TK-D5	Tanks	241-Z Treatment Tank	Process Waste	ZP-2				
242-T-151	Diversion Box	242-T-151 Diversion Box	Process Condensate	TP-5				
244-A RT	Tanks	244-A	Process Waste	PO-3				
244-AR VAULT	Vault	244-AR	Process Waste	PO-3				
244-CR VAULT	Vault	244-CR	Process Waste	PO-3				
244-S RT	Tanks	244-S Receiver Tank		RO-2				
244-TX RT	Tanks	244-TX Receiving Tank		TP-5				
244-TXR	Vault	244-TXR Vault	Tank Farm Waste	TP-5				
2607-E5	Septic System	2607-E-5 Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-E6	Septic System	2607-E6	Sanitary Waste	PO-2				
2607-E7A	Septic System	2607-E-7A Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-EA	Septic System	2607-EA	Sanitary Waste	PO-2				
2607-EC	Septic System	2607-EC	Sanitary Waste	PO-5				
2607-ED	Septic System	2607-ED	Sanitary Waste	PO-3				
2607-EE	Septic System	2607-EL	Sanitary Waste	PO-2				
2607-EG	Septic System	2607-EG	Sanitary Waste	PO-3				
2607-EJ	Septic System	2607-EJ	Sanitary Waste	PO-3				
2607-N	Septic System	2607-N Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-P	Septic System	2607-P Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-R	Septic System	2607-R Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-W1	Septic System	2607-W1 Septic Tank	Sanitary Waste	SS-2				
2607-W2	Septic System	2607-W2 Septic Tank	Sanitary Waste	SS-2				
2607-W3	Septic System	2607-W3 Septic Tank	Sanitary Waste	TP-4				
2607-W4	Septic System	2607-W4 Septic Tank	Sanitary Waste	TP-4				
2607-W6	Septic System	2607-W6	Sanitary Waste	RO-3				
2607-W8	Septic System	2607-W-8 Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WA	Septic System	2607-WA Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WB		2607-WB Septic Tank and Drain Field						
2607-WT	Septic System	2607-WT Septic Tank	Sanitary Waste	TP-5				
2607-WTX	Septic System	2607-WTX Septic Tank	Sanitary Waste	TP-5				
2607-WZ	Septic System	2607-WZ	Sanitary Waste	RO-1				

Table B.3. (page 14 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Plutonium-242, Units (Ci)	Total Uranium, Units (Ci, unless otherwise stated)	Gross Uranium, Units (Ci)	Uranium-235, Units (Ci)
2607-Z	Septic System	2607-Z Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-Z-1		2607-Z-1 Septic Tank and Drain Field						
2704-C-WS-1	French Drain	2704-C-WS-1, 2704-C French Drain, Gatehouse French Drain	Miscellaneous Drainage	SO-1				
2904-S-160	Diversion Box	2904-S-160	Cooling Water	RO-1				
2904-S-170	Diversion Box	2904-S-170	Process Waste	RO-1				
2904-S-171	Diversion Box	2904-S-171	Cooling Water	RO-1				
291-C	Building	291-C Ventilation System	Process Condensate	SO-1				
299-E24-111	Reverse Well	299-E24-111		PO-2				
HSVP	Diversion Box	Semi-Works Valve Pit	Process Waste	SO-1				
UPR-200-E-141		UN-200-E-141	Solution Storage (1)	SO-1				
UPR-200-E-36		UN-200-E-36	Process & Decon Wastes	SO-1				
UPR-200-E-37		UN-200-E-37	Process & Decon Wastes	SO-1				
UPR-200-E-98		UN-200-E-98	Process & Decon Wastes	SO-1				
UPR-200-W-160		UPR-200-W-160 Unplanned Release	Tank Farm Waste	TP-4				
Z PLANT BP	Burial Site	Z Plant Burn Pit	Debris	ZP-3				

Table B.4. Environmental Restoration Waste Site Inventories for Uraniunm-238, Alpha Emitters, Beta Emitters, and Americium-241 (page 1 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
		212-N to 216-N-1 Pipeline						
		212-P Hazardous Waste Staging Area						
		212-P to 216-N-4 Pipeline						
		212-P Transformer Oil Tank						
		212-R to 216-N-6 Pipeline						
		241-C Waste Line Unplanned Release No. 1						
		241-C Waste Line Unplanned Release No. 2						
		241-Z Diversion Box No. 1						
		241-Z Diversion Box No. 2						
		Sanitary Crib						
200-E BP	Burial Site	200-E Burning Pit	Debris	PO-6				
200-E PAP	Burial Site	200-E Ash Pit		SS-1				
200-E PD	Ditches	200 East Powerhouse Ditch	Cooling Water	SO-1				
200-E-4	French Drain	Critical Mass Laboratory Dry Well North	Miscellaneous Drainage	SO-1				
200-N-3	Burial Site	Ballast Pits	Debris	NO-1				
200-W ADB	Burial Site	200-W Ash Disposal Basin	Ash	SS-2				
200-W ADS	Burial Site	200-W Ash Pit Demolition Site	N/A	SS-2				
200-W BP	Burial Site	200-W Burning Pit	Debris	SS-2				
200-W PAP	Burial Site	200-W Powerhouse Ash Pit	Ash	SS-2				
200-W PP	Ponds	200-W Powerhouse Pond	Cooling Water	TP-2				
201-C	Building	201-C Process Building	Process Condensate	SO-1				0.2
207-A		207-A						
207-B	Retention Basin	207-Bb/ Retention Basin	Cooling Water	BP-8				
207-S	Retention Basin	207-S	Cooling Water	RO-2				
207-SL	Retention Basin	207-SL	Lab Waste	RO-3				
207-T	Retention Basin	207-T Retention Basin	Cooling Water	TP-3				
207-Z	Retention Basin	207-Z Retention Basin	Steam Condensate	ZP-2				
209-E-WS-1	French Drain	Critical Mass Laboratory Dry Well East	Miscellaneous Drainage	SO-1				
209-E-WS-2	French Drain	Critical Mass Laboratory Dry Well South	Miscellaneous Drainage	SO-1				
209-E-WS-3	Diversion Box	Critical Mass Laboratory Valve Pit	Process Waste	SO-1				
2101-M POND	Ponds	2101-M Pond	Lab Waste	SS-1				
216-A-1	Cribs	216-A-1	Process Waste	PO-5	0.0516	0.00614	0.17	
216-A-10	Cribs	216-A-10	Process Condensate	PO-2		28.1	360	0.773

Table B.4. (page 2 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
216-A-11	French Drain	216-A-11	Miscellaneous Drainage	PO-2				
216-A-12	French Drain	216-A-12	Miscellaneous Drainage	PO-2				
216-A-13	French Drain	216-A-13	Miscellaneous Drainage	PO-2				
216-A-14	French Drain	216-A-14	Miscellaneous Drainage	PO-2				
216-A-15	French Drain	216-A-15	Process Condensate	PO-2				
216-A-16	French Drain	216-A-16	Chemical Sewer	PO-5				
216-A-17	French Drain	216-A-17	Chemical Sewer	PO-5				
216-A-18	Trench	216-A-18	Process Waste	PO-5	0.472	0.00614	0.172	
216-A-19	Trench	216-A-19	Process Waste	PO-5	13	0.00614	0.17	
216-A-2	Cribs	216-A-2	Process Waste	PO-2	0.0262	7.98	4.71	
216-A-20	Trench	216-A-20	Process Waste	PO-5	0.135	0.00614	0.17	
216-A-21	Cribs	216-A-21	Lab Waste	PO-2	0.0653	9.21	166	
216-A-22	French Drain	216-A-22	Miscellaneous Drainage	PO-2				
216-A-23A	French Drain	216-A-23A	Process Condensate	PO-5				
216-A-23B	French Drain	216-A-23B	Process Condensate	PO-5				
216-A-24	Cribs	216-A-24	Process Condensate	PO-5	0.0168	0.311	552	
216-A-25	Ponds	216-A-25 Pond	Cooling Water	IU-6		27.5	939	0.000528
216-A-26	French Drain	216-A-26	Miscellaneous Drainage	PO-2				
216-A-26A	French Drain	216-A-26A	Miscellaneous Drainage	PO-2				
216-A-27	Cribs	216-A-27	Miscellaneous Drainage	PO-2	0.0228	5.92	112	
216-A-28	Cribs	216-A-28	Process Condensate	PO-2	0.212		0.00747	
216-A-29	Ditches	216-A-29	Chemical Sewer	BP-11				
216-A-3	Cribs	216-A-3	Process Waste	PO-2		0.0123	0.182	
216-A-30	Cribs	216-A-30	Steam Condensate	PO-4		4.64	432	0.198
216-A-31	Cribs	216-A-31	Process Waste	PO-2	0.00686	0.553	162	
216-A-32	Cribs	216-A-32	Miscellaneous Drainage	PO-2				
216-A-33	French Drain	216-A-33	Miscellaneous Drainage	PO-2				
216-A-34	Cribs	216-A-34	Process Condensate	PO-5				
216-A-35	French Drain	216-A-35	Miscellaneous Drainage	PO-2				
216-A-36A	Cribs	216-A-36A	Process Waste	PO-2	0.0486	4.91	3630	
216-A-36B	Cribs	216-A-36B	Process Waste	PO-2		11	1360	0.217
216-A-37-1	Cribs	216-A-37-1	Process Condensate	PO-4		0.00845	0.508	0.000369
216-A-37-2	Cribs	216-A-37-2	Steam Condensate	PO-4		0.105	1.85	0.0982

Table B.4. (page 3 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
216-A-38-1	Cribs	216-A-38-1	N/A	PO-2				
216-A-39	Ditches	216-A-39	Miscellaneous Drainage	PO-3			27.5	
216-A-4	Cribs	216-A-4	Lab Waste	PO-2	0.134	8.6	2.21	
216-A-40	Retention Basin	216-A-40	Steam Condensate	PO-2				
216-A-41	Cribs	216-A-41	Miscellaneous Drainage	PO-2				
216-A-42	Retention Basin	216-A-42	Cooling Water	PO-4				
216-A-45	Cribs	216-A-45	Process Condensate	PO-2		0.0551	0.112	0.11
216-A-5	Cribs	216-A-5	Process Condensate	PO-2	0.0881	3.99	109	
216-A-524	Diversion Box	216-A-524		PO-5				
216-A-6	Cribs	216-A-6	Steam Condensate	PO-4	0.0553	2.19	291	
216-A-7	Cribs	216-A-7	Process Waste	PO-5	0.00228	0.0614	5.29	
216-A-8	Cribs	216-A-8	Process Condensate	PO-5		3.07	1110	
216-A-9	Cribs	216-A-9	Cooling Water	PO-2	0.00008	0.0307	31	
216-B-10A	Cribs	216-B-10A Crib	Lab Waste	BP-6	0.00304	0.602	4.55	0
216-B-10B	Cribs	216-B-10B Crib	Lab Waste	BP-6	0	0.00000291	0.000000531	0
216-B-11A&B	Reverse Well	216-B-11A&B Reverse Wells	Process Condensate	BP-4	0.00456	0.246	44.9	0
216-B-12	Cribs	216-B-12 Crib	Process Condensate	BP-9	7	23	1540	0
216-B-13	French Drain	216-B-13 French Drain	Miscellaneous Drainage	BP-6				
216-B-14	Cribs	216-B-14 Crib	Scavenged Waste	BP-2	0.073	1.53	567	0
216-B-15	Cribs	216-B-15 Crib	Scavenged Waste	BP-2	0.0348	0.307	357	0
216-B-16	Cribs	216-B-16 Crib	Scavenged Waste	BP-2	0.108	0.614	1180	0
216-B-17	Cribs	216-B-17 Crib	Scavenged Waste	BP-2	0.119	0.614	330	0
216-B-18	Cribs	216-B-18 Crib	Scavenged Waste	BP-2	0.0791	0.614	385	0
216-B-19	Cribs	216-B-19 Crib	Scavenged Waste	BP-2	0.0606	0.614	418	0
216-B-2-1	Ditches	216-B-2-1 Ditch[b]	Cooling Water	BP-11		16.2	390	3.96
216-B-2-2	Ditches	216-B-2-2 Ditch[c]	Cooling Water	BP-11	0	0.00258	295	
216-B-2-3	Ditches	216-B-2-3 Ditch	Cooling Water	BP-11			864	
216-B-20	Trench	216-B-20 Trench	Scavenged Waste	BP-2	0.118	0.0798	2000	0
216-B-21	Trench	216-B-21 Trench	Scavenged Waste	BP-2	0.226	0.632	965	0
216-B-22	Trench	216-B-22 Trench	Scavenged Waste	BP-2	0.14	0.16	398	0
216-B-23	Trench	216-B-23 Trench	Scavenged Waste	BP-2	0.0523	0.111	226	0
216-B-24	Trench	216-B-24 Trench	Scavenged Waste	BP-2	0.0825	0.473	0.274	0
216-B-25	Trench	216-B-25 Trench	Scavenged Waste	BP-2	0.513	0.123	229	0

Table B.4. (page 4 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
216-B-26	Trench	216-B-26 Trench	Scavenged Waste	BP-2	0.197	0.0153	1800	0
216-B-27	Trench	216-B-27 Trench	Scavenged Waste	BP-2	0.115	0.043	560	0
216-B-28	Trench	216-B-28 Trench	Scavenged Waste	BP-2	0.101	0.34	121	0
216-B-29	Trench	216-B-29 Trench	Scavenged Waste	BP-2	0.115	0.0675	226	0
216-B-3	Ponds	216-B-3 Ponds/	Cooling Water	BP-11		16.2	390	3.96
216-B-3-1	Ditches	216-B-3-1 Ditchb/	Cooling Water	BP-11				
216-B-3-2	Ditches	216-B-3-2 Ditchc/	Cooling Water	BP-11				
216-B-3-3	Ditches	216-B-3-3 Ditch	Cooling Water	BP-11				
216-B-30	Trench	216-B-30 Trench	Scavenged Waste	BP-2	0.0295	0.129	3540	0
216-B-32	Trench	216-B-32 Trench	Scavenged Waste	BP-2	0.00368	0.16	339	0
216-B-33	Trench	216-B-33 Trench	Scavenged Waste	BP-2	0.0067	0.724	281	0
216-B-34	Trench	216-B-34 Trench	Scavenged Waste	BP-2	0.0285	0.35	51.7	0
216-B-35	Trench	216-B-35 Trench	Tank Farm Waste	BP-3	0.00559	0.0737	549	0
216-B-36	Trench	216-B-36 Trench	Tank Farm Waste	BP-3	0.00532	0.0491	1040	0
216-B-37	Trench	216-B-37 Trench	Process Waste	BP-3	0.00121	0.123	2600	0
216-B-38	Trench	216-B-38 Trench	Tank Farm Waste	BP-3	0.0142	0.0737	1940	0
216-B-39	Trench	216-B-39 Trench	Tank Farm Waste	BP-3	0.00194	0.0927	387	0
216-B-3A	Ponds	216-B-3A Pond	Cooling Water	BP-11				
216-B-3B	Ponds	216-B-3B Pond	Cooling Water	BP-11				
216-B-3C	Ponds	216-B-3C Pond	Cooling Water	BP-11				
216-B-4	Reverse Well	216-B-4 Reverse Well	Miscellaneous Drainage	BP-6	0	0	1	0
216-B-40	Trench	216-B-40 Trench	Tank Farm Waste	BP-3	0.00117	0.0614	523	0
216-B-41	Trench	216-B-41 Trench	Tank Farm Waste	BP-3	0.00251	0.0184	780	0
216-B-42	Trench	216-B-42 Trench	Scavenged Waste	BP-3	0.228	0.614	1010	0
216-B-43	Cribs	216-B-43 Crib	Scavenged Waste	BP-1	0.00456	0.0307	1400	0
216-B-44	Cribs	216-B-44 Crib	Scavenged Waste	BP-1	0.00076	0.921	2990	0
216-B-45	Cribs	216-B-45 Crib	Scavenged Waste	BP-1	0.00228	0.614	3640	0
216-B-46	Cribs	216-B-46 Crib	Scavenged Waste	BP-1	0.0636	1.23	1440	0
216-B-47	Cribs	216-B-47 Crib	Scavenged Waste	BP-1	0.00228	0.307	650	0
216-B-48	Cribs	216-B-48 Crib	Scavenged Waste	BP-1	0.00076	0.307	1490	0
216-B-49	Cribs	216-B-49 Crib	Scavenged Waste	BP-1	0.106	262	2360	0
216-B-5	Reverse Well	216-B-5 Reverse Well	Process Waste	BP-6	0	262	108	0
216-B-50	Cribs	216-B-50 Crib	Process Condensate	BP-1	0.0001	0.0147	105	0

Table B.4. (page 5 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
216-B-51	French Drain	216-B-51 French Drain	Miscellaneous Drainage	BP-4				
216-B-52	Trench	216-B-52 Trench	Scavenged Waste	BP-2	0.01	1.17	317	0
216-B-53A	Trench	216-B-53A Trench	Lab Waste	BP-2	0.076	6.14	0.246	0
216-B-53B	Trench	216-B-53B Trench	Lab Waste	BP-2	0.00303	307	17.2	0
216-B-54	Trench	216-B-54 Trench	Lab Waste	BP-2	0.00303	0.307	0.945	0
216-B-55	Cribs	216-B-55 Crib	Steam Condensate	BP-9		0.0423	40.9	0.0000038
216-B-56	Cribs	216-B-56 Crib	N/A	BP-6				
216-B-57	Cribs	216-B-57 Crib	Process Condensate	BP-1	0.00029	0.0115	437	0
216-B-58	Trench	216-B-58 Trench	Lab Waste	BP-2	0.00305	0.411	19.7	0
216-B-59	Retention Basin	216-B-59 Basin	Cooling Water	BP-8			0.0832	
216-B-6	Reverse Well	216-B-6 Reverse Well	Lab Waste	BP-6	0	0	10	0
216-B-60	Cribs	216-B-60 Crib	Decon Waste	BP-8				
216-B-61	Cribs	216-B-61 Crib	N/A	BP-1				
216-B-62	Cribs	216-B-62 Crib	Process Condensate	BP-9		0.105	418	0.103
216-B-63	Ditches	216-B-63 Trench	Chemical Sewer	BP-11		0.0742	6.32	0.0348
216-B-64	Retention Basin	216-B-64 Basin	N/A	BP-9				
216-B-7A&B	Cribs	216-B-7A&B Crib	Process Waste	BP-4	0.061	264	4490	0
216-B-8	Cribs	216-B-8TF Crib	Process Waste	BP-4	0	1.84	49.3	0
216-B-9	Cribs	216-B-9TF Crib	Process Waste	BP-6	0.0152	10.7	2	0
216-C-1	Cribs	216-C-1 Crib	Process Condensate	SO-1	0.0988			
216-C-10	Cribs	216-C-10 Crib	Process Condensate	SO-1	0.00001			
216-C-2	Reverse Well	216-C-2 Reverse Well	Miscellaneous Drainage	SO-1				
216-C-3	Cribs	216-C-3 Crib	Process Waste	SO-1	0.0153			
216-C-4	Cribs	216-C-4 Crib	Process Waste	SO-1	0.0011			
216-C-5	Cribs	216-C-5 Crib	Process Waste	SO-1	0.0182			
216-C-6	Cribs	216-C-6 Crib	Process Condensate	SO-1	0.0001			
216-C-7	Cribs	216-C-7 Crib	Process Waste	SO-1				
216-C-8	French Drain	216-C-8	Process Waste	PO-3				
216-C-9	Ponds	216-C-9 Pond	Cooling Water	SO-1				
216-E-28	Ponds	216-E-28 Pond	N/A	BP-11				
216-N-1	Ponds	216-N-1 Pond	Cooling Water	NO-1				
216-N-2	Trench	216-N-2 Trench	Cooling Water	NO-1			0.29	
216-N-3	Trench	216-N-3 Trench	Cooling Water	NO-1			0.326	

Table B.4. (page 6 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
216-N-4	Ponds	216-N-4 Pond	Cooling Water	NO-1	0.00152	0.0614	0.3	
216-N-5	Trench	216-N-5 Trench	Cooling Water	NO-1			0.326	
216-N-6	Ponds	216-N-6 Pond	Cooling Water	NO-1	0.00152	0.0614	0.3	
216-N-7	Trench	216-N-7 Trench	Cooling Water	NO-1			0.326	
216-N-8	Ponds	216-N-8 Pond		IU-6				
216-S-1&2	Cribs	216-S-1 & 2	Process Condensate	RO-2		73.7	4750	
216-S-10D	Ditches	216-S-10D	Chemical Sewer	RO-1		0.0244	3.51	0.0152
216-S-10P	Ponds	216-S-10P	Chemical Sewer	RO-1				
216-S-11	Ponds	216-S-11	Chemical Sewer	RO-1		0.00553	1.94	
216-S-12	Trench	216-S-12	Miscellaneous Drainage	RO-3		0.0614	1.66	
216-S-13	Cribs	216-S-13	Process Waste	RO-2		0.491	5.5	
216-S-14	Trench	216-S-14	Process Waste	RO-3				
216-S-15	Ponds	216-S-15	Cooling Water	RO-2				
216-S-16D	Ditches	216-S-16D	Cooling Water	RO-1				
216-S-16P	Ponds	216-S-16P	Cooling Water	RO-1		22.6	148	
216-S-17	Ponds	216-S-17	Cooling Water	RO-1		0.184	56.3	
216-S-172	Diversion Box	216-S-172	Cooling Water	RO-1				
216-S-18	Trench	216-S-18	Debris	RO-2				
216-S-19	Ponds	216-S-19	Lab Waste	RO-1		1.26	5.12	
216-S-20	Cribs	216-S-20	Lab Waste	RO-3		10.5	156	
216-S-22	Cribs	216-S-22	Process Waste	RO-3		0.0062	1.83	
216-S-23	Cribs	216-S-23	Process Condensate	RO-2		0.0611	9.07	
216-S-25	Cribs	216-S-25	Steam Condensate	RO-1		0.012	0.247	
216-S-26	Cribs	216-S-26	Lab Waste	RO-3		0.000763	0.01	0.00058
216-S-3	French Drain	216-S-3	Process Condensate	RO-2		0.0307	43	
216-S-4	French Drain	216-S-4	Process Condensate	UP-2				
216-S-5	Cribs	216-S-5	Cooling Water	RO-1		35.6	159	
216-S-6	Cribs	216-S-6	Cooling Water	RO-1		29	630	
216-S-7	Cribs	216-S-7	Process Condensate	RO-2		27	4180	
216-S-8	Trench	216-S-8	Process Waste	RO-2		0.123	10.5	
216-S-9	Cribs	216-S-9	Process Condensate	RO-2		3.99	753	
216-T-1	Ditches	216-T-1 Ditch	Cooling Water	TP-4	0.0015			
216-T-10	Trench	216-T-10 Trench	Decon Waste	TP-4				

Table B.4. (page 7 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
216-T-11	Trench	216-T-11 Trench	Decon Waste	TP-4				
216-T-12	Trench	216-T-12 Trench	Cooling Water	TP-3	0.0152			
216-T-13	Trench	216-T-13 Trench	Decon Waste	TP-2				
216-T-14	Trench	216-T-14 Trench	Tank Farm Waste	TP-3	0.0102			
216-T-15	Trench	216-T-15 Trench	Tank Farm Waste	TP-3	0.00911			
216-T-16	Trench	216-T-16 Trench	Tank Farm Waste	TP-3	0.00743			
216-T-17	Trench	216-T-17 Trench	Tank Farm Waste	TP-3	0.0068			
216-T-18	Cribs	216-T-18 Crib	Tank Farm Waste	TP-2	0.00911			
216-T-19	Cribs	216-T-19TF Crib and Tile Field	Process Waste	TP-2				0.00982
216-T-2	Reverse Well	216-T-2 Reverse Well	Lab Waste	TP-4				
216-T-20	Trench	216-T-20 Trench	Process Waste	TP-2	0.0167			
216-T-21	Trench	216-T-21 Trench	Tank Farm Waste	TP-1	0.00033			
216-T-22	Trench	216-T-22 Trench	Tank Farm Waste	TP-1	0.00067			
216-T-23	Trench	216-T-23 Trench	Tank Farm Waste	TP-1	0.00034			
216-T-24	Trench	216-T-24 Trench	Tank Farm Waste	TP-1	0.00278			
216-T-25	Trench	216-T-25 Trench	Process Waste	TP-1	0.0003			
216-T-26	Cribs	216-T-26 Crib	Tank Farm Waste	TP-2	0.503			
216-T-27	Cribs	216-T-27 Crib	Lab Waste	TP-2	0.00243			
216-T-28	Cribs	216-T-28 Crib	Decon Waste	TP-2	0.131			
216-T-29	Cribs	216-T-29 Crib	Miscellaneous Drainage	TP-4				
216-T-3	Reverse Well	216-T-3 Reverse Well	Process Waste	TP-4				
216-T-31	French Drain	216-T-31 French Drain	Miscellaneous Drainage	TP-2				
216-T-32	Cribs	216-T-32 Crib	Process Waste	TP-1	0.0076			
216-T-33	Cribs	216-T-33 Crib	Decon Waste	TP-4	0.00152			
216-T-34	Cribs	216-T-34 Crib	Lab Waste	TP-4	0.00138			
216-T-35	Cribs	216-T-35 Crib	Lab Waste	TP-4	0.0164			
216-T-36	Cribs	216-T-36 Crib	Steam Condensate	TP-1	0.00039			
216-T-4-1D	Ditches	216-T-4-1D Ditch	Cooling Water	TP-3				
216-T-4-2	Ditches	216-T-4-2 Ditch	Steam Condensate	TP-3				
216-T-4A	Ponds	216-T-4A Pond	Cooling Water	TP-3				
216-T-4B	Ponds	216-T-4B Pond	Cooling Water	TP-3	0.232			
216-T-5	Trench	216-T-5 Trench	Tank Farm Waste	TP-1	0.00152			
216-T-6	Cribs	216-T-6 Crib	Process Waste	TP-3	0.0076			

Table B.4. (page 8 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
216-T-7	Cribs	216-T-7TF Crib and Tile Field	Tank Farm Waste	TP-1	0.00304			
216-T-8	Cribs	216-T-8 Crib	Lab Waste	TP-4	0.0015			
216-T-9	Trench	216-T-9 Trench	Decon Waste	TP-4				
216-U-1&2	Cribs	216-U-1 & 216-U-2	Process Condensate	UP-2		2.62	12.6	
216-U-10	Ponds	216-U-10	Cooling Water	UP-2		505	44.2	0.492
216-U-11	Ditches	216-U-11	Cooling Water	UP-2				
216-U-12	Cribs	216-U-12	Process Condensate	UP-2		0.105	112	0.00645
216-U-13	Trench	216-U-13 (same as UN-200-W- 125)	Decon Waste	UP-2	0.00012	0.00614	0.176	
216-U-14	Ditches	216-U-14	Cooling Water	UP-2				
216-U-15	Trench	216-U-15	Process Waste	UP-2	0.00076	0.00614	0.18	
216-U-16	Cribs	216-U-16	Process Condensate	UP-2		0.00739	0.0515	
216-U-17	Cribs	216-U-17	Process Condensate	UP-2		0.000195		0.000053
216-U-21		216-U-21			0.0014	0.128	208	
216-U-3	French Drain	216-U-3	Miscellaneous Drainage	UP-2		0.00614	0.1917	
216-U-4	Reverse Well	216-U-4	Lab Waste	UP-2				
216-U-4A	French Drain	216-U-4A	Miscellaneous Drainage	UP-2		0.000553	0.387	
216-U-4B	French Drain	216-U-4B	Miscellaneous Drainage	UP-2		0.00332	0.381	
216-U-5	Trench	216-U-5 & 216-U-6	Process Waste	UP-2	0.122	0.00307	0.0792	
216-U-7	French Drain	216-U-7	Miscellaneous Drainage	UP-2				
216-U-8	Cribs	216-U-8	Process Condensate	UP-2	8.04	22.7	0.65	
216-U-9	Ditches	216-U-9	Cooling Water	RO-1				
216-W-LWC	Cribs	216-W-LWC Crib	Chemical Sewer	SS-2				
216-Z-1&2	Cribs	216-Z-1 & 216-Z-2 Cribs	Process Waste	ZP-2	0.027			
216-Z-10	Reverse Well	216-Z-10 Reverse Well	Process Waste	ZP-2				1
216-Z-11	Ditches	216-Z-11	Cooling Water	UP-2				
216-Z-12	Cribs	216-Z-12 Crib	Process Waste	ZP-2	0.000017			
216-Z-13	French Drain	216-Z-13 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-14	French Drain	216-Z-14 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-15	French Drain	216-Z-15 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-16	Cribs	216-Z-16 Crib	Lab Waste	ZP-2				
216-Z-17	Ditches	216-Z-17 Trench	Lab Waste	ZP-2	0.00005			
216-Z-18	Cribs	216-Z-18 Crib	Process Waste	ZP-2				
216-Z-19	Ditches	216-Z-19	Cooling Water	UP-2				

Table B.4. (page 9 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
216-Z-1A	Cribs	216-Z-1A Tile Field	Process Waste	ZP-2				3432
216-Z-1D	Ditches	216-Z-1D	Cooling Water	UP-2				
216-Z-20	Cribs	216-Z-20	Cooling Water	UP-2		2.22	0.409	1.01
216-Z-21	Retention Basin	216-Z-21 Seepage Basin	Cooling Water	ZP-2				
216-Z-3	Cribs	216-Z-3 Crib	Process Waste	ZP-2	0.000017			
216-Z-4	Trench	216-Z-4 Trench	Process Waste	ZP-2	0.000017			
216-Z-5	Cribs	216-Z-5 Crib	Process Waste	ZP-2	0.000017			
216-Z-6	Cribs	216-Z-6 Crib	Process Waste	ZP-2	0.000017			
216-Z-7	Cribs	216-Z-7 Crib	Lab Waste	ZP-2	0.0015			
216-Z-8	Cribs	216-Z-8 French Drain	Process Waste	ZP-2				1373
216-Z-9	Cribs	216-Z-9 Trench	Process Waste	ZP-2	0.000017			8580
218-C-9	Burial Site	218-C-9 Burial Ground	LLW - SOLID	SO-1				
218-E-1	Burial Site	218-E-1	LLW - SOLID	PO-2	0.134			
218-E-10	Burial Site	218-E-10 Burial Ground	LLW - SOLID	BP-10			430000	
218-E-12A	Burial Site	218-E-12A	LLW - SOLID	PO-6	0.332			
218-E-12B	Burial Site	218-E-12B	LLW - SOLID	PO-6				
218-E-13	Burial Site	218-E-13		PO-2				
218-E-2	Burial Site	218-E-2 Burial Ground	LLW - SOLID	BP-10				
218-E-2A	Burial Site	218-E-2A Burial Ground	LLW - SOLID	BP-10				
218-E-4	Burial Site	218-E-4 Burial Ground	LLW - SOLID	BP-10				
218-E-5	Burial Site	218-E-5 Burial Ground	LLW - SOLID	BP-10				
218-E-5A	Burial Site	218-E-5A Burial Ground	LLW - SOLID	BP-10				
218-E-6	Burial Site	218-E-6 Burial Ground	Debris	BP-6				
218-E-7	Burial Site	218-E-7 Burial Ground	Lab Waste	BP-6				
218-E-8	Burial Site	218-E-8	TRU Solid Waste	PO-6	0.00067			
218-E-9	Burial Site	218-E-9 Burial Ground	LLW - SOLID	BP-10				
218-W-1	Burial Site	218-W-1 Burial Ground	TRU Solid Waste	ZP-3	0.0235			
218-W-11	Burial Site	218-W-11 Burial Ground	LLW - SOLID	ZP-3				
218-W-1A	Burial Site	218-W-1A Burial Ground	LLW - SOLID	ZP-3	0.302			
218-W-2	Burial Site	218-W-2 Burial Ground	TRU Solid Waste	ZP-3	46.9			
218-W-2A	Burial Site	218-W-2A Burial Ground	LLW - SOLID	ZP-3				
218-W-3	Burial Site	218-W-3 Burial Ground	TRU Solid Waste	ZP-3	23.5			
218-W-3A	Burial Site	218-W-3A Burial Ground	TRU Solid Waste	ZP-3				

Table B.4. (page 10 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
218-W-3AE	Burial Site	218-W-3AE Burial Ground	LLW - SOLID	ZP-3				
218-W-4A	Burial Site	218-W-4A Burial Ground	TRU Solid Waste	ZP-3				
218-W-4B	Burial Site	218-W-4B Caissons	TRU Solid Waste	ZP-3				
218-W-4B	Burial Site	218-W-4B Trenches	TRU Solid Waste	ZP-3				
218-W-4C	Burial Site	218-W-4C Burial Ground	TRU Solid Waste	ZP-3				
218-W-5	Burial Site	218-W-5 Burial Ground	TRU Solid Waste	ZP-3				
218-W-6	Burial Site	218-W-6 Burial Ground	LLW - SOLID	ZP-3				
218-W-7	Burial Site	218-W-7	LLW - SOLID	RO-3				
218-W-8	Burial Site	218-W-8 Burial Ground	Lab Waste	TP-4	0.0001			
218-W-9	Burial Site	218-W-9	LLW - SOLID	RO-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
232-Z	Building	232-Z Incinerator		ZP-2				
240-S-151	Diversion Box	240-S-151	LLW - SOLID	RO-3				
240-S-152	Diversion Box	240-S-152	Tank Farm Waste	RO-3				
240-S-302	Tanks	240-S-302	Lab Waste	RO-3				
241-A-151	Diversion Box	241-A-151	Tank Farm Waste	PO-2				
241-A-152	Diversion Box	241-A-152	Process Waste	PO-3				
241-A-153	Diversion Box	241-A-153	Process Waste	PO-3				
241-A-302A	Tanks	241-A-302A		PO-2				
241-A-302B	Tanks	241-A-302B		PO-5				
241-A-350	Tanks	241-A-350	Process Waste	PO-3				
241-A-417	Tanks	241-A-417	Process Waste	PO-3				
241-A-A	Diversion Box	241-A-A	Process Waste	PO-3				
241-A-B	Diversion Box	241-A-B	Process Waste	PO-3				
241-AN-A	Diversion Box	241-AN-A	Process Waste	PO-3				
241-AN-B	Diversion Box	241-AN-B	Process Waste	PO-3				

Table B.4. (page 11 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
241-AP VP	Valve Pit	241-AP	Process Waste	PO-3				
241-AR-151	Diversion Box	241-AR-151	Process Waste	PO-3				
241-AW-A	Diversion Box	241-AW-A	Process Waste	PO-3				
241-AW-B	Diversion Box	241-AW-B	Process Waste	PO-3				
241-AX-151	Diversion Box	241-AX-151	Process Waste	PO-3				
241-AX-152DS	Tanks	241-AX-152DS	Process Waste	PO-3				
241-AX-155	Diversion Box	241-AX-155	Tank Farm Waste	PO-3				
241-AX-501	Valve Pit	241-AX-501		PO-3				
241-AX-A	Diversion Box	241-AX-A		PO-3				
241-AX-B	Diversion Box	241-AX-B		PO-3				
241-AY-151	Diversion Box	241-AY-151	Process Waste	PO-3				
241-AY-152	Diversion Box	241-AY-152	Process Waste	PO-3				
241-AZ-151DS	Diversion Box	241-AZ-151DS		PO-3				
241-AZ-152	Diversion Box	241-AZ-152		PO-3				
241-C-151	Diversion Box	241-C-151		PO-3				
241-C-152	Diversion Box	241-C-152		PO-3				
241-C-153	Diversion Box	241-C-153		PO-3				
241-C-154	Diversion Box	241-C-154 Diversion Box	Process Waste	SO-1				
241-C-252	Diversion Box	241-C-252		PO-3				
241-C-301C	Tanks	241-C-301C		PO-3				
241-CR-151	Diversion Box	241-CR-151		PO-3				
241-CR-152	Diversion Box	241-CR-152		PO-3				
241-CR-153	Diversion Box	241-CR-153		PO-3				
241-CX-TK-70	Tanks	241-CX-70 Storage Tank	Tank Farm Waste	SO-1				
241-CX-TK-71	Tanks	241-CX-71 Storage Tank	Process Condensate	SO-1	0.0988			
241-CX-TK-72	Tanks	241-CX-72 Storage Tank	Process Waste	SO-1	0.000000533			
241-ER-153	Diversion Box	241-ER-153		PO-3				
241-S-151	Diversion Box	241-S-151	LLW - SOLID	RO-2				
241-S-152	Diversion Box	241-S-152	Tank Farm Waste	RO-4				
241-S-302A	Tanks	241-S-302A	Lab Waste	RO-2				
241-S-302B	Tanks	241-S-302B	LLW - SOLID	RO-4				
241-S-A	Diversion Box	241-S-A	Tank Farm Waste	RO-4				
241-S-B	Diversion Box	241-S-B	Tank Farm Waste	RO-4				

Table B.4. (page 12 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
241-S-C	Diversion Box	241-S-C	Tank Farm Waste	RO-4				
241-S-D	Diversion Box	241-S-D	Tank Farm Waste	RO-4				
241-SX-151	Diversion Box	241-SX-151	Tank Farm Waste	RO-4				
241-SX-152	Diversion Box	241-SX-152	Tank Farm Waste	RO-4				
241-SX-302	Tanks	241-SX-302		RO-2				
241-SX-A	Diversion Box	241-SX-A		RO-4				
241-SX-B	Diversion Box	241-SX-B		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-T-151	Diversion Box	241-T-151 Diversion Box	Tank Farm Waste	TP-6				
241-T-152	Diversion Box	241-T-152 Diversion Box	Tank Farm Waste	TP-6				
241-T-153	Diversion Box	241-T-153 Diversion Box	Tank Farm Waste	TP-6				
241-T-252	Diversion Box	241-T-252 Diversion Box	Tank Farm Waste	TP-6				
241-T-301	Tanks	241-T-301 Catch Tank	Tank Farm Waste	TP-6				
241-T-302	Tanks	241-T-302 Catch Tank	Tank Farm Waste	TP-6				
241-T-361	Tanks	241-T-361 Settling Tank	Process Waste	TP-4				
241-TR-152	Diversion Box	241-TR-152 Diversion Box	Tank Farm Waste	TP-6				
241-TR-153	Diversion Box	241-TR-153 Diversion Box	Tank Farm Waste	TP-6				
241-TX-152	Diversion Box	241-TX-152 Diversion Box	Tank Farm Waste	TP-2				
241-TX-153	Diversion Box	241-TX-153 Diversion Box	Tank Farm Waste	TP-5				
241-TX-154	Diversion Box	241-TX-154 Diversion Box	Tank Farm Waste	TP-4				
241-TX-155	Diversion Box	241-TX-155 Diversion Box	Tank Farm Waste	TP-2				
241-TX-302A	Tanks	241-TX-302A Catch Tank	Tank Farm Waste	TP-5				
241-TX-302B	Tanks	241-TX-302B Catch Tank	Tank Farm Waste	TP-2				
241-TX-302C	Tanks	241-TX-302C Catch Tank	Tank Farm Waste	TP-4				
241-TXR-151	Diversion Box	241-TXR-151 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-152	Diversion Box	241-TXR-152 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-153	Diversion Box	241-TXR-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-153	Diversion Box	241-TY-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-302A	Tanks	241-TY-302A Catch Tank	Tank Farm Waste	TP-5				
241-TY-302B	Tanks	241-TY-302B Catch Tank	Tank Farm Waste	TP-5				

