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Hydrogeologic Performance Assessment Analysis of the Low-Level Radioactive Waste Disposal Facility near Sheffield, Illinois

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Pacific Northwest Laboratory
Operated by
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Pacific Northwest Laboratory
Richland, WA 99352

Prepared for
Division of Low-Level Waste Management and Decommissioning
Office of Nuclear Material Safety and Safeguards
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ABSTRACT

A hydrogeologic performance assessment was conducted for the commercial low-level radioactive waste disposal site located about 3 mi southwest of the town of Sheffield, in Bureau County, northwestern Illinois. The site has 21 trenches, which contain about 900,000 m^3, of buried waste and about 60,000 Ci of nuclear by-product material. The disposal trenches cut through a complex series of Quaternary deposits, and are composed primarily of silts, clays, and sands. Ground water beneath the site, which ranges in depth from 1.5 to 14 m, generally moves in two directions: northeast to east toward a strip-mine lake and south to southeast toward small tributary channels belonging to Lawson Creek, which eventually drains into the strip-mine lake southeast of the site.

The results in the performance assessment, which focused on the site ground-water pathway, suggest that tritium, ^{90}Sr, and ^{14}C would be the only radionuclides released from the Sheffield site in any significant concentrations. A comparison of simulated tritium concentrations east of the site in the time frame of the burial history would suggest that model results are greater than the highest measured values by a factor of 2 or 3. The discrepancy between actual and predicted concentrations likely reflects errors in the assumed tritium inventory estimates, availability in the inventory, and/or the actual release from the multitude of waste forms considered in the performance assessment. A comparison of transport results for ^{90}Sr and ^{14}C is not possible since neither has been detected in ground water near the site.

SUMMARY

The Sheffield low-level radioactive waste disposal site is located about 3 mi southwest of the town of Sheffield, in Bureau County, northwestern Illinois. The site is located on 20 acres of rolling terrain owned by the state of Illinois. The site contains 21 trenches, which differ in size and shape. The site first began receiving wastes in 1967 and operated until closure in 1978. The waste buried at the site totaled about 900,000 m^3, containing about 60,000 Ci of nuclear by-products.

The disposal trenches cut through a complex series of Quaternary deposits, which range in thickness from 3 to 20 m and are composed primarily of silts, clays, and sands. Depth to ground water beneath the site ranges from 1.5 to 14 m. Ground-water movement is generally in two predominant directions: northeast to east toward a strip-mine lake and south to southeast toward small tributary channels belonging to Lawson Creek, which eventually drains into the strip-mine lake southeast of the site. The most significant hydrogeologic unit on the site is a pebbly-sand unit in the Toulon Member of the Glasford Formation, which grades into a coarse gravel with sand and pebbles east of the disposal site. East of the site, a narrow, channel-like depression in the Hulick Till Member is filled with coarse gravelly sand of the pebbly-sand unit of the Toulon Member, which hydraulically connects the site and the nearby strip-mine lake.

Results of ground-water sampling on the Sheffield site indicate that only tritium has migrated from the site. Tritium was first observed in 1982 and has since been detected along the entire length of the buried channel leading east of the site to the strip-mine lake and in seeps along the banks of the strip-mine lake. Tritium concentrations do not vary significantly from one location to another along the narrow channel, which suggests that migration in the channel is rapid. Tritium has been found near some trenches along the south end of the burial area; however, with the exception of one well, offsite migration of tritium to the south has not been as extensive as northeast of the site. Tritium concentrations in ground water have ranged from below detection level (about 20 pCi/L) to more than 300,000 pCi/L.

In this study, the performance assessment focused on the site ground-water pathway; thus, no analysis was performed to evaluate the transport of radionuclides along the remaining pathways. Neither was this performance assessment analysis intended to be a basis for determining Sheffield site compliance with the licensing regulations put forth in 10 CFR 61. However, final model results of key radionuclide peak concentrations are compared to dose equivalent concentrations consistent with the regulatory limit of 25-mrem/yr whole body dose.

A two-dimensional ground-water flow model of the site was previously developed in U.S. Geological Survey (USGS) site studies by Garklavs and Healy (1986). The distribution of the high-permeability Toulon pebbly-sand unit proved to be an important factor in the calibration of the USGS two-dimensional ground-water flow model and in the distribution of travel paths

from the trenches inferred from those modeling results. For this reason, a three-dimensional ground-water flow model was constructed that more accurately represented the horizontal and vertical distribution of the pebbly-sand unit.

The three-dimensional model covered approximately the same area as the previous two-dimensional model but consisted of three model layers: Layer 1 consisted of the Cahokia Alluvium, Peoria Loess, Roxana Silt, Berry Clay, and Radnor Till; Layer 2 consisted of the pebbly-sand unit of the Toulon member; and Layer 3 consisted of the other units of the Toulon Member, the Hulick Till, and the Duncan Mills Silt.