Table B.4. (page 13 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
241-Z-361	Tanks	241-Z-361 Settling Tank	Process Waste	ZP-2				
241-Z-TK-8	Tanks	216-Z-8 Settling Tank	Process Waste	ZP-2				
241-Z-TK-D5	Tanks	241-Z Treatment Tank	Process Waste	ZP-2				
242-T-151	Diversion Box	242-T-151 Diversion Box	Process Condensate	TP-5				
244-A RT	Tanks	244-A	Process Waste	PO-3				
244-AR VAULT	Vault	244-AR	Process Waste	PO-3				
244-CR VAULT	Vault	244-CR	Process Waste	PO-3				
244-S RT	Tanks	244-S Receiver Tank		RO-2				
244-TX RT	Tanks	244-TX Receiving Tank		TP-5				
244-TXR	Vault	244-TXR Vault	Tank Farm Waste	TP-5				
2607-E5	Septic System	2607-E-5 Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-E6	Septic System	2607-E6	Sanitary Waste	PO-2				
2607-E7A	Septic System	2607-E-7A Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-EA	Septic System	2607-EA	Sanitary Waste	PO-2				
2607-EC	Septic System	2607-EC	Sanitary Waste	PO-5				
2607-ED	Septic System	2607-ED	Sanitary Waste	PO-3				
2607-EE	Septic System	2607-EL	Sanitary Waste	PO-2				
2607-EG	Septic System	2607-EG	Sanitary Waste	PO-3				
2607-EJ	Septic System	2607-EJ	Sanitary Waste	PO-3				
2607-N	Septic System	2607-N Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-P	Septic System	2607-P Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-R	Septic System	2607-R Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-W1	Septic System	2607-W1 Septic Tank	Sanitary Waste	SS-2				
2607-W2	Septic System	2607-W2 Septic Tank	Sanitary Waste	SS-2				
2607-W3	Septic System	2607-W3 Septic Tank	Sanitary Waste	TP-4				
2607-W4	Septic System	2607-W4 Septic Tank	Sanitary Waste	TP-4				
2607-W6	Septic System	2607-W6	Sanitary Waste	RO-3				
2607-W8	Septic System	2607-W-8 Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WA	Septic System	2607-WA Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WB		2607-WB Septic Tank and Drain Field						
2607-WT	Septic System	2607-WT Septic Tank	Sanitary Waste	TP-5				
2607-WTX	Septic System	2607-WTX Septic Tank	Sanitary Waste	TP-5				
2607-WZ	Septic System	2607-WZ	Sanitary Waste	RO-1				

Table B.4. (page 14 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Uranium-238, Units (Ci)	Alpha Emitters, Units (Ci)	Beta Emitters, Units (Ci)	Americium-241, Units (Ci)
2607-Z	Septic System	2607-Z Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-Z-1		2607-Z-1 Septic Tank and Drain Field						
2704-C-WS-1	French Drain	2704-C-WS-1, 2704-C French Drain, Gatehouse French Drain	Miscellaneous Drainage	SO-1				
2904-S-160	Diversion Box	2904-S-160	Cooling Water	RO-1				
2904-S-170	Diversion Box	2904-S-170	Process Waste	RO-1				
2904-S-171	Diversion Box	2904-S-171	Cooling Water	RO-1				
291-C	Building	291-C Ventilation System	Process Condensate	SO-1				
299-E24-111	Reverse Well	299-E24-111		PO-2				
HSVP	Diversion Box	Semi-Works Valve Pit	Process Waste	SO-1				
UPR-200-E-141		UN-200-E-141	Solution Storage (1)	SO-1				
UPR-200-E-36		UN-200-E-36	Process & Decon Wastes	SO-1				
UPR-200-E-37		UN-200-E-37	Process & Decon Wastes	SO-1				
UPR-200-E-98		UN-200-E-98	Process & Decon Wastes	SO-1				
UPR-200-W-160		UPR-200-W-160 Unplanned Release	Tank Farm Waste	TP-4				
Z PLANT BP	Burial Site	Z Plant Burn Pit	Debris	ZP-3				

Table B.5. Environmental Restoration Waste Site Inventories for Tritium, Cobalt-60, Carbon-14, and Europium-154 (page 1 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
		212-N to 216-N-1 Pipeline						
		212-P Hazardous Waste Staging Area						
		212-P to 216-N-4 Pipeline						
		212-P Transformer Oil Tank						
		212-R to 216-N-6 Pipeline						
		241-C Waste Line Unplanned Release No. 1						
		241-C Waste Line Unplanned Release No. 2						
		241-Z Diversion Box No. 1						
		241-Z Diversion Box No. 2						
		Sanitary Crib						
200-E BP	Burial Site	200-E Burning Pit	Debris	PO-6				
200-E PAP	Burial Site	200-E Ash Pit		SS-1				
200-E PD	Ditches	200 East Powerhouse Ditch	Cooling Water	SO-1				
200-E-4	French Drain	Critical Mass Laboratory Dry Well North	Miscellaneous Drainage	SO-1				
200-N-3	Burial Site	Ballast Pits	Debris	NO-1				
200-W ADB	Burial Site	200-W Ash Disposal Basin	Ash	SS-2				
200-W ADS	Burial Site	200-W Ash Pit Demolition Site	N/A	SS-2				
200-W BP	Burial Site	200-W Burning Pit	Debris	SS-2				
200-W PAP	Burial Site	200-W Powerhouse Ash Pit	Ash	SS-2				
200-W PP	Ponds	200-W Powerhouse Pond	Cooling Water	TP-2				
201-C	Building	201-C Process Building	Process Condensate	SO-1				
207-A		207-A						
207-B	Retention Basin	207-Bb/ Retention Basin	Cooling Water	BP-8				
207-S	Retention Basin	207-S	Cooling Water	RO-2				
207-SL	Retention Basin	207-SL	Lab Waste	RO-3				
207-T	Retention Basin	207-T Retention Basin	Cooling Water	TP-3				
207-Z	Retention Basin	207-Z Retention Basin	Steam Condensate	ZP-2				
209-E-WS-1	French Drain	Critical Mass Laboratory Dry Well East	Miscellaneous Drainage	SO-1				
209-E-WS-2	French Drain	Critical Mass Laboratory Dry Well South	Miscellaneous Drainage	SO-1				
209-E-WS-3	Diversion Box	Critical Mass Laboratory Valve Pit	Process Waste	SO-1				
2101-M POND	Ponds	2101-M Pond	Lab Waste	SS-1				
216-A-1	Cribs	216-A-1	Process Waste	PO-5		0.00179		
216-A-10	Cribs	216-A-10	Process Condensate	PO-2	1850			

Table B.5. (page 2 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
216-A-11	French Drain	216-A-11	Miscellaneous Drainage	PO-2				
216-A-12	French Drain	216-A-12	Miscellaneous Drainage	PO-2				
216-A-13	French Drain	216-A-13	Miscellaneous Drainage	PO-2				
216-A-14	French Drain	216-A-14	Miscellaneous Drainage	PO-2				
216-A-15	French Drain	216-A-15	Process Condensate	PO-2				
216-A-16	French Drain	216-A-16	Chemical Sewer	PO-5				
216-A-17	French Drain	216-A-17	Chemical Sewer	PO-5				
216-A-18	Trench	216-A-18	Process Waste	PO-5		0.00179		
216-A-19	Trench	216-A-19	Process Waste	PO-5		0.00179		
216-A-2	Cribs	216-A-2	Process Waste	PO-2		0.0297		
216-A-20	Trench	216-A-20	Process Waste	PO-5		0.00179		
216-A-21	Cribs	216-A-21	Lab Waste	PO-2		0.471		
216-A-22	French Drain	216-A-22	Miscellaneous Drainage	PO-2				
216-A-23A	French Drain	216-A-23A	Process Condensate	PO-5				
216-A-23B	French Drain	216-A-23B	Process Condensate	PO-5				
216-A-24	Cribs	216-A-24	Process Condensate	PO-5	1400	0.0219		
216-A-25	Ponds	216-A-25 Pond	Cooling Water	IU-6	213			
216-A-26	French Drain	216-A-26	Miscellaneous Drainage	PO-2				
216-A-26A	French Drain	216-A-26A	Miscellaneous Drainage	PO-2				
216-A-27	Cribs	216-A-27	Miscellaneous Drainage	PO-2		0.3		
216-A-28	Cribs	216-A-28	Process Condensate	PO-2				
216-A-29	Ditches	216-A-29	Chemical Sewer	BP-11				
216-A-3	Cribs	216-A-3	Process Waste	PO-2				
216-A-30	Cribs	216-A-30	Steam Condensate	PO-4	16			
216-A-31	Cribs	216-A-31	Process Waste	PO-2		0.00588		
216-A-32	Cribs	216-A-32	Miscellaneous Drainage	PO-2				
216-A-33	French Drain	216-A-33	Miscellaneous Drainage	PO-2				
216-A-34	Cribs	216-A-34	Process Condensate	PO-5				
216-A-35	French Drain	216-A-35	Miscellaneous Drainage	PO-2				
216-A-36A	Cribs	216-A-36A	Process Waste	PO-2		0.71		
216-A-36B	Cribs	216-A-36B	Process Waste	PO-2	507			
216-A-37-1	Cribs	216-A-37-1	Process Condensate	PO-4	1600			
216-A-37-2	Cribs	216-A-37-2	Steam Condensate	PO-4	6			

Table B.5. (page 3 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
216-A-38-1	Cribs	216-A-38-1	N/A	PO-2				
216-A-39	Ditches	216-A-39	Miscellaneous Drainage	PO-3				
216-A-4	Cribs	216-A-4	Lab Waste	PO-2		0.0226		
216-A-40	Retention Basin	216-A-40	Steam Condensate	PO-2				
216-A-41	Cribs	216-A-41	Miscellaneous Drainage	PO-2				
216-A-42	Retention Basin	216-A-42	Cooling Water	PO-4				
216-A-45	Cribs	216-A-45	Process Condensate	PO-2	3850			
216-A-5	Cribs	216-A-5	Process Condensate	PO-2		3.32		
216-A-524	Diversion Box	216-A-524		PO-5				
216-A-6	Cribs	216-A-6	Steam Condensate	PO-4		0.18		
216-A-7	Cribs	216-A-7	Process Waste	PO-5		0.00204		
216-A-8	Cribs	216-A-8	Process Condensate	PO-5	0			
216-A-9	Cribs	216-A-9	Cooling Water	PO-2	4000	0.00583		
216-B-10A	Cribs	216-B-10A Crib	Lab Waste	BP-6	0	0.00099		
216-B-10B	Cribs	216-B-10B Crib	Lab Waste	BP-6	0	0		
216-B-11A&B	Reverse Well	216-B-11A&B Reverse Wells	Process Condensate	BP-4	0	0.00143		
216-B-12	Cribs	216-B-12 Crib	Process Condensate	BP-9	0	0.232		
216-B-13	French Drain	216-B-13 French Drain	Miscellaneous Drainage	BP-6				
216-B-14	Cribs	216-B-14 Crib	Scavenged Waste	BP-2	0	0.103		
216-B-15	Cribs	216-B-15 Crib	Scavenged Waste	BP-2	0	0.109		
216-B-16	Cribs	216-B-16 Crib	Scavenged Waste	BP-2	450	0.103		
216-B-17	Cribs	216-B-17 Crib	Scavenged Waste	BP-2	0	0.0204		
216-B-18	Cribs	216-B-18 Crib	Scavenged Waste	BP-2	0	0.103		
216-B-19	Cribs	216-B-19 Crib	Scavenged Waste	BP-2	0	0.117		
216-B-2-1	Ditches	216-B-2-1 Ditchb/	Cooling Water	BP-11	790			
216-B-2-2	Ditches	216-B-2-2 Ditchc/	Cooling Water	BP-11	0			
216-B-2-3	Ditches	216-B-2-3 Ditch	Cooling Water	BP-11				
216-B-20	Trench	216-B-20 Trench	Scavenged Waste	BP-2	0	0.0899		
216-B-21	Trench	216-B-21 Trench	Scavenged Waste	BP-2	0	0.133		
216-B-22	Trench	216-B-22 Trench	Scavenged Waste	BP-2	0	0.274		
216-B-23	Trench	216-B-23 Trench	Scavenged Waste	BP-2	0	0.137		
216-B-24	Trench	216-B-24 Trench	Scavenged Waste	BP-2	0	0.21		
216-B-25	Trench	216-B-25 Trench	Scavenged Waste	BP-2	0	0.141		

Table B.5. (page 4 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
216-B-26	Trench	216-B-26 Trench	Scavenged Waste	BP-2	0	0.223		
216-B-27	Trench	216-B-27 Trench	Scavenged Waste	BP-2	0	0.177		
216-B-28	Trench	216-B-28 Trench	Scavenged Waste	BP-2	0	0.0537		
216-B-29	Trench	216-B-29 Trench	Scavenged Waste	BP-2	0	0.165		
216-B-3	Ponds	216-B-3 Ponds/	Cooling Water	BP-11	790			
216-B-3-1	Ditches	216-B-3-1 Ditchb/	Cooling Water	BP-11				
216-B-3-2	Ditches	216-B-3-2 Ditchc/	Cooling Water	BP-11				
216-B-3-3	Ditches	216-B-3-3 Ditch	Cooling Water	BP-11				
216-B-30	Trench	216-B-30 Trench	Scavenged Waste	BP-2	0	0.0397		
216-B-32	Trench	216-B-32 Trench	Scavenged Waste	BP-2	0	0.0397		
216-B-33	Trench	216-B-33 Trench	Scavenged Waste	BP-2	0	0.0327		
216-B-34	Trench	216-B-34 Trench	Scavenged Waste	BP-2	0	0.014		
216-B-35	Trench	216-B-35 Trench	Tank Farm Waste	BP-3	0	0.00047		
216-B-36	Trench	216-B-36 Trench	Tank Farm Waste	BP-3	0	0.0011		
216-B-37	Trench	216-B-37 Trench	Process Waste	BP-3	0	0.0157		
216-B-38	Trench	216-B-38 Trench	Tank Farm Waste	BP-3	0	0.00094		
216-B-39	Trench	216-B-39 Trench	Tank Farm Waste	BP-3	0	0.0148		
216-B-3A	Ponds	216-B-3A Pond	Cooling Water	BP-11				
216-B-3B	Ponds	216-B-3B Pond	Cooling Water	BP-11				
216-B-3C	Ponds	216-B-3C Pond	Cooling Water	BP-11				
216-B-4	Reverse Well	216-B-4 Reverse Well	Miscellaneous Drainage	BP-6	0	0		
216-B-40	Trench	216-B-40 Trench	Tank Farm Waste	BP-3	0	0.00031		
216-B-41	Trench	216-B-41 Trench	Tank Farm Waste	BP-3	0	0.00016		
216-B-42	Trench	216-B-42 Trench	Scavenged Waste	BP-3	0	0.179		
216-B-43	Cribs	216-B-43 Crib	Scavenged Waste	BP-1	170	0.0157		
216-B-44	Cribs	216-B-44 Crib	Scavenged Waste	BP-1	450	0.0848		
216-B-45	Cribs	216-B-45 Crib	Scavenged Waste	BP-1	390	0.0899		
216-B-46	Cribs	216-B-46 Crib	Scavenged Waste	BP-1	536	0.0899		
216-B-47	Cribs	216-B-47 Crib	Scavenged Waste	BP-1	0	0.0179		
216-B-48	Cribs	216-B-48 Crib	Scavenged Waste	BP-1	327	0.0179		
216-B-49	Cribs	216-B-49 Crib	Scavenged Waste	BP-1	536	0.0899		
216-B-5	Reverse Well	216-B-5 Reverse Well	Process Waste	BP-6	0	0		
216-B-50	Cribs	216-B-50 Crib	Process Condensate	BP-1	90	0.0283		

Table B.5. (page 5 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
216-B-51	French Drain	216-B-51 French Drain	Miscellaneous Drainage	BP-4				
216-B-52	Trench	216-B-52 Trench	Scavenged Waste	BP-2	0	0.113		
216-B-53A	Trench	216-B-53A Trench	Lab Waste	BP-2	0	0.0335		
216-B-53B	Trench	216-B-53B Trench	Lab Waste	BP-2	0	0.0483		
216-B-54	Trench	216-B-54 Trench	Lab Waste	BP-2	0	0.0059		
216-B-55	Cribs	216-B-55 Crib	Steam Condensate	BP-9	2.68			
216-B-56	Cribs	216-B-56 Crib	N/A	BP-6				
216-B-57	Cribs	216-B-57 Crib	Process Condensate	BP-1	0	0.0147		
216-B-58	Trench	216-B-58 Trench	Lab Waste	BP-2	0	0.198		
216-B-59	Retention Basin	216-B-59 Basin	Cooling Water	BP-6				
216-B-6	Reverse Well	216-B-6 Reverse Well	Lab Waste	BP-6	0	0		
216-B-60	Cribs	216-B-60 Crib	Decon Waste	BP-6				
216-B-61	Cribs	216-B-61 Crib	N/A	BP-1				
216-B-62	Cribs	216-B-62 Crib	Process Condensate	BP-9	14.7			
216-B-63	Ditches	216-B-63 Trench	Chemical Sewer	BP-11	2.12			
216-B-64	Retention Basin	216-B-64 Basin	N/A	BP-9				
216-B-7A&B	Cribs	216-B-7A&B Crib	Process Waste	BP-4	0	0.012		
216-B-8	Cribs	216-B-8TF Crib	Process Waste	BP-4	0	0.009		
216-B-9	Cribs	216-B-9TF Crib	Process Waste	BP-6	0	0.0009		
216-C-1	Cribs	216-C-1 Crib	Process Condensate	SO-1	70	0.002		
216-C-10	Cribs	216-C-10 Crib	Process Condensate	SO-1		0.0113		
216-C-2	Reverse Well	216-C-2 Reverse Well	Miscellaneous Drainage	SO-1				
216-C-3	Cribs	216-C-3 Crib	Process Waste	SO-1		0.0014		
216-C-4	Cribs	216-C-4 Crib	Process Waste	SO-1		0.0018		
216-C-5	Cribs	216-C-5 Crib	Process Waste	SO-1		0.0018		
216-C-6	Cribs	216-C-6 Crib	Process Condensate	SO-1		0.0025		
216-C-7	Cribs	216-C-7 Crib	Process Waste	SO-1				
216-C-8	French Drain	216-C-8	Process Waste	PO-3				
216-C-9	Ponds	216-C-9 Pond	Cooling Water	SO-1				
216-E-28	Ponds	216-E-28 Pond	N/A	BP-11				
216-N-1	Ponds	216-N-1 Pond	Cooling Water	NO-1				
216-N-2	Trench	216-N-2 Trench	Cooling Water	NO-1				
216-N-3	Trench	216-N-3 Trench	Cooling Water	NO-1				

Table B.5. (page 6 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
216-N-4	Ponds	216-N-4 Pond	Cooling Water	NO-1				
216-N-5	Trench	216-N-5 Trench	Cooling Water	NO-1				
216-N-6	Ponds	216-N-6 Pond	Cooling Water	NO-1				
216-N-7	Trench	216-N-7 Trench	Cooling Water	NO-1				
216-N-8	Ponds	216-N-8 Pond		IU-6				
216-S-1&2	Cribs	216-S-1 & 2	Process Condensate	RO-2				
216-S-10D	Ditches	216-S-10D	Chemical Sewer	RO-1				
216-S-10P	Ponds	216-S-10P	Chemical Sewer	RO-1				
216-S-11	Ponds	216-S-11	Chemical Sewer	RO-1				
216-S-12	Trench	216-S-12	Miscellaneous Drainage	RO-3				
216-S-13	Cribs	216-S-13	Process Waste	RO-2				
216-S-14	Trench	216-S-14	Process Waste	RO-3				
216-S-15	Ponds	216-S-15	Cooling Water	RO-2				
216-S-16D	Ditches	216-S-16D	Cooling Water	RO-1				
216-S-16P	Ponds	216-S-16P	Cooling Water	RO-1				
216-S-17	Ponds	216-S-17	Cooling Water	RO-1				
216-S-172	Diversion Box	216-S-172	Cooling Water	RO-1				
216-S-18	Trench	216-S-18	Debris	RO-2				
216-S-19	Ponds	216-S-19	Lab Waste	RO-1	0.187			
216-S-20	Cribs	216-S-20	Lab Waste	RO-3				
216-S-22	Cribs	216-S-22	Process Waste	RO-3				
216-S-23	Cribs	216-S-23	Process Condensate	RO-2				
216-S-25	Cribs	216-S-25	Steam Condensate	RO-1	148			
216-S-26	Cribs	216-S-26	Lab Waste	RO-3				
216-S-3	French Drain	216-S-3	Process Condensate	RO-2				
216-S-4	French Drain	216-S-4	Process Condensate	UP-2	0.02			
216-S-5	Cribs	216-S-5	Cooling Water	RO-1				
216-S-6	Cribs	216-S-6	Cooling Water	RO-1				
216-S-7	Cribs	216-S-7	Process Condensate	RO-2				
216-S-8	Trench	216-S-8	Process Waste	RO-2				
216-S-9	Cribs	216-S-9	Process Condensate	RO-2				
216-T-1	Ditches	216-T-1 Ditch	Cooling Water	TP-4				
216-T-10	Trench	216-T-10 Trench	Decon Waste	TP-4				

Table B.5. (page 7 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
216-T-11	Trench	216-T-11 Trench	Decon Waste	TP-4				
216-T-12	Trench	216-T-12 Trench	Cooling Water	TP-3		0.0341		
216-T-13	Trench	216-T-13 Trench	Decon Waste	TP-2				
216-T-14	Trench	216-T-14 Trench	Tank Farm Waste	TP-3	0.8	0.236		
216-T-15	Trench	216-T-15 Trench	Tank Farm Waste	TP-3	0.8	0.186		
216-T-16	Trench	216-T-16 Trench	Tank Farm Waste	TP-3	0.8	0.204		
216-T-17	Trench	216-T-17 Trench	Tank Farm Waste	TP-3	0.6	0.0157		
216-T-18	Cribs	216-T-18 Crib	Tank Farm Waste	TP-2	0.8	0.137		
216-T-19	Cribs	216-T-19TF Crib and Tile Field	Process Waste	TP-2	4.25			
216-T-2	Reverse Well	216-T-2 Reverse Well	Lab Waste	TP-4				
216-T-20	Trench	216-T-20 Trench	Process Waste	TP-2				
216-T-21	Trench	216-T-21 Trench	Tank Farm Waste	TP-1	0.4	0.314		
216-T-22	Trench	216-T-22 Trench	Tank Farm Waste	TP-1	1.2	0.0157		
216-T-23	Trench	216-T-23 Trench	Tank Farm Waste	TP-1	1.2	0.0157		
216-T-24	Trench	216-T-24 Trench	Tank Farm Waste	TP-1	1.2	0.0157		
216-T-25	Trench	216-T-25 Trench	Process Waste	TP-1	2.4	0.00157		
216-T-26	Cribs	216-T-26 Crib	Tank Farm Waste	TP-2		0.0189		
216-T-27	Cribs	216-T-27 Crib	Lab Waste	TP-2		0.067		
216-T-28	Cribs	216-T-28 Crib	Decon Waste	TP-2		0.319		
216-T-29	Cribs	216-T-29 Crib	Miscellaneous Drainage	TP-4				
216-T-3	Reverse Well	216-T-3 Reverse Well	Process Waste	TP-4				
216-T-31	French Drain	216-T-31 French Drain	Miscellaneous Drainage	TP-2				
216-T-32	Cribs	216-T-32 Crib	Process Waste	TP-1		0.00827		
216-T-33	Cribs	216-T-33 Crib	Decon Waste	TP-4		0.0515		
216-T-34	Cribs	216-T-34 Crib	Lab Waste	TP-4		0.585		
216-T-35	Cribs	216-T-35 Crib	Lab Waste	TP-4		0.298		
216-T-36	Cribs	216-T-36 Crib	Steam Condensate	TP-1		0.0487		
216-T-4-1D	Ditches	216-T-4-1D Ditch	Cooling Water	TP-3				
216-T-4-2	Ditches	216-T-4-2 Ditch	Steam Condensate	TP-3				
216-T-4A	Ponds	216-T-4A Pond	Cooling Water	TP-3				
216-T-4B	Ponds	216-T-4B Pond	Cooling Water	TP-3				
216-T-5	Trench	216-T-5 Trench	Tank Farm Waste	TP-1		0.0899		
216-T-6	Cribs	216-T-6 Crib	Process Waste	TP-3		0.0305		

Table B.5. (page 8 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
216-T-7	Cribs	216-T-7TF Crib and Tile Field	Tank Farm Waste	TP-1		0.0142		
216-T-8	Cribs	216-T-8 Crib	Lab Waste	TP-4		0.00099		
216-T-9	Trench	216-T-9 Trench	Decon Waste	TP-4				
216-U-1&2	Cribs	216-U-1 & 216-U-2	Process Condensate	UP-2		0.00157		
216-U-10	Ponds	216-U-10	Cooling Water	UP-2	196			
216-U-11	Ditches	216-U-11	Cooling Water	UP-2				
216-U-12	Cribs	216-U-12	Process Condensate	UP-2	0.00188			
216-U-13	Trench	216-U-13 (same as UN-200-W- 125)	Decon Waste	UP-2		0.00179		
216-U-14	Ditches	216-U-14	Cooling Water	UP-2				
216-U-15	Trench	216-U-15	Process Waste	UP-2		0.00233		
216-U-16	Cribs	216-U-16	Process Condensate	UP-2	0.233			
216-U-17	Cribs	216-U-17	Process Condensate	UP-2	69.7			
216-U-21		216-U-21				0.3333		
216-U-3	French Drain	216-U-3	Miscellaneous Drainage	UP-2		0.00157		
216-U-4	Reverse Well	216-U-4	Lab Waste	UP-2				
216-U-4A	French Drain	216-U-4A	Miscellaneous Drainage	UP-2				
216-U-4B	French Drain	216-U-4B	Miscellaneous Drainage	UP-2				
216-U-5	Trench	216-U-5 & 216-U-6	Process Waste	UP-2		0.0006		
216-U-7	French Drain	216-U-7	Miscellaneous Drainage	UP-2				
216-U-8	Cribs	216-U-8	Process Condensate	UP-2		0.00204		
216-U-9	Ditches	216-U-9	Cooling Water	RO-1				
216-W-LWC	Cribs	216-W-LWC Crib	Chemical Sewer	SS-2				
216-Z-1&2	Cribs	216-Z-1 & 216-Z-2 Cribs	Process Waste	ZP-2		0.0171		
216-Z-10	Reverse Well	216-Z-10 Reverse Well	Process Waste	ZP-2				
216-Z-11	Ditches	216-Z-11	Cooling Water	UP-2				
216-Z-12	Cribs	216-Z-12 Crib	Process Waste	ZP-2		0.00515		
216-Z-13	French Drain	216-Z-13 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-14	French Drain	216-Z-14 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-15	French Drain	216-Z-15 French Drain	Miscellaneous Drainage	ZP-2				
216-Z-16	Cribs	216-Z-16 Crib	Lab Waste	ZP-2				
216-Z-17	Ditches	216-Z-17 Trench	Lab Waste	ZP-2				
216-Z-18	Cribs	216-Z-18 Crib	Process Waste	ZP-2				
216-Z-19	Ditches	216-Z-19	Cooling Water	UP-2				

Table B.5. (page 9 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
216-Z-1A	Cribs	216-Z-1A Tile Field	Process Waste	ZP-2				
216-Z-1D	Ditches	216-Z-1D	Cooling Water	UP-2				
216-Z-20	Cribs	216-Z-20	Cooling Water	UP-2				
216-Z-21	Retention Basin	216-Z-21 Seepage Basin	Cooling Water	ZP-2				
216-Z-3	Cribs	216-Z-3 Crib	Process Waste	ZP-2				
216-Z-4	Trench	216-Z-4 Trench	Process Waste	ZP-2				
216-Z-5	Cribs	216-Z-5 Crib	Process Waste	ZP-2		0.0026		
216-Z-6	Cribs	216-Z-6 Crib	Process Waste	ZP-2		0.00048		
216-Z-7	Cribs	216-Z-7 Crib	Lab Waste	ZP-2		0.0765		
216-Z-8	Cribs	216-Z-8 French Drain	Process Waste	ZP-2				
216-Z-9	Cribs	216-Z-9 Trench	Process Waste	ZP-2		0.00395		
218-C-9	Burial Site	218-C-9 Burial Ground	LLW - SOLID	SO-1			0.000001	
218-E-1	Burial Site	218-E-1	LLW - SOLID	PO-2				
218-E-10	Burial Site	218-E-10 Burial Ground	LLW - SOLID	BP-10		2470		
218-E-12A	Burial Site	218-E-12A	LLW - SOLID	PO-6				
218-E-12B	Burial Site	218-E-12B	LLW - SOLID	PO-6				
218-E-13	Burial Site	218-E-13		PO-2				
218-E-2	Burial Site	218-E-2 Burial Ground	LLW - SOLID	BP-10				
218-E-2A	Burial Site	218-E-2A Burial Ground	LLW - SOLID	BP-10				
218-E-4	Burial Site	218-E-4 Burial Ground	LLW - SOLID	BP-10				
218-E-5	Burial Site	218-E-5 Burial Ground	LLW - SOLID	BP-10				
218-E-5A	Burial Site	218-E-5A Burial Ground	LLW - SOLID	BP-10				
218-E-6	Burial Site	218-E-6 Burial Ground	Debris	BP-6				
218-E-7	Burial Site	218-E-7 Burial Ground	Lab Waste	BP-6				
218-E-8	Burial Site	218-E-8	TRU Solid Waste	PO-6				
218-E-9	Burial Site	218-E-9 Burial Ground	LLW - SOLID	BP-10				
218-W-1	Burial Site	218-W-1 Burial Ground	TRU Solid Waste	ZP-3				
218-W-11	Burial Site	218-W-11 Burial Ground	LLW - SOLID	ZP-3				
218-W-1A	Burial Site	218-W-1A Burial Ground	LLW - SOLID	ZP-3				
218-W-2	Burial Site	218-W-2 Burial Ground	TRU Solid Waste	ZP-3				
218-W-2A	Burial Site	218-W-2A Burial Ground	LLW - SOLID	ZP-3		0.33		
218-W-3	Burial Site	218-W-3 Burial Ground	TRU Solid Waste	ZP-3				
218-W-3A	Burial Site	218-W-3A Burial Ground	TRU Solid Waste	ZP-3	178000	9840	1.74	0.145

Table B.5. (page 10 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
218-W-3AE	Burial Site	218-W-3AE Burial Ground	LLW - SOLID	ZP-3	19500	299	0.321	0.141
218-W-4A	Burial Site	218-W-4A Burial Ground	TRU Solid Waste	ZP-3				
218-W-4B	Burial Site	218-W-4B Caissons	TRU Solid Waste	ZP-3	786	76000		0.211
218-W-4B	Burial Site	218-W-4B Trenches	TRU Solid Waste	ZP-3	68500			
218-W-4C	Burial Site	218-W-4C Burial Ground	TRU Solid Waste	ZP-3	25.1	221000	7.85	288
218-W-5	Burial Site	218-W-5 Burial Ground	TRU Solid Waste	ZP-3	15200	3410	4.29	108
218-W-6	Burial Site	218-W-6 Burial Ground	LLW - SOLID	ZP-3				
218-W-7	Burial Site	218-W-7	LLW - SOLID	RO-3				
218-W-8	Burial Site	218-W-8 Burial Ground	Lab Waste	TP-4				
218-W-9	Burial Site	218-W-9	LLW - SOLID	RO-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2				
231-W-151	Vault	231-Z-151 Sump		ZP-2				
232-Z	Building	232-Z Incinerator		ZP-2				
240-S-151	Diversion Box	240-S-151	LLW - SOLID	RO-3				
240-S-152	Diversion Box	240-S-152	Tank Farm Waste	RO-3				
240-S-302	Tanks	240-S-302	Lab Waste	RO-3				
241-A-151	Diversion Box	241-A-151	Tank Farm Waste	PO-2				
241-A-152	Diversion Box	241-A-152	Process Waste	PO-3				
241-A-153	Diversion Box	241-A-153	Process Waste	PO-3				
241-A-302A	Tanks	241-A-302A		PO-2				
241-A-302B	Tanks	241-A-302B		PO-5				
241-A-350	Tanks	241-A-350	Process Waste	PO-3				
241-A-417	Tanks	241-A-417	Process Waste	PO-3				
241-A-A	Diversion Box	241-A-A	Process Waste	PO-3				
241-A-B	Diversion Box	241-A-B	Process Waste	PO-3				
241-AN-A	Diversion Box	241-AN-A	Process Waste	PO-3				
241-AN-B	Diversion Box	241-AN-B	Process Waste	PO-3				

Table B.5. (page 11 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
241-AP VP	Valve Pit	241-AP	Process Waste	PO-3				
241-AR-151	Diversion Box	241-AR-151	Process Waste	PO-3				
241-AW-A	Diversion Box	241-AW-A	Process Waste	PO-3				
241-AW-B	Diversion Box	241-AW-B	Process Waste	PO-3				
241-AX-151	Diversion Box	241-AX-151	Process Waste	PO-3				
241-AX-152DS	Tanks	241-AX-152DS	Process Waste	PO-3				
241-AX-155	Diversion Box	241-AX-155	Tank Farm Waste	PO-3				
241-AX-501	Valve Pit	241-AX-501		PO-3				
241-AX-A	Diversion Box	241-AX-A		PO-3				
241-AX-B	Diversion Box	241-AX-B		PO-3				
241-AY-151	Diversion Box	241-AY-151	Process Waste	PO-3				
241-AY-152	Diversion Box	241-AY-152	Process Waste	PO-3				
241-AZ-151DS	Diversion Box	241-AZ-151DS		PO-3				
241-AZ-152	Diversion Box	241-AZ-152		PO-3				
241-C-151	Diversion Box	241-C-151		PO-3				
241-C-152	Diversion Box	241-C-152		PO-3				
241-C-153	Diversion Box	241-C-153		PO-3				
241-C-154	Diversion Box	241-C-154 Diversion Box	Process Waste	SO-1				
241-C-252	Diversion Box	241-C-252		PO-3				
241-C-301C	Tanks	241-C-301C		PO-3				
241-CR-151	Diversion Box	241-CR-151		PO-3				
241-CR-152	Diversion Box	241-CR-152		PO-3				
241-CR-153	Diversion Box	241-CR-153		PO-3				
241-CX-TK-70	Tanks	241-CX-70 Storage Tank	Tank Farm Waste	SO-1				
241-CX-TK-71	Tanks	241-CX-71 Storage Tank	Process Condensate	SO-1	70	0.002		
241-CX-TK-72	Tanks	241-CX-72 Storage Tank	Process Waste	SO-1				
241-ER-153	Diversion Box	241-ER-153		PO-3				
241-S-151	Diversion Box	241-S-151	LLW - SOLID	RO-2				
241-S-152	Diversion Box	241-S-152	Tank Farm Waste	RO-4				
241-S-302A	Tanks	241-S-302A	Lab Waste	RO-2				
241-S-302B	Tanks	241-S-302B	LLW - SOLID	RO-4				
241-S-A	Diversion Box	241-S-A	Tank Farm Waste	RO-4				
241-S-B	Diversion Box	241-S-B	Tank Farm Waste	RO-4				

Table B.5. (page 12 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
241-S-C	Diversion Box	241-S-C	Tank Farm Waste	RO-4				
241-S-D	Diversion Box	241-S-D	Tank Farm Waste	RO-4				
241-SX-151	Diversion Box	241-SX-151	Tank Farm Waste	RO-4				
241-SX-152	Diversion Box	241-SX-152	Tank Farm Waste	RO-4				
241-SX-302	Tanks	241-SX-302		RO-2				
241-SX-A	Diversion Box	241-SX-A		RO-4				
241-SX-B	Diversion Box	241-SX-B		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-A	Diversion Box	241-SY-A		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-SY-B	Diversion Box	241-SY-B		RO-4				
241-T-151	Diversion Box	241-T-151 Diversion Box	Tank Farm Waste	TP-6				
241-T-152	Diversion Box	241-T-152 Diversion Box	Tank Farm Waste	TP-6				
241-T-153	Diversion Box	241-T-153 Diversion Box	Tank Farm Waste	TP-6				
241-T-252	Diversion Box	241-T-252 Diversion Box	Tank Farm Waste	TP-6				
241-T-301	Tanks	241-T-301 Catch Tank	Tank Farm Waste	TP-6				
241-T-302	Tanks	241-T-302 Catch Tank	Tank Farm Waste	TP-6				
241-T-361	Tanks	241-T-361 Settling Tank	Process Waste	TP-4				
241-TR-152	Diversion Box	241-TR-152 Diversion Box	Tank Farm Waste	TP-6				
241-TR-153	Diversion Box	241-TR-153 Diversion Box	Tank Farm Waste	TP-6				
241-TX-152	Diversion Box	241-TX-152 Diversion Box	Tank Farm Waste	TP-2				
241-TX-153	Diversion Box	241-TX-153 Diversion Box	Tank Farm Waste	TP-5				
241-TX-154	Diversion Box	241-TX-154 Diversion Box	Tank Farm Waste	TP-4				
241-TX-155	Diversion Box	241-TX-155 Diversion Box	Tank Farm Waste	TP-2				
241-TX-302A	Tanks	241-TX-302A Catch Tank	Tank Farm Waste	TP-5				
241-TX-302B	Tanks	241-TX-302B Catch Tank	Tank Farm Waste	TP-2				
241-TX-302C	Tanks	241-TX-302C Catch Tank	Tank Farm Waste	TP-4				
241-TXR-151	Diversion Box	241-TXR-151 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-152	Diversion Box	241-TXR-152 Diversion Box	Tank Farm Waste	TP-5				
241-TXR-153	Diversion Box	241-TXR-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-153	Diversion Box	241-TY-153 Diversion Box	Tank Farm Waste	TP-5				
241-TY-302A	Tanks	241-TY-302A Catch Tank	Tank Farm Waste	TP-5				
241-TY-302B	Tanks	241-TY-302B Catch Tank	Tank Farm Waste	TP-5				

Table B.5. (page 13 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
241-Z-361	Tanks	241-Z-361 Settling Tank	Process Waste	ZP-2				
241-Z-TK-8	Tanks	216-Z-8 Settling Tank	Process Waste	ZP-2				
241-Z-TK-D5	Tanks	241-Z Treatment Tank	Process Waste	ZP-2				
242-T-151	Diversion Box	242-T-151 Diversion Box	Process Condensate	TP-5				
244-A RT	Tanks	244-A	Process Waste	PO-3				
244-AR VAULT	Vault	244-AR	Process Waste	PO-3				
244-CR VAULT	Vault	244-CR	Process Waste	PO-3				
244-S RT	Tanks	244-S Receiver Tank		RO-2				
244-TX RT	Tanks	244-TX Receiving Tank		TP-5				
244-TXR	Vault	244-TXR Vault	Tank Farm Waste	TP-5				
2607-E5	Septic System	2607-E-5 Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-E6	Septic System	2607-E6	Sanitary Waste	PO-2				
2607-E7A	Septic System	2607-E-7A Septic Tank and Drain Field	Sanitary Waste	SO-1				
2607-EA	Septic System	2607-EA	Sanitary Waste	PO-2				
2607-EC	Septic System	2607-EC	Sanitary Waste	PO-5				
2607-ED	Septic System	2607-ED	Sanitary Waste	PO-3				
2607-EE	Septic System	2607-EL	Sanitary Waste	PO-2				
2607-EG	Septic System	2607-EG	Sanitary Waste	PO-3				
2607-EJ	Septic System	2607-EJ	Sanitary Waste	PO-3				
2607-N	Septic System	2607-N Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-P	Septic System	2607-P Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-R	Septic System	2607-R Septic Tank/Drain Field	Sanitary Waste	NO-1				
2607-W1	Septic System	2607-W1 Septic Tank	Sanitary Waste	SS-2				
2607-W2	Septic System	2607-W2 Septic Tank	Sanitary Waste	SS-2				
2607-W3	Septic System	2607-W3 Septic Tank	Sanitary Waste	TP-4				
2607-W4	Septic System	2607-W4 Septic Tank	Sanitary Waste	TP-4				
2607-W6	Septic System	2607-W6	Sanitary Waste	RO-3				
2607-W8	Septic System	2607-W-8 Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WA	Septic System	2607-WA Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-WB		2607-WB Septic Tank and Drain Field						
2607-WT	Septic System	2607-WT Septic Tank	Sanitary Waste	TP-5				
2607-WTX	Septic System	2607-WTX Septic Tank	Sanitary Waste	TP-5				
2607-WZ	Septic System	2607-WZ	Sanitary Waste	RO-1				