The resulting estimates of travel time and distance traveled by ground water beneath the disposal area, as predicted by both the two-dimensional and three-dimensional ground-water flow models, were used in conjunction with the site waste inventory to simulate site performance. The transport code used in the analysis, TRANSS, is based on the one-dimensional stochastic-convective theory of transport and relates dispersion directly to the variation observed in the travel time from the source. The code also features radioactive decay of contaminants in both the waste source and the ground-water system, a general empirical description of contaminant release, and a choice of three optional release models: 1) a constant fractional release rate, 2) a concentration-limited release based on chemical solubility, and 3) an adsorption equilibrium release based on the K_d value for each nuclide.

After some initial screening calculations, radionuclides with insufficient characterization, low activity totals, or short half-lives (e.g., ^{26}Al, ^{63}Ni, ^{94}Nb, ^{99}Tc, ^{99m}Tc, ^{36}Cl, ^{51}Cr, ^{110m}Ag, ^{131}I, ^{125}I, ^{32}P, ^{35}S, ^{75}Se, ^{237}Np, ^{252}Cf, and ^{134}Cs) were not included in the analysis. Other radionuclides with affinities for adsorption onto site soil and short half-lives (e.g., ^{137}Cs, ^{60}Co, ^{241}Pu, ^{241}Am, ^{59}Mn, ^{59}Fe, ^{65}Zn, and ^{95}Zr) were also not considered.

The results in the performance assessment suggest that tritium, ^{90}Sr, and ^{14}C would be the only radionuclides released from the Sheffield site in any significant concentrations. To date, no radionuclides, with the exception of tritium, have been detected outside the Sheffield disposal area. Tritium concentrations detected east of the site do not vary significantly from one location to another along the narrow channel, and range from 10,000 to 100,000 pCi/L. A comparison of simulated tritium concentrations east of the site in the time frame of the burial history would suggest that model results are greater than the highest measured values by a factor of 2 or 3. Model results within a 20-yr time frame give tritium concentrations on the order of two or three hundred thousand pCi/L, whereas measured concentrations range from 10,000 to 100,000 pCi/L. The discrepancy between actual and predicted concentrations likely reflects errors in the assumed tritium inventory estimates, availability in the inventory, and/or the actual release from the multitude of waste forms considered in the performance assessment. The arrival and the magnitude of the modeled peak concentrations from Trench 23 suggests that the observed tritium migration may be due to tritium leaching from this trench waste alone. The release of tritium in the model suggests that tritium concentrations in the channel will increase as tritium leached from other trenches in the burial area migrate off site.

A comparison of transport results for ^{90}Sr and ^{14}C was not possible since neither has been detected in ground water near the site. Predicted peak concentrations of ^{90}Sr using a conservative distribution coefficient of 3.4 ml/g were well above regulatory standards. Predicted ^{90}Sr concentrations were as high as 6790 pCi/L after about 120 yr. These predicted concentrations are significantly reduced if more moderate K_ds (4 to 7 ml/g) and/or waste form containment are considered. For their given range in uncertainty, transport results were far more sensitive to changes in these parameters than others such as the infiltration rate and effective porosity.

Carbon-14 concentrations of just under 1 pCi/L, as controlled by a prescribed solubility limit in the model analysis, are predicted to occur within a period of about 10 yr. A peak value of just over 60 pCi/L is estimated after 110 to 120 yr. The levels of ^{14}C predicted offsite before 10 yrs are inconsistent with the lack of ^{14}C detection in site ground water. The lack of measured ^{14}C in ground water suggests that actual leaching and migration of ^{14}C is much less than was assumed in the model analysis.

Results between the two-dimensional and three-dimensional ground-water models were very similar for all three radionuclides, although results from the three-dimensional model produced slightly lower peak concentrations and the release occurred over a short period of time. For this site and the approach used in the three-dimensional model, changes in the overall performance assessment were not significant.

ABBREVIATIONS

ASLB	Atomic Safety and Licensing Board
BNL	Brookhaven National Laboratory
CFEST	Coupled Flow, Energy, and Solute Transport
CFR	Cumulative fractional release
DOT	U.S. Department of Transportation
EPA	U.S. Environmental Protection Agency
IDNS	Illinois Department of Nuclear Safety
ISGS	Illinois State Geological Survey
IFR	Incremental fractional release
LLW	Low-level radioactive waste
MSE	Mean square error
NECO	Nuclear Engineering Company
NMSS	Office of Nuclear Material Safety and Safeguards (NRC)
NRC	U.S. Nuclear Regulatory Commission
PNL	Pacific Northwest Laboratory
RSR	Radioactive shipping records
SNM	Special nuclear material
TRU	Transuranic
USE	U.S. Ecology, Inc.
USGS	U.S. Geological Survey
WRD	Water Resources Division

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FIGURES

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1.0 INTRODUCTION

In December 1982, the U.S. Nuclear Regulatory Commission (NRC) published regulations establishing specific evaluation criteria for licensing the land disposal of low-level radioactive waste (LLW). These regulations are contained in Title 10 of the Code of Federal Regulations, Part 61 (10 CFR 61), "Licensing Requirements for Land Disposal of Radioactive Waste." A specific requirement of these regulations indicated in Section 61.5, Subpart D, states that any future LLW site..." shall be capable of being characterized, modeled, analyzed, and monitored...." Implicit in these requirements is that applications for future LLW sites include sufficient information and analyses to provide reasonable assurance that performance objectives in the regulations will be met. These analyses will likely include the use of transport models for prediction of radionuclide transport along the ground-water pathway from the prospective facility. To date, no LLW sites have been licensed under 10 CFR 61; thus, the NRC technical staff can only speculate on the degree of complexity needed to model the performance of future LLW sites along the ground-water pathway with respect to the evaluation criteria of 10 CFR 61.