Table B.5. (page 14 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	(Tritium) Hydrogen-3, Units (Ci)	Cobalt-60, Units (Ci)	Carbon-14, Units (Ci)	Europium-154, Units (Ci)
2607-Z	Septic System	2607-Z Septic Tank and Drain Field	Sanitary Waste	ZP-2				
2607-Z-1		2607-Z-1 Septic Tank and Drain Field						
2704-C-WS-1	French Drain	2704-C-WS-1, 2704-C French Drain, Gatehouse French Drain	Miscellaneous Drainage	SO-1				
2904-S-160	Diversion Box	2904-S-160	Cooling Water	RO-1				
2904-S-170	Diversion Box	2904-S-170	Process Waste	RO-1				
2904-S-171	Diversion Box	2904-S-171	Cooling Water	RO-1				
291-C	Building	291-C Ventilation System	Process Condensate	SO-1				
299-E24-111	Reverse Well	299-E24-111		PO-2				
HSVP	Diversion Box	Semi-Works Valve Pit	Process Waste	SO-1				
UPR-200-E-141		UN-200-E-141	Solution Storage (1)	SO-1				
UPR-200-E-36		UN-200-E-36	Process & Decon Wastes	SO-1				
UPR-200-E-37		UN-200-E-37	Process & Decon Wastes	SO-1				
UPR-200-E-98		UN-200-E-98	Process & Decon Wastes	SO-1				
UPR-200-W-160		UPR-200-W-160 Unplanned Release	Tank Farm Waste	TP-4				
Z PLANT BP	Burial Site	Z Plant Bum Pit	Debris	ZP-3				

Table B.6. Environmental Restoration Waste Site Inventories for Promethium-147, Tin (Sn-113), and Iodine-129 (page 1 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
		212-N to 216-N-1 Pipeline					
		212-P Hazardous Waste Staging Area					
		212-P to 216-N-4 Pipeline					
		212-P Transformer Oil Tank					
		212-R to 216-N-6 Pipeline					
		241-C Waste Line Unplanned Release No. 1					
		241-C Waste Line Unplanned Release No. 2					
		241-Z Diversion Box No. 1					
		241-Z Diversion Box No. 2					
		Sanitary Crib					
200-E BP	Burial Site	200-E Burning Pit	Debris	PO-6			
200-E PAP	Burial Site	200-E Ash Pit		SS-1			
200-E PD	Ditches	200 East Powerhouse Ditch	Cooling Water	SO-1			
200-E-4	French Drain	Critical Mass Laboratory Dry Well North	Miscellaneous Drainage	SO-1			
200-N-3	Burial Site	Ballast Pits	Debris	NO-1			
200-W ADB	Burial Site	200-W Ash Disposal Basin	Ash	SS-2			
200-W ADS	Burial Site	200-W Ash Pit Demolition Site	N/A	SS-2			
200-W BP	Burial Site	200-W Burning Pit	Debris	SS-2			
200-W PAP	Burial Site	200-W Powerhouse Ash Pit	Ash	SS-2			
200-W PP	Ponds	200-W Powerhouse Pond	Cooling Water	TP-2			
201-C	Building	201-C Process Building	Process Condensate	SO-1			
207-A		207-A					
207-B	Retention Basin	207-Bb/ Retention Basin	Cooling Water	BP-8			
207-S	Retention Basin	207-S	Cooling Water	RO-2			
207-SL	Retention Basin	207-SL	Lab Waste	RO-3			
207-T	Retention Basin	207-T Retention Basin	Cooling Water	TP-3			
207-Z	Retention Basin	207-Z Retention Basin	Steam Condensate	ZP-2			
209-E-WS-1.	French Drain	Critical Mass Laboratory Dry Well East	Miscellaneous Drainage	SO-1			
209-E-WS-2	French Drain	Critical Mass Laboratory Dry Well South	Miscellaneous Drainage	SO-1			
209-E-WS-3	Diversion Box	Critical Mass Laboratory Valve Pit	Process Waste	SO-1			
2101-M POND	Ponds	2101-M Pond	Lab Waste	SS-1			
216-A-1	Cribs	216-A-1	Process Waste	PO-5			
216-A-10	Cribs	216-A-10	Process Condensate	PO-2	0.312		0.107

Table B.6. (page 2 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
216-A-11	French Drain	216-A-11	Miscellaneous Drainage	PO-2			
216-A-12	French Drain	216-A-12	Miscellaneous Drainage	PO-2			
216-A-13	French Drain	216-A-13	Miscellaneous Drainage	PO-2			
216-A-14	French Drain	216-A-14	Miscellaneous Drainage	PO-2			
216-A-15	French Drain	216-A-15	Process Condensate	PO-2			
216-A-16	French Drain	216-A-16	Chemical Sewer	PO-5			
216-A-17	French Drain	216-A-17	Chemical Sewer	PO-5			
216-A-18	Trench	216-A-18	Process Waste	PO-5			
216-A-19	Trench	216-A-19	Process Waste	PO-5			
216-A-2	Cribs	216-A-2	Process Waste	PO-2			
216-A-20	Trench	216-A-20	Process Waste	PO-5			
216-A-21	Cribs	216-A-21	Lab Waste	PO-2			
216-A-22	French Drain	216-A-22	Miscellaneous Drainage	PO-2			
216-A-23A	French Drain	216-A-23A	Process Condensate	PO-5			
216-A-23B	French Drain	216-A-23B	Process Condensate	PO-5			
216-A-24	Cribs	216-A-24	Process Condensate	PO-5			
216-A-25	Ponds	216-A-25 Pond	Cooling Water	IU-6			
216-A-26	French Drain	216-A-26	Miscellaneous Drainage	PO-2			
216-A-26A	French Drain	216-A-26A	Miscellaneous Drainage	PO-2			
216-A-27	Cribs	216-A-27	Miscellaneous Drainage	PO-2			
216-A-28	Cribs	216-A-28	Process Condensate	PO-2			
216-A-29	Ditches	216-A-29	Chemical Sewer	BP-11			
216-A-3	Cribs	216-A-3	Process Waste	PO-2			
216-A-30	Cribs	216-A-30	Steam Condensate	PO-4	0.429	0.00315	
216-A-31	Cribs	216-A-31	Process Waste	PO-2			
216-A-32	Cribs	216-A-32	Miscellaneous Drainage	PO-2			
216-A-33	French Drain	216-A-33	Miscellaneous Drainage	PO-2			
216-A-34	Cribs	216-A-34	Process Condensate	PO-5			
216-A-35	French Drain	216-A-35	Miscellaneous Drainage	PO-2			
216-A-36A	Cribs	216-A-36A	Process Waste	PO-2			
216-A-36B	Cribs	216-A-36B	Process Waste	PO-2	1.99	0.000579	0.00842
216-A-37-1	Cribs	216-A-37-1	Process Condensate	PO-4	0.0919	0.00252	0.00426
216-A-37-2	Cribs	216-A-37-2	Steam Condensate	PO-4	0.196	0.00157	

Table B.6. (page 3 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
216-A-38-1	Cribs	216-A-38-1	N/A	PO-2			
216-A-39	Ditches	216-A-39	Miscellaneous Drainage	PO-3			
216-A-4	Cribs	216-A-4	Lab Waste	PO-2			
216-A-40	Retention Basin	216-A-40	Steam Condensate	PO-2			
216-A-41	Cribs	216-A-41	Miscellaneous Drainage	PO-2			
216-A-42	Retention Basin	216-A-42	Cooling Water	PO-4			
216-A-45	Cribs	216-A-45	Process Condensate	PO-2	0.0421	0.0000656	0.011
216-A-5	Cribs	216-A-5	Process Condensate	PO-2			
216-A-524	Diversion Box	216-A-524		PO-5			
216-A-6	Cribs	216-A-6	Steam Condensate	PO-4			
216-A-7	Cribs	216-A-7	Process Waste	PO-5			
216-A-8	Cribs	216-A-8	Process Condensate	PO-5			
216-A-9	Cribs	216-A-9	Cooling Water	PO-2			
216-B-10A	Cribs	216-B-10A Crib	Lab Waste	BP-6			
216-B-10B	Cribs	216-B-10B Crib	Lab Waste	BP-6			
216-B-11A&B	Reverse Well	216-B-11A&B Reverse Wells	Process Condensate	BP-4			
216-B-12	Cribs	216-B-12 Crib	Process Condensate	BP-9			
216-B-13	French Drain	216-B-13 French Drain	Miscellaneous Drainage	BP-6			
216-B-14	Cribs	216-B-14 Crib	Scavenged Waste	BP-2			
216-B-15	Cribs	216-B-15 Crib	Scavenged Waste	BP-2			
216-B-16	Cribs	216-B-16 Crib	Scavenged Waste	BP-2			
216-B-17	Cribs	216-B-17 Crib	Scavenged Waste	BP-2			
216-B-18	Cribs	216-B-18 Crib	Scavenged Waste	BP-2			
216-B-19	Cribs	216-B-19 Crib	Scavenged Waste	BP-2			
216-B-2-1	Ditches	216-B-2-1 Ditchb/	Cooling Water	BP-11			
216-B-2-2	Ditches	216-B-2-2 Ditchc/	Cooling Water	BP-11			
216-B-2-3	Ditches	216-B-2-3 Ditch	Cooling Water	BP-11			
216-B-20	Trench	216-B-20 Trench	Scavenged Waste	BP-2			
216-B-21	Trench	216-B-21 Trench	Scavenged Waste	BP-2			
216-B-22	Trench	216-B-22 Trench	Scavenged Waste	BP-2			
216-B-23	Trench	216-B-23 Trench	Scavenged Waste	BP-2			
216-B-24	Trench	216-B-24 Trench	Scavenged Waste	BP-2			
216-B-25	Trench	216-B-25 Trench	Scavenged Waste	BP-2			

Table B.6. (page 4 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
216-B-26	Trench	216-B-26 Trench	Scavenged Waste	BP-2			
216-B-27	Trench	216-B-27 Trench	Scavenged Waste	BP-2			
216-B-28	Trench	216-B-28 Trench	Scavenged Waste	BP-2			
216-B-29	Trench	216-B-29 Trench	Scavenged Waste	BP-2			
216-B-3	Ponds	216-B-3 Pond[a]	Cooling Water	BP-11			
216-B-3-1	Ditches	216-B-3-1 Ditch[b]	Cooling Water	BP-11			
216-B-3-2	Ditches	216-B-3-2 Ditch[c]	Cooling Water	BP-11			
216-B-3-3	Ditches	216-B-3-3 Ditch	Cooling Water	BP-11			
216-B-30	Trench	216-B-30 Trench	Scavenged Waste	BP-2			
216-B-32	Trench	216-B-32 Trench	Scavenged Waste	BP-2			
216-B-33	Trench	216-B-33 Trench	Scavenged Waste	BP-2			
216-B-34	Trench	216-B-34 Trench	Scavenged Waste	BP-2			
216-B-35	Trench	216-B-35 Trench	Tank Farm Waste	BP-3			
216-B-36	Trench	216-B-36 Trench	Tank Farm Waste	BP-3			
216-B-37	Trench	216-B-37 Trench	Process Waste	BP-3			
216-B-38	Trench	216-B-38 Trench	Tank Farm Waste	BP-3			
216-B-39	Trench	216-B-39 Trench	Tank Farm Waste	BP-3			
216-B-3A	Ponds	216-B-3A Pond	Cooling Water	BP-11			
216-B-3B	Ponds	216-B-3B Pond	Cooling Water	BP-11			
216-B-3C	Ponds	216-B-3C Pond	Cooling Water	BP-11			
216-B-4	Reverse Well	216-B-4 Reverse Well	Miscellaneous Drainage	BP-6			
216-B-40	Trench	216-B-40 Trench	Tank Farm Waste	BP-3			
216-B-41	Trench	216-B-41 Trench	Tank Farm Waste	BP-3			
216-B-42	Trench	216-B-42 Trench	Scavenged Waste	BP-3			
216-B-43	Cribs	216-B-43 Crib	Scavenged Waste	BP-1			
216-B-44	Cribs	216-B-44 Crib	Scavenged Waste	BP-1			
216-B-45	Cribs	216-B-45 Crib	Scavenged Waste	BP-1			
216-B-46	Cribs	216-B-46 Crib	Scavenged Waste	BP-1			
216-B-47	Cribs	216-B-47 Crib	Scavenged Waste	BP-1			
216-B-48	Cribs	216-B-48 Crib	Scavenged Waste	BP-1			
216-B-49	Cribs	216-B-49 Crib	Scavenged Waste	BP-1			
216-B-5	Reverse Well	216-B-5 Reverse Well	Process Waste	BP-6			
216-B-50	Cribs	216-B-50 Crib	Process Condensate	BP-1			

Table B.6. (page 5 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
216-B-51	French Drain	216-B-51 French Drain	Miscellaneous Drainage	BP-4			
216-B-52	Trench	216-B-52 Trench	Scavenged Waste	BP-2			
216-B-53A	Trench	216-B-53A Trench	Lab Waste	BP-2			
216-B-53B	Trench	216-B-53B Trench	Lab Waste	BP-2			
216-B-54	Trench	216-B-54 Trench	Lab Waste	BP-2			
216-B-55	Cribs	216-B-55 Crib	Steam Condensate	BP-9			
216-B-56	Cribs	216-B-56 Crib	N/A	BP-6			
216-B-57	Cribs	216-B-57 Crib	Process Condensate	BP-1			
216-B-58	Trench	216-B-58 Trench	Lab Waste	BP-2			
216-B-59	Retention Basin	216-B-59 Basin	Cooling Water	BP-6			
216-B-6	Reverse Well	216-B-6 Reverse Well	Lab Waste	BP-6			
216-B-60	Cribs	216-B-60 Crib	Decon Waste	BP-6			
216-B-61	Cribs	216-B-61 Crib	N/A	BP-1			
216-B-62	Cribs	216-B-62 Crib	Process Condensate	BP-9			
216-B-63	Ditches	216-B-63 Trench	Chemical Sewer	BP-11			
216-B-64	Retention Basin	216-B-64 Basin	N/A	BP-9			
216-B-7A&B	Cribs	216-B-7A&B Crib	Process Waste	BP-4			
216-B-8	Cribs	216-B-8TF Crib	Process Waste	BP-4			
216-B-9	Cribs	216-B-9TF Crib	Process Waste	BP-6			
216-C-1	Cribs	216-C-1 Crib	Process Condensate	SO-1			
216-C-10	Cribs	216-C-10 Crib	Process Condensate	SO-1			
216-C-2	Reverse Well	216-C-2 Reverse Well	Miscellaneous Drainage	SO-1			
216-C-3	Cribs	216-C-3 Crib	Process Waste	SO-1			
216-C-4	Cribs	216-C-4 Crib	Process Waste	SO-1			
216-C-5	Cribs	216-C-5 Crib	Process Waste	SO-1			
216-C-6	Cribs	216-C-6 Crib	Process Condensate	SO-1			
216-C-7	Cribs	216-C-7 Crib	Process Waste	SO-1			
216-C-8	French Drain	216-C-8	Process Waste	PO-3			
216-C-9	Ponds	216-C-9 Pond	Cooling Water	SO-1			
216-E-28	Ponds	216-E-28 Pond	N/A	BP-11			
216-N-1	Ponds	216-N-1 Pond	Cooling Water	NO-1			
216-N-2	Trench	216-N-2 Trench	Cooling Water	NO-1			
216-N-3	Trench	216-N-3 Trench	Cooling Water	NO-1			

Table B.6. (page 6 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
216-N-4	Ponds	216-N-4 Pond	Cooling Water	NO-1			
216-N-5	Trench	216-N-5 Trench	Cooling Water	NO-1			
216-N-6	Ponds	216-N-6 Pond	Cooling Water	NO-1			
216-N-7	Trench	216-N-7 Trench	Cooling Water	NO-1			
216-N-8	Ponds	216-N-8 Pond		IU-6			
216-S-1&2	Cribs	216-S-1 & 2	Process Condensate	RO-2			
216-S-10D	Ditches	216-S-10D	Chemical Sewer	RO-1			
216-S-10P	Ponds	216-S-10P	Chemical Sewer	RO-1			
216-S-11	Ponds	216-S-11	Chemical Sewer	RO-1			
216-S-12	Trench	216-S-12	Miscellaneous Drainage	RO-3			
216-S-13	Cribs	216-S-13	Process Waste	RO-2			
216-S-14	Trench	216-S-14	Process Waste	RO-3			
216-S-15	Ponds	216-S-15	Cooling Water	RO-2			
216-S-16D	Ditches	216-S-16D	Cooling Water	RO-1			
216-S-16P	Ponds	216-S-16P	Cooling Water	RO-1			
216-S-17	Ponds	216-S-17	Cooling Water	RO-1			
216-S-172	Diversion Box	216-S-172	Cooling Water	RO-1			
216-S-18	Trench	216-S-18	Debris	RO-2			
216-S-19	Ponds	216-S-19	Lab Waste	RO-1			
216-S-20	Cribs	216-S-20	Lab Waste	RO-3			
216-S-22	Cribs	216-S-22	Process Waste	RO-3			
216-S-23	Cribs	216-S-23	Process Condensate	RO-2			
216-S-25	Cribs	216-S-25	Steam Condensate	RO-1			
216-S-26	Cribs	216-S-26	Lab Waste	RO-3			
216-S-3	French Drain	216-S-3	Process Condensate	RO-2			
216-S-4	French Drain	216-S-4	Process Condensate	UP-2			
216-S-5	Cribs	216-S-5	Cooling Water	RO-1			
216-S-6	Cribs	216-S-6	Cooling Water	RO-1			
216-S-7	Cribs	216-S-7	Process Condensate	RO-2			
216-S-8	Trench	216-S-8	Process Waste	RO-2			
216-S-9	Cribs	216-S-9	Process Condensate	RO-2			
216-T-1	Ditches	216-T-1 Ditch	Cooling Water	TP-4			
216-T-10	Trench	216-T-10 Trench	Decon Waste	TP-4			

Table B.6. (page 7 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
216-T-11	Trench	216-T-11 Trench	Decon Waste	TP-4			
216-T-12	Trench	216-T-12 Trench	Cooling Water	TP-3			
216-T-13	Trench	216-T-13 Trench	Decon Waste	TP-2			
216-T-14	Trench	216-T-14 Trench	Tank Farm Waste	TP-3			
216-T-15	Trench	216-T-15 Trench	Tank Farm Waste	TP-3			
216-T-16	Trench	216-T-16 Trench	Tank Farm Waste	TP-3			
216-T-17	Trench	216-T-17 Trench	Tank Farm Waste	TP-3			
216-T-18	Cribs	216-T-18 Crib	Tank Farm Waste	TP-2			
216-T-19	Cribs	216-T-19TF Crib and Tile Field	Process Waste	TP-2			
216-T-2	Reverse Well	216-T-2 Reverse Well	Lab Waste	TP-4			
216-T-20	Trench	216-T-20 Trench	Process Waste	TP-2			
216-T-21	Trench	216-T-21 Trench	Tank Farm Waste	TP-1			
216-T-22	Trench	216-T-22 Trench	Tank Farm Waste	TP-1			
216-T-23	Trench	216-T-23 Trench	Tank Farm Waste	TP-1			
216-T-24	Trench	216-T-24 Trench	Tank Farm Waste	TP-1			
216-T-25	Trench	216-T-25 Trench	Process Waste	TP-1			
216-T-26	Cribs	216-T-26 Crib	Tank Farm Waste	TP-2			
216-T-27	Cribs	216-T-27 Crib	Lab Waste	TP-2			
216-T-28	Cribs	216-T-28 Crib	Decon Waste	TP-2			
216-T-29	Cribs	216-T-29 Crib	Miscellaneous Drainage	TP-4			
216-T-3	Reverse Well	216-T-3 Reverse Well	Process Waste	TP-4			
216-T-31	French Drain	216-T-31 French Drain	Miscellaneous Drainage	TP-2			
216-T-32	Cribs	216-T-32 Crib	Process Waste	TP-1			
216-T-33	Cribs	216-T-33 Crib	Decon Waste	TP-4			
216-T-34	Cribs	216-T-34 Crib	Lab Waste	TP-4			
216-T-35	Cribs	216-T-35 Crib	Lab Waste	TP-4			
216-T-36	Cribs	216-T-36 Crib	Steam Condensate	TP-1			
216-T-4-1D	Ditches	216-T-4-1D Ditch	Cooling Water	TP-3			
216-T-4-2	Ditches	216-T-4-2 Ditch	Steam Condensate	TP-3			
216-T-4A	Ponds	216-T-4A Pond	Cooling Water	TP-3			
216-T-4B	Ponds	216-T-4B Pond	Cooling Water	TP-3			
216-T-5	Trench	216-T-5 Trench	Tank Farm Waste	TP-1			
216-T-6	Cribs	216-T-6 Crib	Process Waste	TP-3			

Table B.6. (page 8 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
216-T-7	Cribs	216-T-7TF Crib and Tile Field	Tank Farm Waste	TP-1			
216-T-8	Cribs	216-T-8 Crib	Lab Waste	TP-4			
216-T-9	Trench	216-T-9 Trench	Decon Waste	TP-4			
216-U-1&2	Cribs	216-U-1 & 216-U-2	Process Condensate	UP-2			
216-U-10	Ponds	216-U-10	Cooling Water	UP-2			
216-U-11	Ditches	216-U-11	Cooling Water	UP-2			
216-U-12	Cribs	216-U-12	Process Condensate	UP-2			
216-U-13	Trench	216-U-13 (same as UN-200-W- 125)	Decon Waste	UP-2			
216-U-14	Ditches	216-U-14	Cooling Water	UP-2			
216-U-15	Trench	216-U-15	Process Waste	UP-2			
216-U-16	Cribs	216-U-16	Process Condensate	UP-2			
216-U-17	Cribs	216-U-17	Process Condensate	UP-2			
216-U-21		216-U-21					
216-U-3	French Drain	216-U-3	Miscellaneous Drainage	UP-2			
216-U-4	Reverse Well	216-U-4	Lab Waste	UP-2			
216-U-4A	French Drain	216-U-4A	Miscellaneous Drainage	UP-2			
216-U-4B	French Drain	216-U-4B	Miscellaneous Drainage	UP-2			
216-U-5	Trench	216-U-5 & 216-U-6	Process Waste	UP-2			
216-U-7	French Drain	216-U-7	Miscellaneous Drainage	UP-2			
216-U-8	Cribs	216-U-8	Process Condensate	UP-2			
216-U-9	Ditches	216-U-9	Cooling Water	RO-1			
216-W-LWC	Cribs	216-W-LWC Crib	Chemical Sewer	SS-2			
216-Z-1&2	Cribs	216-Z-1 & 216-Z-2 Cribs	Process Waste	ZP-2			
216-Z-10	Reverse Well	216-Z-10 Reverse Well	Process Waste	ZP-2			
216-Z-11	Ditches	216-Z-11	Cooling Water	UP-2			
216-Z-12	Cribs	216-Z-12 Crib	Process Waste	ZP-2			
216-Z-13	French Drain	216-Z-13 French Drain	Miscellaneous Drainage	ZP-2			
216-Z-14	French Drain	216-Z-14 French Drain	Miscellaneous Drainage	ZP-2			
216-Z-15	French Drain	216-Z-15 French Drain	Miscellaneous Drainage	ZP-2			
216-Z-16	Cribs	216-Z-16 Crib	Lab Waste	ZP-2			
216-Z-17	Ditches	216-Z-17 Trench	Lab Waste	ZP-2			
216-Z-18	Cribs	216-Z-18 Crib	Process Waste	ZP-2			
216-Z-19	Ditches	216-Z-19	Cooling Water	UP-2			

Table B.6. (page 9 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
216-Z-1A	Cribs	216-Z-1A Tile Field	Process Waste	ZP-2			
216-Z-1D	Ditches	216-Z-1D	Cooling Water	UP-2			
216-Z-20	Cribs	216-Z-20	Cooling Water	UP-2			
216-Z-21	Retention Basin	216-Z-21 Seepage Basin	Cooling Water	ZP-2			
216-Z-3	Cribs	216-Z-3 Crib	Process Waste	ZP-2			
216-Z-4	Trench	216-Z-4 Trench	Process Waste	ZP-2			
216-Z-5	Cribs	216-Z-5 Crib	Process Waste	ZP-2			
216-Z-6	Cribs	216-Z-6 Crib	Process Waste	ZP-2			
216-Z-7	Cribs	216-Z-7 Crib	Lab Waste	ZP-2			
216-Z-8	Cribs	216-Z-8 French Drain	Process Waste	ZP-2			
216-Z-9	Cribs	216-Z-9 Trench	Process Waste	ZP-2			
218-C-9	Burial Site	218-C-9 Burial Ground	LLW - SOLID	SO-1			
218-E-1	Burial Site	218-E-1	LLW - SOLID	PO-2			
218-E-10	Burial Site	218-E-10 Burial Ground	LLW - SOLID	BP-10			
218-E-12A	Burial Site	218-E-12A	LLW - SOLID	PO-6			
218-E-12B	Burial Site	218-E-12B	LLW - SOLID	PO-6			
218-E-13	Burial Site	218-E-13		PO-2			
218-E-2	Burial Site	218-E-2 Burial Ground	LLW - SOLID	BP-10			
218-E-2A	Burial Site	218-E-2A Burial Ground	LLW - SOLID	BP-10			
218-E-4	Burial Site	218-E-4 Burial Ground	LLW - SOLID	BP-10			
218-E-5	Burial Site	218-E-5 Burial Ground	LLW - SOLID	BP-10			
218-E-5A	Burial Site	218-E-5A Burial Ground	LLW - SOLID	BP-10			
218-E-6	Burial Site	218-E-6 Burial Ground	Debris	BP-6			
218-E-7	Burial Site	218-E-7 Burial Ground	Lab Waste	BP-6			
218-E-8	Burial Site	218-E-8	TRU Solid Waste	PO-6			
218-E-9	Burial Site	218-E-9 Burial Ground	LLW - SOLID	BP-10			
218-W-1	Burial Site	218-W-1 Burial Ground	TRU Solid Waste	ZP-3			
218-W-11	Burial Site	218-W-11 Burial Ground	LLW - SOLID	ZP-3			
218-W-1A	Burial Site	218-W-1A Burial Ground	LLW - SOLID	ZP-3			
218-W-2	Burial Site	218-W-2 Burial Ground	TRU Solid Waste	ZP-3			
218-W-2A	Burial Site	218-W-2A Burial Ground	LLW - SOLID	ZP-3			
218-W-3	Burial Site	218-W-3 Burial Ground	TRU Solid Waste	ZP-3			
218-W-3A	Burial Site	218-W-3A Burial Ground	TRU Solid Waste	ZP-3			

Table B.6. (page 10 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit*	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
218-W-3AE	Burial Site	218-W-3AE Burial Ground	LLW - SOLID	ZP-3			
218-W-4A	Burial Site	218-W-4A Burial Ground	TRU Solid Waste	ZP-3			
218-W-4B	Burial Site	218-W-4B Caissons	TRU Solid Waste	ZP-3			
218-W-4B	Burial Site	218-W-4B Trenches	TRU Solid Waste	ZP-3			
218-W-4C	Burial Site	218-W-4C Burial Ground	TRU Solid Waste	ZP-3			
218-W-5	Burial Site	218-W-5 Burial Ground	TRU Solid Waste	ZP-3			
218-W-6	Burial Site	218-W-6 Burial Ground	LLW - SOLID	ZP-3			
218-W-7	Burial Site	218-W-7	LLW - SOLID	RO-3			
218-W-8	Burial Site	218-W-8 Burial Ground	Lab Waste	TP-4			
218-W-9	Burial Site	218-W-9	LLW - SOLID	RO-2			
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2			
231-W-151	Vault	231-Z-151 Sump		ZP-2			
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2			
231-W-151	Vault	231-Z-151 Sump		ZP-2			
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2			
231-W-151	Vault	231-Z-151 Sump		ZP-2			
231-W-151	Diversion Box	231-Z-151 Sump		ZP-2			
231-W-151	Vault	231-Z-151 Sump		ZP-2			
232-Z	Building	232-Z Incinerator		ZP-2			
240-S-151	Diversion Box	240-S-151	LLW - SOLID	RO-3			
240-S-152	Diversion Box	240-S-152	Tank Farm Waste	RO-3			
240-S-302	Tanks	240-S-302	Lab Waste	RO-3			
241-A-151	Diversion Box	241-A-151	Tank Farm Waste	PO-2			
241-A-152	Diversion Box	241-A-152	Process Waste	PO-3			
241-A-153	Diversion Box	241-A-153	Process Waste	PO-3			
241-A-302A	Tanks	241-A-302A		PO-2			
241-A-302B	Tanks	241-A-302B		PO-5			
241-A-350	Tanks	241-A-350	Process Waste	PO-3			
241-A-417	Tanks	241-A-417	Process Waste	PO-3			
241-A-A	Diversion Box	241-A-A	Process Waste	PO-3			
241-A-B	Diversion Box	241-A-B	Process Waste	PO-3			
241-AN-A	Diversion Box	241-AN-A	Process Waste	PO-3			
241-AN-B	Diversion Box	241-AN-B	Process Waste	PO-3			

Table B.6. (page 11 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
241-AP VP	Valve Pit	241-AP	Process Waste	PO-3			
241-AR-151	Diversion Box	241-AR-151	Process Waste	PO-3			
241-AW-A	Diversion Box	241-AW-A	Process Waste	PO-3			
241-AW-B	Diversion Box	241-AW-B	Process Waste	PO-3			
241-AX-151	Diversion Box	241-AX-151	Process Waste	PO-3			
241-AX-152DS	Tanks	241-AX-152DS	Process Waste	PO-3			
241-AX-155	Diversion Box	241-AX-155	Tank Farm Waste	PO-3			
241-AX-501	Valve Pit	241-AX-501		PO-3			
241-AX-A	Diversion Box	241-AX-A		PO-3			
241-AX-B	Diversion Box	241-AX-B		PO-3			
241-AY-151	Diversion Box	241-AY-151	Process Waste	PO-3			
241-AY-152	Diversion Box	241-AY-152	Process Waste	PO-3			
241-AZ-151DS	Diversion Box	241-AZ-151DS		PO-3			
241-AZ-152	Diversion Box	241-AZ-152		PO-3			
241-C-151	Diversion Box	241-C-151		PO-3			
241-C-152	Diversion Box	241-C-152		PO-3			
241-C-153	Diversion Box	241-C-153		PO-3			
241-C-154	Diversion Box	241-C-154 Diversion Box	Process Waste	SO-1			
241-C-252	Diversion Box	241-C-252		PO-3			
241-C-301C	Tanks	241-C-301C		PO-3			
241-CR-151	Diversion Box	241-CR-151		PO-3			
241-CR-152	Diversion Box	241-CR-152		PO-3			
241-CR-153	Diversion Box	241-CR-153		PO-3			
241-CX-TK-70	Tanks	241-CX-70 Storage Tank	Tank Farm Waste	SO-1			
241-CX-TK-71	Tanks	241-CX-71 Storage Tank	Process Condensate	SO-1			
241-CX-TK-72	Tanks	241-CX-72 Storage Tank	Process Waste	SO-1			
241-ER-153	Diversion Box	241-ER-153		PO-3			
241-S-151	Diversion Box	241-S-151	LLW - SOLID	RO-2			
241-S-152	Diversion Box	241-S-152	Tank Farm Waste	RO-4			
241-S-302A	Tanks	241-S-302A	Lab Waste	RO-2			
241-S-302B	Tanks	241-S-302B	LLW - SOLID	RO-4			
241-S-A	Diversion Box	241-S-A	Tank Farm Waste	RO-4			
241-S-B	Diversion Box	241-S-B	Tank Farm Waste	RO-4			

Table B.6. (page 12 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
241-S-C	Diversion Box	241-S-C	Tank Farm Waste	RO-4			
241-S-D	Diversion Box	241-S-D	Tank Farm Waste	RO-4			
241-SX-151	Diversion Box	241-SX-151	Tank Farm Waste	RO-4			
241-SX-152	Diversion Box	241-SX-152	Tank Farm Waste	RO-4			
241-SX-302	Tanks	241-SX-302		RO-2			
241-SX-A	Diversion Box	241-SX-A		RO-4			
241-SX-B	Diversion Box	241-SX-B		RO-4			
241-SY-A	Diversion Box	241-SY-A		RO-4			
241-SY-A	Diversion Box	241-SY-A		RO-4			
241-SY-B	Diversion Box	241-SY-B		RO-4			
241-SY-B	Diversion Box	241-SY-B		RO-4			
241-T-151	Diversion Box	241-T-151 Diversion Box	Tank Farm Waste	TP-6			
241-T-152	Diversion Box	241-T-152 Diversion Box	Tank Farm Waste	TP-6			
241-T-153	Diversion Box	241-T-153 Diversion Box	Tank Farm Waste	TP-6			
241-T-252	Diversion Box	241-T-252 Diversion Box	Tank Farm Waste	TP-6			
241-T-301	Tanks	241-T-301 Catch Tank	Tank Farm Waste	TP-6			
241-T-302	Tanks	241-T-302 Catch Tank	Tank Farm Waste	TP-6			
241-T-361	Tanks	241-T-361 Settling Tank	Process Waste	TP-4			
241-TR-152	Diversion Box	241-TR-152 Diversion Box	Tank Farm Waste	TP-6			
241-TR-153	Diversion Box	241-TR-153 Diversion Box	Tank Farm Waste	TP-6			
241-TX-152	Diversion Box	241-TX-152 Diversion Box	Tank Farm Waste	TP-2			
241-TX-153	Diversion Box	241-TX-153 Diversion Box	Tank Farm Waste	TP-5			
241-TX-154	Diversion Box	241-TX-154 Diversion Box	Tank Farm Waste	TP-4			
241-TX-155	Diversion Box	241-TX-155 Diversion Box	Tank Farm Waste	TP-2			
241-TX-302A	Tanks	241-TX-302A Catch Tank	Tank Farm Waste	TP-5			
241-TX-302B	Tanks	241-TX-302B Catch Tank	Tank Farm Waste	TP-2			
241-TX-302C	Tanks	241-TX-302C Catch Tank	Tank Farm Waste	TP-4			
241-TXR-151	Diversion Box	241-TXR-151 Diversion Box	Tank Farm Waste	TP-5			
241-TXR-152	Diversion Box	241-TXR-152 Diversion Box	Tank Farm Waste	TP-5			
241-TXR-153	Diversion Box	241-TXR-153 Diversion Box	Tank Farm Waste	TP-5			
241-TY-153	Diversion Box	241-TY-153 Diversion Box	Tank Farm Waste	TP-5			
241-TY-302A	Tanks	241-TY-302A Catch Tank	Tank Farm Waste	TP-5			
241-TY-302B	Tanks	241-TY-302B Catch Tank	Tank Farm Waste	TP-5			

Table B.6. (page 13 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
241-Z-361	Tanks	241-Z-361 Settling Tank	Process Waste	ZP-2			
241-Z-TK-8	Tanks	216-Z-8 Settling Tank	Process Waste	ZP-2			
241-Z-TK-D5	Tanks	241-Z Treatment Tank	Process Waste	ZP-2			
242-T-151	Diversion Box	242-T-151 Diversion Box	Process Condensate	TP-5			
244-A RT	Tanks	244-A	Process Waste	PO-3			
244-AR VAULT	Vault	244-AR	Process Waste	PO-3			
244-CR VAULT	Vault	244-CR	Process Waste	PO-3			
244-S RT	Tanks	244-S Receiver Tank		RO-2			
244-TX RT	Tanks	244-TX Receiving Tank		TP-5			
244-TXR	Vault	244-TXR Vault	Tank Farm Waste	TP-5			
2607-E5	Septic System	2607-E-5 Septic Tank and Drain Field	Sanitary Waste	SO-1			
2607-E6	Septic System	2607-E6	Sanitary Waste	PO-2			
2607-E7A	Septic System	2607-E-7A Septic Tank and Drain Field	Sanitary Waste	SO-1			
2607-EA	Septic System	2607-EA	Sanitary Waste	PO-2			
2607-EC	Septic System	2607-EC	Sanitary Waste	PO-5			
2607-ED	Septic System	2607-ED	Sanitary Waste	PO-3			
2607-EE	Septic System	2607-EL	Sanitary Waste	PO-2			
2607-EG	Septic System	2607-EG	Sanitary Waste	PO-3			
2607-EJ	Septic System	2607-EJ	Sanitary Waste	PO-3			
2607-N	Septic System	2607-N Septic Tank/Drain Field	Sanitary Waste	NO-1			
2607-P	Septic System	2607-P Septic Tank/Drain Field	Sanitary Waste	NO-1			
2607-R	Septic System	2607-R Septic Tank/Drain Field	Sanitary Waste	NO-1			
2607-W1	Septic System	2607-W1 Septic Tank	Sanitary Waste	SS-2			
2607-W2	Septic System	2607-W2 Septic Tank	Sanitary Waste	SS-2			
2607-W3	Septic System	2607-W3 Septic Tank	Sanitary Waste	TP-4			
2607-W4	Septic System	2607-W4 Septic Tank	Sanitary Waste	TP-4			
2607-W6	Septic System	2607-W6	Sanitary Waste	RO-3			
2607-W8	Septic System	2607-W-8 Septic Tank and Drain Field	Sanitary Waste	ZP-2			
2607-WA	Septic System	2607-WA Septic Tank and Drain Field	Sanitary Waste	ZP-2			
2607-WB		2607-WB Septic Tank and Drain Field					
2607-WT	Septic System	2607-WT Septic Tank	Sanitary Waste	TP-5			
2607-WTX	Septic System	2607-WTX Septic Tank	Sanitary Waste	TP-5			
2607-WZ	Septic System	2607-WZ	Sanitary Waste	RO-1			

Table B.6. (page 14 of 14)

Site Code	Waste Management Unit Type	Waste Management Unit	Waste Type	Operable Unit	Promethium-147, Units (Ci)	(Tin) Sn-113, Units (Ci)	Iodine-129, Units (Ci)
2607-Z	Septic System	2607-Z Septic Tank and Drain Field	Sanitary Waste	ZP-2			
2607-Z-1		2607-Z-1 Septic Tank and Drain Field					
2704-C-WS-1	French Drain	2704-C-WS-1, 2704-C French Drain, Gatehouse French Drain	Miscellaneous Drainage	SO-1			
2904-S-160	Diversion Box	2904-S-160	Cooling Water	RO-1			
2904-S-170	Diversion Box	2904-S-170	Process Waste	RO-1			
2904-S-171	Diversion Box	2904-S-171	Cooling Water	RO-1			
291-C	Building	291-C Ventilation System	Process Condensate	SO-1			
299-E24-111	Reverse Well	299-E24-111		PO-2			
HSVP	Diversion Box	Semi-Works Valve Pit	Process Waste	SO-1			
UPR-200-E-141		UN-200-E-141	Solution Storage (1)	SO-1			
UPR-200-E-36		UN-200-E-36	Process & Decon Wastes (1)	SO-1			
UPR-200-E-37		UN-200-E-37	Process & Decon Wastes (1)	SO-1			
UPR-200-E-98		UN-200-E-98	Process & Decon Wastes (3)	SO-1			
UPR-200-W-160		UPR-200-W-160 Unplanned Release	Tank Farm Waste	TP-4			
Z PLANT BP	Burial Site	Z Plant Burn Pit	Debris	ZP-3			

Appendix C

Environmental Restoration Sites Without Inventories

Appendix C

Environmental Restoration Sites Without Inventories

Appendix C is a spreadsheet obtained from the Environmental Restoration Contractor (ERC), Bechtel Hanford, Inc. This appendix consists of one table that lists 363 environmental restoration sites on the 200 Area Plateau at Hanford for which inventories have not been assigned. The table presents the following available information for each site listed: waste volume, waste description, type of waste (radioactive, chemical, or mixed), and an evaluation of whether the release constitutes a potentially significant source.

Table C.1. Environmental Restoration Sites Without Inventories (page 1 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
200-E PD					
200-E-14			Process Effluent		
200-E-18					
200-E-26			Contaminated Soil	Chemical	
200-E-4			Steam Condensate	Radioactive	
200-E-8			Oil	Chemical	
200-W CSLA					
200-W PP					
200-W-16			Storage Tank	Mixed	
200-W-7			Equipment	Radioactive	
200-W-9					
207-A-NORTH					
207-A-SOUTH					
207-B					
207-S			Process Effluent	Mixed	
207-SL			Process Effluent	Mixed	
207-T			Steam Condensate	Radioactive	
207-Z			Steam Condensate		
209-E-WS-1			Steam Condensate	Radioactive	
209-E-WS-2			Steam Condensate	Radioactive	
209-E-WS-3			Process Effluent	Radioactive	
2101-M POND				Nonradioactive	
212-N to 216-N-1 Pipeline					
212-P to 216-N-4 Pipeline					
212-R to 216-N-6 Pipeline					
216-A-11		100	Steam Condensate	Mixed	
216-A-12		100	Steam Condensate	Mixed	
216-A-13		100	Water	Mixed	
216-A-14		1	Steam Condensate	Mixed	
216-A-15		10000	Process Effluent	Mixed	
216-A-22		10			
216-A-26					
216-A-26A		1			
216-A-29		10400312			

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
216-A-32		4			
216-A-33			Cooling water		
216-A-34					
216-A-35		10			
216-A-38-1					
216-A-40		946	Steam Condensate	Radioactive	
216-A-41		10			
216-A-42					
216-A-524			Process Effluent	Radioactive	
216-B-13		21	Process Effluent	Mixed	
216-B-3-1		149000000			
216-B-3-2		149000000			
216-B-3-3					
216-B-3A					
216-B-3B					
216-B-3C					
216-B-4		10			
216-B-51		1	Process Effluent	Mixed	
216-B-56					
216-B-6		6000			
216-B-61			Steam Condensate		
216-B-64					
216-BY-201					
216-C-2					
216-C-9 Pond Diversion Box					
216-E-28					
216-N-1		946000			
216-N-8					
216-S-10P					
216-S-14			Process Effluent	Mixed	
216-S-16D		400000	Process Effluent	Mixed	
216-S-172			Process Effluent	Mixed	
216-S-18			Chemicals	Mixed	
216-S-4		1000	Process Effluent	Mixed	

Table C.1. (page 3 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
216-T-10			Process Effluent	Nondangerous/nonradioactive	
216-T-11			Process Effluent	Nondangerous/nonradioactive	
216-T-13			Process Effluent	Mixed	
216-T-2		6000	Process Effluent	Mixed	
216-T-29		74	Steam Condensate	Mixed	
216-T-30					
216-T-31			Steam Condensate	Radioactive	
216-T-4-1D			Steam Condensate	Mixed	
216-T-4-2			Steam Condensate	Radioactive	
216-T-4A		42500000	Steam Condensate	Mixed	
216-T-9			Process Effluent	Nondangerous/nonradioactive	
216-TY-201					
216-U-11			Process Effluent	Mixed	
216-U-14					
216-U-4			Process Effluent	Mixed	
216-U-9					
216-W-LWC		1200000	Water	Radioactive	
216-Z-11			Process Effluent	Radioactive	
216-Z-13			Steam Condensate	Nondangerous/nonradioactive	
216-Z-14			Steam Condensate	Radioactive	
216-Z-15			Process Effluent	Mixed	
216-Z-19			Process Effluent	Radioactive	
216-Z-1D		1000	Process Effluent	Radioactive	
216-Z-21			Steam Condensate		
218-E-2A					
224-B			Chemicals	Mixed	
231-W-151					
231-Z-151 Sump					
240-S-151			Chemicals	Mixed	
240-S-152			Chemicals	Mixed	
240-S-302			Storage Tank	Mixed	
241-A-151			Process Effluent	Radioactive	
241-A-152CT Catch Tank			Chemicals	Mixed	
241-A-302A			Process Effluent	Radioactive	

Table C.1. (page 4 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
241-A-302B			Process Effluent	Radioactive	
241-B-154			Chemicals	Mixed	
241-B-302B			Chemicals	Mixed	
241-BX-154			Chemicals	Mixed	
241-BX-155			Chemicals	Mixed	
241-BX-302B			Chemicals	Mixed	
241-BX-302C			Chemicals	Mixed	
241-C Waste Line Unplanned Release No. 1					
241-C Waste Line Unplanned Release No. 2					
241-C-154			Chemicals	Mixed	
241-C-302A Catch Tank					
241-CX-TK-70					
241-ER-151			Chemicals	Mixed	
241-ER-152			Chemicals	Mixed	
241-ER-311			Chemicals	Mixed	
241-ER-311A					
241-S-151			Chemicals	Mixed	
241-S-302A			Storage Tank	Mixed	
241-SX-302			Process Effluent	Mixed	
241-TX-152			Chemicals	Mixed	
241-TX-154			Chemicals	Mixed	
241-TX-155			Chemicals	Mixed	
241-TX-302B			Storage Tank	Mixed	
241-TX-302BR			Storage Tank	Mixed	
241-TX-302C			Chemicals	Mixed	
241-U-151			Chemicals	Mixed	
241-U-152			Chemicals	Mixed	
241-U-302					
241-UX-154			Chemicals	Mixed	
241-WR VAULT			Process Effluent	Mixed	
241-Z Diversion Box No. 1					
241-Z Diversion Box No. 2					
241-Z-TK-D5					

Table C.1. (page 5 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
244-S RT			Storage Tank	Mixed	
2704-C-WS-1					
270-E CNT			Chemicals	Mixed	
270-W					
2718-E-WS-1					
276-S-TK-141			Chemicals	Mixed	
276-S-TK-142			Chemicals	Mixed	
2904-S-160			Process Effluent	Radioactive	
2904-S-170			Process Effluent	Radioactive	
2904-S-171			Process Effluent	Radioactive	
291-C-1			Demolition and Inert Waste	Radioactive	
299-E24-111				Radioactive	
600 NSTFUT			Sanitary Sewage	Nondangerous/nonradioactive	
600-25					
CTFN 2703-E			Process Effluent	Nondangerous/nonradioactive	
HSVP			Chemicals	Mixed	
UN-200-E-161					
UPR-200-E-1			Process Effluent	Radioactive	from 221-B, 1946
UPR-200-E-10					Remediated
UPR-200-E-103			Process Effluent	Mixed	Pipe leak
UPR-200-E-11					Remediated
UPR-200-E-110			Process Effluent	Mixed	contamination
UPR-200-E-112			Chemicals	Mixed	train wheel, remediated
UPR-200-E-114			Process Effluent	Radioactive	Assume remediated
UPR-200-E-117			Process Effluent	Radioactive	volume or inventory
UPR-200-E-12					
UPR-200-E-13					remediated
UPR-200-E-138			Process Effluent	Mixed	released in 1970
UPR-200-E-14			Process Effluent	Unknown	see Appendix B
UPR-200-E-140			Chemicals	Chemical	
UPR-200-E-141			Chemicals	Mixed	
UPR-200-E-142			Oil	Chemical	
UPR-200-E-143			Animal Waste	Radioactive	Remediated
UPR-200-E-144					

Table C.1. (page 6 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
UPR-200-E-145			Soil	Radioactive	
UPR-200-E-15			Process Effluent	Radioactive	crib
UPR-200-E-17			Process Effluent	Radioactive	crib
UPR-200-E-18			Process Effluent	Mixed	
UPR-200-E-19			Process Effluent	Radioactive	Minor ground contamination
UPR-200-E-2			Process Effluent	Mixed	particle releases
UPR-200-E-20			Process Effluent	Radioactive	Minor spot contamination
UPR-200-E-21					assume minor
UPR-200-E-22			Chemical Release		
UPR-200-E-24					
UPR-200-E-25					from steam release
UPR-200-E-26					Remediated
UPR-200-E-28					steam
UPR-200-E-29			Process Effluent	Radioactive	
UPR-200-E-3	[redacted]		Process Effluent	Radioactive	154 tank, no volume
UPR-200-E-30			Barrels/Drums/Buckets/Cans	Radioactive	
UPR-200-E-31	[redacted]		Process Effluent	Radioactive	Unknown volume or inventory
UPR-200-E-32			Process Effluent	Radioactive	see Appendix B
UPR-200-E-33			Barrels/Drums/Buckets/Cans	Radioactive	Minor spot contamination
UPR-200-E-34			Process Effluent	Radioactive	see Appendix B
UPR-200-E-35					Appendix B
UPR-200-E-36			Process Effluent	Radioactive	
UPR-200-E-37			Process Effluent	Radioactive	
UPR-200-E-39			Process Effluent	Mixed	Minor vapor release
UPR-200-E-40			Process Effluent	Radioactive	Minor spot contamination
UPR-200-E-41			Process Effluent	Radioactive	Remediated
UPR-200-E-42			Process Effluent	Radioactive	
UPR-200-E-44			Process Effluent	Radioactive	line leak
UPR-200-E-45			Process Effluent	Radioactive	Remediated
UPR-200-E-49			Process Effluent	Radioactive	
UPR-200-E-50			Equipment	Radioactive	
UPR-200-E-51					216-B-3 Pond, see Appendix B
UPR-200-E-52					probably <100 L
UPR-200-E-53					

Table C.1. (page 7 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
UPR-200-E-54					Remediated
UPR-200-E-55					Remediated
UPR-200-E-56					
UPR-200-E-58					
UPR-200-E-59					
UPR-200-E-60					
UPR-200-E-61			Process Effluent	Radioactive	Remediated
UPR-200-E-62			Process Effluent	Radioactive	
UPR-200-E-63			Vegetation	Radioactive	removed
UPR-200-E-64					Minor specks
UPR-200-E-65			Process Effluent	Radioactive	
UPR-200-E-66			Process Effluent	Radioactive	
UPR-200-E-67			Misc. Trash and Debris	Radioactive	
UPR-200-E-69			Barrels/Drums/Buckets/Cans	Radioactive	from equipment on flat car
UPR-200-E-7			Process Effluent	Radioactive	waste, no volume information
UPR-200-E-77			Process Effluent	Radioactive	1 Ci process effluent leak
UPR-200-E-78			Process Effluent	Radioactive	volume information
UPR-200-E-80			Process Effluent	Mixed	volume information
UPR-200-E-83			Animal Waste	Radioactive	cotaminated rabbit and coyote
UPR-200-E-84			Process Effluent	Mixed	leak
UPR-200-E-85			Process Effluent	Mixed	effluent
UPR-200-E-87			Process Effluent	Mixed	contaminated 4-m2 area
UPR-200-E-88			Process Effluent	Radioactive	
UPR-200-E-89			Process Effluent	Mixed	from BY tank farm, minor
UPR-200-E-9			Process Effluent	Mixed	Remediated
UPR-200-E-90			Process Effluent	Radioactive	No release
UPR-200-E-92			Chemicals	Mixed	tumbleweeds
UPR-200-E-93			Chemicals	Mixed	from tumbleweeds
UPR-200-E-95			Chemicals	Mixed	contamination
UPR-200-E-96			Process Effluent	Radioactive	
UPR-200-E-97			Soil	Unknown	
UPR-200-E-98			Chemicals	Mixed	
UPR-200-N-1					contamination
UPR-200-N-2					remediated

Table C.1. (page 8 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
UPR-200-W-101			Chemicals	Mixed	
UPR-200-W-102			Chemicals	Mixed	
UPR-200-W-103			Process Effluent	Mixed	
UPR-200-W-104			Process Effluent	Mixed	
UPR-200-W-105			Process Effluent	Mixed	
UPR-200-W-106			Process Effluent	Mixed	
UPR-200-W-107			Process Effluent	Mixed	
UPR-200-W-108			Process Effluent	Mixed	
UPR-200-W-109			Process Effluent	Mixed	
UPR-200-W-11				Radioactive	
UPR-200-W-110			Chemicals	Mixed	
UPR-200-W-111			Sludge	Mixed	
UPR-200-W-112			Sludge	Mixed	
UPR-200-W-113					
UPR-200-W-114			Chemicals	Mixed	
UPR-200-W-115					
UPR-200-W-116			Chemicals	Mixed	
UPR-200-W-117			Chemicals	Mixed	
UPR-200-W-118			Chemicals	Mixed	
UPR-200-W-123			Chemicals	Mixed	
UPR-200-W-124			Process Effluent	Mixed	
UPR-200-W-125					
UPR-200-W-127			Process Effluent	Mixed	
UPR-200-W-13			Steam Condensate	Radioactive	
UPR-200-W-130			Process Effluent	Mixed	
UPR-200-W-131					
UPR-200-W-132			Process Effluent	Mixed	
UPR-200-W-134			Barrels/Drums/Buckets/Cans	Mixed	
UPR-200-W-135			Process Effluent	Mixed	
UPR-200-W-137					
UPR-200-W-138			Process Effluent	Mixed	
UPR-200-W-139					
UPR-200-W-14			Steam Condensate	Unknown	
UPR-200-W-140			Process Effluent	Mixed	

Table C.1. (page 9 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
UPR-200-W-141			Process Effluent	Mixed	
UPR-200-W-15			Steam Condensate	Radioactive	
UPR-200-W-158					
UPR-200-W-159			Chemicals	Mixed	
UPR-200-W-16					
UPR-200-W-160			Process Effluent	Mixed	
UPR-200-W-161			Chemicals	Mixed	
UPR-200-W-162			Chemicals	Mixed	
UPR-200-W-163			Process Effluent	Mixed	
UPR-200-W-164			Process Effluent	Radioactive	
UPR-200-W-165			Chemicals	Mixed	
UPR-200-W-166			Soil	Radioactive	
UPR-200-W-167			Chemicals	Mixed	
UPR-200-W-18					
UPR-200-W-19					
UPR-200-W-2					
UPR-200-W-20					
UPR-200-W-21					
UPR-200-W-23					
UPR-200-W-26					
UPR-200-W-27					
UPR-200-W-28					
UPR-200-W-29			Process Effluent	Radioactive	
UPR-200-W-3				Radioactive	
UPR-200-W-30			Process Effluent	Unknown	
UPR-200-W-32			Chemical Release	Radioactive	
UPR-200-W-33					
UPR-200-W-34			Process Effluent	Unknown	
UPR-200-W-35			Chemical Release		
UPR-200-W-36			Process Effluent	Radioactive	
UPR-200-W-37			Miscellaneous Trash and Debris	Radioactive	
UPR-200-W-38			Process Effluent	Radioactive	
UPR-200-W-39				Unknown	
UPR-200-W-4					

Table C.1. (page 10 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
UPR-200-W-40			Process Effluent	Mixed	
UPR-200-W-41					
UPR-200-W-42					
UPR-200-W-43					
UPR-200-W-44					
UPR-200-W-45					
UPR-200-W-46			Equipment		
UPR-200-W-47					
UPR-200-W-48					
UPR-200-W-49					
UPR-200-W-5					
UPR-200-W-50					
UPR-200-W-51					
UPR-200-W-52					
UPR-200-W-53			Equipment	Radioactive	
UPR-200-W-55					
UPR-200-W-56					
UPR-200-W-57					
UPR-200-W-58					
UPR-200-W-59					
UPR-200-W-60					
UPR-200-W-61					
UPR-200-W-63					
UPR-200-W-65					
UPR-200-W-67					
UPR-200-W-68					
UPR-200-W-69					
UPR-200-W-7					
UPR-200-W-70					
UPR-200-W-72					
UPR-200-W-73					
UPR-200-W-74					
UPR-200-W-75					
UPR-200-W-77					

Table C.1. (page 11 of 11)

Site Code	Significant	Volume (m^3)	Waste Description	Waste Type	Notes
UPR-200-W-78					
UPR-200-W-79					
UPR-200-W-8			Miscellaneous Trash and Debris	Radioactive	
UPR-200-W-82			Chemicals	Mixed	
UPR-200-W-83			Chemicals	Mixed	
UPR-200-W-84			Chemicals	Mixed	
UPR-200-W-85			Process Effluent	Mixed	
UPR-200-W-86					
UPR-200-W-87			Chemicals	Mixed	
UPR-200-W-88			Chemicals	Mixed	
UPR-200-W-89			Chemicals	Mixed	
UPR-200-W-90			Chemicals	Mixed	
UPR-200-W-91			Chemicals	Mixed	
UPR-200-W-95			Process Effluent	Mixed	
UPR-200-W-96			Process Effluent	Mixed	
UPR-200-W-98			Process Effluent	Mixed	
UPR-200-W-99			Chemicals	Radioactive	
UPR-216-W-25					
UPR-600-12			Chemicals	Mixed	Remediated
UPR-600-16					Remediated
UPR-600-18					Will be removed
UPR-600-19					Nonradioactive
UPR-600-20			Process Effluent	Mixed	Minor specks
UPR-600-21					from tumbleweeds
WBL			Miscellaneous Trash and Debris	Chemical	

Appendix D

Hanford Composite Analysis Source Term Release Models

Appendix D

Hanford Composite Analysis Source-Term Release Models

G. P. Streile

D.1 Types of Contaminant Sources, and Source Zone Attributes

There are many different types of contaminant sources at Hanford that release, or could release, contamination to the vadose zone. Consequently, many different types of quantitative release models could be required to perform a detailed release analysis for every type of source zone. However, for the scope of this effort, only five idealized generic types of contaminant source zones (i.e., generic waste form types) were considered for the conceptual model of release: soil-debris, cake waste, glass waste, cement waste, and reactor block waste. Each source zone at the Hanford Site was characterized in terms of its generic waste form type, contaminant inventories, volume, and horizontal cross-sectional area. Only radionuclide contaminants are considered in the present analyses.

D.1.1 Soil-Debris Waste Form Type

The first generic waste form type consists of unconsolidated wastes mixed with soil material, and is referred to as the "soil-debris" type of waste form. Source zones composed of this waste form type are permeable to percolating water; and thus all surfaces of the waste come into contact with the percolating water as it passes through the zone in a manner similar to how infiltrating water passes through natural vadose zone material. If contaminant inventories in the source zone are high enough, leaching of contaminant out of the bottom of the source zone is controlled by the solubility of the contaminant in the percolating water. Otherwise, the leaching is controlled by partitioning of the contaminant between aqueous and sorbed phases. Unconsolidated wastes in this waste form type could be further subdivided into those having either high or low surface-area-to-volume (S/V) ratios. Contaminants from wastes in the low S/V category (e.g., waste containers, personal protection equipment, and metal process equipment) readily leach into the surrounding soil; and therefore their release from the source zone is controlled by the properties of the surrounding soil in the source zone. Contaminants from wastes in the high S/V category (e.g., sludge, soil, and spent filters/ adsorbents) can have high surface adsorption coefficients. Therefore, their release from the source zone is controlled by the properties of the waste material itself. However, because the availability of physical and chemical data regarding these wastes is limited, and because of the scope of this effort, it has been assumed that the properties of the surrounding soil can also be used to calculate release in this case. (This assumption is understood to be conservative because the sorptive properties of the surrounding soil would be lower than that of the waste material.)

D.1.2 Cake Waste Form Type

The second generic waste form type consists of consolidated waste that is permeable to water, and that dissolves over time because some major structural component of the solid waste dissolves in the water percolating through the waste form. Tank waste consisting of salt cake and sludge is a waste form of this type. This is referred to as the "cake" type of waste form. As the solid waste form dissolves (at a constant rate controlled by the aqueous solubility of the major structural component)

all of the contaminants associated with the portion of the waste form that dissolved are released into the percolating water congruently at constant rates related to their concentration in the waste form.

D.1.3 Glass Waste Form Type

The third generic waste form type consists of solidified wastes whose permeability is much lower than that of the surrounding soil, and is also so low that contaminant mobility within the waste form is essentially zero. This is referred to as the "glass" type of waste form. It is assumed that this waste form is composed of pieces of "glass" that are roughly cubical in shape, and that only the "glass" surfaces are exposed to water percolating through the source zone. Furthermore, the waste form is assumed to be slowly dissolving (from the exterior surfaces of the "glass") with time; i.e., over time the pieces of "glass" are slowly shrinking in size. The overall rate of dissolution of the waste form changes over time because the surface area of the waste form (exposed to the percolating water) changes as the pieces of "glass" shrink. All of the contaminants associated with the portion of the waste form that dissolved are released into the percolating water congruently at rates related to their concentration in the waste form and the overall waste form dissolution rate at the given time.

D.1.4 Cement Waste Form Type

The fourth generic waste form type consists of solidified wastes whose permeability is much lower than that of the surrounding soil (i.e., low enough that advective water flow within the waste form is essentially zero), but is sufficiently high to allow some contaminant mobility within the waste form. This is referred to as the "cement" type of waste form. Percolating water tends to move around this type of waste form, and contaminants are only leached from the waste form's outer surface. As this occurs, contaminants inside the waste form are assumed to diffuse toward the outer surface. Therefore, overall contaminant release from the source zone is assumed to be controlled by the contaminant's effective diffusion coefficient in the waste form.

D.1.5 Reactor Block Waste Form Type

The fifth generic waste form type consists of irradiated solids that release contaminants into the water percolating past them via unspecified loss processes from the solid matrix as well as via corrosion of the solid components themselves over time. This is referred to as the "reactor block" type of waste form. Because of the lack of information regarding the conceptual and mathematical description of the actual processes occurring, release of contaminants is assumed to be described by rates calculated from experimental leach test data.

D.1.6 Assumptions Made About Waste Form Types

Contaminants released from cake, glass, cement, and reactor block waste form types may initially enter some kind of soil material surrounding them, if these waste forms are present in a larger, overall source zone that also contains soil. It is possible that the ultimate release from the overall source zone could be limited by the release from this surrounding soil. However, the analyses now assume that the release from the waste form itself is the limiting step in the total release process. This could be modified in the future to compare the release rate from the specific waste form type to that from the surrounding soil, and then use the lower of these two values.

Analyses now assume that cement waste forms stay intact for all time during the simulation. This could be modified to allow the waste form to catastrophically fail at some specified time, after which the source zone acts like a soil-debris waste form type.

In addition to the primary waste forms and surrounding soil, the source zone may initially contain other material, such as facilities/buildings, waste containers, waste-zone structural components (e.g., asphalt pads, and plywood sheets separating layers of waste containers). In these analyses, no credit is taken for the ability of these other materials to inhibit contaminant release (i.e., the analyses now assume that these components degrade rapidly and offer no protection for the five generic waste form types for essentially the entire simulation time).

Each source zone on the Hanford Site considered in this Composite Analysis was categorized into one of the above five generic waste form types. The inventories of all relevant radionuclide contaminants for that source zone were compiled. If a source zone contained more than one of the waste form types, the contaminant inventories were appropriately apportioned among the different waste form types; separate release calculations were performed for each waste form type; and the resulting losses into the vadose zone for any specific contaminant were summed.

For the soil-debris type of waste form, the overall volume of the source zone was used to obtain contaminant concentrations (needed for the mathematical model of release) from inventories. For cake, glass, and cement waste forms, it is possible that some of the source zone also contains soil material (which is not considered in the mathematical model of contaminant release). So, for these waste form types, the actual volume of the cake, glass, or cement waste form in the source zone was used to obtain concentrations from inventories. The release model associated with the reactor block type of waste form does not contain these volume and concentration considerations.

For the soil-debris type of waste form, the horizontal cross-sectional area of the overall source zone (the perpendicular area seen by water percolating through the source zone) was used to calculate the water and contaminant fluxes for leaching losses. (This area is also needed by the vadose zone transport component of the Composite Analysis.) For the other four generic waste form types, the effective horizontal cross-sectional area used to calculate the water and contaminant fluxes for leaching losses may be less than that of the overall source zone if it were determined that part of the water percolating through the overall source zone did not really come into contact with the waste form.

D.2 Contaminant Release Models

In all cases, the Composite Analysis assumes that the (radionuclide) contaminants are lost from the source zone only via radioactive decay within the source zone and leaching from the bottom of the source zone along with water percolating through or around the waste form types. Additional potential contaminant loss processes (e.g., volatilization, wind suspension of contaminated particles, and water erosion of contaminated particles) are not considered in the primary analyses.

The release model appropriate to a specific source zone depends on the overall waste form type and the potential for geochemical controls to limit the release. The soil-debris, cake, glass, and reactor block source zones are assumed to be so-called "well-mixed reactors"; i.e., the properties (contaminant concentrations) are assumed to be spatially uniform throughout. The cement source zone is assumed to contain concentration gradients within the waste form.

The mathematical approach to the entire release and transport problem is as follows. It is assumed that the impact of progeny products is negligible, and that the ingrowth and transport of progeny products need not be accounted for. Because of this assumption, each contaminant can be analyzed individually. Furthermore, with this conceptualization, the mathematical problem of leaching release coupled with radioactive decay in the source zone (as well as transport coupled with decay in the vadose zone and aquifer) can be reduced to an associated mathematical problem that

considers only leaching release and vadose zone/aquifer transport of a nondecaying species. After this associated scenario is analyzed to produce concentration breakthrough curves at receptor points for the nondecaying species, the actual concentration breakthrough curves for the original problem scenario can be obtained by decaying the contaminant concentrations at the receptor point based on contaminant arrival time and decay half-life. This simplifies the source zone release and transport models, and reduces the number of transport simulations that must be done.

The primary required output from the source-term release component of the Hanford Composite Analysis is the fraction of initial inventory remaining in the source zone for each contaminant as a function of time. This function is used as an input boundary condition to the vadose zone transport component of the Composite Analysis. The different source zone release models described below calculate the fraction remaining for the nondecaying species of the associated mathematical problem. Table D.1 defines the source-term release model notation used in the following sections.

D.2.1 Variable Transform Method of Formulating the Mathematical Problem

Mathematical expressions for contaminant release from the source zone are based on the total activity of the radionuclide in the source zone:

$$M_i^{\bullet}(t) = \int_v C_{Ti}^{\bullet}(t)\, dV \qquad (D.1)$$

where $M_i^{\bullet}$ = the total activity of contaminant *i* in the source zone for the original mathematical problem (that includes decay) (Ci)

t = the time since initial condition of the source zone (yr)

$C_{Ti}^{\bullet}$ = the total concentration of contaminant *i* (in all forms) in the source zone for the original mathematical problem (that includes decay) (Ci cm^{-3})

V = the volume of the source zone (cm^3).

(In Equation D.1 and in all of the following equations, the symbol "•" in the superscript of a variable related to radionuclide quantity means that the variable is associated with the original mathematical problem [i.e., the real-world situation where decay as well as leaching is occurring].) At this point in the mathematical development, it is not necessary to assume that the total concentration of the contaminant is spatially uniform throughout the source zone.

With leaching and decay being the only loss processes assumed, the rate of change of contaminant activity in the source zone can be expressed as

$$\frac{dM_i^{\bullet}}{dt} = \left[\frac{dM_i^{\bullet}}{dt}\right]_{leach} + \left[\frac{dM_i^{\bullet}}{dt}\right]_{decay} = \left[\frac{dM_i^{\bullet}}{dt}\right]_{leach} - \lambda_i M_i^{\bullet} \qquad (D.2)$$

where λ_i = the first-order decay coefficient for contaminant i (yr^{-1}).

Equation D.2 implicitly assumes that the first-order decay coefficient of the contaminant is independent of the phase (aqueous, sorbed, or precipitated) in which the contaminant resides (this is strictly true for radionuclides).

The flux density of a contaminant entering the vadose zone below the source zone (because of leaching from the source zone) can be expressed as

$$q^{\bullet}_{ci}(t) = -\frac{1}{A}\left[\frac{dM^{\bullet}_{i}}{dt}\right]_{leach} \qquad (D.3)$$

where $q^{\bullet}_{ci}$ = the flux density of contaminant *i* entering the vadose zone below the source zone for the original mathematical problem (that includes decay) (Ci cm^{-2} yr^{-1})

A = the effective horizontal cross-sectional area of the contaminant source zone (cm^2).

Note that Equation D.3 can be interpreted as a definition of an average leaching flux density over the effective horizontal cross-sectional area, A, of the bottom of the source zone. The source-term calculations need a single value of flux density at this vertical location (at any particular time) because a one-dimensional, vertical transport scenario is assumed for the vadose zone below the source zone. Therefore, at this point in the mathematical development, it is still not necessary to assume that the total concentration of the contaminant in the source zone is spatially uniform. However, if the flux density variable is assumed to be an actual, horizontally uniform value (rather than a horizontally spatially averaged value), then the source zone should now be assumed to be horizontally spatially uniform.

This flux density, which is a function of time, is a necessary input to (i.e., a boundary condition for) the mathematical transport problem in the vadose zone that must be solved subsequently. However, in order to avoid complications that arise from using discrete time intervals in the solution to the coupled source-zone/vadose-zone problem, the two zones are linked via a function that represents the fraction of contaminant remaining in the source zone over time (rather than the flux density of contaminant out of the source zone over time):

$$f^{\bullet}_{ri}(t) = \frac{M^{\bullet}_{i}(t)}{M^{\bullet}_{oi}} \qquad (D.4)$$

where $f^{\bullet}_{ri}$ = the fraction of the initial inventory of contaminant *i* remaining in the source zone for the original mathematical problem that includes decay (unitless)

$M^{\bullet}_{oi}$ = the initial total activity of contaminant *i* in the source zone for the original mathematical problem that includes decay (Ci).

If Equation D.2 is rearranged and substituted into Equation D.3, and then Equation D.4 is rearranged and substituted into that equation, it can be shown that the relationship between the flux density and the fraction remaining is

$$q^{\bullet}_{ci} = -\frac{M^{\bullet}_{oi}}{A}\left(\frac{df^{\bullet}_{ri}}{dt} + \lambda_i f^{\bullet}_{ri}\right) \qquad (D.5)$$

Now, let us define the following variable transformation:

$$M_i^*(t) \equiv M_i(t)\, e^{-\lambda_i t} \tag{D.6}$$

where M_i = the total activity of contaminant *i* in the source zone for the transformed mathematical problem (Ci).

If Equation D.6 is differentiated with respect to time, and if that expression is then substituted into Equation D.2, and if the resulting expression is then simplified, we obtain the expression

$$\frac{dM_i}{dt} = \left[\frac{dM_i}{dt}\right]_{leach} \tag{D.7}$$

In deriving Equation D.7, the leaching loss term (on the right hand side of the equation) was explicitly defined to be

$$\left[\frac{dM_i}{dt}\right]_{leach} \equiv e^{\lambda_i t}\left[\frac{dM_i^*}{dt}\right]_{leach} \tag{D.8}$$

Furthermore, if Equation D.8 is rearranged and substituted into Equation D.3, and if the resulting expression is simplified, we obtain the expression

$$q_{ci}(t) = -\frac{1}{A}\left[\frac{dM_i}{dt}\right]_{leach} \tag{D.9}$$

where q_{ci} = the flux density of contaminant *i* entering the vadose zone below the source zone for the transformed mathematical problem (Ci cm^{-2} yr^{-1}).

In deriving Equation D.9, we made use of the definition:

$$q_{ci}^*(t) \equiv q_{ci}(t)\, e^{-\lambda_i t} \tag{D.10}$$

Furthermore, if Equation D.6 is substituted into Equation D.4, and if we make use of the fact that the initial total activities of the radionuclide must be the same for the two mathematical problems, i.e.,

$$M_{oi}^* \equiv M_{oi} \tag{D.11}$$

where M_{oi} = the initial total activity of contaminant *i* in the source zone for the transformed mathematical problem that includes decay (Ci)

we obtain the expression

$$f_{ri}(t) = \frac{M_i(t)}{M_{oi}} \tag{D.12}$$

where f_{ri} = the fraction of the initial mass of contaminant *i* remaining in the source zone for the transformed mathematical problem (unitless).

In deriving Equation D.12, we made use of the definition:

$$f^{\bullet}_{ri}(t) \equiv f_{ri}(t)\, e^{-\lambda_i t} \tag{D.13}$$

Furthermore, if Equations D.10, D.11, and D.13 are substituted into Equation D.5, and if the resulting equation is then simplified, we obtain the expression

$$q_{ci} = -\frac{M_{oi}}{A}\,\frac{df_{ri}}{dt} \tag{D.14}$$

The original mathematical problem (describing the actual real-world situation of loss by leaching and decay) given by Equations D.2, D.3, D.4, and D.5 has been transformed into an analogous problem given by Equations D.7, D.9, D.12, and D.14 (by using the variable transform definitions in Equations D.6, D.8, D.10, and D.13). Note further that Equations D.7, D.9, D.12, and D.14 have the proper mathematical form to describe a situation where the contaminants do not decay, and loss occurs only by leaching. To illustrate this, just let λ_i go to zero in Equations D.2, D.3, D.4, and D.5, and they reduce to the mathematical forms of Equations D.7, D.9, D.12 and D.14. In other words, the variable transform mathematics show that we really only need to solve the source zone release problem assuming that the contaminants do not decay. As long as we develop the leaching loss expression as the product of the actual real-world expression and the exponential factor (according to Equation D.8), we can use the variable transform definitions in Equations D.6, D.10, and D.13 to obtain the actual values of source zone activity, flux density, and fraction remaining if we so desire.

Next, a similar procedure can be applied to the mathematical problems of reactive transport in the vadose zone and aquifer. Beginning with the vadose zone, the differential equation of transport can be written as

$$\frac{\partial}{\partial t}\left(\theta_w C^{\bullet}_{wi} + \beta C^{\bullet}_{si}\right) = -\frac{\partial}{\partial z}\left(-\theta_w D_{si}\frac{\partial C^{\bullet}_{wi}}{\partial z} + q_w C^{\bullet}_{wi}\right) - \lambda_i\left(\theta_w C^{\bullet}_{wi} + \beta C^{\bullet}_{si}\right) \tag{D.15}$$

where θ_w = the volumetric water content of the source zone soil or vadose zone soil (unitless; $cm^3\ cm^{-3}$)

$C^{\bullet}_{wi}$ = the concentration of contaminant *i* in the aqueous phase for the original mathematical problem (that includes decay) (Ci cm^{-3})

β = the bulk density of the source zone soil or vadose zone soil (g cm^{-3})

$C^{\bullet}_{si}$ = the concentration of contaminant *i* in the sorbed phase for the original mathematical problem (that includes decay) (Ci cm^{-3})

z = the vertical spatial coordinate (cm)

D_{si} = the effective diffusion coefficient of contaminant *i* in the soil ($cm^2\ yr^{-1}$)

q_w = the Darcy flux density of water flowing through the source zone or vadose zone (cm yr^{-1}).

It is assumed that the contaminant sorption process is linear, reversible, and at equilibrium, i.e., that it can be described by a single sorption coefficient, K_{di}, for each contaminant for each porous medium.

Therefore:

$$C^{\bullet}_{si} = K_{di}\, C^{\bullet}_{wi} \tag{D.16}$$

where K_{di} = the linear equilibrium sorption coefficient for contaminant *i* to the source zone soil or vadose zone soil ($cm^3\ g^{-1}$).

Substituting Equation D.16 in Equation D.15, and assuming that

$$v_p = \frac{q_w}{\theta_w} \tag{D.17}$$

where v_p = the pore water velocity ($cm\ yr^{-1}$).

Equation D.15 can be simplified to

$$\frac{\partial C^{\bullet}_{wi}}{\partial t} = \frac{D_{si}}{R_i}\frac{\partial^2 C^{\bullet}_{wi}}{\partial z^2} - \frac{v_p}{R_i}\frac{\partial C^{\bullet}_{wi}}{\partial z} - \lambda_i\, C^{\bullet}_{wi} \tag{D.18}$$

where R_i = the retardation factor, or phase apportionment factor, for contaminant *i* (unitless)

and is given by

$$R_i = 1 + \left(\frac{\beta K_{di}}{\theta_w}\right) \tag{D.19}$$

Equation D.18 is the transport differential equation solved for the vadose zone. To complete the specification of the mathematical problem, the initial condition is given by

$$C^{\bullet}_{wi}(z,0) = 0 \tag{D.20}$$

the upper boundary condition (at the bottom of the source zone) is given by

$$\left[-\theta_w D_{si}\frac{\partial C^{\bullet}_{wi}}{\partial z} + q_w C^{\bullet}_{wi}\right]_{z=z_{sz}} = q^{\bullet}_{ci} \tag{D.21}$$

where z_{sz} = the location of the bottom of the source zone (cm)

and the lower boundary condition (at the water table) is given by

$$\left[\frac{\partial C^{\bullet}_{wi}}{\partial z}\right]_{z=z_{wt}} = 0 \tag{D.22}$$

where z_{wt} = the location of the water table (cm).

Now, let us define the following variable transformation:

$$C^{*}_{wi}(t) \equiv C_{wi}(t)\, e^{-\lambda_i t} \tag{D.23}$$

where C_{wi} = the concentration of contaminant *i* in the aqueous phase for the transformed mathematical problem (Ci cm^{-3}).

This definition is consistent with the variable transform definition for total activity in the source zone given by Equation D.6. In fact, if there were no precipitated-phase contaminant in the source zone, we could have used Equation D.23 as the variable transform definition for the source zone; and then derived Equation D.6 from Equation D.23 using Equations D.1 and D.16, along with the standard relationship for the total concentration in terms of the concentrations in aqueous and sorbed phases. By substituting Equation D.23 into Equations D.18, D.20, D.21, and D.22 (and then differentiating, rearranging, and simplifying as was done for the source zone equations), we obtain the following set of equations for the transformed mathematical problem:

$$\frac{\partial C_{wi}}{\partial t} = \frac{D_{si}}{R_i}\frac{\partial^2 C_{wi}}{\partial z^2} - \frac{v_p}{R_i}\frac{\partial C_{wi}}{\partial z} \tag{D.24}$$

$$C_{wi}(z,0) = 0 \tag{D.25}$$

$$\left[-\theta_w D_{si}\frac{\partial C_{wi}}{\partial z} + q_w C_{wi} \right]_{z=z_{sz}} = q_{ci} \tag{D.26}$$

$$\left[\frac{\partial C_{wi}}{\partial z}\right]_{z=z_{wt}} = 0 \tag{D.27}$$

Equations D.24 through D.27 have the proper mathematical form to describe a transport situation where the contaminants do not decay. (To illustrate this, just let λ_i go to zero in Equations D.18, D.20, D.21, and D.22, and they reduce to the mathematical forms of Equations D.24 through D.27.) Once this set of equations is solved (using the transformed leaching flux density from the source zone calculations in Equation D.26), we can use the variable transform definition in Equation D.23 to obtain the actual value of aqueous concentration if we so desire. We can also then calculate the contaminant flux density versus time function at the water table, which represents contaminant input to the groundwater aquifer. Furthermore, by using Equation D.14 (i.e., the relationship between the flux density and the fraction remaining), the vadose zone problem could also be formulated to use the fraction remaining as part of the upper boundary condition.

Next, a similar procedure can be applied to the mathematical problem of reactive transport in the groundwater aquifer. The approach and the equation development are very similar to that described above for the vadose zone problem. The differential equation includes a three-dimensional representation of dispersion, and the boundary conditions are more numerous and slightly different; but the logic is the same. Therefore, the full derivation is not presented. Suffice it to say that the

transformed groundwater transport problem can be solved to obtain contaminant concentrations as functions of time at desired receptor points. The actual (real-world) concentrations at the receptors can be obtained from these values by applying the transform definition in Equation D.23.

With the above understanding of the mathematical approach to the problem, all that is left to do to calculate the release from the source zone is to derive the specific transformed fraction remaining functions (Equation D.12) based on the leaching loss terms (Equations D.7 and D.8) appropriate to each specific type of waste form category.

D.2.2. Equations Used for the Soil-Debris Waste Form Type

The source zone is conceptualized as unconsolidated porous material, and the contaminants are assumed to be uniformly distributed throughout for all times. Because only radionuclide contaminants are considered, the conceptual model assumes that no organic liquid phase (immiscible with the aqueous phase) is present. It is also assumed that there is no competition between contaminants for sorption sites, and no other significant chemical interaction between contaminants. Because of these assumptions, partitioning of the mass/activity of a specific contaminant between phases depends only on the amount of that contaminant itself, rather than on the amounts of all contaminants jointly. Furthermore, it is also assumed that partitioning of the radionuclides into the vapor phase is negligible. Therefore, only aqueous and sorbed phases, and possibly a precipitated phase, are assumed to exist.

The maximum amount of contaminant that can be accommodated in the aqueous and sorbed phases of a source zone (without a precipitated phase) can be expressed as

$$M^{\bullet}_{maxi} = \left(\theta_w C^{sol}_{wi} + \beta K_{di} C^{sol}_{wi}\right) \bullet V \qquad (D.28)$$

where $M^{\bullet}_{maxi}$ = the maximum amount of contaminant *i* possible in the source zone without a precipitated phase (Ci)

C^{sol}_{wi} = the aqueous solubility of contaminant *i* (Ci cm^{-3}).

If the volume of the source zone is given by

$$V = A\,h \qquad (D.29)$$

where h = the average vertical thickness of the contaminant source zone (cm)

Equation D.28 can be rewritten as

$$M^{\bullet}_{maxi} = \theta_w R_i C^{sol}_{wi} A\,h \qquad (D.30)$$

If more than this amount ($M^{\bullet}_{maxi}$) of contaminant mass/activity exists in the source zone, a precipitated phase is assumed to be present and the aqueous concentration of the contaminant in the source zone is assumed to be solubility-controlled. In other words, if the following condition is true

$$M^{\bullet}_{i} - \theta_w R_i C^{sol}_{wi} A\,h > 0 \qquad (D.31)$$

the actual aqueous concentration in the source zone is given by

$$C^{*}_{wi} = C^{sol}_{wi} \tag{D.32}$$

Using the variable transform definition in Equation D.23, this means that the transformed aqueous concentration in the source zone is given by

$$C_{wi} = C^{sol}_{wi}\, e^{\lambda_i t} \tag{D.33}$$

(This is an unusual and important point to remember, and arises from the fact that the solubility limit must be applied to the actual scenario and not the transformed one.) If the condition in Equation D.31 is false, the transformed aqueous concentration of the contaminant in the source zone is assumed to be desorption-controlled, and given by

$$C_{wi} = \frac{M_i}{\theta_w R_i A h} \tag{D.34}$$

Because of the similarity/interdependence of variable transform definitions (Equations D.6 and D.23) for total activity and aqueous concentration when no precipitated phase is present, the mathematical form of Equation D.34 is identical to the analogous expression for the original (real-world) variables.