In May 1987, Pacific Northwest Laboratory (PNL), under a program being sponsored by the Division of Waste Management, Geotechnical Branch (WMGT) of the NRC, began providing technical assistance to the NRC to develop performance assessment capabilities. These capabilities will enable the NRC to evaluate the adequacy of future LLW sites currently being sited following the congressional mandate in the Low-Level Radioactive Waste Policy Amendments Act of 1985 (LLRWPAA). As a part of this program, PNL is applying commonly used performance assessment models at existing commercial LLW sites to gain experience and insight in regulatory performance assessments. Application of these models to existing commercial LLW sites should contribute significantly to understanding the level of detail and conservatism that will be needed to meet these regulatory requirements and to identify geologic, hydrogeologic, geochemical, and geophysical issues that affect NRC licensing decisions and evaluations of LLW disposal sites under 10 CFR 61.

The overall objective of PNL's program is to provide technical support to the NRC on the geologic, hydrogeologic, geochemical, and geophysical aspects of LLW disposal. Specific goals of this project include the following:

- develop detailed reports describing the geoscience characteristics and general disposal design for existing commercial LLW disposal sites; sites initially being evaluated are the commercial LLW disposal facilities at Sheffield, Illinois, and West Valley, New York

- develop a computerized geoscience database system for these existing commercial LLW disposal sites; this database system will be used by the NRC staff to analyze ground-water flow and transport at existing sites and to provide prelicensing guidance to states based on past experience for LLW disposal

- select and apply hydrogeologic codes to existing commercial LLW disposal sites to gain insight into modeling strategies, data needs, and levels of detail needed to adequately model proposed LLW sites with respect to 10 CFR 61

- recommend and supply modeling codes as needed for NRC staff to evaluate the performance of proposed LLW sites; these performance assessment capabilities will be used to support findings on specific licensing requirements of 10 CFR 61.

The purpose of this report is to present a summary of past investigations conducted at the Sheffield LLW site that are pertinent to site performance assessment and to describe the performance assessment approach that was used to evaluate the long-term transport of radionuclides through the subsurface at the site. The report summarizes relevant environmental characteristics and describes past performance of the Sheffield LLW site. This report also includes a review of pertinent past investigations and studies that have been conducted at the Sheffield site by the site operations contractor and various state and federal agencies. The information compiled in this report was assembled in a computerized geoscience database developed for the Sheffield site. This information provided the basis for the selection and application of appropriate codes for analysis of the ground-water pathway at the Sheffield site.

2.0 DESCRIPTION OF DISPOSAL FACILITY

2.1 PHYSICAL SETTING

The Sheffield LLW disposal site is located about 5 km (3 mi) southwest of the town of Sheffield, in Bureau County, northwestern Illinois (Figure 2.1). Sheffield is south of Interstate 80, approximately 195 km (120 mi) west of Chicago, and 70 km (45 mi) east of Moline, Illinois. The site is located on 20 acres of rolling terrain owned by the state of Illinois (Figure 2.2). The site contains 21 trenches, which differ in size and shape; the depth of the trenches generally ranges from 3.5 to 8 m. The site first began receiving wastes in 1967 and operated until it was closed in 1978. The waste buried at the site totaled about 90,000 m^3, containing about 60,000 Ci