The leaching process is assumed to occur by advective transport of the aqueous-phase contaminant out of the bottom face of the source zone along with the percolating vadose zone water. Hence, the leaching flux is given by the product of the volumetric flux of water out of the source zone face and the aqueous concentration in the water at that time. The volumetric water flux is assumed to be in steady state, and is equal to the product of the Darcy water flux density and the horizontal cross-sectional area of the source zone. Therefore, the rate of loss of mass/activity from the source zone by leaching at any time is given by

$$\frac{dM_i}{dt} = - q_w A C_{wi} \tag{D.35}$$

In Equation D.35, the right-hand side of the equation is the leaching loss term given by the right-hand side of Equation D.7; and it is equal to the right hand side of Equation D.8, as it is supposed to be. Furthermore, in Equation D.35, C_{wi} is given by the expression in either Equation D.33 or D.34, depending on whether the system is solubility- or desorption-controlled.

If the condition given by Equation D.31 is false at time t=0, the leaching process is desorption-controlled for all times. In this instance, Equation D.35 (along with Equation D.34) can be rearranged and integrated to obtain the following expression for the contaminant activity remaining in the source zone as a function of time:

$$M_i(t) = M_{0i} \exp\left[- \frac{q_w t}{\theta_w R_i h} \right] \tag{D.36}$$

To obtain Equation D.36, the lower limits of the integrals involved are M_{oi} for contaminant activity and 0 for time. To obtain an expression for the fraction of contaminant activity remaining in the source zone as a function of time, Equation D.36 is divided by the initial inventory of the contaminant. Therefore, when the initial contaminant inventory is low enough that the leaching is always desorption-controlled, the fraction remaining is given by

$$f_{ri}(t) = \exp\left[-\frac{q_w t}{\theta_w R_i h}\right] \tag{D.37}$$

If the condition given by Equation D.5 is true at time t=0, there will be a period of time when the contaminant is leaching from the source zone via solubility control. For this time period, Equation D.35 (along with Equation D.33) can be rearranged and integrated to obtain the following expression for the contaminant activity remaining in the source zone as a function of time:

$$M_i(t) = M_{oi} - q_w A C_{wi}^{sol}\left(\frac{e^{\lambda_i t} - 1}{\lambda_i}\right) \tag{D.38}$$

For radionuclides with sufficiently long half-lives (i.e., λ_i approaching zero), the expression in Equation D.38 can be taken to the limit to obtain

$$M_i(t) \approx M_{oi} - q_w A C_{wi}^{sol} t \tag{D.39}$$

which is the expression we would expect for a nondecaying contaminant.

The function given by Equation D.38 is only valid up until the time, t_{co}, when the source zone changes over to a desorption-controlled leaching regime. In other words, the type of leaching described by Equation D.38 will last until the activity of the contaminant in the source zone is reduced to the amount specified by Equation D.30. Thus, this change-over time can be calculated by first substituting Equation D.6 into Equation D.38 (to obtain a expression for the actual activity), then rearranging the resulting expression, then setting M_i^* and t in that expression to M_{maxi}^* and t_{co}, respectively, to obtain

$$M_{maxi}^* = M_{oi} e^{-\lambda_i t_{co}} - q_w A C_{wi}^{sol}\left(\frac{1 - e^{-\lambda_i t_{co}}}{\lambda_i}\right) \tag{D.40}$$

where t_{co} = the time at which leaching changes from solubility- to desorption-controlled (yr).

Equation D.40 must then be solved for t. This must be done by some type of root-finding algorithm. Again, for radionuclides with sufficiently long half-lives (i.e., λ_i approaching zero), the expression in Equation D.40 can be taken to the limit, and then the resulting expression can be explicitly solved for t_{co} to obtain

$$t_{co} = \frac{\left(M_{oi} - M_{maxi}^*\right)}{q_w A C_{wi}^{sol}} \tag{D.41}$$

For times greater than t_{co}, the leaching is desorption-controlled. In this instance, Equation D.35 (along with Equation D.34) can be rearranged and integrated to obtain the following expression for the contaminant activity remaining in the source zone as a function of time:

$$M_i(t) = M^{\bullet}_{maxi} \exp\left[- \frac{q_w (t - t_{co})}{\theta_w R_i h} \right] \qquad (D.42)$$

To obtain Equation D.42, the lower limits of the integrals involved are $M^{\bullet}_{maxi}$ for contaminant activity and t_{co} for time (rather than M_{oi} and 0, which were used to obtain Equation D.36).

To obtain expressions for the fraction of contaminant activity remaining in the source zone as a function of time, Equations D.38 and D.42 are divided by the initial inventory of the contaminant. Therefore, when the initial contaminant inventory is high enough that a period of solubility-controlled release exists, the fraction remaining is given by

$$f_{ri}(t) = \begin{cases} 1 - \left(\frac{q_w A C^{sol}_{wi}}{M_{oi}} \right) \left(\frac{e^{\lambda_i t} - 1}{\lambda_i} \right) & \text{for } 0 \le t < t_{co} \\ \frac{M^{\bullet}_{maxi}}{M_{oi}} \exp\left[- \frac{q_w (t - t_{co})}{\theta_w R_i h} \right] & \text{for } t \ge t_{co} \end{cases} \qquad (D.43)$$

where t_{co} is given by the solution of Equation D.40. Again, for radionuclides with sufficiently long half-lives (i.e., λ_i approaching zero), the fraction remaining could be approximated by

$$f_{ri}(t) = \begin{cases} 1 - \left(\frac{q_w A C^{sol}_{wi}}{M_{oi}} \right) t & \text{for } 0 \le t < t_{co} \\ \frac{M^{\bullet}_{maxi}}{M_{oi}} \exp\left[- \frac{q_w (t - t_{co})}{\theta_w R_i h} \right] & \text{for } t \ge t_{co} \end{cases} \qquad (D.44)$$

where t_{co} is given by Equation D.41.

D.2.3 Equations Used for the Cake Waste Form Type

The source zone is conceptualized as consolidated porous material; and the contaminants are assumed to be uniformly distributed throughout for all times. The cake type of waste form also contains a given initial inventory, M_{msco}, of some major structural component that is also assumed to be uniformly distributed throughout for all times. The dissolution of this major structural component (via solubility control) into the water that percolates through the cake controls the dissolution of the overall cake. Therefore, the rate of loss of mass of the major structural component from the source zone by leaching at any time is given by

$$\frac{dM_{msc}}{dt} = - q_w A C^{sol}_{wmsc} \qquad (D.45)$$

where M_{msc} = the mass of the major structural component in the source zone (g)

C^{sol}_{wmsc} = the aqueous solubility of the major structural component (g cm^{-3}).

Equation D.45 can be rearranged and integrated to obtain the following expression for the mass of the major structural component remaining in the source zone as a function of time:

$$M_{msc}(t) = M_{msco} - q_w A C^{sol}_{wmsc} t \tag{D.46}$$

where M_{msco} = the initial mass of the major structural component in the source zone (g).

The time that it takes the percolating water to completely dissolve the cake can be calculated by letting $M_{msc}(t)$ be equal to 0 in Equation D.46, and solving the resulting equation for t. Doing so produces the expression

$$t_{cd} = \frac{M_{msco}}{q_w A C^{sol}_{wmsc}} \tag{D.47}$$

where t_{cd} = the cake dissolution time (yr).

Because all of the contaminants in the waste form are leached congruently with the dissolving cake, the initial inventory of each nondecaying contaminant is lost at a constant rate over the time period t_{cd}. In other words,

$$\left[\frac{dM_i}{dt}\right]_{leach} = \frac{M_{oi}}{t_{cd}} = C_{Ti}\left(\frac{V_{cao}}{t_{cd}}\right) \tag{D.48}$$

where V_{cao} = the initial volume of the cake source zone (cm^3).

In Equation D.48, the term V_{cao}/t_{cd} can be considered to be the volumetric rate of dissolution of the cake. For the real-world scenario, where the contaminant is decaying, the volumetric cake dissolution rate would be the same, but the total contaminant concentration in the remaining cake would be decreasing over time. This would be accounted for by including an exponential decay factor to the leaching loss term for the actual scenario; which means that the theory presented here is indeed consistent with the condition in Equation D.8. Therefore, the resulting expression for the activity of a nondecaying contaminant remaining in the source zone as a function of time is given by

$$M_i(t) = M_{oi} - \left(\frac{M_{oi}}{t_{cd}}\right) t \tag{D.49}$$

Substituting Equation D.47 into Equation D.49 results in

$$M_i(t) = M_{oi}\left(1 - \frac{q_w A C^{sol}_{wmsc} t}{M_{msco}}\right) \tag{D.50}$$

To obtain an expression for the fraction of contaminant mass/activity remaining in the source zone as a function of time, Equation D.50 is divided by the initial inventory of the contaminant. Therefore, the fraction remaining is given by

$$f_{ri}(t) = 1 - \left(\frac{q_w A C_{wmsc}^{sol}}{M_{msco}} \right) t \qquad \text{(D.51)}$$

D.2.4 Equations Used for the Glass Waste Form Type

For the glass waste form types, the contaminant release mechanism is dissolution of the glass at the outer surface of the waste form as water percolates past it. Contaminants are assumed to be uniformly distributed throughout the glass for all times. The conceptual and mathematical models for release used for the Hanford Composite Analysis are the same as those used for the interim performance assessment of Hanford low-level tank waste (Mann et al. 1997). Therefore, the rate of loss of activity from the source zone by leaching at any time is given by

$$\frac{dM_i}{dt} = - \frac{r_g A_{sg} M_{oi}}{V_{go}} \qquad \text{(D.52)}$$

where r_g = the volumetric dissolution rate of glass per area of surface (cm yr^{-1})

A_{sg} = the total external surface area of the glass waste form in the source zone (cm^2)

V_{go} = the initial volume of the glass waste form in the source zone (cm^3).

In Equation D.52, M_{oi}/V_{go} represents the volumetric total concentration of the nondecaying contaminant in the glass. Hence, Equation D.52 describes the contaminant mass loss rate as the product of the volumetric glass dissolution rate per area of surface, the surface area, and the contaminant concentration in the glass. For the real-world scenario, where the contaminant is decaying, the total concentration in the remaining glass would be decreasing over time. This would be accounted for by including an exponential decay factor to the leaching loss term for the actual scenario; which means that the theory presented here is indeed consistent with the condition in Equation D.8.

A_{sg} is not constant, but instead is changing over time as the glass dissolves. Therefore, Equation D.52 must be further developed by substituting into it an appropriate expression for how the surface area changes as a function of time. To develop this expression, the initial shape of the waste form is assumed to be roughly cubical. For this shape, the time-dependent surface area of a single waste form (six square sides) is given by

$$A_{sg} = 6 \left(L_o - 2 r_g t \right)^2 \qquad \text{(D.53)}$$

where L_o = the initial linear dimension of the cubical glass waste form (cm).

The time that it takes the percolating water to completely dissolve the glass can be calculated by letting A_{sg} be equal to 0 in Equation D.53, and solving the resulting equation for t. Doing so produces the expression

$$t_{gd} = \frac{L_o}{2\, r_g} \tag{D.54}$$

where t_{gd} = the glass dissolution time (yr).

By solving Equation D.54 for L_o and substituting the resulting expression into Equation D.53, the surface area of the waste form can be expressed as

$$A_{sg} = 24\, r_g^2 \left(t_{gd} - t\right)^2 \tag{D.55}$$

By substituting Equation D.55 into Equation D.52, and then expressing the initial volume of the cube in terms of the initial linear dimension (L_o), and then expressing L_o in terms of t_{gd} (via Equation D.54), the rate of loss of mass/activity from the source zone by leaching at any time can be expressed as

$$\frac{dM_i}{dt} = -\frac{3\left(t_{gd} - t\right)^2 M_{oi}}{t_{gd}^3} \tag{D.56}$$

Now, an expression for the initial fractional contaminant release rate can be derived from Equation D.56, and is given by

$$F_{rrgoi} = -\frac{1}{M_{oi}}\left[\frac{dM_i}{dt}\right]_{t=0} = \frac{3}{t_{gd}} \tag{D.57}$$

where F_{rrgoi} = the initial fractional release rate from a glass waste form for contaminant *i* (yr^{-1}).

The Request for Proposal (RFP) for the glass waste form to be produced for use at Hanford specifies the maximum average initial fractional release rate that will be allowed for different radionuclides (i.e, one value is specified for the release of technetium-99, and one value is specified for the release of all non-technetium-99 radionuclides). The Composite Analysis calculations assume that the initial fractional contaminant release rate will be equal to the value specified in the RFP. Therefore, by solving Equation D.57 for t_{gd}, and then substituting the resulting expression into Equation D.56, the rate of loss of activity from the source zone by leaching at any time can be expressed as

$$\frac{dM_i}{dt} = -\frac{F_{rrgoi}^3 \left(\frac{3}{F_{rrgoi}} - t\right)^2 M_{oi}}{9} \tag{D.58}$$

Equation D.58 can now be rearranged and integrated to obtain the following expression for the contaminant mass/activity remaining in the source zone as a function of time:

$$M_i(t) = M_{oi}\left(1 - \frac{F_{rrgoi}\, t}{3}\right)^3 \tag{D.59}$$

To obtain an expression for the fraction of contaminant mass/activity remaining in the source zone as a function of time, Equation D.59 is divided by the initial inventory of the contaminant. Therefore, the fraction remaining is given by

$$f_{ri}(t) = \left(1 - \frac{F_{rrgoi}\, t}{3}\right)^3 \qquad \text{(D.60)}$$

D.2.5 Equations Used for the Cement Waste Form Type

For cement waste form types, the contaminant release mechanism to the leaching pathway is diffusion through the solidified waste material to the outer surface of the waste form where it is carried away by the water percolating past the surface. For this conceptualization, the rate of loss of activity from the source zone by leaching at any time is assumed to be given by

$$\frac{dM_i}{dt} = - M_{oi} \left(\frac{A_{sc}}{V_{ce}}\right) \sqrt{\frac{D_{ci}}{\pi\, t}} \qquad \text{(D.61)}$$

where A_{sc} = the total external surface area of the cement waste form in the source zone (cm2)

V_{ce} = the volume of the cement waste form in the source zone (cm^3)

D_{ci} = the effective diffusion coefficient of contaminant *i* within a cement waste form (cm^2 yr-1).

Equation D.61 is actually derived from the solution to the diffusion equation for mass/activity lost through an infinite plane that bounds a semi-infinite solid source when no decay occurs (Godbee et al. 1980). In Equation D.61, M_{oi}/V_{ce} represents the total concentration of the nondecaying contaminant in the cement waste form. For the real-world scenario, where the contaminant is decaying, the total concentration in the cement would be decreasing over time. This would be accounted for by including an exponential decay factor to the leaching loss term for the actual scenario; which means that the theory presented here is indeed consistent with the condition in Equation D.8.

To go from an expression for flux density of contaminant lost from an infinite plane to the expression (Equation D.61) for total flux lost from the finite cement waste form, the assumption is made that the flux density expression can merely be multiplied by the total external surface area of the cement in the source zone. The flux calculated by Equation D.61 is approximately equal to that coming from a finite solid source for early times. However, at later times, Equation D.61 will overpredict the flux by an increasing amount as time goes on. Furthermore, the larger the cement waste form, and the smaller the effective diffusion coefficient, the longer the flux given by Equation D.61 will be approximately equal to that diffusing from a finite waste form. In spite of the approximate nature of the above expression, this idealized approach was taken because more accurate flux expressions for finite solids would strongly depend on the specific shape of the cement waste forms. There could likely be a variety of waste form shapes encountered in the Composite Analysis (meaning that a number of different, more complicated, expressions would need to be derived for the diffusive release), and these were not known a priori.

It is also worth noting that Equation D.61 depends on the activity in the source zone initially, rather than on the activity in the source zone at any given time (as in the flux expressions used for the

other generic types of source zone). This arises from the fact that the conceptual model used here requires that a spatial gradient in concentration be present within the waste form. This is in direct opposition to the "well-mixed reactor" assumption used to derive loss flux expressions in all of the other scenarios dealt with. In addition, because Equation D.61 depends on M_{oi} rather than M_i, partial failure of a cement waste form cannot be simulated. However, total failure of a cement waste form at some designated time can, in principle, be simulated. In this case the leaching flux expression would revert back to that for a soil-debris waste form (with contaminant activity at the initial time of that phase equal to the mass/activity remaining in the cement waste form when it failed). However, the analyses now assume that the cement waste form never fails; and so Equation D.61 is used for all times.

Equation D.61 can now be rearranged and integrated to obtain the following expression for the contaminant mass/activity remaining in the source zone as a function of time:

$$M_i(t) = M_{oi}\left[1 - 2\left(\frac{A_{sc}}{V_{ce}}\right)\sqrt{\frac{D_{ci}\,t}{\pi}}\right] \qquad (D.62)$$

To obtain an expression for the fraction of contaminant mass/activity remaining in the source zone as a function of time, Equation D.62 is divided by the initial inventory of the contaminant. Therefore, the fraction remaining is given by

$$f_{ri}(t) = 1 - 2\left(\frac{A_{sc}}{V_{ce}}\right)\sqrt{\frac{D_{ci}\,t}{\pi}} \qquad (D.63)$$

D.2.6 Equations Used for the Reactor Block Waste Form Type

This generic waste form type was developed to apply to the loss of radionuclides from irradiated graphite reactor blocks disposed of in the vadose zone. The conceptual and mathematical models of release are based on those reported in the draft environmental impact statement (EIS) for the decommissioning of the eight surplus production reactors at Hanford (DOE 1989). For the dosimetric analysis in that EIS, it was assumed that half of the released carbon-14 was leached to the vadose zone and half was volatilized to the atmosphere. However, for the Composite Analysis, it is assumed that all released contaminants are leached to the vadose zone.

The blocks release contaminants into the water percolating past them via unspecified loss processes from the solid graphite matrix as well as via corrosion of the solid graphite matrix and irradiated metal components over time. The surplus reactor EIS (DOE 1989) reports that no specific data are available regarding radionuclide release rates from the irradiated metal components; and so the EIS assumed that release from the metal was the same as from the graphite material of the reactor block. Several experimental studies of the loss of carbon-14 from graphite reactor block material indicate that there is an initial period of high release followed by a longer period of approximately steady-state release that is approximately two orders of magnitude lower. The surplus reactor EIS uses laboratory data to derive a correlation equation between carbon-14 release rate and time, and then uses the volume-to-surface area ratio of the Hanford reactor blocks to obtain a correlation equation that is equivalent to the following equation for predicting the loss of carbon-14 from the reactor block as a function of time and temperature:

$$\frac{dM_i}{dt} = - M_{oi} (365) \left[565 \left(1 + 100 \, e^{-(0.08)(365)t} \right) e^{-6440/T} \right] \qquad (D.64)$$

where T = the absolute temperature of the reactor block (K).

In Equation D.64, the factors of 365 have been added to convert time from the units used in the surplus reactor EIS (days) to those used here (years).

To put the time dependence of Equation D.64 in perspective, note that the release flux will fall to within 1% of its ultimate steady-state value at approximately 0.3 yr. Compared to the length of the Composite Analysis simulation time (which is on the order of 1000 to 2000 yr), the initial period of transient release is assumed to be insignificant. In addition, carbon-14 is the only radionuclide for which temperature-dependent release information is available. Because of this fact, the surplus reactor EIS assumed that the temperature of the reactor blocks was constant at 22°C. By assuming constant temperature and steady-state release flux conditions, Equation D.64 reduces to a form that is identical to the form of the release model used in the surplus reactor EIS for other radionuclides. Specifically, for other radionuclides, the only information available was limited laboratory data on the steady-state fractional release rates at ambient temperature. These single values were corrected for the volume-to-surface area ratio of the Hanford reactor blocks to produce a table (which was reported in the surplus reactor EIS) of Hanford-specific fractional release rates for certain specific radionuclides. Furthermore, for certain additional radionuclides for which there were no release data, the surplus reactor EIS recommends using the fractional release rate values of specific tabulated radionuclides that are assumed to behave similarly.

Based on these considerations, the rate of loss of activity from the source zone by leaching at any time is assumed to be given by

$$\frac{dM_i}{dt} = - M_{oi} F_{rri} \qquad (D.65)$$

where F_{rri} = the fractional release rate from a reactor block waste form for contaminant *i* (yr^{-1}).

Equation D.65 can now be rearranged and integrated to obtain the following expression for the contaminant activity remaining in the source zone as a function of time:

$$M_i(t) = M_{oi}(1 - F_{rri}t) \qquad (D.66)$$

To obtain an expression for the fraction of contaminant activity remaining in the source zone as a function of time, Equation D.66 is divided by the initial inventory of the contaminant. Therefore, the fraction remaining is given by

$$f_{ri}(t) = 1 - F_{rri}t \qquad (D.67)$$

D.3 Rationale for Choosing Values for Radionuclide-Related Parameters in the Release Model Equations

The radionuclide-related parameters required by the source zone release model are decay coefficient, aqueous solubility, distribution or sorption coefficient, initial fractional release rate from glass, effective diffusion coefficient in cement, and fractional release rate from reactor blocks. Some

of these parameters were input directly while others were calculated from other parameters. Table D.2 presents a list of the radionuclides considered in the source zone calculations, along with the input values used for each parameter for each nuclide.

The values of some of these parameters would, in general, be specific to the conditions at a particular source site. In some cases, where it was believed that reasonable "Hanford Site-specific" values were known, these values were used in the calculations. However, note that the value used for a specific parameter for a specific nuclide was the same for all Hanford source sites (i.e., because of the scope of this effort, no attempt was made to examine physico-chemical conditions at each source site and determine a different individual value of a parameter for each site). Most parameter values are based on actual data. However, some values are based on assumed similarity in behavior with other radionuclides, and some values are set equal to "default" values when no other information is available.

D.3.1 Decay Coefficient

Radioactive decay coefficients are actually calculated from decay half-lives by the source zone release model, according to the equation

$$\lambda_i = \frac{\ln 2}{t_{1/2i}} \qquad \text{(D.68)}$$

where $t_{1/2i}$ = the decay half-life of contaminant i (yr).

Values of radioactive decay half-lives for different radionuclides are unambiguous and well known. The specific values of half-life used for the source-term calculations were the values originally reported in a U.S. Environmental Protection Agency (EPA) Guidance report for radionuclides (Eckerman, Wolbarst, and Richardson 1988). These values had previously been incorporated into a computer database known as the Multimedia-Modeling Environmental Database and Editor (MMEDE) (Warren and Strenge 1994). For the source-term calculation effort of the Hanford Composite Analysis project, the MMEDE database was queried to produce an electronic file of tabulated half-lives for relevant radionuclides (that was subsequently incorporated into the source-term calculation spreadsheet).

D.3.2 Aqueous Solubility

First, the MMEDE database (Warren and Strenge 1994) was queried for values of aqueous solubility for each radionuclide. (The database contains a reference for each solubility value it contains.) Unfortunately, other than for tritium, the database does not contain a value for aqueous solubility for the radionuclides considered here.

However, as part of recent prior efforts on preparation of the Waste Isolation Pilot Plant Supplementary Environmental Impact Statement (WIPP SEIS) (DOE 1997), the solubility values for some of these radionuclides were estimated based on geochemical calculations (using the MINTEQA2 computer code [Allison, Brown, and Novo-Gradoc 1991]) for Hanford Site-specific conditions. The specific radionuclides chosen for estimation were based on a screening of the WIPP SEIS contaminants to determine which were most likely to be solubility-controlled and have a major influence on ultimate risk. The screening process and geochemical calculations are described, and the resulting solubility values are reported, in Buck et al. (1996). These values were adopted for use in the Composite Analysis calculations. For all remaining radionuclides (for which there were no

specific values available), the aqueous solubility was fixed at an arbitrarily high default value (1 x 10^{10} mg L^{-1}) so that the source zone release model would automatically select algorithms for desorption control rather than solubility control in these cases.

The source zone release model actually needs aqueous solubility values measured in units of Ci cm^{-3}. Values measured in units of mg L^{-1} were converted to units of Ci cm^{-3} by multiplying by the specific activity of each radionuclide (along with appropriate units conversion factors). The specific activity, in turn, was calculated from the decay half-life and the atomic mass according to the formula (DOHEW 1970):

$$a_{spi} = \frac{3.578 \times 10^{5}}{t_{1/2i}\ m_{ai}} \qquad \text{(D.69)}$$

where a_{spi} = the specific activity of contaminant *i* (Ci g^{-1})

m_{ai} = the atomic mass of contaminant *i* (g mol^{-1}).

Therefore, Table D.2 also includes values of specific activity and atomic mass.

D.3.3 Sorption Coefficient

A set of Hanford Site-specific K_d values were developed specifically for the Composite Analysis project in a separate effort. The Hanford data used and the approach taken for developing the K_d values are discussed in detail in Appendix E of the Composite Analysis.

D.3.4 Initial Fractional Release Rate from Glass

As stated previously, the conceptual and mathematical models for release from glass waste form types used for Hanford Composite Analysis are the same as those used for the interim performance assessment of Hanford low-level waste (Mann et al. 1997). Mann et al. (1997) also specify the initial fractional release rate to be used in the calculations. This value is part of the specifications for the waste form reported in the waste-form privatization RPF, and is the same for all radionuclides for these calculations.

D.3.5 Effective Diffusion Coefficient in Cement

First, specific values of effective diffusion coefficient in cement type waste forms for each radionuclide were chosen to be the values originally reported by Serne et al. (1989). These values had previously been incorporated into a computer database known as the MMEDE (Warren and Strenge 1994). For the source-term calculation effort of the Hanford Composite Analysis project, the MMEDE database was queried to produce an electronic file of tabulated diffusion coefficients for relevant radionuclides (which was subsequently incorporated into the source-term calculation spreadsheet).

However, as part of recent prior efforts on preparation of the WIPP SEIS, the diffusion coefficient values for some of these radionuclides were improved. The rationale for modifying the diffusion coefficients is described, and the resulting diffusion coefficient values are reported in Buck et al.

(1996). These values were adopted for use in the Composite Analysis calculations. For some radionuclides (for which there were no specific values available), the diffusion coefficient was fixed at a reasonable conservatively high default value (5×10^{-8} cm^2 s^{-1}).

D.3.6 Fractional Release Rate from Reactor Blocks

As stated previously, the conceptual and mathematical models of release from reactor blocks are based on those reported in the EIS for the decommissioning of the eight surplus production reactors at Hanford (DOE 1989). The surplus reactor EIS also reported values for fractional release rate (based on data) of some specific radionuclides from Hanford reactor blocks. It also made recommendations for what values to use for certain other radionuclides based on assumed similarity in behavior to radionuclides with measured data. These values were adopted for use in the Composite Analysis calculations.

D.4.0 References

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Table D.1. Definition of Source-Term Release Model Notation

Notation	Definition	Units
a_{spi}	specific activity of contaminant *i*	Ci g^{-1}
A	effective horizontal cross-sectional area of the contaminant source zone	cm^2
A_{sc}	total external surface area of the cement waste form in the source zone	cm^2
A_{sg}	total external surface area of the glass waste form in the source zone	cm^2
$C^{\bullet}_{si}$	concentration of contaminant *i* in the sorbed phase for the original mathematical problem (that includes decay)	Ci cm^{-3}
$C^{\bullet}_{T}$	total concentration of contaminant *i* (in all forms) in the source zone for the original mathematical problem (that includes decay)	Ci cm^{-3}
C_{wi}	concentration of contaminant *i* in the aqueous phase for the transformed mathematical problem	Ci cm^{-3}
C^{sol}_{w}	aqueous solubility of contaminant *i*	Ci cm^{-3}
$C^{\bullet}_{w}$	concentration of contaminant *i* in the aqueous phase for the original mathematical problem (that includes decay)	Ci cm^{-3}
C^{sol}_{wms}	aqueous solubility of the major structural component	g cm^{-3}
D_{ci}	effective diffusion coefficient of contaminant *i* within a cement waste form	cm^2 yr^{-1}
D_{si}	effective diffusion coefficient of contaminant *i* in the soil	$cm^2 \cdot yr^{-1}$
f_{ri}	fraction of the initial mass of contaminant *i* remaining in the source zone for the transformed mathematical problem	unitless
$f^{\bullet}_{ri}$	fraction of the initial inventory of contaminant *i* remaining in the source zone for the original mathematical problem (that includes decay)	unitless
F_{rmi}	fractional release rate from a reactor block waste form for contaminant *i*	yr^{-1}
F_{rrgoi}	initial fractional release rate from a glass waste form for contaminant *i*	yr^{-1}
h	average vertical thickness of the contaminant source zone	cm
K_{di}	linear equilibrium sorption coefficient for contaminant *i* to the source zone soil or vadose zone soil	cm^3 g^{-1}
L_o	initial linear dimension of the cubical glass waste form	cm
m_{ai}	atomic mass of contaminant *i*	g mol^{-1}
M_i	total activity of contaminant *i* in the source zone for the transformed mathematical problem	Ci

Table D.1. (contd)

Notation	Definition	Units
M_i^*	total activity of contaminant *i* in the source zone for the original mathematical problem (that includes decay)	Ci
M_{max}^*	maximum amount of contaminant *i* possible in the source zone without a precipitated phase	Ci
M_{msc}	mass of the major structural component in the source zone	g
M_{msco}	initial mass of the major structural component in the source zone	g
M_{oi}	initial total activity of contaminant *i* in the source zone for the transformed mathematical problem (that includes decay)	Ci
M_{oi}^*	initial total activity of contaminant *i* in the source zone for the original mathematical problem (that includes decay)	Ci
q_{ci}	flux density of contaminant *i* entering the vadose zone below the source zone for the transformed mathematical problem	Ci cm^{-2} yr^{-1}
q_{ci}^*	flux density of contaminant *i* entering the vadose zone below the source zone for the original mathematical problem (that includes decay)	Ci cm^{-2} yr^{-1}
q_w	Darcy flux density of water flowing through the source zone or vadose zone	cm yr^{-1}
r_g	volumetric dissolution rate of glass per area of surface	cm yr^{-1}
R_i	retardation factor, or phase apportionment factor, for contaminant *i*	unitless
t	time since initial condition of the source zone	yr
$t_{1/2i}$	decay half-life of contaminant *i*	yr
t_{cd}	cake dissolution time	yr
t_{co}	time at which leaching changes from solubility- to desorption-controlled	yr
t_{gd}	glass dissolution time	yr
T	absolute temperature of the reactor block	K
V	volume of the source zone	cm^3
V_{cao}	initial volume of the cake source zone	cm^3
V_{ce}	volume of the cement waste form in the source zone	cm^3
V_{go}	initial volume of the glass waste form in the source zone	cm^3
v_p	Pore water velocity	Cm yr^{-1}
z	vertical spatial coordinate	cm
z_{sz}	location of the bottom of the source zone	cm

Table D.1. (contd)

Notation	Definition	Units
z_{wt}	location of the water table	cm
β	bulk density of the source zone soil or vadose zone soil	g cm^{-3}
λ_i	first-order decay coefficient for contaminant *i*	yr^{-1}
θ_w	volumetric water content of the source zone soil or vadose zone soil	unitless (cm^3 cm^{-3})

Table D.2. Radionuclide-Specific Properties Needed as Inputs for the Composite Analysis

		Aqueous Solubility		Sorption K_d	Fractional Release Rate		Cement Diffusion Coefficient		Fractional Release Rate		Decay Half-Life		Specific Activity	Atomic Mass
Radionuclide	ID Code	(mg/L)	(Ci/cm^3)	(cm3/g)	(1/s)	(1/yr)	(cm^2/s)	(cm^2/yr)	(1/d)	(1/yr)	(d)	(yr)	(Ci/g)	(g/mol)
Actinium-224	AC224	1.00E+10	4.82E+10	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.21E-01	3.31E-04	4.82E+06	224
Actinium-225	AC225	1.00E+10	5.81E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.00E+01	2.74E-02	5.81E+04	225
Actinium-226	AC226	1.00E+10	4.35E+09	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.33E+00	3.64E-03	4.35E+05	226
Actinium-227	AC227	1.00E+10	7.24E+05	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			7.95E+03	2.18E+01	7.24E+01	227
Actinium-228	AC228	1.00E+10	2.25E+10	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.55E-01	6.98E-04	2.25E+06	228
Americium-237	AM237	1.00E+10	1.09E+11	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	5.07E-02	1.39E-04	1.09E+07	237
Americium-238	AM238	1.00E+10	8.06E+10	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	6.81E-02	1.86E-04	8.06E+06	238
Americium-239	AM239	1.00E+10	1.10E+10	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	4.96E-01	1.36E-03	1.10E+06	239
Americium-240	AM240	1.00E+10	2.57E+09	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	2.12E+00	5.80E-03	2.57E+05	240
Americium-241	AM241	1.00E+10	3.43E+04	40	1.4E-13	4.31E-06	5.00E-13	1.58E-05	8.00E-05	2.92E-02	1.58E+05	4.33E+02	3.43E+00	241
Americium-242	AM242	1.00E+10	8.06E+09	40	1.4E-13	4.31E-06	5.00E-13	1.58E-05	8.00E-05	2.92E-02	6.70E-01	1.83E-03	8.06E+05	242
Americium-242M	AM242M	1.00E+10	9.73E+04	40	1.4E-13	4.31E-06	5.00E-13	1.58E-05	8.00E-05	2.92E-02	5.55E+04	1.52E+02	9.73E+00	242
Americium-243	AM243	1.00E+10	1.99E+03	40	1.4E-13	4.31E-06	5.00E-13	1.58E-05	8.00E-05	2.92E-02	2.70E+06	7.39E+03	1.99E-01	243
Carbon-14	C-14	1.00E+10	4.47E+04		1.4E-13	4.31E-06	1.00E-12	3.16E-05			2.09E+06	5.72E+03	4.47E+00	14
Cerium-134	CE134	1.00E+10	3.25E+09	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03			3.00E+00	8.21E-03	3.25E+05	134
Cerium-135	CE135	1.00E+10	1.32E+10	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03			7.33E-01	2.01E-03	1.32E+06	135
Cerium-137	CE137	1.00E+10	2.54E+10	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03			3.75E-01	1.03E-03	2.54E+06	137
Cerium-137M	CE137M	1.00E+10	6.67E+09	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03			1.43E+00	3.92E-03	6.67E+05	137
Cerium-139	CE139	1.00E+10	6.81E+07	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03			1.38E+02	3.78E-01	6.81E+03	139
Cerium-144	CE144	1.00E+10	3.20E+07	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03			2.84E+02	7.78E-01	3.20E+03	144
Cesium-130	CS130	1.00E+10	4.83E+11	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	2.08E-02	5.69E-05	4.83E+07	130
Cesium-132	CS132	1.00E+10	1.53E+09	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	6.48E+00	1.77E-02	1.53E+05	132
Cesium-134	CS134	1.00E+10	1.30E+07	40	1.4E-13	4.31E-06	5.00E-10	1.58E-02	3.00E-05	1.10E-02	7.53E+02	2.06E+00	1.30E+03	134
Cesium-135	CS135	1.00E+10	1.15E+01	40	1.4E-13	4.31E-06	5.00E-10	1.58E-02	3.00E-05	1.10E-02	8.40E+08	2.30E+06	1.15E-03	135
Cesium-136	CS136	1.00E+10	7.34E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	1.31E+01	3.59E-02	7.34E+04	136
Cesium-137+D	CS137	1.00E+10	8.67E+05	40	1.4E-13	4.31E-06	5.00E-10	1.58E-02	3.00E-05	1.10E-02	1.10E+04	3.01E+01	8.67E+01	137
Cesium-138	CS138	1.00E+10	4.23E+11	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	2.24E-02	6.13E-05	4.23E+07	138
Chlorine-36	CL36	1.00E+10	3.30E+02	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00	1.00E-06	3.65E-04	1.10E+08	3.01E+05	3.30E-02	36
Cobalt-55	CO55	1.00E+10	3.25E+10	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	7.31E-01	2.00E-03	3.25E+06	55
Cobalt-56	CO56	1.00E+10	2.98E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	7.88E+01	2.16E-01	2.98E+04	56
Cobalt-57	CO57	1.00E+10	8.46E+07	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03	3.00E-05	1.10E-02	2.71E+02	7.42E-01	8.46E+03	57
Cobalt-58	CO58	1.00E+10	3.18E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	7.08E+01	1.94E-01	3.18E+04	58
Cobalt-58M	CO58M	1.00E+10	5.91E+10	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	3.81E-01	1.04E-03	5.91E+06	58
Cobalt-60	CO60	1.00E+10	1.13E+07	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03	3.00E-05	1.10E-02	1.93E+03	5.28E+00	1.13E+03	60
Curium-240	CM240	1.00E+10	2.02E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.70E+01	7.39E-02	2.02E+04	240
Curium-242	CM242	1.00E+10	3.31E+07	40	1.4E-13	4.31E-06	5.00E-13	1.58E-05			1.63E+02	4.46E-01	3.31E+03	242
Curium-243	CM243	1.00E-03	5.17E-08	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03			1.04E+04	2.85E+01	5.17E+01	243
Curium-244	CM244	1.00E-03	8.10E-08	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03			6.61E+03	1.81E+01	8.10E+01	244
Curium-245	CM245	1.00E+10	1.72E+03	40	1.4E-13	4.31E-06	5.00E-13	1.58E-05			3.10E+06	8.49E+03	1.72E-01	245
Curium-246	CM246	1.00E+10	3.07E+03	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.73E+06	4.74E+03	3.07E-01	246
Curium-248	CM248	1.00E+10	4.25E+01	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.24E+08	3.39E+05	4.25E-03	248
Europium-145	EU145	1.00E+10	1.52E+09	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	5.94E+00	1.63E-02	1.52E+05	145
Europium-146	EU146	1.00E+10	1.94E+09	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	4.61E+00	1.26E-02	1.94E+05	146
Europium-147	EU147	1.00E+10	3.70E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	2.40E+01	6.57E-02	3.70E+04	147
Europium-148	EU148	1.00E+10	1.62E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	5.45E+01	1.49E-01	1.62E+04	148
Europium-149	EU149	1.00E+10	9.42E+07	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	9.31E+01	2.55E-01	9.42E+03	149

Table D.2. (contd)