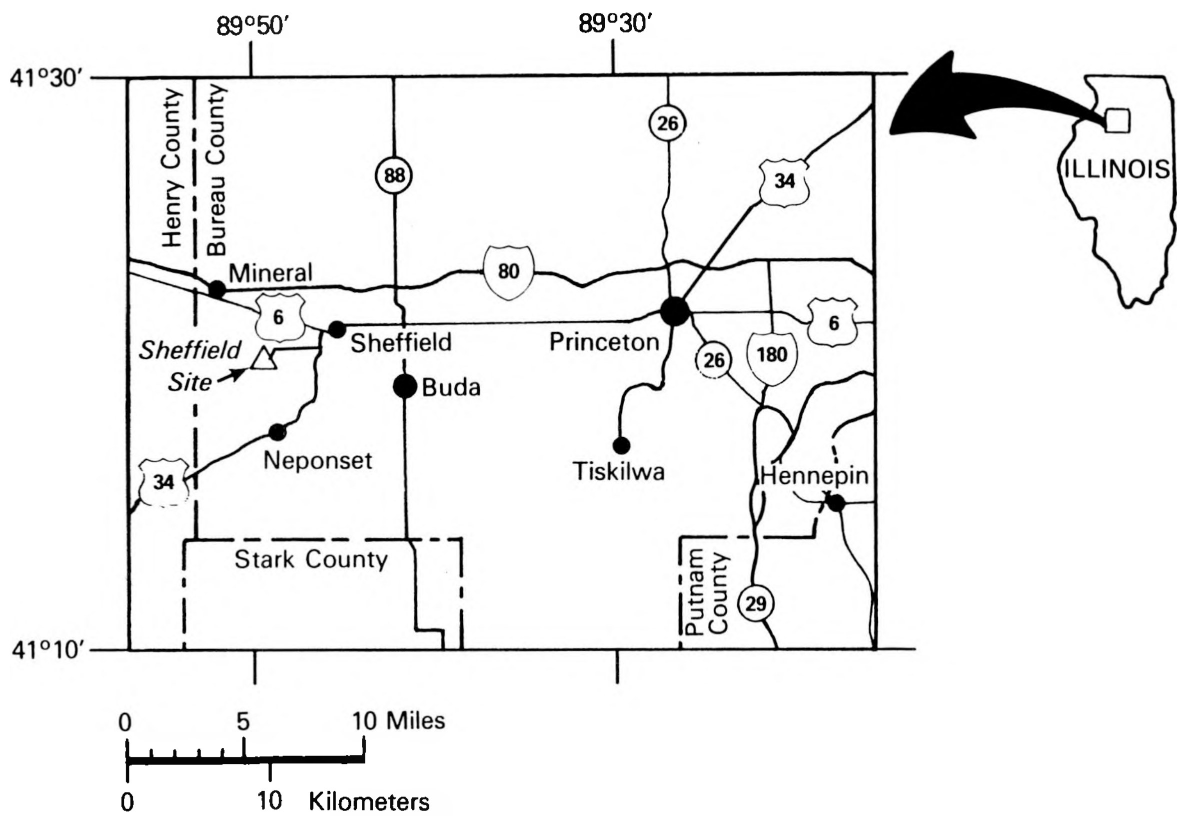


FIGURE 2.1. Location of the Commercial Low-Level Radioactive Waste Disposal Facility Near Sheffield, Illinois

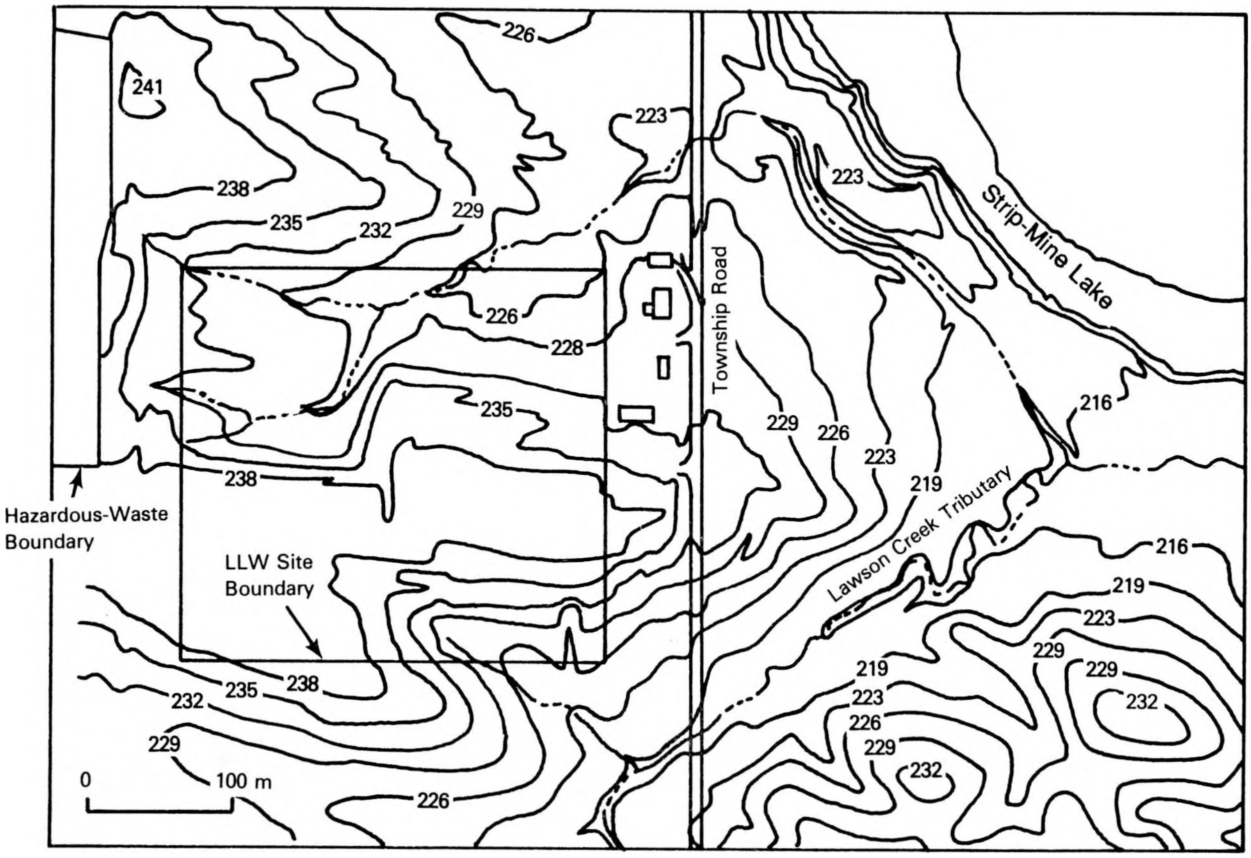


FIGURE 2.2. Topography of the Sheffield Site

of nuclear by-products. The area surrounding the site is owned by the site operator, and is used for industrial waste disposal (16 ha) and farm leases (48.8 ha). To the north, the site is bordered by rolling terrain, abandoned strip mines, and an industrial waste disposal site operated by US Ecology (USE). A 61-m buffer zone separates the LLW disposal site from the industrial waste disposal site. Farther to the north and to the northeast and northwest are abandoned coal pits, now filled with water.