		Aqueous Solubility		Sorption K_d	Fractional Release Rate		Cement Diffusion Coefficient		Fractional Release Rate		Decay Half-Life		Specific Activity	Atomic Mass
Radionuclide	ID Code	(mg/L)	(Ci/cm^3)	(cm3/g)	(1/s)	(1/yr)	(cm^2/s)	(cm^2/yr)	(1/d)	(1/yr)	(d)	(yr)	(Ci/g)	(g/mol)
Europium-150A	EU150A	1.00E+10	1.66E+10	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	5.25E-01	1.44E-03	1.66E+06	150
Europium-150B	EU150B	1.00E+10	6.97E+05	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	1.25E+04	3.42E+01	6.97E+01	150
Europium-152	EU152	1.00E+10	1.77E+06	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03	8.00E-05	2.92E-02	4.87E+03	1.33E+01	1.77E+02	152
Europium-154	EU154	1.00E+10	2.64E+06	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03	8.00E-05	2.92E-02	3.21E+03	8.79E+00	2.64E+02	154
Europium-155	EU155	1.00E+10	4.66E+06	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03	8.00E-05	2.92E-02	1.81E+03	4.96E+00	4.66E+02	155
Iodine-129	I129	1.00E+10	1.77E+00	0.6	1.4E-13	4.31E-06	5.00E-08	1.58E+00			5.73E+09	1.57E+07	1.77E-04	129
Iodine-131	I131	1.00E+10	1.24E+09	0.6	1.4E-13	4.31E-06	5.00E-08	1.58E+00			8.04E+00	2.20E-02	1.24E+05	131
Iodine-135	I135	1.00E+10	3.52E+10	0.6	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.75E-01	7.53E-04	3.52E+06	135
Lead-203	PB203	2.00E-01	5.93E-02	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.17E+00	5.94E-03	2.97E+05	203
Lead-209	PB209	2.00E-01	9.20E-01	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.36E-01	3.72E-04	4.60E+06	209
Lead-210	PB210	2.00E-01	1.53E-05	40	1.4E-13	4.31E-06	1.00E-11	3.16E-04			8.15E+03	2.23E+01	7.64E+01	210
Lead-211	PB211	2.00E-01	4.94E+00	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.51E-02	6.87E-05	2.47E+07	211
Lead-212	PB212	2.00E-01	2.78E-01	40	1.4E-13	4.31E-06	1.00E-11	3.16E-04			4.43E-01	1.21E-03	1.39E+06	212
Neptunium-237	NP237	9.00E+02	6.35E-07	10	1.4E-13	4.31E-06	5.00E-10	1.58E-02			7.82E+08	2.14E+06	7.05E-04	237
Neptunium-238	NP238	1.00E+10	2.59E+09	10	1.4E-13	4.31E-06	5.00E-13	1.58E-05			2.12E+00	5.80E-03	2.59E+05	238
Neptunium-239	NP239	1.00E+10	2.32E+09	10	1.4E-13	4.31E-06	5.00E-13	1.58E-05			2.36E+00	6.46E-03	2.32E+05	239
Nickel-56	NI56	1.00E+10	3.63E+09	40	1.4E-13	4.31E-06	5.00E-10	1.58E-02	1.00E-05	3.65E-03	6.10E+00	1.67E-02	3.83E+05	56
Nickel-57	NI57	1.00E+10	1.53E+10	40	1.4E-13	4.31E-06	5.00E-10	1.58E-02	1.00E-05	3.65E-03	1.50E+00	4.11E-03	1.53E+06	57
Nickel-59	NI59	1.00E+10	8.08E+02	40	1.4E-13	4.31E-06	5.00E-10	1.58E-02	1.00E-05	3.65E-03	2.74E+07	7.50E+04	8.08E-02	59
Nickel-63	NI63	1.00E+10	5.91E+05	40	1.4E-13	4.31E-06	5.00E-10	1.58E-02	1.00E-05	3.65E-03	3.51E+04	9.61E+01	5.91E+01	63
Niobium-88	NB88	1.00E+10	1.50E+12	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			9.93E-03	2.72E-05	1.50E+08	88
Niobium-89A	NB89A	1.00E+10	3.21E+11	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			4.58E-02	1.25E-04	3.21E+07	89
Niobium-89B	NB89B	1.00E+10	1.74E+11	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			8.46E-02	2.32E-04	1.74E+07	89
Niobium-90	NB90	1.00E+10	2.39E+10	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			6.08E-01	1.66E-03	2.39E+06	90
Niobium-93M	NB93M	1.00E+10	2.83E+06	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			4.97E+03	1.36E+01	2.83E+02	93
Niobium-94	NB94	1.00E+10	1.88E+03	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			7.41E+06	2.03E+04	1.88E-01	94
Niobium-95	NB95	1.00E+10	3.91E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			3.52E+01	9.64E-02	3.91E+04	95
Plutonium-234	PU234	1.00E+10	1.52E+10	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	3.67E-01	1.00E-03	1.52E+06	234
Plutonium-235	PU235	1.00E+10	3.16E+11	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	1.76E-02	4.82E-05	3.16E+07	235
Plutonium-236	PU236	1.00E+10	5.32E+06	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	1.04E+03	2.85E+00	5.32E+02	236
Plutonium-237	PU237	1.00E+10	1.22E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	4.53E+01	1.24E-01	1.22E+04	237
Plutonium-238	PU238	7.00E+01	1.20E-03	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03	8.00E-05	2.92E-02	3.20E+04	8.76E+01	1.72E+01	238
Plutonium-239	PU239	7.00E+01	4.35E-06	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03	8.00E-05	2.92E-02	8.79E+06	2.41E+04	6.22E-02	239
Plutonium-240	PU240	7.00E+01	1.59E-05	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03	8.00E-05	2.92E-02	2.39E+06	6.54E+03	2.28E-01	240
Plutonium-241	PU241	1.00E+10	1.03E+06	40	1.4E-13	4.31E-06	5.00E-11	1.58E-03	8.00E-05	2.92E-02	5.26E+03	1.44E+01	1.03E+02	241
Plutonium-242	PU242	1.00E+10	3.94E+01	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00	8.00E-05	2.92E-02	1.37E+08	3.75E+05	3.94E-03	242
Protactinium-231	PA231	1.00E+10	4.71E+02	10	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.20E+07	3.29E+04	4.71E-02	231
Protactinium-233	PA233	1.00E+10	2.08E+08	10	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.70E+01	7.39E-02	2.08E+04	233
Radium-223	RA223	1.00E+10	5.14E+08	10	1.4E-13	4.31E-06	5.00E-11	1.58E-03			1.14E+01	3.12E-02	5.14E+04	223
Radium-224	RA224	1.00E+10	1.59E+09	10	1.4E-13	4.31E-06	5.00E-08	1.58E+00			3.66E+00	1.00E-02	1.59E+05	224
Radium-225	RA225	1.00E+10	3.92E+08	10	1.4E-13	4.31E-06	5.00E-11	1.58E-03			1.48E+01	4.05E-02	3.92E+04	225
Radium-226	RA226	1.00E+10	9.90E+03	10	1.4E-13	4.31E-06	5.00E-11	1.58E-03			5.84E+05	1.60E+03	9.90E-01	226
Radium-228	RA228	1.00E+10	2.73E+06	10	1.4E-13	4.31E-06	5.00E-11	1.58E-03			2.10E+03	5.75E+00	2.73E+02	228
Ruthenium-103	RU103	1.00E+10	3.23E+08	10	1.4E-13	4.31E-06	5.00E-08	1.58E+00			3.93E+01	1.08E-01	3.23E+04	103
Ruthenium-105	RU105	1.00E+10	6.73E+10	10	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.85E-01	5.07E-04	6.73E+06	105
Ruthenium-106	RU106	1.00E+10	3.35E+07	10	1.4E-13	4.31E-06	5.00E-08	1.58E+00			3.68E+02	1.01E+00	3.35E+03	106

Table D.2. (contd)

		Aqueous Solubility		Sorption K_d	Fractional Release Rate		Cement Diffusion Coefficient		Fractional Release Rate		Decay Half-Life		Specific Activity	Atomic Mass
Radionuclide	ID Code	(mg/L)	(Ci/cm^3)	(cm3/g)	(1/s)	(1/yr)	(cm^2/s)	(cm^2/yr)	(1/d)	(1/yr)	(d)	(yr)	(Ci/g)	(g/mol)
Selenium-70	SE70	1.00E+10	6.55E+11	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.85E-02	7.80E-05	6.55E+07	70
Selenium-73	SE73	1.00E+10	6.01E+10	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.98E-01	8.16E-04	6.01E+06	73
Selenium-73M	SE73M	1.00E+10	6.61E+11	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.71E-02	7.42E-05	6.61E+07	73
Selenium-75	SE75	1.00E+10	1.45E+08	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.20E+02	3.29E-01	1.45E+04	75
Selenium-79	SE79	1.00E+10	6.98E+02	0	1.4E-13	4.31E-06	2.00E-10	6.31E-03			2.37E+07	6.49E+04	6.98E-02	79
Selenium-81	SE81	1.00E+10	1.26E+12	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.28E-02	3.50E-05	1.26E+08	81
Strontium-85	SR85	1.00E+10	2.37E+08	10	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	6.48E+01	1.77E-01	2.37E+04	85
Strontium-89	SR89	1.00E+10	2.91E+08	10	1.4E-13	4.31E-06	5.00E-11	1.58E-03	3.00E-05	1.10E-02	5.05E+01	1.38E-01	2.91E+04	89
Strontium-90	SR90	1.00E+10	1.37E+06	10	1.4E-13	4.31E-06	5.00E-11	1.58E-03	3.00E-05	1.10E-02	1.06E+04	2.90E+01	1.37E+02	90
Strontium-91	SR91	1.00E+10	3.63E+10	10	1.4E-13	4.31E-06	5.00E-08	1.58E+00	3.00E-05	1.10E-02	3.96E-01	1.08E-03	3.63E+06	91
Technetium-101	TC101	1.00E+10	1.31E+12	0	1.4E-13	4.31E-06	1.00E-08	3.16E-01			9.86E-03	2.70E-05	1.31E+08	101
Technetium-97	TC97	1.00E+10	1.42E+01	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00			9.49E+08	2.60E+06	1.42E-03	97
Technetium-98	TC98	1.00E+10	8.72E+00	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.53E+09	4.19E+06	8.72E-04	98
Technetium-99	TC99	1.00E+10	1.70E+02	0	1.4E-13	4.31E-06	1.00E-09	3.16E-02			7.78E+07	2.13E+05	1.70E-02	99
Thorium-227	TH227	1.00E+10	3.08E+08	40	1.4E-13	4.31E-06	1.00E-12	3.16E-05			1.87E+01	5.12E-02	3.08E+04	227
Thorium-228	TH228	1.00E+10	8.20E+06	40	1.4E-13	4.31E-06	1.00E-12	3.16E-05			6.99E+02	1.91E+00	8.20E+02	228
Thorium-229	TH229	1.00E+10	2.13E+03	40	1.4E-13	4.31E-06	1.00E-12	3.16E-05			2.68E+06	7.34E+03	2.13E-01	229
Thorium-230	TH230	1.00E+10	2.02E+02	40	1.4E-13	4.31E-06	1.00E-12	3.16E-05			2.81E+07	7.69E+04	2.02E-02	230
Thorium-231	TH231	1.00E+10	5.34E+09	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.06E+00	2.90E-03	5.34E+05	231
Thorium-232	TH232	1.00E+10	4.00E-01	40	1.4E-13	4.31E-06	1.00E-12	3.16E-05			1.41E+10	3.86E+07	4.00E-05	232
Thorium-234	TH234	1.00E+10	2.32E+08	40	1.4E-13	4.31E-06	1.00E-12	3.16E-05			2.41E+01	6.60E-02	2.32E+04	234
Tin-113	SN113	1.00E+10	1.01E+08	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.15E+02	3.15E-01	1.01E+04	113
Tin-121	SN121	1.00E+10	9.56E+09	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.13E+00	3.09E-03	9.56E+05	121
Tin-123	SN123	1.00E+10	8.24E+07	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			1.29E+02	3.53E-01	8.24E+03	123
Tin-125	SN125	1.00E+10	1.08E+09	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			9.64E+00	2.64E-02	1.08E+05	125
Tin-126	SN126	1.00E+10	2.84E+02	40	1.4E-13	4.31E-06	5.00E-08	1.58E+00			3.65E+07	9.99E+04	2.84E-02	126
Tritium (Elemental)	H3-EL	3.80E-03	3.69E-05	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00	1.00E-06	3.65E-04	4.49E+03	1.23E+01	9.70E+03	3
Tritium (H3)	H3	1.00E+10	9.70E+07	0	1.4E-13	4.31E-06	5.00E-08	1.58E+00	1.00E-06	3.65E-04	4.49E+03	1.23E+01	9.70E+03	3
Uranium-232	U232	1.00E+10	2.14E+05	0.6	1.4E-13	4.31E-06	5.00E-08	1.58E+00			2.63E+04	7.20E+01	2.14E+01	232
Uranium-233	U233	1.00E+10	9.69E+01	0.6	1.4E-13	4.31E-06	1.00E-12	3.16E-05			5.79E+07	1.59E+05	9.69E-03	233
Uranium-234	U234	1.00E+10	6.25E+01	0.6	1.4E-13	4.31E-06	1.00E-12	3.16E-05			8.93E+07	2.44E+05	6.25E-03	234
Uranium-235	U235	1.00E+10	2.16E-02	0.6	1.4E-13	4.31E-06	1.00E-12	3.16E-05			2.57E+11	7.04E+08	2.16E-06	235
Uranium-236	U236	1.00E+10	6.48E-01	0.6	1.4E-13	4.31E-06	1.00E-11	3.16E-04			8.55E+09	2.34E+07	6.48E-05	236
Uranium-238	U238	1.00E+10	3.37E-03	0.6	1.4E-13	4.31E-06	1.00E-11	3.16E-04			1.63E+12	4.46E+09	3.37E-07	238
Uranium-239	U239	1.00E+10	3.35E+11	0.6	1.4E-13	4.31E-06	1.00E-11	3.16E-04			1.63E-02	4.46E-05	3.35E+07	239
Zirconium-89	ZR89	1.00E+10	4.49E+09	40	1.4E-13	4.31E-06	1.00E-12	3.16E-05			3.27E+00	8.95E-03	4.49E+05	89
Zirconium-93	ZR93	1.00E+10	2.51E+01	40	1.4E-13	4.31E-06	1.00E-12	3.16E-05			5.59E+08	1.53E+06	2.51E-03	93

Appendix E

Distribution Coefficient (K_d) Selection for the Composite Analysis

Appendix E

Distribution Coefficient (K_d) Selection for the Composite Analysis

D. I. Kaplan, R. J. Serne, V. G. Johnson, and C. T. Kincaid

E.1 Background

The purpose of the Composite Analysis is to estimate the projected cumulative impacts of all radioactive material in the ground that may interact with projected releases from existing or planned LLW disposal facilities. Guidance was issued for the Composite Analysis to examine how the variety of wastes to be permanently disposed at a DOE site might commingle and might exceed health protective limits.

The requirement to analyze, in a single analysis all wastes that will remain at the Hanford Site forced the examination of the numerous previous analyses of individual facilities and reconciliation of the conceptual models selected and model parameters applied to those analyses. The purpose of this appendix is to document the selections made for the geochemical adsorption/desorption distribution coefficients for the Composite Analysis.

E.2 Approach

For the Composite Analysis, several assumptions were made regarding the characteristics of sorption and the model that was employed. Adsorption was assumed to be fully reversible. Thus, a single distribution coefficient was used to represent both sorption and desorption. Because of its use in previous analyses at the Hanford Site (environmental impact statements, performance assessments, and CERCLA risk assessments), the linear sorption isotherm model was selected to represent the adsorption process. Other adsorption models exist, but their large data requirements cannot be met for the suite of radionuclides examined in the Composite Analysis. A distribution coefficient (K_d) defined by the following equation:

K_d = mass of solute on solid phase per unit mass of solid phase/concentration of solute in solution

was selected for elements and applied to all isotopes of that element. Thus, the same K_d value was applied to all isotopes of uranium considered in the analysis. The K_d values assembled here are based on experiments on saturated sediments. While research is underway to study the dependence of adsorption on moisture content, results are not available for a general model and the suite of radionuclides of interest.

In deriving the K_d values for elements considered in the Composite Analysis, previous analyses were examined. Specifically, K_d values from the following analyses were reviewed and evaluated:

- Performance assessments for the 200 West Area solid waste burial ground (Wood et al. 1995b); the 200 East Area solid waste burial ground (Wood et al. 1996); and the interim performance assessment for low-activity waste from Hanford tanks (Mann et al. 1996) and the remedial investigation/ feasibility study report for the Environmental Restoration Disposal Facility (ERDF) (DOE 1994; Wood et al. 1995a).

- Environmental impact statements completed for the surplus production reactors (DOE 1989, 1992), environmental restoration (DOE 1996), and the Tank Waste Remediation System (TWRS) program (DOE and State of Washington Department of Ecology [Ecology] 1996).

- The closure plan for the commercial LLW disposal site operated at Hanford by US Ecology (Grant, Environmental, Chase Environmental Group, and US Ecology, Inc. 1996).

K_d values used in these previous analyses are summarized in Table E.1. This table illustrates that a consistent suite of K_d values was not selected and used by the different programs evaluating LLW disposal at Hanford.

Because of the inconsistent definition of K_d values from the previous analyses at Hanford, it was necessary to evaluate all of the available data and derive a consistent set of values to use in the Composite Analysis. The first attempt at deriving a consistent set of K_d values involved use of a single K_d for each element. In an effort to minimize the number of simulations that must be conducted, radionuclides were assigned a K_d value that is less than or equal to its actual K_d value. However, the results of the source term release model demonstrated that a single K_d approach did not adequately represent the complexity of the disposal environment and natural subsurface system.

In the Composite Analysis, K_d values were assigned in a manner designed to recognize the impacts of waste chemistry and background chemistry. The concentrations of chelating agents, salts, and organic phases as well as pH have been demonstrated to greatly affect the magnitude of K_d values measured in the laboratory or derived from field observations. To account for the impacts of waste chemistry manifested through these factors, the sources were first categorized according to their waste compositions. The six source term categories used in the Composite Analysis are described in Table E.2.

The K_d values used in the Composite Analysis were further categorized based on the estimated impacts of background chemistry (Table E.3). Three distribution coefficient zones were established to represent changing geochemical conditions away from the source: 1) the high-impact zone near the source in the vadose zone, 2) an intermediate-impact zone away from the source, but still in the vadose zone, and 3) the groundwater zone. The high-impact zone is defined as the zone where the geochemistry of the vadose zone is greatly affected by the chemical composition of the waste source. The intermediate-impact zone differs from the high-impact zone in that the effect, if any, of the source-term pH on K_d values has disappeared; the effects of salts and organics, if present, continue to affect K_d values. The

intermediate-zone was defined in the vadose zone, before contaminants reach the groundwater. The groundwater zone is defined as the zone where K_d values are not affected by the chemical composition of the waste source. The background chemical composition of the groundwater zone is assumed to be greatly diluted and does not affect K_d values. The presence of chelates in the waste source represents the only aqueous constituent that could influence K_d values in the groundwater zone.

To accommodate the different waste source-term categories and K_d zones, K_d values had to be assigned to fill in a matrix of the six source types and three zones (Table E.4). Unique K_d values were not needed for all eighteen categories generated from the matrix of the three zones and six source types. Many of the categories were effectively the same and only eight different categories of K_d values were needed. The category identified in Table E.4 as "F" represents the far-field K_d values; category "C" represents the far-field K_d values affected by chelates.

Once the K_d categories were established, the geochemistry literature was reviewed to identify measured values to assign to the matrix. A range of K_d values was selected for each cell in the matrix. Generally, the lowest value of the range was used to represent the conservative estimate of K_d for each element. "Best"-estimate values for K_d were also identified. In some cases, values other than the lowest value in the range were assigned to the conservative value. In these cases expert judgement was applied to make the assignment. Where "best" estimates were based only on expert judgement, they are identified.

E.3 K_d Values for the Eight Source-Zone Categories

K_d values for each of the eight source-zone categories identified in Table E.4 are presented in Tables E.5 through E.12. Conservative (low), "best," and likely range of K_d values are included in the table. Additionally, a brief outline of the justification and references used to make these estimates are also provided. The "best" estimates are presented to provide guidance on what the most likely K_d value is for a given condition. Table E.10 provides K_d values for typical groundwater conditions at Hanford.

E.4 Summary Tables

Tables E.13 through E.17 provide summaries of the K_d values presented in Tables E.5 through E.12, but without the justifications and references. Table E.13 is a summary of the best estimate values used in the Composite Analysis. Table E.14 presents the summary of best estimate K_d values, adjusted for the maximum value of 40 mL/g replacement of all values greater than 40 mL/g. This adjustment was made to reduce the number of K_d values that had to be modeled. Constituents with K_d values of 40 mL/g and greater are considered immobile in the vadose zone and groundwater. Table E.15 is a summary of the conservative values. Table E.16 presents the summary of conservative K_d values, adjusted for the maximum value of 40 mL/g replacement of all values greater than 40 mL/g. Table E.17 provides a summary of the ranges of K_d values.

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Table E.1. Summary of Distribution Coefficients (mL/g) Previously Assigned to Radionuclides

Element	Distribution Coefficients Assigned in Previous Studies								
	Surplus Reactors[a]	ERDF[b]	200 East SWBG[c]	TWRS EIS[d]	HRA EIS[e]	US Ecology[f]	TWRS ILAW[g]	Low K_d[h]	High K_d[h]
Group of Highly Mobile Elements Assigned a K_d of 0 mL/g									
H	0	0	0		0				
Cl	0		0		0	0			
Se			0	0	0		0	0	0.78
Tc	0	0	0	0	0	0	0	0	1.3
Group of Somewhat Mobile Elements Assigned a K_d of 0.6 mL/g									
I			-	-	-	-	3	0.04	18
U	0	0	0	0	0 to 250	0	0.6	0.08	79.3
Group of Moderately Immobile Elements Assigned a K_d of 10 mL/g									
Np		2	10	0	0 to 500		15	2.4	29.1
Pa				1	50		6	10	1000
Ra		10	10	10	20	200	15	24	100
Ru				0	0			27	274
Sr	0.64	10	10	10	10	0.64	3	5	173
Group of Highly Immobile Elements Assigned a K_d of 40 mL/g									
Ac				50			40	7	1330
Am	76	100	100	50	50	810	40	67	>1200
Bi				1	100				
Ce							100	100	>2000
Cm				50	50		100	106	1330
Co	100	1	1		12		100	1200	12,500
Cs	26	100	100	50	30		100	540	3180
Eu		10	10	50			100	100	228
K		10	10		0.2	0			
Nb					100	350	40	50	100
Ni	100	100	100	1	12	100	40	50	2350
Pb				10	100		100	13,000	79,000
Po					100				
Pu	71	100	100	10	1 to 200	73	40	80	>1980
Re			0						
Sn				10			100	100	230
Th		100	100	10	50	40	40	40	100
Y				50	100				
Zr	2000			50	50		40	90	>2000
Special Case Elements									
C[i]	0	0	0	0	0	0	6	0	4

(a) From DOE (1989).
(b) From DOE (1994).
(c) From Wood et al. (1996).
(d) From DOE and Ecology (1996).
(e) From DOE (1996).
(f) From Grant Environmental, Chase Environmental Group, and US Ecology, Inc. (1996).
(g) From Mann et al. (1997).
(h) From Kaplan and Serne (1995) and Kaplan, Serne, and Piepho (1995).
(i) Recent work by Martin (1996) suggests carbon-14 undergoes attenuation in the environment because of isotopic exchange or dilution through recrystallization of minerals.

Table E.2. Source Term Categories

Category	Description	Examples/Comments
High Organic/Very Acidic	Selected plutonium and organic-rich condensate and process wastes	Z Plant, Carbon tetrachloride (with TBP, DBBP or lard oil) and aqueous waste streams from same facility
High Organic/Near Neutral pH	Organic rich process condensate and process wastes	REDOX, PUREX, Z Plant. Organics include hexone, carbon tetrachloride, TBP, and DBBP
Very High Salt/Very Basic	Tank wastes and wastes associated with small tanks, lines, pits, and boxes	Tank waste can contain chelators but the high pH tends to diminish impacts of organic chelators on K_d values
Chelates/High salts	Tank wastes with organic chelating or complexing agents	By cribs, waste with ferrocyanide (used to remove cesium) or EDTA additives (used to remove strontium)
Low Organic/Low Salts/Very Acidic	Uranium-rich process condensate	Uranium recovery from bismuth-phosphate wastes; PUREX; REDOX; and S-1/2, S-9, U-1/2, U-8, and U-12 cribs received acid waste
Low Organic/Low Salts/Near Neutral pH	• Plutonium-rich process condensate and process wastes. Characterized as neutral-basic wastes without organics, from Z Plant • General process condensate and process wastes. Characterized as small inventories of low-salt, neutral-basic wastes • Steam condensate. Characterized as small inventories of low-salt, neutral-basic wastes with high volumes • Chemical sewers. Characterized as small inventories of low-salt, neutral-basic wastes with high volumes • Cooling water. Characterized as small inventories of low-salt, neutral-basic wastes with high volumes • Chemical laboratory wastes. Characterized as low-salt, neutral-basic wastes • Miscellaneous wastes. Characterized as low-salt, neutral-basic wastes	

TBP = Tributyl phosphate
DBBP = Dibutyl butyl phosphate
REDOX = Reduction-Oxidation (S Plant)
PUREX = Plutonium-uranium extraction
EDTA = Ethylenediaminetetraacetic acid

Table E.3. Distribution Coefficient (K_d) Zone Categories

Zone Category	Description	Generalized Effect on K_d
High Impact	This zone is located in the vadose zone near the disposal facility inlet or ground surface. The liquid phase is greatly affected by the chemical composition of the contaminated liquid source. Organic compounds, pH, and salt, when present in the source term may affect K_d values.	Lowest K_d values
Intermediate Impact	This zone is located in the vadose zone immediately below the high-impact zone. The upper boundary is defined as the depth where the excessive acidic or basic nature of the waste has been neutralized by the buffering capacity of the natural soil. No pH effects of the plume remain.	Intermediate K_d values
Groundwater	This zone is in the unconfined aquifer where K_d values are not affected by the chemical composition of the plume. The waste source chemical compositions in this zone are assumed to be so greatly diluted they do not affect K_d values.	Largest K_d values

Table E.4. Source and Distribution Coefficient (K_d) Zone Categories[(a)]

	Zone Category		
Source Category	High Impact	Intermediate Impact	Groundwater
High Organic/Very Acidic	A	B	F
High Organic/Near Neutral	B	B	F
Very High Salt/Very Basic	D	E	F
Chelates/High Salts	G	G	C
Low Organic/Low Salts/Acidic	H	F	F
Low Organic/Low Salts/ Near Neutral	F	F	F

(a) Categories with the same letters have similar background chemistries and, therefore, similar K_d values.

Table E.5. Distribution Coefficient (K_d) Values for Source-Zone Category A[a]

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Tritium, Cl, Tc	0 (0)	0 to 1	No adsorption studies pertinent to these conditions were found in the literature. Tritium moves with water. A very slight degree of adsorption (K_d ~0.1 mL/g) has been reported resulting from HO^- sorbing to iron-oxides or tritiated water exchanging for regular water on clay surfaces[b,c]. Chlorine and technetium exist in groundwater primarily as anions. These anions tend not to sorb to mineral surfaces.
Ac, Am, Ce, Cm, Eu	0 (0.3)	0 to 2	These elements have +3 valences and quite similar sorption behavior. Laboratory column studies with Hanford soils and organic phase consisting of americium and plutonium in carbon tetrachloride (70-80%) and TBP or DBP/DBBP (20-30%), as well as depth distribution observations beneath the Z-1A crib suggest K_d are on the order of 0 to 1 mL/g[d]. Hajek and Knoll (1966) indicate that the spent waste consisting of the degraded TBP would have a K_d of 0.6 for americium (and 0 for plutonium). These column experiments were attempts to simulate the behavior of the spent process liquids containing carbon tetrachloride, high salt, acidic wastes discharged to the Z-1A, Z-9, Z18, Z-3 cribs. Batch experiments conducted with effluent simulating TBP process waste had americium K_d between 0 to 2[e].
C	0 (0)	0 to 0.4	Carbon-14 is introduced into the source in an inorganic form. The assumed dominant species are CO_2 gas under these acidic conditions, and C-Organic compound, HCO_3^- under neutral pH conditions. Carbon chemistry is very complicated in such mixed systems. It can become complexed with inorganic metals, enter into the structure of organic compounds, be volatilized out of solutions as CO_2 gas, or (co)precipitate into natural calcite minerals existing in the aquifer. The net effect of these conflicting processes is difficult to quantify because of a lack of experimental data. In an experiment conducted with Hanford groundwater spiked with $HCO2^-$ (no organics in liquid phase) and Hanford sediments in which the calcite coatings were removed with acid, K_d values were measured for carbon-14 between 0.27 to 0.38 mL/g[f].
Co	0 (0)	0 to 2	Cobalt did not sorb to a Hanford sediment when it was in the presence of bismuth phosphate-uranium-recovery scavenged waste containing ferrocyanide process effluent[g,h]. Cobalt is soluble in acid and readily complexed. Tests conducted to simulate uranium-recovery scavenged waste containing ferrocyanide process waste moving through soil columns showed that cobalt-60 was not removed by adsorption on the soil. The presence of nonexchangeable cobalt-60 became a limiting factor in the disposal of some wastes[g,h].
Cs	5 (7)	5 to 50	Cesium K_d values generally decrease as pH decreases[b]. K_d values of 5 to 50 in Hanford sediments have been reported for cesium.[e]
I	0 (0)	0	Iodine is a soluble anion. K_d values were estimated.[i]
Ni, Sn, Nb	2 (4)	0 to 10	The K_d values were estimated.[i]

Table E.5. (contd)

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Np, Pa	0.1 (0.2)	0.1 to 1	The effects of organic phase are unknown. They are assumed to behave similar to plutonium. The assumed dominant protactinium species is PaO_2^+ and NpO_2^+ is assumed to be a reasonable analog.[j]
Pb	0 (0)	0 to 10	The K_d values were estimated.[i]
Pu	0.1 (0.4)	0.1 to 1	See discussion above for actinium, americium, cerium, curium, and europium. The range for plutonium in Hanford sediments is 0.1 to 1 mL/g in liquid phases.[e,k]
Ra, Sr	0.1 (0.4)	0.1 to 5	K_d values were based on strontium sorption experiments.[e]
Ru	0.1 (0.4)	0 to 10	K_d values were estimated.[i] Nitrates and nitrites tend to decrease ruthenium sorption.
Se	0 (0)	0	K_d values were estimated.[i]
Th, Zr	1 (5)	1 to 20	K_d values were estimated.[i,l]
U	0.1 (0.2)	0.1 to 1	K_d values were estimated.[i] The effects of organic phase are unknown. They are assumed to behave similarly to plutonium.

(a) Category A is defined in Table E.4.
(b) From Ames and Rai (1978).
(c) From Thibault, Sheppard, and Smith (1990).
(d) From Hajek and Knoll (1966).
(e) From Knoll (1969).
(f) From Martin (1996).
(g) From Haney (1957).
(h) From Rhodes and Nelson (1957).
(i) From Ames and Serne (1991).
(j) From Pourbaix (1966).
(k) From Benson (1960).
(l) From Prout (1959).

TBP = Tributyl phosphate
DBP = Dibutyl phosphate
DBBP = Dibutyl butyl phosphate

Table E.6. Distribution Coefficient (K_d) Values for Source-Zone Category B[a]

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
tritium, Cl, Tc	0 (0)	0 to 1	Tritium moves with water. A very slight degree of adsorption (K_d ~0.1 mL/g) has been reported resulting from HO^- sorbing to iron oxides or tritiated water exchanging for regular water on clay surfaces.[b,c] Chlorine and technetium exist in groundwater primarily as anions.[b,c] The K_d value for TcO_4^- K_d is -0.6 to 0.02 in organic rich solid phase.[d]
Ac, Am, Ce, Cm, Eu	10 (20)	20 to >200	Hajek and Knoll[d] conducted column breakthrough tests simulating the behavior of the spent process liquids containing carbon tetrachloride, high salt, acidic wastes discharged to the Z-1A, Z-9, Z18, Z-3 cribs. When waste liquid was neutralized, the americium K_d value went up to over 200 mL/g (in acid solutions the K_d for americium was ~1 mL/g). When neutralized waste mixed with 20% by volume organics (carbon tetrachloride: TBP/DBBP mix), the K_d for americium dropped to 40. Once the americium (or plutonium) was adsorbed on the soil column, the organic mixture was not effective in removing it. The only PFP crib in this category is the Z-12 crib that received low-salt, neutralized waste containing americium (and plutonium) and some amounts of organic (carbon tetrachloride, TBP). Because of the neutralization, the americium and plutonium had a high affinity for the soil either resulting from sorption or resulting from filtering of particulate phases that may have formed prior to disposal.
C	0	0 to 10	The assumed dominant species are CO_2 and C-Organic compound, HCO_3^-. Carbon chemistry is complex in these mixed systems. Carbon can become complexed with inorganic metals, enter into the structure of organic compounds, be volatilized out of solutions as CO_2 gas, or (co)precipitate into natural calcite minerals existing in the aquifer. The net effect of these conflicting processes are difficult to quantify because of the lack of experimental data. The K_d is an estimate.[e]
Co	0.1 (3)	0.1 to 10	(f,g,h)
Cs	5 (10)	3 to 300	(g,h)
I	0 (0.1)	0 to 1	Iodine is a soluble anion. The K_d value was estimated.[i]
Ni, Sn, Nb,	3 (4)	0 to 30	The K_d value was estimated.[j,n]
Np, Pa	0.1 (0.2)	0.1 to 5	Neptunium is assumed to behave like plutonium, for which more data are available under these groundwater conditions. The dominant protactinium species is PaO_2^+ and that NpO_2^+ is a reasonable analog.[k]
Pb	0 (4)	0 to 10	Pb^{2+} forms stronger complexes with cyanide than Co^{2+}.[l] Cobalt mobility is greatly increased in the subsurface when cyanide is present.

Table E.6. (contd)

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Pu	15 (25)	15 to 50	See discussion above for actinium, americium, cerium, curium, and europium. Depth distribution studies show that over 95% of the inventory is within the upper 2-3 m (6-10 ft) beneath crib bottoms. Some has been found deeper, which may reflect a small fraction present as colloids or complexed. Based on simple one-dimensional unit gradient (steady-state flow) predictions of migration depth and observed maximum depths,[m] a K_d of 25 for plutonium would account for the most mobile (greatest depth of penetration) fraction of the inventory. Based on the above, the maximum depth of inventory is 0.25 m (1% of inventory), and >90% of the inventory is at 0.01 m depth. The K_d for this intermediate-impact zone of 25 mL/g is believed to be the best estimate. A conservative estimate of 15 mL/g is suggested.. Subsequent (groundwater zone) are assumed to be the same because of natural pH of around 8 in soil moisture and groundwater.
Ra, Sr	5 (7)	5 to 20	(n,o,h)
Ru	0.1 (2)	0 to 30	(p,q,h)
Se	0 (0)	0 to 1	Selenium is a soluble anion. The K_d value was estimated.[b]
Th, Zr	20 (40)	20 to 200	These elements are strong absorbers. The K_d values were estimated.[b]
U	0.2 (0.2)	0.2 to 10	Carbonate complexes are anionic. The K_d value was estimated.[b,r]

(a) Category B is defined in Table E.4.
(b) From Ames and Rai (1978).
(c) From Thibault, Sheppard, and Smith (1990).
(d) From Hajek and Knoll (1966).
(e) From Martin (1996).
(f) From Haney (1957).
(g) From Barney (1978).
(h) From Ames and Serne (1991).
(i) From Brown (1967).
(j) From Prout (1959).
(k) From Pourbaix (1966).
(l) From Smith and Martell (1976).
(m) From Johnson (1993).
(n) From Rhodes (1956).
(o) From Routson et al. (1981).
(p) From Raymond (1964).
(q) From Raymond (1965).
(r) From Kaplan and Serne (1995) and Kaplan, Serne, and Piepho (1995).

TBP = Tributyl phosphate
DBBP = Dibutyl butyl phosphate
PFP = Plutonium Finishing Plant (Z Plant)

Table E.7. Distribution Coefficient (K_d) Values for Source-Zone Category C[a,b]

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Co	0 (0)	0 to 3	Cobalt is likely complexed with EDTA and/or cyanide. Field data suggest that the cobalt-chelate complexed species exists and moves rapidly.
Sr, Pb, Ni, Sn	2 (4)	2 to 20	A strontium K_d of 0.4 mL/g has been measured in one Hanford soil (soil P) and 1.5 mL/g in another Hanford soil (soil S) in an aqueous system containing high concentrations of salts and medium to high concentrations of complexing agents, such as EDTA and HEDTA.[c] A slightly higher K_d value than these is likely to exist in the Hanford Site because the complexing agent concentrations will likely be appreciably lower. It is also anticipated that an appreciable amount of microbial degradation will occur to the organic complexes during their extended travel time to the far field.[d,e]
Pu	20 (40)	20 to >1980	A plutonium K_d of 21 mL/g has been measured in one Hanford soil (soil P) and 26 mL/g in another Hanford soil (soil S) in an aqueous system containing high concentrations of salts and medium to high concentrations of complexing agents, such as EDTA and HEDTA.[c] A slightly higher K_d value than these is likely because the complexing agent concentrations will likely be appreciably lower and it is anticipated that an appreciable amount of microbial degradation will occur to the organic complexes during their extended travel time to the far field.[d,e]
Np, Pa	2 (5)	2 to 15	A K_d of 8.7 mL/g has been measured for neptunium in one Hanford soil (soil P) and 12 mL/g in another Hanford soil (soil S) in an aqueous system containing high concentrations of salts and medium to high concentrations of complexing agents, such as EDTA and HEDTA.[c] Slightly higher K_d values than these are likely to exist because the complexing agent concentrations will likely be appreciably lower, and it is anticipated that an appreciable amount of microbial degradation will occur to the organic complexes during their extended travel time to the far field.[d,e] The assumed dominant protactinium species is PaO_2^+ and that NpO_2^+ is a reasonable analog.[f]
Ac, Am, Ce, Cm, Eu	10 (50)	10 to 500	A K_d of 5.6 mL/g has been measured for americium in one Hanford soil (soil P) and 10 mL/g in another Hanford soil (soil S) in an aqueous system containing high concentrations of salts and medium to high concentrations of complexing agents, such as EDTA and HEDTA.[c] Slightly higher K_d values than these are likely to exist because the complexing agent concentration will likely be appreciably lower and it is anticipated that an appreciable amount of microbial degradation will occur to the organic complexes during their extended travel time to the far field. Actinium, cerium, and curium also have +3 valance.[d,e]

(a) All K_d values not reported in this table are identical to those in Table E.10 for the far field groundwater.
(b) Category C is defined in Table E.4.
(c) From Delegard and Barney (1983) (see Table 11, "Dilute complexed" data)
(d) From Serne et al. (1995).
(e) From Ames and Rai (1978).
(f) From Pourbaix (1966).

EDTA = ethylenediaminetetraacetic acid
HEDTA = N-(2-hydroxyethyl) ethylene diaminetetraacetic acid

Table E.8. Distribution Coefficient (K_d) Values for Source-Zone Category D[a]

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Tritium, Cl, Tc, I, Se, Ru, C	0 (0)	0 to 0.2	Technetium, carbon, iodine, selenium, and chlorine are anionic. Tritium will move with water. Ruthenium has often been suggested as being coincident with water in tank-leak scenarios based on gamma borehole logging. Carbon as carbonate in high-pH tank environments is insoluble and combines with alkaline earth elements. To account for insolubility a K_d value > 0 is appropriate, but to keep carbon from getting stuck permanently in this source (high impact) zone the value must be set at 0.[b,c,d]
Ac, Am, Ce, Cm, Eu,	2 (5)	2 to 10	K_d values were estimated.[c,e]
Cs	1 (1.5)	1 to 25	Based on observations at Tank T-106, cesium-137 seemed to peak at about 3 m (10 ft) below the base (elevation) of the tank and nitrate seemed to peak at about 24 m (80 ft). This implies an in situ retardation factor of about 8 or K_d in the range of 1 - 2 during the initial tank leak. The lack of cesium in groundwater beneath tanks suggests it may not have broken through and more likely than not has a K_d that approaches the default value for neutral, high salt at greater distances from the source. Serne and Burke[f] measured a K_d of 26 mL/g for a simulated REDOX tank liquor. But the results are not consistent with inferred cesium migration using gamma borehole logging at SX tank farm.[g]
Co, Ni, Nb, Np, Pa, Sn	0.1 (0.2)	0.1 to 4	The K_d values were estimated.[e]
Sr, Ra	4 (10)	4 to 20	Strontium is known to be rather insoluble in tank liquors and does not migrate through soils in tank liquor as rapidly as other cations.[e]
Th, Zr, Pb, Pu	5 (10)	5 to 100	The K_d values were estimated.[c,e]
U	5 (20)	10 to 800	Kaplan et al.[h] reported that uranium K_d values increased from ~2 to >400 mL/g when the pH of a Hanford sediment/groundwater slurry increased from 8.3 to > 10.5. The extremely high K_d was attributed to uranium (co)precipitation either as uranium phases or as calcite phases. Over a 1000-year period, it is anticipated that the solutions pH of any near field would eventually decrease. Thus, over time, the K_d values would be expected to decrease as the pH increased above ~10.5 and the uranium dissolved from the solid phase.

(a) Category D is defined in Table E.4.
(b) From Ames and Rai (1978).
(c) From Thibault, Sheppard, and Smith (1990).
(d) From Martin (1996).
(e) From Ames and Serne (1991),
(f) From Serne and Burke (1997).
(g) From Hartman and Dresel (1997).
(h) From Kaplan et al. (1996).

Table E.9. Distribution Coefficient (K_d) Values for Source-Zone Category E[a]

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Tritium, Cl, Tc	0 (0)	0 to 0.1	Technetium and chlorine are anionic. Tritium will move with H_2O.
Ac, Am, Ce, Cm, Eu	100 (350)	280 to >1200	Americium in a calcium-dominated system has K_d values >1200 mL/g. In a sodium-dominated system, americium has a K_d value of 280 mL/g.[b]
C	0 (0)	0 to 10	K_d values were estimated.[c]
Co	50 (50)	222 to 4760	In a sodium-dominated system, K_d values are 1060 to 4760 mL/g.[d] In a calcium-dominated system, K_d values are 222 to 640 mL/g.[d] Cobalt forms complexes, especially with organics.
Cs	64 (500)	64 to 1360	In a sodium-dominated system, K_d values are 64 to 1170 mL/g [d]. In a calcium-dominated system, K_d values are 790 to 1360 mL/g [d]. Cesium does not form complexes.
I	0 (0)	0 to 2	Iodine is an anion. K_d values were estimated.[e,i]
Ni, Sn, Nb	30 (50)	3 to 40	Nickel is similar to cobalt but adsorbs slightly less possibly because of moderate complexing. K_d values were estimated.[e,f]
Np, Pa	0.2 (0.8)	0.4 to 4	K_d values range from 0.4 to 4 mL/g.[b] The dominant protactinium species is assumed to be PaO_2^+, and NpO_2^+ is assumed to be a reasonable analog.[g]
Pb	20 (100)	20 to 1000	Lead is a good absorber, it is insoluble. The K_d values were estimated.[f]
Pu	5 (20)	5 to >98	The K_d value is >98 mL/g.[h]
Ra, Sr	0.2 (0.5)	0.3 to 42	In a sodium-dominated system, K_d values range from 1.7 to 42 mL/g for strontium. In a calcium-dominated system, K_d value range from 0.3 to 1.6 mL/g for strontium.
Ru	0 (1)	0 to 500	This element may form RuO_4^{2-} and/or anionic complexes with nitrates and nitrites. The K_d values were estimated.[e,i,j]
Se	0 (0)	0 to 4	Selenium is anionic. The K_d values were estimated.[f]
Th, Zr	40 (50)	40 to 470	In sandy soil, thorium has K_d values ranging from 40 to 470 mL/g.[k]
U	0 (0.3)	0 to 3	Uranium is anionic and forms neutral carbonate and hydroxide species. K_d values were estimated.[e,f]

(a) Category E is defined in Table E.4.
(b) From Routson, Jansen, and Robinson (1976).
(c) From Martin (1996).
(d) From Routson, Barney, and Seil (1978).
(e) From Ames and Serne (1991).
(f) From Kaplan et al. (1996).
(g) From Pourbaix (1966).
(h) From Rhodes (1957b).
(i) From Ames and Rai (1978).
(j) From Barney (1978).
(k) From Sheppard, Kittrick, and Hardt (1976).

Table E.10. Distribution Coefficient (K_d) Values for Source-Zone Category F[a]

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Tritium, Cl, Tc	0 (0)	-2.8 to 0.6	Technetium exists predominantly as TcO_4^-. K_d values have been reported for technetium in Hanford sediments ranging from –2.8 to 0.6 mL/g for 15 observations with a median of 0.1 mL/g.[b] Later studies did not change this range but did decrease the median slightly to -0.1 mL/g.[c] Negative K_d values are physically possible and may not be an experimental artifact.[c] Tritium is expected to move along with water. Chlorine is expected to behave as a dissolved anionic species.
Ac, Am, Ce, Cm, Eu	100 (300)	67 to 1330	Two ranges for K_d values for americium have been reported: 67 to >1200 mL/g.[d] and 125 to 833 mL/g.[e]
C	0.5 (5, see Justification)	0.5 to 1000	The assumed dominant species for carbon is HCO_3^-. Three processes will be acting on carbon to take it out of solution: 1) adsorption onto the calcite surface, 2) volatilization as CO_2 gas, and 3) precipitation into the calcite structure. The latter process is largely irreversible, therefore it is not well represented by the K_d construct (K_d assumes that adsorption occurs as readily as desorption). Volatilization is entirely removed from the definition of the K_d construct. In systems that contain higher concentrations of carbonate minerals, such as the calcrete layer in the 200 West Area, an appreciably higher K_d should be used to account for the isotopic dilution/precipitation reaction that may occur, a K_d of 100 mL/g would be appropriate for such a system. Since most of the 100 and 200 Area Plateau contains <1% carbonate, lower K_d values are warranted for these areas, such as 0.5 mL/g. K_d values for carbon-14 of >250 mL/g have been measured in calcite.[f] At the 100K Area, the carbon-14 is widely distributed downgradient from a crib associated with reactor operations.[g,h,i,j,k] The range of K_d values was estimated.
Co	1200 (1200)	1200 to 12500	In a sodium-dominated system, the K_d values range from 1290 to 2120 mL/g[l] In a calcium-dominated system, the K_d values range from 2000 to 3870 mL/g [l] In the Hanford sediment/groundwater system, the K_d values range from 11600 to 12500 mL/g.[m]
Cs	540 (1500)	540 to 3180	In a sodium-dominated system, the K_d values range from 1410 to 1590 mL/g[l] In the Hanford sediment/groundwater system, the K_d values range from 540 to 3180 mL/g[m]
I	0.3 (0.5)	0.2 to 15	A review of K_d values for iodine in Hanford sediments showed a range of 0.7 to 15 mL/g for 9 observations; the median was 0.7 mL/g.[b] Later studies increased this range to 0.2 to 15 mL/g; the median was decreased to 0.3 mL/g.[c]
Ni, Sn, Nb	50 (300)	50 to 2350	In the Hanford sediment/groundwater system, K_d values for nickel ranged from 440 to 2350 mL/g.[m] In a broad range of sediments, including those from Hanford, K_d values for nickel ranged from 50 to 340 mL/g.[n]

Table E.10. (contd)

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Np, Pa	10 (15)	2.4 to 21.9	A review of neptunium K_d values for Hanford sediments showed range of 2.4 to 21.7 mL/g for 4 observations; the median was 17.8 mL/g.[b] Later studies increased the K_d values slightly to 2.2 to 21.7 mL/g; the median was slightly lowered, 15 mL/g.[c] The dominant protactinium species is assumed to be PaO_2^+ and NpO_2^+ is assumed to be a reasonable analog.[i]
Pb	2,000 (6,000)	13,000 to 79,000	In a system where the pH is 6 and there are no competing ions, the K_d values range from 13,000 to 79,000 mL/g.[o]
Pu	80 (200)	80 to >1980	For plutonium (V, VI) where the pH is 4 to 12, the K_d values range from 80 to >1980 mL/g.[p]
Ra, Sr	8 (20)	5 to 173	For a sodium-dominated system, the strontium K_d values range from 173 mL/g, and 49 to 50 mL/g[l] For a calcium-dominated system, the strontium K_d values range from 8 to 13 mL/g, 5 to 19 mL/g[l], 5 to 120 mL/g[q], and 19.1 to 21.5 mL/g[m] For a sodium-dominated system, where the pH is 7 to 11, the strontium K_d values range from 14.9 to 25.1 mL/g.[r]
Ru	10 (20)	10 to 1,000	K_d values were estimated.[p,i]
Se	0 (0)	-3.44 to 0.78	In the Hanford groundwater/sediment system, the K_d values range from -3.44 to 0.78 mL/g[m]
Th, Zr	40 (1000)	40 to >2000	K_d values were estimated. For zirconium, when the pH is 6 to 12, the K_d values range from 90 to >2000 mL/g[p]
U	0.6 (3)	0.1 to 79.3	A review of Hanford sediment uranium K_d values showed range of 0.1 to 79.3 mL/g for 13 observations; the median was 0.6 mL/g.[b] Results from later studies support the range.[c] In all reported data, some uranium was adsorbed by Hanford sediments and >90% of the values were between 0.6 and 4 mL/g.

(a) Category F is defined in Table E.4.
(b) From Kaplan and Serne (1995).
(c) From Kaplan et al. (1996).
(d) From Routson, Jansen, and Robinson (1976)
(e) From Sheppard, Kittrick, and Hardt (1976)
(f) From Martin (1996).
(g) From Striegl and Armstrong (1990).
(h) From Garnier (1985).
(i) From Pourbaix (1966).
(j) From Mozeto, Fritz, and Reardon (1983).
(k) From Zhang, Quay, and Wilbur (1995).
(l) From Routson, Barney, and Seil (1978).
(m) From Serne et al. (1993).
(n) From Serne and Relyea (1983).
(o) From Rhodes et al. (1992)
(p) From Rhodes (1957b).
(q) From Rhodes (1957a).
(r) From Nelson (1959).

Table E.11. Distribution Coefficients (K_d) Values for Source-Zone Category G[a]

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Tritium, Cl, Tc, C, Co, I, Se	0 (0)	0 to 0.5	Technetium, iodine, selenium, and chlorine are anions. Cobalt forms an unusually strong complex with EDTA by virtue of unique chemical reactions, namely the Co(II) converts to Co(III) through an auto-oxidation process and the Co(III) forms very strong complexes with the EDTA. Tritium is assumed to behave like water. The others do not complex with chelators and their low K_d values are controlled by virtue of their anionic nature.[b]
Ac, Am, Ce, Cm, Eu	3 (3)	3 to 50	A K_d for americium of 5.6 mL/g has been measured in one Hanford soil (soil P) and 24 mL/g in another Hanford soil (soil S) in an aqueous system containing high concentrations of salts and high concentrations of complexing agents, such as EDTA and HEDTA.[c] Additionally, bore hole data beneath 216-Z-1A Crib suggest that americium moves appreciably slower than carbon tetrachloride.[d] If carbon tetrachloride is considered a conservative tracer, then it would appear that americium behaves as if it has a nonzero K_d value, i.e., that it is retarded. Curium, cerium, and europium have a +3 valence and were assumed to behave like americium.[b,e]
Cs	6 (10)	6 to 18	These estimates are based on column breakthrough curves using actual uranium recovery scavenged waste.[f] The lack of cesium in the groundwater beneath the cribs suggests it has not broken through and more likely than not has a K_d value that approaches the default value (Table E.10).
Np, Pa	2 (5)	2 to 10	A K_d value of 3.9 mL/g has been measured for neptunium in one Hanford soil (soil P) and 6.8 mL/g in another (soil S) using an aqueous system containing high concentrations of salts and high concentrations of complexing agents, such as EDTA and HEDTA.[c] It is also assumed that protactinium speciation is predominately PaO_2^+ and that NpO_2^+ is a reasonable analog.[g]
U	0.2 (0.4)	0.2 to 3	K_d values were estimated.[b,h]
Ra, Sr, Pb, Ru, Ni, Nb, Sn	0.4 (5)	0 to 30	A K_d value of 0.02 mL/g has been measured for strontium in one Hanford soil (soil P) and 1.5 mL/g in another Hanford soil (soil S) in an aqueous system containing high concentrations of salts and high concentrations of complexing agents, such as EDTA and HEDTA.[c] These organic complexants are likely to be degraded by microbes over time, thereby converting the radionuclides into a more adsorbing species. Strontium is used as an analogue because of its similar +2 valence.

Table E.11. (contd)

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Th, Zr, Pu	0.5 (3)	0.6 to 100	A K_d of 0.6 mL/g has been measured for plutonium in one Hanford soil (soil P) and 2.6 mL/g in another (soil S) with an aqueous system containing high concentrations of salts and high concentrations of complexing agents, such as EDTA and HEDTA.[c] Additionally, bore hole data beneath 216-Z-1A Crib suggest that plutonium and americium move appreciably slower than carbon tetrachloride.[d] If carbon tetrachloride is considered a conservative tracer, then it would appear that both actinides behave as if they have nonzero K_d values.

(a) Category G is defined in Table E.4.
(b) From Serne et al. (1995).
(c) From Delegard and Barney (1983) (see Table 1.1, "Highly complexed" data).
(d) From Price et al. (1979).
(e) From Delegard and Barney (1983).
(f) From Rhodes and Nelson (1957).
(g) From Pourbaix (1966).
(h) From Ames and Rai (1978).

EDTA = ethylenediaminetetraacetic acid
HEDTA = N-(2-hydroxyethyl) ethylenediaminetetraacetic acid

Table E.12. Distribution Coefficients (K_d) Values for Source-Zone Category H.[a]

Element	Conservative and ("Best") K_d Estimate (mL/g)	Range K_d Estimate (mL/g)	Justification/References
Tritium	0 (0)	0	
Ac, Am, Ce, Cm, Eu	25 (50)	50 to 200	K_d values were estimated.[b,c,d,e]
C	0.1 (0.2)	0.1 to 5	In an experiment conducted with HCO_3^- spiked Hanford groundwater (no organics in liquid phase) and Hanford sediments in which the calcite coatings were removed with acid, K_d values between 0.27 to 0.38 mL/g were measured for carbon.[f] Under acid conditions inorganic carbon is predominately CO_2 gas. The main reason for the nonzero K_d value is that anions tend to sorb more to sediments in acid environments than in basic environments.
Co	0.2 (5)	0.2 to 20	K_d values were estimated.[e]
Cs	10 (30)	10 to 100	K_d values were estimated.[e, g, h]
I, Cl, Tc, Se	0.1 (0.2)	0.1 to 2	Anions sorb to iron oxides and kaolinite at lower pH levels.[e,i]
Ni, Sn, Nb	10 (20)	10 to 1,000	K_d values were estimated.[e]
U	20 (30)	20 to 200	K_d values were estimated.[e]
Np, Pa	3 (5)	--	K_d values were estimated.[e] It was also assumed that the dominant protectinium species is PaO_2^+ and that NpO_2^+ is a reasonable analog.[j]
Pb	25 (50)	--	K_d values were estimated.[e]
Pu	20 (50)	20 to 200	K_d values were estimated.[e, k]
Ra, Sr	10 (50)	50 to 200	K_d values were estimated.[e]
Ru	10 (20)	10 to 1000	K_d values were estimated.[e, l]
Th, Zr	30 (100)	30 to 5000	K_d values were estimated.

(a) Category H is defined in Table E.4.
(b) From Benson (1960).
(c) From Routson, Jansen, and Robinson (1976).
(d) From Sheppard, Kittrick, and Hardt (1976).
(e) From Ames and Serne (1991).
(f) From Martin (1996).
(g) From McHenry (1954).
(h) From Rhodes and Nelson (1957).
(i) From Ames and Rai (1978).
(j) From Pourbaix (1966).
(k) From Rhodes (1957b).
(l) From Rhodes (1957a).

Table E.13. Summary of Best-Estimate K_d Values Used in the Composite Analysis

Sources	Source Category	Elements																									
		H	Cl	Tc	Ac	Am	Ce	Cm	Eu	C	Co	Cs	I	Ni	Sn	Nb	Np	Pa	Pb	Pu	Ra	Sr	Ru	Se	Th	Zr	U
High Organic	A	0	0	0	0.3	0.3	0.3	0.3	0.3	0	0	7	0	4	4	4	0.2	0.2	0	0.4	0.4	0.4	0.4	0	5	5	0.2
Very Acidic	B	0	0	0	20	20	20	20	20	0	3	10	0.1	4	4	4	0.2	0.2	4	25	7	7	2	0	40	40	0.2
	F	0	0	0	300	300	300	300	300	5	1200	1500	0.5	300	300	300	15	15	6000	200	20	20	20	0	1000	1000	3
High Organic	B	0	0	0	20	20	20	20	20	0	3	10	0.1	4	4	4	0.2	0.2	4	25	7	7	2	0	40	40	0.2
Near Neutral	B	0	0	0	20	20	20	20	20	0	3	10	0.1	4	4	4	0.2	0.2	4	25	7	7	2	0	40	40	0.2
	F	0	0	0	300	300	300	300	300	5	1200	1500	0.5	300	300	300	15	15	6000	200	20	20	20	0	1000	1000	3
Very High Salts	D	0	0	0	5	5	5	5	5	0	0.2	1.5	0	0.2	0.2	0.2	0.2	0.2	10	10	10	10	0	0	10	10	20
Very Basic	E	0	0	0	350	350	350	350	350	0	50	500	0	50	50	50	0.8	0.8	100	20	0.5	0.5	1	0	50	50	0.3
	F	0	0	0	300	300	300	300	300	5	1200	1500	0.5	300	300	300	15	15	6000	200	20	20	20	0	1000	1000	3
Chelates	G	0	0	0	3	3	3	3	3	0	0	10	0	5	5	5	5	5	5	3	5	5	5	0	3	3	0.4
High Salts	G	0	0	0	3	3	3	3	3	0	0	10	0	5	5	5	5	5	5	3	5	5	5	0	3	3	0.4
	C	0	0	0	50	50	50	50	50	5	0	1500	0.5	4	4	300	5	5	4	40	20	4	20	0	1000	1000	3
Low Organic	H	0	0.2	0.2	50	50	50	50	50	0.2	5	30	0.2	20	20	20	5	5	50	50	50	50	20	0.2	100	100	30
Low Salts	F	0	0	0	300	300	300	300	300	5	1200	1500	0.5	300	300	300	15	15	6000	200	20	20	20	0	1000	1000	3
Acidic	F	0	0	0	300	300	300	300	300	5	1200	1500	0.5	300	300	300	15	15	6000	200	20	20	20	0	1000	1000	3
Low Organic	F	0	0	0	300	300	300	300	300	5	1200	1500	0.5	300	300	300	15	15	6000	200	20	20	20	0	1000	1000	3
Low Salts	F	0	0	0	300	300	300	300	300	5	1200	1500	0.5	300	300	300	15	15	6000	200	20	20	20	0	1000	1000	3
Near Neutral	F	0	0	0	300	300	300	300	300	5	1200	1500	0.5	300	300	300	15	15	6000	200	20	20	20	0	1000	1000	3

Table E.14. Summary of Best-Estimate K_d Values Used in the Composite Analysis, Adjusted to a Maximum of 40 mL/g

Sources	Source Category	Elements																								
		H	Cl	Tc	Ac	Am	Ce	Cm	Eu	C	Co	Cs	I	Ni	Sn	Nb	Np	Pa	Pb	Pu	Ra	Sr	Ru	Se	Th	Zr	U
High Organic	A	0	0	0	0.3	0.3	0.3	0.3	0.3	0	0	7	0	4	4	4	0.2	0.2	0	0.4	0.4	0.4	0.4	0	5	5	0.2
Very Acidic	B	0	0	0	20	20	20	20	20	0	3	10	0.1	4	4	4	0.2	0.2	4	25	7	7	2	0	40	40	0.2
	F	0	0	0	40	40	40	40	40	5	40	40	0.5	40	40	40	15	15	40	40	20	20	20	0	40	40	3
High Organic	B	0	0	0	20	20	20	20	20	0	3	10	0.1	4	4	4	0.2	0.2	4	25	7	7	2	0	40	40	0.2
Near Neutral	B	0	0	0	20	20	20	20	20	0	3	10	0.1	4	4	4	0.2	0.2	4	25	7	7	2	0	40	40	0.2
	F	0	0	0	40	40	40	40	40	5	40	40	0.5	40	40	40	15	15	40	40	20	20	20	0	40	40	3
Very High Salts	D	0	0	0	5	5	5	5	5	0	0.2	1.5	0	0.2	0.2	0.2	0.2	0.2	10	10	10	10	0	0	10	10	20
Very Basic	E	0	0	0	40	40	40	40	40	0	40	40	0	40	40	40	0.8	0.8	40	20	0.5	0.5	1	0	40	40	0.3
	F	0	0	0	40	40	40	40	40	5	40	40	0.5	40	40	40	15	15	40	40	20	20	20	0	40	40	3
Chelates	G	0	0	0	3	3	3	3	3	0	0	10	0	5	5	5	5	5	5	3	5	5	5	0	3	3	0.4
High Salts	G	0	0	0	3	3	3	3	3	0	0	10	0	5	5	5	5	5	5	3	5	5	5	0	3	3	0.4
	C	0	0	0	40	40	40	40	40	5	0	40	0.5	4	4	40	5	5	4	40	20	4	20	0	40	40	3
Low Organic	H	0	0.2	0.2	40	40	40	40	40	0.2	5	30	0.2	20	20	20	5	5	40	40	40	40	20	0.2	40	40	30
Low Salts	F	0	0	0	40	40	40	40	40	5	40	40	0.5	40	40	40	15	15	40	40	20	20	20	0	40	40	3
Acidic	F	0	0	0	40	40	40	40	40	5	40	40	0.5	40	40	40	15	15	40	40	20	20	20	0	40	40	3
Low Organic	F	0	0	0	40	40	40	40	40	5	40	40	0.5	40	40	40	15	15	40	40	20	20	20	0	40	40	3
Low Salts	F	0	0	0	40	40	40	40	40	5	40	40	0.5	40	40	40	15	15	40	40	20	20	20	0	40	40	3
Near Neutral	F	0	0	0	40	40	40	40	40	5	40	40	0.5	40	40	40	15	15	40	40	20	20	20	0	40	40	3

Table E.15. Summary of Conservative K_d Values Used in the Composite Analysis

Sources	Source Category	Elements																								
		H	Cl	Tc	Ac	Am	Ce	Cm	Eu	C	Co	Cs	I	Ni	Sn	Nb	Np	Pa	Pb	Pu	Ra	Sr	Ru	Se	Th	Zr	U
High Organic	A	0	0	0	0	0	0	0	0	0	0	5	0	2	2	2	0.1	0.1	0	0.1	0.1	0.1	0.1	0	1	1	0.1
Very Acidic	B	0	0	0	10	10	10	10	10	0	0.1	5	0	3	3	3	0.1	0.1	0	15	5	5	0.1	0	20	20	0.2
	F	0	0	0	100	100	100	100	100	0.5	1200	540	0.3	50	50	50	10	10	2000	80	8	8	10	0	40	40	0.8
High Organic	B	0	0	0	10	10	10	10	10	0	0.1	5	0	3	3	3	0.1	0.1	0	15	5	5	0.1	0	20	20	0.2
Near Neutral	B	0	0	0	10	10	10	10	10	0	0.1	5	0	3	3	3	0.1	0.1	0	15	5	5	0.1	0	20	20	0.2
	F	0	0	0	100	100	100	100	100	0.5	1200	540	0.3	50	50	50	10	10	2000	80	8	8	10	0	40	40	0.8
Very High Salts	D	0	0	0	2	2	2	2	2	0	0.1	1	0	0.1	0.1	0.1	0.1	0.1	5	5	4	4	0	0	5	5	5
Very Basic	E	0	0	0	100	100	100	100	100	0	50	64	0	30	30	30	0.2	0.2	20	5	0.2	0.2	0	0	40	40	0
	F	0	0	0	100	100	100	100	100	0.5	1200	540	0.3	50	50	50	10	10	2000	80	8	8	10	0	40	40	0.8
Chelates	G	0	0	0	3	3	3	3	3	0	0	6	0	0.4	0.4	0.4	2	2	0.4	0.5	0.4	0.4	0.4	0	0.5	0.5	0.2
High Salts	G	0	0	0	3	3	3	3	3	0	0	6	0	0.4	0.4	0.4	2	2	0.4	0.5	0.4	0.4	0.4	0	0.5	0.5	0.2
	C	0	0	0	10	10	10	10	10	0.5	0	540	0.3	2	2	50	2	2	2	20	8	2	10	0	40	40	0.8
Low Organic	H	0	0.1	0.1	25	25	25	25	25	0.1	0.2	10	0.1	10	10	10	3	3	25	20	10	10	10	0.1	30	30	20
Low Salts	F	0	0	0	100	100	100	100	100	0.5	1200	540	0.3	50	50	50	10	10	2000	80	8	8	10	0	40	40	0.8
Acidic	F	0	0	0	100	100	100	100	100	0.5	1200	540	0.3	50	50	50	10	10	2000	80	8	8	10	0	40	40	0.8
Low Organic	F	0	0	0	100	100	100	100	100	0.5	1200	540	0.3	50	50	50	10	10	2000	80	8	8	10	0	40	40	0.8
Low Salts	F	0	0	0	100	100	100	100	100	0.5	1200	540	0.3	50	50	50	10	10	2000	80	8	8	10	0	40	40	0.8
Near Neutral	F	0	0	0	100	100	100	100	100	0.5	1200	540	0.3	50	50	50	10	10	2000	80	8	8	10	0	40	40	0.8

Table E.16. Summary of Conservative K_d Values for the Composite Analysis, Adjusted to a Maximum of 40 mL/g

Sources	Source Category	Elements																									
		H	Cl	Tc	Ac	Am	Ce	Cm	Eu	C	Co	Cs	I	Ni	Sn	Nb	Np	Pa	Pb	Pu	Ra	Sr	Ru	Se	Th	Zr	U
High Organic	A	0	0	0	0	0	0	0	0	0	0	5	0	2	2	2	0.1	0.1	0	0.1	0.1	0.1	0.1	0	1	1	0.1
Very Acidic	B	0	0	0	10	10	10	10	10	0	0.1	5	0	3	3	3	0.1	0.1	0	15	5	5	0.1	0	20	20	0.2
	F	0	0	0	40	40	40	40	40	0.5	40	40	0.3	40	40	40	10	10	40	40	8	8	10	0	40	40	0.6
High Organic	B	0	0	0	10	10	10	10	10	0	0.1	5	0	3	3	3	0.1	0.1	0	15	5	5	0.1	0	20	20	0.2
Near Neutral	B	0	0	0	10	10	10	10	10	0	0.1	5	0	3	3	3	0.1	0.1	0	15	5	5	0.1	0	20	20	0.2
	F	0	0	0	40	40	40	40	40	0.5	40	40	0.3	40	40	40	10	10	40	40	8	8	10	0	40	40	0.6
Very High Salts	D	0	0	0	2	2	2	2	2	0	0.1	1	0	0.1	0.1	0.1	0.1	0.1	5	5	4	4	0	0	5	5	5
Very Basic	E	0	0	0	40	40	40	40	40	0	40	40	0	30	30	30	0.2	0.2	20	5	0.2	0.2	0	0	40	40	0
	F	0	0	0	40	40	40	40	40	0.5	40	40	0.3	40	40	40	10	10	40	40	8	8	10	0	40	40	0.6
Chelates	G	0	0	0	3	3	3	3	3	0	0	6	0	0.4	0.4	0.4	2	2	0.4	0.5	0.4	0.4	0.4	0	0.5	0.5	0.2
High Salts	G	0	0	0	3	3	3	3	3	0	0	6	0	0.4	0.4	0.4	2	2	0.4	0.5	0.4	0.4	0.4	0	0.5	0.5	0.2
	C	0	0	0	10	10	10	10	10	0.5	0	40	0.3	2	2	40	2	2	2	20	8	2	10	0	40	40	0.6
Low Organic	H	0	0.1	0.1	25	25	25	25	25	0.1	0.2	10	0.1	10	10	10	3	3	25	20	10	10	10	0.1	30	30	20
Low Salts	F	0	0	0	40	40	40	40	40	0.5	40	40	0.3	40	40	40	10	10	40	40	8	8	10	0	40	40	0.6
Acidic	F	0	0	0	40	40	40	40	40	0.5	40	40	0.3	40	40	40	10	10	40	40	8	8	10	0	40	40	0.6
Low Organic	F	0	0	0	40	40	40	40	40	0.5	40	40	0.3	40	40	40	10	10	40	40	8	8	10	0	40	40	0.6
Low Salts	F	0	0	0	40	40	40	40	40	0.5	40	40	0.3	40	40	40	10	10	40	40	8	8	10	0	40	40	0.6
Near Neutral	F	0	0	0	40	40	40	40	40	0.5	40	40	0.3	40	40	40	10	10	40	40	8	8	10	0	40	40	0.6

Table E.17. Summary of Ranges of K_d Values for the Composite Analysis

Sources	Source Category	Elements																									
		H	Cl	Tc	Ac	Am	Ce	Cm	Eu	C	Co	Cs	I	Ni	Sn	Nb	Np	Pa	Pb	Pu	Ra	Sr	Ru	Se	Th	Zr	U
High Organic Very Acidic	A - min	0	0	0	0	0	0	0	0	0	0	5	0	0	0	0	0.1	0.1	0	0.1	0.1	0.1	0	0	1	1	0.1
	A - max	1	1	1	2	2	2	2	2	0.4	2	50	0	10	10	10	1	1	10	1	5	5	10	0	20	20	1
	B - min	0	0	0	20	20	20	20	20	0	0.1	3	0	0	0	0	0.1	0.1	0	15	5	5	0	0	20	20	0.2
	B - max	1	1	1	>200	>200	>200	>200	>200	10	10	300	1	30	30	30	5	5	10	50	20	20	30	1	200	200	10
	F - min	-2.8	-2.8	-2.8	67	67	67	67	67	0.5	1200	540	0.2	50	50	50	2.4	2.4	13000	80	5	5	10	-3.44	40	40	0.1
	F- max	0.6	0.6	0.6	1330	1330	1330	1330	1330	1000	12500	3180	15	2350	2350	2350	21.9	21.9	79000	>1980	173	173	1000	0.78	>2000	>2000	79.3
High Organic Near Neutral	B - min	0	0	0	20	20	20	20	20	0	0.1	3	0	0	0	0	0.1	0.1	0	15	5	5	0	0	20	20	0.2
	B - max	1	1	1	>200	>200	>200	>200	>200	10	10	300	1	30	30	30	5	5	10	50	20	20	30	1	200	200	10
	B - min	0	0	0	20	20	20	20	20	0	0.1	3	0	0	0	0	0.1	0.1	0	15	5	5	0	0	20	20	0.2
	B - max	1	1	1	>200	>200	>200	>200	>200	10	10	300	1	30	30	30	5	5	10	50	20	20	30	1	200	200	10
	F - min	-2.8	-2.8	-2.8	67	67	67	67	67	0.5	1200	540	0.2	50	50	50	2.4	2.4	13000	80	5	5	10	-3.44	40	40	0.1
	F- max	0.6	0.6	0.6	1330	1330	1330	1330	1330	1000	12500	3180	15	2350	2350	2350	21.9	21.9	79000	>1980	173	173	1000	0.78	>2000	>2000	79.3
Very High Salts Very Basic	D - min	0	0	0	2	2	2	2	2	0	0.1	1	0	0.1	0.1	0.1	0.1	0.1	5	5	4	4	0	0	5	5	10
	D - max	0.2	0.2	0.2	10	10	10	10	10	0.2	4	25	0.2	4	4	4	4	4	100	100	20	20	0.2	0.2	100	100	800
	E - min	0	0	0	280	280	280	280	280	0	222	64	0	3	3	3	0.4	0.4	20	5	0.3	0.3	0	0	40	40	0
	E - max	0.1	0.1	0.1	>1200	>1200	>1200	>1200	>1200	10	4760	1360	2	40	40	40	4	4	1000	>98	42	42	500	4	470	470	3
	F - min	-2.8	-2.8	-2.8	67	67	67	67	67	0.5	1200	540	0.2	50	50	50	2.4	2.4	13000	80	5	5	10	-3.44	40	40	0.1
	F- max	0.6	0.6	0.6	1330	1330	1330	1330	1330	1000	12500	3180	15	2350	2350	2350	21.9	21.9	79000	>1980	173	173	1000	0.78	>2000	>2000	79.3
Chelates High Salts	G - min	0	0	0	3	3	3	3	3	0	0	6	0	0	0	0	2	2	0	0.5	0	0	0	0	0.6	0.6	0.2
	G - max	0.5	0.5	0.5	50	50	50	50	50	0.5	0.5	18	0.5	30	30	30	10	10	30	100	30	30	30	0.5	100	100	3
	G - min	0	0	0	3	3	3	3	3	0	0	6	0	0	0	0	2	2	0	0.5	0	0	0	0	0.6	0.6	0.2
	G - max	0.5	0.5	0.5	50	50	50	50	50	0.5	0.5	18	0.5	30	30	30	10	10	30	100	30	30	30	0.5	100	100	3
	C - min	-2.8	-2.8	-2.8	10	10	10	10	10	0.5	0	540	0.2	2	2	0	2	2	2	20	5	2	10	-3.44	40	40	0.1
	C - max	0.6	0.6	0.6	500	500	500	500	500	1000	3	3180	15	20	20	3	15	15	20	>1980	173	20	1000	0.78	>2000	>2000	79.3
Low Organic Low Salts Acidic	H - min	0	0.1	0.1	50	50	50	50	50	0.1	0.2	10	0.1	10	10	10	(a)	(a)	(a)	20	50	50	10	0.1	30	30	20
	H - max	0	2	2	200	200	200	200	200	5	20	100	2	1000	1000	1000	(a)	(a)	(a)	200	200	200	1000	2	5000	5000	200
	F - min	-2.8	-2.8	-2.8	67	67	67	67	67	0.5	1200	540	0.2	50	50	50	2.4	2.4	13000	80	5	5	10	-3.44	40	40	0.1
	F- max	0.6	0.6	0.6	1330	1330	1330	1330	1330	1000	12500	3180	15	2350	2350	2350	21.9	21.9	79000	>1980	173	173	1000	0.78	>2000	>2000	79.3
	F - min	-2.8	-2.8	-2.8	67	67	67	67	67	0.5	1200	540	0.2	50	50	50	2.4	2.4	13000	80	5	5	10	-3.44	40	40	0.1
	F- max	0.6	0.6	0.6	1330	1330	1330	1330	1330	1000	12500	3180	15	2350	2350	2350	21.9	21.9	79000	>1980	173	173	1000	0.78	>2000	>2000	79.3
Low Organic Low Salts Near Neutral	F - min	-2.8	-2.8	-2.8	67	67	67	67	67	0.5	1200	540	0.2	50	50	50	2.4	2.4	13000	80	5	5	10	-3.44	40	40	0.1
	F- max	0.6	0.6	0.6	1330	1330	1330	1330	1330	1000	12500	3180	15	2350	2350	2350	21.9	21.9	79000	>1980	173	173	1000	0.78	>2000	>2000	79.3
	F - min	-2.8	-2.8	-2.8	67	67	67	67	67	0.5	1200	540	0.2	50	50	50	2.4	2.4	13000	80	5	5	10	-3.44	40	40	0.1
	F- max	0.6	0.6	0.6	1330	1330	1330	1330	1330	1000	12500	3180	15	2350	2350	2350	21.9	21.9	79000	>1980	173	173	1000	0.78	>2000	>2000	79.3
	F - min	-2.8	-2.8	-2.8	67	67	67	67	67	0.5	1200	540	0.2	50	50	50	2.4	2.4	13000	80	5	5	10	-3.44	40	40	0.1
	F- max	0.6	0.6	0.6	1330	1330	1330	1330	1330	1000	12500	3180	15	2350	2350	2350	21.9	21.9	79000	>1980	173	173	1000	0.78	>2000	>2000	79.3

(a) No estimated range provided, see best estimate and conservative sheets.

Appendix F

Evaluation of Unit Dose Factors

Appendix F

Evaluation of Unit Dose Factors

D. L. Strenge

This appendix provides a description of unit dose factors (UDFs) used for evaluation of radiation dose rate[(a)] and chemical health impacts for the *Composite Analysis for Low-Level Waste Disposal in the 200 Area Plateau of the Hanford Site* (Composite Analysis). The UDFs were used to provide an estimate of radiation dose rate or chemical health impact (in this case, from uranium) per unit concentration in a medium (e.g., groundwater). The dose rates were evaluated for an individual exposed via pathways associated with contact to that medium.

Exposure scenarios defined for the Hanford Site Risk Assessment Methodology (HSRAM) (DOE 1995) were used as the basis for determining the pathways and contact rates for each medium. The HSRAM scenarios were developed for the Hanford Site as a guide to performing evaluations of dose and risk related to the Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) remedial investigations and the Resource Conservation and Recovery Act (RCRA) facility investigations. The four HSRAM exposure scenarios are referred to as recreational, industrial, residential, and agricultural. These scenarios, associated parameters, and UDF values are described in this appendix. The general methods described in Strenge and Chamberlain (1994) were used in the Composite Analysis to evaluate the UDFs.

F.1 Radionuclides and Chemicals of Interest

The UDFs were evaluated for the radionuclides of interest and uranium (Table F.1). This list is based on the radionuclides that were considered to be the most likely to result in radiation or chemical exposure of individuals from releases to the environment. In developing this list of radionuclides, consideration was given to past analyses at the Hanford Site. The list includes the progeny radionuclides that are generated with time and are potentially present at exposure locations after release and transport. Dose results presented in this analysis are limited to a subset of those for which UDFs were developed. Those dose results reported include only the key mobile radionuclides, carbon-14, chlorine-36, selenium-79, technetium-99, iodine-129, and uranium.

(a) All dose rates in the Composite Analysis (except where noted) are in units of mrem effective dose equivalent (EDE) in a year.

F.2 Unit Dose Factors for Radionuclides

The UDFs for radionuclides were used to provide an estimate of the annual radiation dose received by an individual exposed, as defined for the specific HSRAM scenarios. The dose is expressed in units of rem per year and represents the committed effective dose equivalent for one year of intake or exposure. The UDFs were evaluated for a unit concentration in a specific exposure medium. For example, when groundwater is the transport medium, the UDF is expressed per pCi/L in the groundwater. When air is the transport medium, the UDF is expressed per pCi/m^3 in air. In all cases, the concentrations in the transport medium were assumed to be constant over the exposure duration. The concentrations were also assumed to be constant over a period of time prior to the exposure period during which the deposited contaminant (from irrigation or atmospheric deposition, if appropriate to the scenario) was allowed to reach equilibrium in the soil. Equilibrium was assumed when the deposition rate was equal to leaching and radioactive decay losses from the soil.

F.3 Radiation Dosimetry Factors

The evaluation of annual radiation dose was completed for the Composite Analysis based on radiation dose conversion factors published in Federal Guidance Reports No. 11 and 12 (Eckerman, Wolbarst, and Richardson 1988; Eckerman and Ryman 1993). These dose factors are based on recommendations of the International Commission on Radiological Protection (ICRP) as given in ICRP Publication 30 (ICRP 1979a, 1979b). The resulting doses represent the effective dose equivalent received over a commitment period of 50 years following intake in the first year.

The evaluation of annual radiation dose as the endpoint in the analysis is a deviation from the guidance in the HSRAM (DOE 1995) report. The HSRAM guidance is for evaluation of the lifetime cancer incidence risk from radionuclides using slope factors. The slope factors relate intake (pCi) to the lifetime cancer incidence risk. However, the Composite Analysis required evaluation of annual radiation dose. Therefore, the use of radiation dose conversion factors replaced slope factors in the Composite Analysis.

F.4 Unit Dose Factors for Uranium

For chemicals, the unit dose factor is the hazard quotient defined by the U.S. Environmental Protection Agency (EPA) as the average daily intake of a chemical (in this case uranium) divided by the Reference Dose (RfD) for that chemical (EPA 1994). The hazard quotient was evaluated for both inhalation exposures and ingestion exposures with an RfD determined for each route.

F.5 Uranium Reference Doses

The EPA evaluates RfD values for selected chemicals and reports the values in the Integrated Risk Information System (IRIS) and the Health Effects Assessment Summary Tables (HEAST). At this time, neither IRIS nor HEAST provide estimates of RfD values for uranium. In order to generate an estimate of the health impact level from exposure to uranium (as a chemical toxicant), it was necessary to develop approximate values for the uranium RfDs (ingestion and inhalation intake routes).

The ingestion reference dose value was taken from a previous version of the EPA HEAST report (EPA 1994). The previous report gave a value of 0.003 mg/(kg-d) average daily intake. Although this value was withdrawn by EPA, it was used in the Composite Analysis as the best available value for ingestion exposure.

The EPA has not presented a value for the inhalation reference dose for uranium. A value was estimated for the Composite Analysis based on the Threshold Limit Value (TLV) published by the American Conference of Governmental Industrial Hygienists (ACGIH 1991). The TLV value for occupational inhalation exposure to uranium (as soluble and insoluble compounds of natural uranium) is 0.2 mg/m^3. This value was applied to workers exposed continuously during a 40-hour work week. In contrast, the RfD value was intended to represent safe levels for exposure of members of the public under continuous exposure (24 hours/day, 365 days/year). The TLV value can be converted to an inhalation RfD by consideration of relative inhalation rates, times of exposure, and sensitivity of members of the public relative to occupational workers. The conversion factor (Strenge, Peterson, and Sager 1989) used in the Composite Analysis was 0.007 mg/(kg-d) per mg/m^3, which includes a safety factor of 10 to account for sensitive members of the public. The resulting RfD was 0.0014 mg/(kg-d) for inhalation exposure to uranium compounds (with the intake expressed in mg of uranium).