2.2 SITE HISTORY

The Sheffield site was licensed for the disposal of LLW in 1967 and was originally operated by California Nuclear, Inc. But in 1968 the site license was transferred to the Nuclear Engineering Company (NECO) (Clancy et al. 1981). NECO, now known as USE, operated the site from 1968 to the present. No waste has been accepted since the site was closed in April of 1978. The possession and disposal of source, by-product, and special nuclear material at the site was licensed by the NRC until June 1, 1987. At that time, Illinois became an Agreement State, and licensing authority was transferred to the state of Illinois. The state had previously issued a license to USE for the possession and disposal of radioactive material not regulated under the Atomic Energy Act of 1954 (Clancy et al. 1981).

Radioactive waste at Sheffield was buried between August 1967 and April 1978. According to Clancy et al. (1981), approximately 90,000 m^3 of solid LLW containing over 60,200 Ci of by-product material was disposed at the site. The historic annual additions and total volume of LLW buried at Sheffield are summarized in Table 2.1.

Anticipating running out of licensed disposal area, USE filed an application to NRC and the state in 1976 for site expansion from 20 acres to 188 acres. The last available trench was filled in April 1978. An Atomic Safety and Licensing Board (ASLB) was established to review the NRC application for site expansion. After a delay of almost 2 yr, USE requested suspension of the licensing proceeding in December 1978. USE was allowed to withdraw the application for expansion by the ASLB but not for renewal of the operating license. In March 1979, USE attempted to unilaterally terminate the NRC license, the state license, and the state lease, in order to abandon the site. Both the NRC and the state ordered USE to return to the site; however, USE took the position that it was under no obligation to obey the orders, since it was no longer a licensee of the state and NRC. The state filed suit and won a preliminary injunction ordering USE back to the site until a final settlement is developed regarding the transfer of the site to the state and the conditions of the site when it is transferred. Although the legal issues between the state of Illinois and the NRC were resolved in 1987, the lawsuit between the state of Illinois and USE is still pending.

Ground water from selected wells and trench sumps is still being monitored at the Sheffield site for radionuclides and organic and inorganic constituents. Most of these recent data have not been interpreted or published. Closure (grouting) of onsite wells and a subtrench research tunnel installed by the U.S. Geological Survey (USGS) began in the fall of 1987. Onsite wells and trench sumps installed by USE are still available for monitoring. Post-closure plans for the site will include construction of a compacted clay cap that will extend over the entire LLW site. Offsite wells, installed by the USGS, have been transferred to the state of Illinois. The Illinois

TABLE 2.1. Historic Annual Additions of Low-Level Radioactive Waste Buried at Sheffield (Fischer 1986)

Year	Volume (thousands of m^3)
1975	55.5
1976	13.5
1977	17.7
1978	1.8
1979	--[a] (closed in March 1979)
	88.5

(a) -- no data available.

Department of Nuclear Safety and the U.S. Environmental Protection Agency (EPA) will continue to monitor ground water in the vicinity of the site.

2.3 TRENCH CONSTRUCTION

A total of 21 trenches have been used for waste disposal at the Sheffield site (Figure 2.3). The dimensions and volumes of each trench are shown in Table 2.2. The dimensions vary as follows: the lengths range from 11 to 177 m; the widths from 2.5 to 21 m; and the depths from 3.5 to 8 m. If Trenches 8A and 8B are disregarded, then the average length, width, and depth of the trench is 105 m x 15 m x 6 m. Trenches 8A and 8B are unusually small trenches containing waste forms referred to in Kahle and Rowlands (1981) as Anafco tubs.

All of the trenches, except 14 and 14A, were constructed in a cut and cover operation. Trenches 14 and 14A were constructed partially above grade by compacted fill. The trenches were excavated roughly parallel to one another with approximately 3 m separating the trench side walls. The