F.6 Evaluation of Unit Dose Factors

Unit dose factors were calculated for radionuclides and chemicals with the exposure assessment component of the Multimedia Environmental Constituent Assessment System (MEPAS) code (Buck et al. 1995; Strenge and Chamberlain 1995; Droppo and Buck 1996). The evaluation was performed using equations and parameters for each exposure pathway as defined in the HSRAM report and modified for the Composite Analysis. The equations were structured to take advantage of the summary intake factor (SIF) concept presented in the HSRAM report. The concept of SIFs involved structuring the intake equations for each exposure pathway in such a way that constituent-independent parameters are separated from constituent-specific parameters and the initial media concentration. Each exposure pathway model was described as the product of three factors: a media concentration, an SIF independent of constituent, and a factor composed of all constituent-specific parameters. A general expression was used to calculate the dose or hazard quotient, as follows:

$$\textit{Dose or Hazard Quotient} = C_{mi}\, PF_{mix}\, SIF_{smyx} \qquad \text{(F.1)}$$

where Dose = annual radiation dose from intake or exposure to a radionuclide (rem/yr)

Hazard Quotient = hazard quotient from intake of a chemical toxicant based on the average daily intake over the one-year exposure period (dimensionless)

C_{mi} = concentration of constituent i in medium m (mg or pCi per unit quantity of medium L, kg, m^3, or m^2)

PF_{mix} = constituent specific factors for medium m, constituent i, and exposure pathway x (units specific to analysis)

SIF_{smyx} = summary intake factor for scenario s, medium m, constituent type y, an exposure pathway x (units specific to analysis).

The SIF values were evaluated for each toxicity type, radionuclides and noncarcinogenic chemicals. Carcinogenic effects were not included in the Composite Analysis because no carcinogenic chemicals were identified in the source inventory.

The MEPAS exposure component used in the present analysis allowed the user to provide SIF values as input. The SIF values were precalculated for each scenario, exposure pathway, constituent type (chemical and radionuclide) and medium (air and groundwater).

F.7 Exposure Scenario Descriptions

The four HSRAM exposure scenarios (DOE 1995), industrial, recreational, residential, and agricultural, were used as the basis for the UDF evaluations performed for the Composite Analysis. These exposure scenarios were adopted for the Composite Analysis because they are the current scenarios agreed upon by the U.S. Department of Energy (DOE), the State of Washington, and the EPA for Hanford Site evaluations of risk. They are routinely applied under the environmental restoration program. With one exception, the HSRAM scenarios and exposure parameter values were used as published. The HSRAM scenarios were defined for exposure over an extended duration (20 years for the industrial scenario and 30 years for other scenarios). The Composite Analysis required evaluation of the annual radiation dose received by potentially exposed individuals. The HSRAM scenarios were modified to reflect exposure for a one-year period, instead of a longer exposure duration.

As a result of this change, and because the total exposure to individuals from all exposure pathways was needed, the analysis did not include exposure of children. The HSRAM scenarios include exposures of children for a few pathways in which the child may receive a higher intake than adults. Some of the HSRAM scenarios involve exposure of a child for 6 years followed by exposure of an adult for 24 years. This example represents exposure for a 30-year period with partial intake as a child and partial intake as an adult. For the Composite Analysis, it was assumed to not be possible for an individual to be both a child and an adult during the one-year exposure period. The resulting dose estimates represent exposure of an individual as an adult, with contributions from all defined exposure pathways summed to give a total annual dose.

The first iteration Composite Analysis is an examination of radioactive waste disposal on the 200 Area Plateau. It is envisioned the DOE site boundary will shrink to include only the 200 Area Plateau as Hanford Site closure approaches. Historically agricultural land use and the groundwater have been the exposure scenario and environmental pathway yielding the maximum dose. Accordingly, no attempt was made in the Composite Analysis to model or estimate future contaminant concentrations in the Columbia River. Hence, the surface water medium and its associated exposure pathways (e.g., swimming and fish consumption) are omitted from this analysis.

Contaminant concentrations in groundwater prior to entering the river are greatly diluted in the Columbia River because of the mixing that occurs as water from the unconfined aquifer enters and is

entrained into the river. Consequently, dose estimates based on groundwater use and consumption in the immediate vicinity of the shore of the Columbia River are higher than those based on use and consumption of surface water from the Columbia River. For this reason, the exclusion of surface water exposure pathways and use of groundwater-based exposure pathways is appropriate in the Composite Analysis.

F.7.1 Industrial Scenario

The industrial scenario is intended to represent potential exposures to workers in a commercial or industrial setting. The scenario involves mainly indoor activities. Exposure to radioactive contamination and uranium as a chemical contaminant is limited to that originating with the groundwater, air, and soil (air deposition) transport media. The specific exposure pathways are listed in Table F.2 for the industrial scenario. Consistent with HSRAM (DOE 1995), the third pathway, dermal contact (e.g., bathing), was only applied in the chemical hazard analysis. The workers are assumed to wear no protective clothing. The scenario is not intended to represent exposure of remediation workers.

F.7.2 Recreational Scenario

The recreational scenario is intended to represent exposure to individuals engaging in seven days of recreational activity on the central portion of the Hanford Site. Exposure pathways included those associated with the groundwater, air, and soil (air deposition) transport media. The specific exposure pathways are listed in Table F.3 for the recreational scenario. Consistent with HSRAM (DOE 1995), the third exposure pathway, dermal contact through bathing, was only applied in the chemical hazard analysis. Elements of the HSRAM recreational scenario (DOE 1995) that involved surface water activities or sources of food (e.g., fish) were omitted from the scenario because of the exclusion of surface water exposure pathways from the Composite Analysis.

F.7.3 Residential Scenario

The residential scenario is intended to represent potential exposures to an individual who may take up residence of the Hanford Site in the future. The exposures are assumed to be continuous throughout the year, but limited to those originating with the groundwater, air, and soil (air deposition) transport media. Exposure pathways associated with residence on the Hanford Site and with both radionuclides and uranium as a chemical hazard are listed in Table F.4. Consistent with HSRAM (DOE 1995), the fourth exposure pathway, dermal contact through bathing, was only applied in the chemical hazard analysis. Surface water recreational activities are omitted from the scenario.

F.7.4 Agricultural Scenario

The agricultural scenario is intended to represent potential exposure to an individual who may reside on a small family farm on the Hanford Site in the future. The exposures are assumed to be continuous throughout the year. Exposures accrue from air, soil (air deposition), and groundwater transport media. The individual was assumed to take up residence on the Hanford Site and grow vegetables, fruit, and raise

meat and milk animals. As in the other scenarios, surface water recreational activities and fish consumption are excluded. The specific exposure pathways included in the agricultural scenario are listed in Table F.5 for radionuclides and chemicals. This scenario is unique because of the inclusion of contaminated meat and milk in the diet.

The only radionuclides identified in prior analyses as potentially significant in the air and soil (air transport) media were tritium, carbon-14, and radon-222. Of these, only tritium and carbon-14 were identified as potentially significant in a single source, i.e., the graphite cores of the surplus production reactors. Inclusion of the air and soil media will be seen to have negligible impact on overall dose.

F.8 Exposure Scenario Parameters

Parameter values used to calculate unit dose factors are presented in Tables F.6, F.7, F.8, and F.9 for the industrial, recreational, residential, and agricultural exposure scenarios, respectively. These tables indicate the intake or exposure rate, exposure frequency (days/year), unit conversion factors, and other factors used for specific exposure pathways. The tables also include a list of the resulting values for summary intake factors for chemicals and radionuclides. The parameter values in Tables F.6 through F.9 are taken directly from the HSRAM report (DOE 1995).

F.9 Exposure Pathways

A total of 17 different exposure pathways were considered in the UDF analyses. The pathways included in a specific analysis depend on the transport medium, scenario, and constituent type (radionuclide or uranium treated as a chemical), as indicated in the previous section. Details of each exposure pathway by transport medium are described below. In general, the parameter values for a pathway were derived from the HSRAM report (DOE 1995).

F.9.1 Soil (Air Deposition) Transport Medium

Deposition of airborne activity to soil allows exposure to individuals who come in contact with the soil, breathe suspended particles from the soil, or eat foods grown in the soil. The contamination deposited onto soil was modeled as a concentration per unit area of soil. All UDF values were normalized to the area soil concentration in units of mg/m^2 or pCi/m^2. Some of the soil exposure pathways required that concentrations be expressed in units of soil mass (mg/kg or pCi/kg dry soil). For these pathways, the conversion to soil mass was made using the conversion factor 60 kg/m^2, based on uniform distribution of the contaminant in the top 4 cm of soil having a density of 1.5 g/cm^3. This thickness is representative of the distribution of contaminants in residential soil (e.g., gardens or lawns) for deposition occurring over extended periods (several years).

The parameter values for each exposure pathway related to soil as a medium are summarized in Table F.9 for the agricultural exposure scenario. The assumptions for each of the exposure pathways are presented below.

F.9.1.1 Soil Ingestion

The individual was assumed to inadvertently ingest contaminated soil as part of daily activities defined for the scenarios. Residential and agricultural individuals ingest soil at 100 mg/d for the entire year, while the industrial worker ingests 50 mg/d while on the job for 146 days per year. The worker is assumed to be exposed to soil for only 146 of the 250 work days per year. The recreational visitor is exposed for 7 days per year at 100 mg per day.

F.9.1.2 Soil External Exposure

Radionuclides deposited onto soil may cause external radiation exposure to individuals near the contamination. The industrial worker is assumed to be exposed 8 hours per day for 146 days per year. The recreational visitor is exposed 8 hours per day for 7 days per year. The residential and agricultural scenario individuals are exposed 24 hours per day for 365 days per year.

F.9.1.3 Soil Dermal Contact

The dermal contact pathway was evaluated only for chemicals (as recommended in the HSRAM report). The individuals were assumed to have one contact event per day with soil adhering to the skin at a surface density of 0.2 mg/cm^2 of skin. The area of skin contacted was assumed to be 5000 cm^2. The industrial worker is exposed 146 days per year, the recreational visitor is exposed 7 days per year, and the residential and agricultural individuals are exposed 180 days per year.

F.9.1.4 Soil Resuspension Inhalation

Material deposited on the ground is assumed to be available for resuspension and inhalation by individuals close to the contamination. All exposure scenarios were based on the assumption that the individual inhales 20 m^3 of contaminated air per day. The airborne concentration of soil was evaluated using a resuspension factor to give an air concentration equivalent to 50 $\mu m/m^3$ of soil in air. The effective resuspension factor was assumed to be 8.33×10^{-10} m^{-1}, based on the soil density conversion factor of 60 kg/m^2 as described in Section F.9.1, "Soil (Air Deposition) Transport Medium."

F.9.1.5 Food Crops

Food crops are evaluated as fruits and vegetables. The crops were assumed to be contaminated when soil contamination (from airborne deposition) transfers to the edible parts of the plant by root uptake. Food crops were assumed to be eaten by the agricultural individual. The individual was assumed to consume 42 g/d of fruit and 80 g/d of vegetables throughout the year. The soil concentration was based on a soil mixing or plow depth of 15 cm and a soil density of 1.5 g/cm^3, which is equivalent to an areal soil density of 225 kg/m^2.

F.9.1.6 Game (Deer)

For the recreational and agricultural scenarios the individual was assumed to hunt and kill one deer in the year. The deer becomes contaminated when foraging on plants grown in contaminated soil. The HSRAM scenario applies a hunter success rate of 19% for a season. This is appropriate when the exposure duration is many years (30 years for HSRAM), but was not appropriate for the one-year period considered in the Composite Analysis. The annual dose analysis was based on the assumption that the hunter is successful (success rate = 100% for the year of exposure). Also, the HSRAM intake rate for deer meat is based on the amount of animal fat in the consumed meat. While this may be appropriate for the organic chemical constituents that are lipophilic, it is not generally appropriate for radionuclides. Also, the exposure pathway models for radionuclides evaluate the activity in the edible meat, not fat. The intake rate for deer meat was adjusted to represent the amount of meat ingested. This value was assumed to be 15 g/d, as reported for the recreational scenario of the Columbia River Comprehensive Impact Assessment project (Napier et al. 1996).

F.9.1.7 Meat and Milk Ingestion

Individuals in the agricultural scenario were assumed to ingest 75 g/d of meat (other than game), and 300 g/d of dairy products (represented as milk). The animals were assumed to be exposed from eating feed crops contaminated by root uptake from contaminated soil.

F.9.2 Air Transport Medium

Airborne activity may result in inhalation exposure plus direct transfer to plant surfaces resulting in intake of contaminated food crops and animal products (from animals that eat contaminated feed crops). The UDFs for air transport were based on unit air concentrations expressed in units of mg/m^3 (chemicals) and pCi/m^3 (radionuclides).

The parameter values for each exposure pathway related to air as a medium are presented in Tables F.6 through F.9 for the exposure scenarios. The assumptions associated with each of the exposure pathways are presented below.

F.9.2.1 Inhalation

For all scenarios, the individual inhales 20 m^3 of air during the time the individual is present. For the industrial worker and the recreational visitor, this volume of air is inhaled during an 8-hour period, during which the individuals are engaged in enhanced physical activity. For the residential and agricultural individuals, the air is inhaled during a 24-hour period at average daily inhalation rates. The industrial worker is exposed 250 days per year, the recreational visitor is exposed 7 days per year, and the residential and agricultural individuals are exposed 365 days per year.

F.9.2.2 Food Crops

Food crops are evaluated as fruits and vegetables. The crops were assumed to be contaminated from transfer of airborne contamination directly to the plant surfaces and the edible portions of the plant. Food crops were assumed to be eaten by the residential and the agricultural individuals. The individuals were assumed to consume 42 g/d of fruit and 80 g/d of vegetables throughout the year.

F.9.2.3 Game (Deer)

The dose for this pathway was evaluated as described in Section F.9.1 "Soil (Air Deposition) Transport Medium." Deer are assumed to be contaminated from the air transport pathway when they eat plants contaminated by direct air deposition onto plant surfaces.

F.9.2.4 Meat and Milk Ingestion

Individuals in the agricultural scenario were assumed to ingest 75 g/d of meat (other than game), and 300 g/d of dairy products (represented as milk). The animals were assumed to be exposed from eating feed crops contaminated by direct air deposition.

F.9.3 Groundwater Transport Medium

Groundwater contamination may result in exposure from domestic uses of the water (drinking and showering), and from ingestion of food crops and animal products. Food crops were assumed to become contaminated when groundwater is used for irrigation. Animal products were assumed to become contaminated when animals are fed crops irrigated with groundwater or when animals drink groundwater. The UDFs were based on unit water concentrations expressed in units of mg/L (chemicals) and pCi/L (radionuclides).

Groundwater quality at a moment in time and point in space was defined by the maximum concentration observed for each radionuclide at any of the vertical nodes associated with a spatial surface point in the aquifer model. For example, assume a point in space represents the location of a groundwater well. The nodes aligned vertically below the land surface represent pints in the aquifer providing water to the well. The maximum concentration for one radionuclide (e.g., technetium-99) might be associated with an upper node in the profile of nodes representing the nine hydrostratigraphic units of the groundwater model. The maximum concentration of another radionuclide (e.g., iodine-129) might be associated with a lower node in the profile. The maximum concentration of each radionuclide, regardless of its nodal location in the vertical profile, is used to describe the water quality at that point in space and moment in time. This groundwater quality is then applied in the exposure scenarios.

The parameter values for each exposure pathway related to groundwater as a medium are presented in Tables F.6 through F.9 for the four exposure scenarios. The assumptions for each of the exposure pathways are described below.

F.9.3.1 Drinking Water Ingestion

The industrial worker was assumed to drink 1 L/d of water while on the job 250 days per year. The recreational visitor was assumed to drink 2 L/d during 7 days per year. The residential and agricultural individuals were assumed to drink 2 L/d for the entire year. In this exposure pathway scenario, no contaminants were assumed to be removed by a treatment process.

F.9.3.2 Shower Dermal Contact

Domestic use of groundwater for bathing was included for all scenarios. Individuals are assumed to take one 10-minute shower each day of exposure. The industrial workers were assumed to be exposed 250 days per year, the recreational visitor was assumed to be exposed 7 days per year, and the residential and agricultural individuals were assumed to be exposed 365 days per year. The skin surface area for exposure was set to 20,000 cm^2. Dermal contact exposure was evaluated only for chemicals (as recommended in the HSRAM report).

F.9.3.3 Indoor Air Inhalation

Domestic use of water indoors (e.g., for showers, laundry, dishwashing, and cooking) was assumed to result in release of volatile chemicals and radionuclides into the indoor air. Individuals are then subject to inhalation exposure while indoors. This exposure pathway was considered for all exposure scenarios except the recreational visitor. (Even though the recreational visitor was assumed to use groundwater for showering at the recreational facilities, the visitor was not assumed to remain indoors for extended periods of time to allow significant inhalation exposure.) The air concentration was related to the water concentration using a volatilization factor, assumed to be 0.1 L/m^3 for radon-222, the only constituent considered to be volatile. The daily inhaled air volume for exposure to indoor contamination was assumed to be 15 m^3 for the agricultural and residential individual and 20 m^3 for the industrial worker (consistent with the daily inhalation rate for general air inhalation). The agricultural and residential individual inhalation rates were reduced slightly because the individual was not assumed to be indoors all day.

F.9.3.4 Food Crop Ingestion

Food crops are evaluated as fruits and vegetables. Food crops were assumed to become contaminated when groundwater is used for crop irrigation. The exposure pathway parameters are as described above for Section F.9.1 "Soil (Air Deposition) Transport Medium." For the groundwater irrigation route, irrigation water was assumed to be applied at a rate of 150 L/m^2/month, which corresponds to an annual average application rate of 90 cm/yr.

F.9.3.5 Game (Deer) Ingestion

For the recreational and agricultural scenarios, the individual was assumed to hunt and kill one deer in the year. The dose for this pathway was evaluated as described in Section F.9.1 "Soil (Air Deposition) Transport Medium." Deer meat was assumed to become contaminated when the deer drink contaminated seep or spring water (groundwater). Deer were not assumed to eat irrigated crops.

F.9.3.6 Meat and Milk Ingestion

Individuals in the agricultural scenario were assumed to ingest 75 g/d of meat (other than game), and 300 g/d of dairy products (represented as milk). The animals were assumed to be exposed to contaminants from drinking contaminated water and eating feed crops contaminated by irrigation deposition.

F.10 Equilibrium Analysis

Several pathways involved accumulation of radionuclides in soil over a period of time resulting from airborne deposition or from irrigation water applied on crops. Material deposited onto soil was subject to losses by leaching from the soil and radioactive decay. At equilibrium, these losses were assumed to equal the rate of deposition. The UDF values for such pathways were evaluated at a time when equilibrium had been attained.

An analysis was performed for the radionuclides of interest to determine the time necessary for equilibrium to be attained. In the analysis, two soil types were considered: agricultural soil and native soils. Properties of agricultural soil were used for the industrial, residential, and agricultural scenarios, while the native soil properties were used for recreational scenario and for game meat ingestion (agricultural scenario). The primary parameter affecting the time to equilibrium was the value used for the radionuclide distribution coefficient, K_d. Soil parameters (e.g., density) also affected the time to equilibrium, but were constant for all constituents. The agricultural soil was assumed to be represented by a 15-cm thick layer with an average bulk density of 1.5 g/cm^3. The soil moisture content was assumed to be 10%, and an average annual infiltration rate of 15 cm/yr was used. The native soil properties were assumed to be the same except the infiltration rate is set to 0.2 cm/yr.

Losses from the surface soil layer were assumed to occur by radioactive decay and leaching. A liquid or water leaching rate constant i, i.e., λ_{wi}, was evaluated from the K_d value and the parameters mentioned above as follows.

$$\lambda_{wi} = \frac{i}{h\,\theta\,(1+\frac{\beta_d}{\theta}K_{di})}$$

where i = total infiltration rate (cm/yr)
h = thickness of the surface-soil layer (cm)
θ = moisture content of the surface-soil layer (fraction)
β_d = bulk density of the surface-soil layer (g/cm^3)
K_{di} = distribution coefficient for constituent i (mL/g).

Values used for the distribution coefficient were selected to give low leach rate constants that will result in long soil retention times. This selection resulted in a conservative (high) estimate of radiation dose for those exposure pathways that involve accumulation in soil. The parameters for agricultural soil were used for all exposure pathways (except recreational activities and game ingestion), as a simplification to the analysis and a further conservatism for the residential exposure pathways. Residential soil was

expected to involve mixing in a smaller depth (represented in the above equation by parameter h). A smaller value for soil depth would result in a faster leach rate and lower equilibrium concentrations. Because of lawn irrigation, residential and industrial soils were assumed to be subject to the same infiltration rate as agricultural lands. For the recreational pathways, the infiltration rate was assumed to be 0.2 cm/yr. This value is in the midrange of reported values for Hanford (Kincaid et al. 1995). Other parameters were assumed to be unchanged with the 15-cm depth representing a nominal depth for plant roots.

Results of the equilibrium analysis are presented in Table F.10. This table indicates the time to reach equilibrium (95% of the final value) when the radionuclide is deposited at a constant rate during prior years. The analysis is based on calculation of UDFs for each radionuclide. For radionuclides that have decay progeny, the dose from the progeny was included in the analysis. In some cases, the ingrowth of progeny (none are present at time zero or in the water) may extend the time to reach the equilibrium dose. Table F.10 also presents the ratio of the annual dose at 50 years to the equilibrium dose (or the dose at 1000 years, the time considered for this iteration of the Composite Analysis).

The UDFs generated (presented in Section F.11) were based on the dose received after the equilibrium time period. For those radionuclides that take more than 50 years to reach equilibrium, the dose was evaluated at 50 years. This simplification was necessary because the UDFs are based on constant deposition (i.e., constant air or water concentration) over the deposition period. The simplification was also needed in order to precalculate the UDF values. The precalculated UDF values were evaluated for a unit concentration or deposition rate and did not include consideration of parameter variation with time.

F.11 Unit Dose Factors and Hazard Indices

Unit dose factors were generated for the media contributing to each of the four exposure scenarios. The groundwater transport medium plays the dominant role. However, the air and soil transport media also contribute to the agricultural scenario where air and soil contaminant levels are estimated (i.e., atmospheric releases from the buried graphite cores of the production reactors). The UDFs are summed over all exposure pathways for a medium. Table F.11 presents the summed UDF results for all radionuclides for which doses were calculated. It also presents the summed UDF results for chemical health impacts of uranium for each exposure scenario. These latter values provide the hazard index for uranium exposures. A hazard index of 1.0 indicates the exposure is just at the safe level. Hazard index values greater than 1.0 indicate higher exposures. The radionuclide UDF values provide annual radiation dose expressed in mrem.

F.12 References

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Table F.1. Radionuclides and Chemicals of Interest in the Composite Analysis

Radionuclides	Progeny
^{3}H	
^{14}C	
^{36}Cl	
^{40}K	
^{63}Ni	
^{79}Se	
^{90}Sr	^{90}Y
^{99}Tc	
^{126}Sn	^{126}Sb
^{129}I	
^{187}Re	
^{234}U	
^{235}U	^{231}Pa, ^{227}Ac, ^{227}Th, ^{223}Ra
^{238}U	^{234}Th, ^{230}Th, ^{226}Ra, ^{222}Rn, ^{210}Pb, ^{210}Bi, ^{210}Po
^{244}Cm	^{240}Pu
Chemicals	
Uranium	

Table F.2. Industrial Scenario Exposure Pathways

Transport Medium	Exposure Pathway	Chemical	Radioactive
Soil (air deposition)	Ingestion	Yes	Yes
	External	No	Yes
	Dermal Contact	Yes	No
	Suspension – Inhalation	Yes	Yes
Air	Inhalation	Yes	Yes
Groundwater	Ingestion	Yes	Yes
	Dermal Contact	Yes	No

Table F.3. Recreational Scenario Exposure Pathways

Transport Medium	Exposure Pathway	Chemical	Radioactive
Soil (air deposition)	Ingestion	Yes	Yes
	External	No	Yes
	Dermal Contact	Yes	No
	Suspension - Inhalation	Yes	Yes
	Biota - game (deer)	Yes	Yes
Air	Inhalation	Yes	Yes
	Biota - game (deer)	Yes	Yes
Groundwater	Ingestion	Yes	Yes
	Biota - game (deer)	Yes	Yes
	Dermal Contact (bathing)	Yes	No

Table F.4. Residential Scenario Exposure Pathways

Transport Medium	Exposure Pathway	Chemical	Radioactive
Soil (air deposition)	Ingestion	Yes	Yes
	External	No	Yes
	Dermal Contact	Yes	No
	Biota – Fruit	Yes	Yes
	Biota – Vegetables	Yes	Yes
	Suspension - Inhalation	Yes	Yes
Air	Inhalation	Yes	Yes
	Biota – Fruit	Yes	Yes
	Biota – Vegetables	Yes	Yes
Groundwater	Ingestion	Yes	Yes
	Dermal Contact (bathing)	Yes	No
	Biota – Fruit	Yes	Yes
	Biota – Vegetables	Yes	Yes

Table F.5. Agricultural Scenario Exposure Pathways for Radionuclides and Chemicals

Transport Medium	Exposure Pathway	Chemical	Radioactive
Soil (air deposition)	Ingestion	yes	yes
	External	no	yes
	Dermal Contact	yes	no
	Biota – Dairy	yes	yes
	Biota – Meat	yes	yes
	Biota - Game (deer)	yes	yes
	Biota – Fruit	yes	yes
	Biota – Vegetables	yes	yes
	Suspension – Inhalation	yes	yes
Air	Inhalation	yes	yes
	Biota – Dairy	yes	yes
	Biota – Meat	yes	yes
	Biota - Game (deer)	yes	yes
	Biota – Fruit	yes	yes
	Biota – Vegetables	yes	yes
Groundwater	Ingestion	yes	yes
	Dermal Contact (bathing)	yes	no
	Biota – Dairy	yes	yes
	Biota – Meat	yes	yes
	Biota - Game (deer)	yes	yes
	Biota – Fruit	yes	yes
	Biota – Vegetables	yes	yes
	Inhalation indoor	yes	yes (Rn)

Table F.6. Industrial Scenario Exposure Factors

Pathway		Exposure Parameters[a]				Summary Intake Factor	
Media	Exposure Route	Intake Rate	Exposure Frequency (d/yr)	Conversion Factors	Other Factors	Chemical Noncarcinogens	Radionuclides
Soil[b]	Ingestion	50 mg/d	146	1E-06 kg/mg 60 kg/m^2 [c]	--	4.76E-09 m^2 soil/(kg-d)	1.22E-04 m^2 soil
	External	8 h/d	146	--	0.8	--	9.34E+02 h
	Dermal	0.2 mg/cm^2-d	146	1E-06 kg/mg 60 kg/m^2	5000 cm^2	9.52E-08 m^2 soil/(kg-d)	--
	Inhalation	20 m^3/d	250	1E-09 kg/µg 60 kg/m^2	50 µg/m^3	1.63E-10 m^2 soil/(kg-d)	4.16E-06 m^2 soil
Air[d]	Inhalation	20 m^3/d	250	--	--	1.96E-01 m^3 / (kg-d)	5.00E+03 m^3 air
Groundwater[e]	Ingestion	1 L/d	250	--	--	9.78E-03 L / (kg-d)	2.50E+02 L
	Inhalation	20 m^3/d	250	--	0.5 L/m^3 chemicals 0.1 L/m^3 radon	9.78E-02 L / (kg-d)	5.00E+02 L
	Dermal	0.17 h/d	250	1E-03 L/cm^3	20,000 cm^2	3.33E-02 L h / (kg-d-cm)	--

(a) For all cases, the body weight is 70 kg and exposure is for 1 year.
(b) Units for soil concentration are pCi/kg dry soil for radionuclides, and mg/kg for chemicals.
(c) The factor 60 kg/m^2 is to convert soil concentration between mass (kg) and area (m^2).
(d) Units for air concentration are pCi/m^3 for radionuclides, and mg/m^3 for chemicals.
(e) Units for water concentration are pCi/L for radionuclides, and mg/L for chemicals.

Table F.7. Recreational Scenario Exposure Factors

Pathway		Exposure Parameters[a]				Summary Intake Factor	
Media	Exposure Route	Intake Rate	Exposure Frequency (d/yr)	Conversion Factors	Other Factors	Chemical Noncarcinogens	Radionuclides
Soil[b]	Ingestion	100 mg/d	7	1E-06 kg/mg 60 kg/m^2 [c]	--	4.57E-10 m^2 soil/(kg-d)	1.17E-05 m^2 soil
	External	8 h/d	7	--	0.8	--	4.49E+01 h
	Dermal	0.2 mg/cm^2-d	7	1E-06 kg/mg 60 kg/m^2	5000 cm^2	4.57E-09 m^2 soil/(kg-d)	--
	Inhalation	20 m^3/d	7	1E-09 kg/µg 60 kg/m^2	50 µg/m^3	4.57E-12 m^2 soil/(kg-d)	1.17E-08 m^2 soil
Air[d]	Inhalation	20 m^3/d	7	--	--	5.49E-03 m^3 / (kg-d)	1.40E+02 m^3 air
Groundwater[e]	Ingestion	2 L/d	7	--	--	5.49E-04 L / (kg-d)	1.40E+01 L
	Dermal	0.17 h/d	7	1E-03 L/cm^3	20,000 cm^2	9.33E-04 L h / (kg-d-cm)	--
Biota[f]	Deer	15 g/d	365	1E-03 kg/g	--	2.14E-04 kg deer/(kg-d)	5.48 kg deer/(kg-d)

(a) For all cases, the body weight is 70 kg and exposure is for 1 year.
(b) Units for soil concentration are pCi/kg dry soil for radionuclides, and mg/kg for chemicals.
(c) The factor 60 kg/m^2 is to convert soil concentration between mass (kg) and area (m^2).
(d) Units for air concentration are pCi/m^3 for radionuclides, and mg/m^3 for chemicals.
(e) Units for water concentration are pCi/L for radionuclides, and mg/L for chemicals.
(f) Units for food concentration are pCi/kg wet food for radionuclides, and mg/kg for chemicals.

Table F.8. Residential Scenario Exposure Factors

Pathway		Exposure Parameters[a]				Summary Intake Factor	
Media	Exposure Route	Intake Rate	Exposure Frequency (d/yr)	Conversion Factors	Other Factors	Chemical Noncarcinogens	Radionuclides
Soil[b]	Ingestion	100 mg/d	365	1E-06 kg/mg 60 kg/m^2 [c]	--	2.38E-08 m^2 soil/(kg-d)	6.09E-04 m^2 soil
	External	24 h/d	365	--	0.8	--	7.03E+03 h
	Dermal	0.2 mg/cm^2-d	180	1E-06 kg/mg 60 kg/m^2	5000 cm^2	1.17E-07 m^2 soil/(kg-d)	--
	Inhalation	20 m^3/d	365	1E-09 kg/µg 60 kg/m^2	50 µg/m^3	2.38E-10 m^2 soil/(kg-d)	6.08E-06 m^2 soil
Air[d]	Inhalation	20 m^3/d	365	--	--	2.86E-01 m^3 / (kg-d)	7.31E+03 m^3 air
Groundwater[e]	Ingestion	2 L/d	365	--	--	2.86E-02 L/ (kg-d)	7.31E+02 L
	Inhalation	15 m^3/d	365	--	0.5 L/m^3 chemicals 0.1 L/m^3 radon	1.07E-01 L/ (kg-d)	5.48E+02 L
	Dermal	0.17 h/d	365	1E-03 L/cm^3	20,000 cm^2	4.86E-02 L h/ (kg-d-cm)	--
Biota[f]	Fruit	42 g/d	365	1E-03 kg/g	--	6.00E-04 kg food /(kg-d)	1.53E+01 kg food
	Vegetable	80 g/d	365	1E-03 kg/g	--	1.14E-03 kg food /(kg-d)	2.92E+01 kg food

(a) For all cases, the body weight is 70 kg and exposure is for 1 year.
(b) Units for soil concentration are pCi/kg dry soil for radionuclides, and mg/kg for chemicals.
(c) The factor 60 kg/m^2 is to convert soil concentration between mass (kg) and area (m^2).
(d) Units for air concentration are pCi/m^3 for radionuclides, and mg/m^3 for chemicals.
(e) Units for water concentration are pCi/L for radionuclides, and mg/L for chemicals.
(f) Units for food concentration are pCi/kg wet food for radionuclides, and mg/kg for chemicals.

Table F.9. Agricultural Scenario Exposure Factors

Pathway		Exposure Parameters[a]				Summary Intake Factor	
Media	Exposure Route	Intake Rate	Exposure Frequency (d/yr)	Conversion Factors	Other Factors	Chemical Noncarcinogens	Radionuclides
Soil[b]	Ingestion	100 mg/d	365	1E-06 kg/mg 60 kg/m^2 [c]	--	2.38E-08 m^2 soil/(kg-d)	6.09E-04 m^2 soil
	External	24 h/d	365	--	0.8	--	7.03E+03 h
	Dermal	0.2 mg/cm^2-d	180	1E-06 kg/mg 60 kg/m^2	5000 cm^2	1.17E-07 m^2 soil/(kg-d)	--
	Inhalation	20 m^3/d	365	1E-09 kg/µg 60 kg/m^2	50 µg/m^3	2.38E-10 m^2 soil/(kg-d)	6.08E-06 m^2 soil
Air[d]	Inhalation	20 m^3/d	365	--	--	2.86E-01 m^3 /(kg-d)	7.31E+03 m^3 air
Groundwater[e]	Ingestion	2 L/d	365	--	--	2.86E-02 L /(kg-d)	7.31E+02 L
	Inhalation	15 m^3/d	365	--	0.5 L/m^3 chemicals 0.1 L/m^3 radon	1.07E-01 L /(kg-d)	5.48E+02 L
	Dermal	0.17 h/d	365	1E-03 L/cm^3	20,000 cm^2	4.86E-02 L h /(kg-d-cm)	--
Biota[f]	Dairy	300 g/d	365	1E-03 kg/g	--	4.29E-03 kg food /(kg-d)	1.10E+02 kg food
	Beef	75 g/d	365	1E-03 kg/g	--	1.07E-03 kg food /(kg-d)	2.74E+01 kg food
	Game	15 g/d	365	1E-03 kg/g	--	2.14E-04 kg food /(kg-d)	5.48E+0 kg food
	Fruit	42 g/d	365	1E-03 kg/g	--	6.00E-04 kg food /(kg-d)	1.53E+01 kg food
	Vegetable	80 g/d	365	1E-03 kg/g	--	1.14E-03 kg food /(kg-d)	2.92E+01 kg food

(a) For all cases, the body weight is 70 kg and exposure is for 1 year.
(b) Units for soil concentration are pCi/kg dry soil for radionuclides, and mg/kg for chemicals.
(c) The factor 60 kg/m^2 is to convert soil concentration between mass (kg) and area (m^2).
(d) Units for air concentration are pCi/m^3 for radionuclides, and mg/m^3 for chemicals.
(e) Units for water concentration are pCi/L for radionuclides, and mg/L for chemicals.
(f) Units for food concentration are pCi/kg wet food for radionuclides, and mg/kg for chemicals.

Table F.10. Time to Reach the Equilibrium Annual Dose

	Time to Reach 95% of Equilibrium Dose, in years (Ratio of 50-Year Dose to Equilibrium Year Dose, or 1000-year dose)			
	Agricultural Soil Retention		Native Soil Retention	
Constituent	Soil (air) Pathways	Agricultural (air) Pathways	Recreational Soil Pathways	Game Meat Ingestion
Uranium	16 (1.0)	31 (0.99)	>1000 (0.13)	>1000 (0.088)
^{3}H	1 (1.0)	zero dose	27 (1.0)	zero dose
^{14}C	31 (0.99)	zero dose	>1000 (0.090)	zero dose
^{36}Cl	5 (1.0)	5 (1.0)	>1000 (0.34)	>1000 (0.034)
^{40}K	44 (0.97)	45 (0.96)	>1000 (0.074)	>1000 (0.075)
^{79}Se	9 (1.0)	9 (1.0)	>1000 (0.20)	>1000 (0.020)
^{99}Tc	9 (1.0)	9 (1.0)	>1000 (0.20)	>1000 (0.020)
^{129}I	70 (0.89)	70 (0.89)	>1000 (0.066)	>1000 (0.066)
^{233}U	>1000 (0.35)	>1000 (0.21)	>1000 (0.044)	>1000 (0.078)
^{234}U	31 (0.99)	31 (0.99)	>1000 (0.086)	>1000 (0.086)
^{235}U	>1000 (0.89)	>1000 (0.94)	>1000 (0.075)	>1000 (0.076)
^{236}U	31 (0.99)	32 (0.99)	>1000 (0.086)	>1000 (0.086)
^{238}U	32 (0.99)	31 (0.99)	>1000 (0.086)	>1000 (0.086)
^{237}Np	120 (0.73)	120 (0.74)	>1000 (0.060)	>1000 (0.060)
^{234}Th	1 (1.0)	7 (1.0)	1 (1.0)	>1000 (0.38)
^{231}Th	>1000 (0.49)	>1000 (0.035)	>1000 (0.29)	>1000 (0.031)
^{230}Th	>1000 (0.0095)	>1000 (0.0027)	>1000 (0.012)	>1000 (0.0017)
^{229}Th	>1000 (0.056)	>1000 (0.056)	>1000 (0.053)	>1000 (0.053)
^{227}Th	1 (1.0)	1 (1.0)	1 (1.0)	1 (1.0)
^{223}Pa	1 (1.0)	>1000 (0.49)	1 (1.0)	>1000 (0.18)
^{231}Pa	>1000 (0.034)	>1000 (0.035)	>1000 (0.032)	>1000 (0.031)
^{225}Ra	1 (1.0)	1 (1.0)	1 (1.0)	1 (1.0)
^{227}Ac	100 (0.80)	100 (0.80)	100 (0.80)	100 (0.80)
^{225}Ac	1 (1.0)	1 (1.0)	1 (1.0)	1 (1.0)
^{226}Ra	>1000 (0.10)	>1000 (0.059)	>1000 (0.060)	>1000 (0.036)
^{223}Ra	1 (1.0)	1 (1.0)	1 (1.0)	1 (1.0)
^{210}Pb	100 (0.80)	100 (0.79)	100 (0.79)	100 (0.79)
^{210}Bi	2 (1.0)	2 (1.0)	2 (1.0)	2 (1.0)
^{210}Po	2 (1.0)	2 (1.0)	2 (1.0)	2 (1.0)

Table F.11. Unit Dose Factors for Radionuclides Contributing to Dose and Uranium as a Chemical Hazard

	Scenario					
	Residential	Industrial	Recreational	Agricultural		
Radionuclide	Groundwater [mrem/pCi/L]	Groundwater [mrem/pCi/L]	Groundwater [mrem/pCi/L]	Groundwater [mrem/pCi/L]	Air [mrem/pCi/m^3]	Soil [mrem/pCi/m^2]
Tritium	4.85E-05	1.57E-05	1.05E-06	5.69E-05	1.25E-03	4.18E-11
Carbon-14	1.52E-02	5.22E-04	2.99E-05	4.09E-02	2.53E-01	2.93E-08
Chlorine-36	1.76E-02	7.58E-04	5.29E-05	1.08E-01		
Selenium-79	6.77E-03	2.17E-03	1.28E-04	1.21E-02		
Strontium-90	2.53E-01	3.58E-02	2.01E-03	3.12E-01		
Technetium-99	1.36E-03	3.65E-04	2.10E-05	3.66E-03		
Iodine-129	2.27E-01	6.90E-02	3.95E-03	6.19E-01		
Uranium[a]	[mrem/(µg/L)] 1.69E-01	[mrem/(µg/L)] 5.27E-02	[mrem/(µg/L)] 2.96E-03	[mrem/(µg/L)] 1.86E-01		
Uranium[b]	[hazard index/µg/L] 1.08E-02	[hazard index/µg/L] 3.48E-03	[hazard index/µg/L] 1.89E-04	[hazard index/µg/L] 1.19E-02		

(a) Uranium modeled as contributing to radiation dose.
(b) Uranium modeled as contributing to chemical hazard.

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