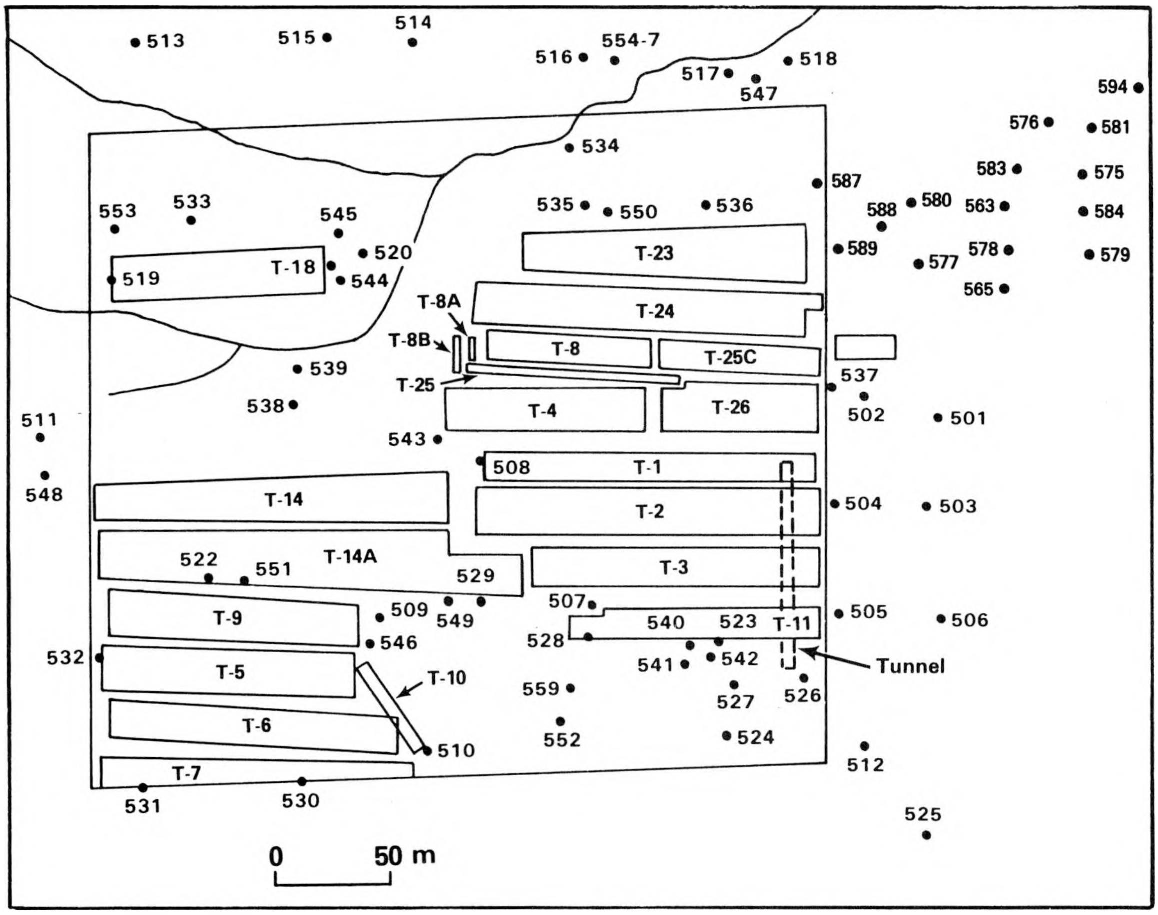


FIGURE 2.3. Location of Wells and Trenches at the Sheffield LLW Site

TABLE 2.2. Trench Data from the Sheffield LLW Site (Kahle and Rowlands 1981)

Trench	Date Opened/Closed	Trench Dimensions L-W-D (m)	Trench Volume[a] (m^3)	Waste Volume (m^3)	Waste Containment
1	8-67/8-68	137-12-6[b]	10,195	4,101	General[c]
2	8-68/3-71	140-18-7.6	19,541	6,549	General[c]
3	3-71/5-72	122-17-7.6	15,576	5,415	General[c]
4	5-72/4-73	85-18-6	9,516	5,604	General[c]
5	4-73/8-73	107-15-6	9,912	3,863	General[c]
6	8-73/3-74	119-14-6	9,940	5,995	General[c]
7	3-74/6-74	122-09-6	7,171	3,787	General[c]
8	7-74/8-74	61-14-3.5	2,931	1,398	General[c]
9	7-74/2-75	107-17.7-6.7	12,648	5,246	General[c]
10	7-74/1-75	40-05-4.6	939	395	Anafco tubs
11	12-74/6-75	107-12-6	5,188	2,617	General[c]
25	2-75/5-75	30.5-2.4-5.5	1,223	411	General[c]
26	5-75/8-75	55-19.5-7	7,503	4,705	General[c]
8A	5-75/5-75	10.7-2.4-5.5	143	90	Anafco tubs
8B	5-75/6-75	15.5-2.4-5.5	208	75	Anafco tubs
24	6-75/5-76	138.7-19-5.2	13,581	6,448	General[c]
18	3-76/12-76	97.5-21-4.6	9,516	3,417	Concrete tanks
25C	4-76/8-76	66.4-10.7-5.5	3,889	1,857	Metal drums
23	8-76/1-77	134-16.5-5.5	12,112	5,224	General[c]
14	1-77/9-77	177-16.5-7.6	22,175	11,169	General[c]
14A	8-77/4-78	145-17.7-8	20,286	9,965	General[c]
Totals			197,025	88,332	

(a) Calculated from dimensions.
(b) Depth estimated, data not given.
(c) Cardboard, plywood, wood and boxes, metal drums, etc.

exceptions to this parallel spacing are several slit trenches; 8A, 8B, and 10. The trench tops are above the probable maximum flood elevation, and the trench bottoms (except for Trench 18) are above the maximum ground-water elevation. The trench bottoms are sloped toward one end and are equipped with French drains that lead to sumps and riser pipes for sampling (Clancy et al. 1981).

Initially, the trenches were totally excavated with the soil placed along their sides. As the waste filled the trench, backfill soil was added until the wastes were within 1 m of the original grade. A soil cover cap was then placed and mounded above grade to provide for runoff (Kahle and Rowlands 1981). According to Herzog et al. (1982), most of the trench covers at Sheffield are basically the same. Generally, material excavated from the trenches was used for the cover. The material was mounded to a minimum depth of 1 to 3 m, and the covers were compacted by heavy earth-moving equipment, then vegetated for erosion control. However, no compaction was documented on any of the trenches except 14, 14A, and 18 (Kahle and Rowlands 1981).

Trench 18 was excavated below the ground-water elevation, and the bottom had to be recompacted up to a required grade before it could be filled with waste. Trenches 14 and 14A were built above grade. The outside walls were brought to a uniform grade by placing a compacted fill of clay shale and loess blend over the natural loess surface soil. The compaction of the trench caps in 14 and 14A was tested to meet 90% of the maximum density as determined by the Modified Proctor test.

2.4 DESCRIPTION OF WASTE INVENTORY

This section contains estimates on the volume and types of wastes buried at the Sheffield site. As background information, a discussion of the containers used to package the waste is presented first. The terms "fuel-cycle waste" and "non-fuel-cycle waste" used throughout this discussion were derived from definitions used by MacKenzie et al. (1985). They define fuel-cycle waste as that waste connected with the operation of commercial nuclear power plants. This definition includes waste associated with fuel fabrication, which is mostly source material or special nuclear material (SNM). All other types of waste are defined as non-fuel-cycle wastes.

Buried wastes were mainly contained in steel drums, fiberboard boxes, fiberboard drums, and steel liners. The principal container used for both fuel-cycle and non-fuel-cycle waste was the 17H carbon steel 208-L drum (MacKenzie et al. 1985). In addition to steel drums, low specific activity waste was also shipped in 127-L fiberboard boxes and 153-L fiberboard drums. Large cylindrical steel structures, known as liners, were used only for fuel-cycle waste. The volumes of these liners range between 1.4 to 11.3 m^3. Dewatered resin and solidified waste were often shipped in these containers. Some wastes were solidified on the site for disposal.

Although the fiberboard boxes and drums were used for low specific activity waste, a few shipments of large Ci amounts were made in these containers according to MacKenzie et al. (1985). The bulk of the fuel-cycle and non-fuel-cycle waste was buried in the standard Department of Transportation (DOT) 17H carbon steel drums. Several types of special containers were also used for non-fuel-cycle waste. Two types of concrete vaults were used; those with 15.2-cm walls (1.5 m x 1.5 m x 1.8 m) and those with 30.4-cm walls (1.2 m x 1.5 m x 2.4 m). The tops of the vaults were coated with a layer of asphalt, subjecting only five of the six surfaces to water infiltration. A series of concrete vaults containing nearly 450 Ci of tritium, was shipped to Trench 14A in 1977. This is nearly half of the tritium in Trench 14A and approximately 20% of all the tritium covered in the MacKenzie et al. (1985) study.

Several 114-L, 7.6-cm lead-lined drums, containing ^{137}Cs and ^{90}Sr contaminated waste, were shipped between 1967 and 1971. These containers were deposited in Trenches 1 and 2 with a crane, which probably eliminated accidental breaching of the drums. In 1970 relatively high specific activity waste (353 Ci/m^3) was shipped in aluminum tubes placed in standard 17H steel drums (MacKenzie et al. 1985). The tubes, referred to as MAP tubes in MacKenzie et al. (1985), were not sealed but had press-fit lids. Although these tubes would be highly resistant to corrosion, water could infiltrate the unsealed lids of the tubes. This is assuming that water has already breached the steel drum.

A summary of the contents of the Sheffield trenches is presented in Table 2.3. Some of the values in Table 2.3 vary depending on the source of the information. MacKenzie et al. (1985) evaluated the radioactive shipment record (RSR) for eight of the trenches to determine the relative quantities of the different radionuclides in the LLW at Sheffield. These investigators estimated the inventory of tritium, ^{14}C, ^{129}I, and selected isotopes with half-lives greater than 50 yr. The authors stress that the inventory

TABLE 2.3. Summary of Estimated Contents of Sheffield LLW Site Trenches (MacKenzie et al. 1985 and Clancy et al. 1981). Please note that discrepancies do exist between these two sources.

Trench	Date Open/Closed	By-Product (Ci)	Special Nuclear Material (g)	Source Material (kg)	Volume Buried (m^3)
1	08-67/08-68	6,157.30	2,592.28	7,119.46	4,089.95
2	08-68/03-71	10,451.15	12,695.86	15,485.81	6,690.21
3	03-71/05-72	7,758.19	8,339.91	2,062.43	5,414.82
4	05-72/04-73	4,443.43	4,863.65	1,805.67	5,604.48
5	04-73/08-73	1,167.66	3,187.33	2,342.39	3,426.70
6	08-73/03-74	1,372.49	7,040.17	215.61	5,589.40
7	03-74/06-74	635.76	1,640.73	615.08	3,786.65
8	07-74/08-74	354.96	0	0	1,398.01
8A	05-74/08-74	237.99	0	0	90.01
8B	05-75/06-75	250.67	0	0	75.14
9	07-74/02-75	1,385.02	912.94	13,432.82	5,245.93
10	08-74/01-75	381.93	0	0	394.93
11	12-74/06-75	1,428.14	683.33	14,922.41	2,651.34
14	01-77/09-77	7,197.06	1,791.51	133,162.78	11,169.40
14A	08-77/04-78	6,322.16	4,741.96	53,699.86	9,962.80
18	03-76/12-76	131.30	100.80	89.81	3,416.97
23	08-76/01-77	4,388.55	211.27	2,983.63	5,102.19
24	06-75/05-76	5,109.38	4,285.61	10,942.52	8,653.99
25	02-75/05-75	195.89	0	0	411.36
25C	04-76/08-76	863.86	177.58	282.25	1,857.22
26	05-75/08-75	991.20	1,087.80	13,431.91	4,705.03
Totals		61,224	54,353	272,594	89,737

developed was based entirely on the records of the waste shippers and the operators of the burial site. No judgments regarding the accuracy of the data were made.

The estimated tritium inventory of 2359 Ci ranks third next to ^{60}Co (6084 Ci) and ^{137}Cs (6321 Ci) in terms of the total amount of activity buried at the Sheffield LLW site. Tritium was not listed on RSRs as a constituent of the fuel-cycle waste and was not being measured by power plant personnel during the period of Sheffield's operation. McKenzie et al. (1985) estimated the amounts of tritium contributed by fuel-cycle waste (Table 2.4). Their results show that power plant wastes tend to contain a few tenths of a percent of tritium, usually 0.1% or less. They concluded that the fuel-cycle tritium in the eight trenches at Sheffield was probably about 20 Ci and, therefore, could be neglected for the purposes of release scenario development. For non-fuel-cycle wastes (Table 2.5), the major portion of the tritium was buried in a solid form (approximately 97%). Approximately 30% of the solid tritium waste is in the form of tritiated zinc sulfide, used in the manufacture of luminous dials.

A summary of the ^{14}C inventories in eight trenches at Sheffield is given in Table 2.6. Carbon-14 estimates made by researchers at Brookhaven National Laboratory (BNL) contained more uncertainty than any of the other relevant isotopes (MacKenzie et al. 1985). Usually ^{14}C was listed with other isotopes on the RSRs, and no individual amounts were given. Fairly good estimates were made for Trenches 2, 14A, 23, and 24 because ^{14}C was listed separately. The "mixed" solids and liquids in Table 2.6 represent ^{14}C mixed with other isotopes. The actual ^{14}C contents of these mixtures were not specified on the shipping records. Roughly 99% of the ^{14}C waste is in solid form.

TABLE 2.4. Comparison of Tritium Inventories with Total By-Product Material (Table 2.12 in MacKenzie et al. 1985)

	Estimated Tritium Inventory (Ci)			
Trench	In Shipments ≥ 1 Ci	In Shipments ≤ 1 Ci	Total Tritium	Total By-Product Material (Ci)[a]
1	305.8	4.6	310.4	6157.3
2	621.6	11.8	633.4	10451.1
7	30.7	4.5	35.2	635.8
11	31.9	4.5	36.4	1428.1
14A	713.4	11.7	725.1	6322.2
23	181.6	7.2	188.8	4388.5
24	306.6	8.7	315.3	5109.4
25C	110.9	3.5	114.4	863.9
Totals	2302.5	56.5	2359.0	

(a) From NUS Corporation (in MacKenzie et al. 1985).

TABLE 2.5. Non-Fuel-Cycle Tritium Amounts Compiled From Shipments ≥1 Ci (MacKenzie et al. 1985)

Trench	Form	Amount (Ci)	Total by Trench (Ci)
1	Solid	287.6	
	Gas	18.2	305.8
2	Solid	551.35	
	Liquid Absorbed on Vermiculites	≤21.4	
	Liquid	5.0	
	Gas	25.0	581.35
7	Liquid	9.6	
	Solid	19.5	29.1
11	Solid	21.7	
	Liquid	<1.1	21.7
14A	Solid	534.7	
	Solid, Tritiated Zinc Sulfide[(a)]	149.5	684.2
23	Solid	35.2	
	Solid, Tritiated Zinc Sulfide	127.2	162.4
24	Solid	30.2	
	Solid, Tritiated Zinc Sulfide	256.2	286.4
25C	Solid, Tritiated Zinc Sulfide	67.5	
	Solid	9.7	
	Liquid	7.0	
	H_2O in Toluol	8.0	92.2

(a) Zinc sulfide is generally insoluble in water; however, zinc sulfide is soluble in dilute mineral acid.

A summary of the amounts of ^{129}I, ^{90}Sr, ^{137}Cs, and ^{60}Co buried at the site (MacKenzie et al. 1985) is given in Table 2.7. Four trenches received ^{129}I, mostly in the form of sources. These sources generally contained μCi amounts of ^{129}I. One shipment to Trench 2 was unlike all the others in both magnitude and form. This shipment contained approximately 1 mCi of ^{129}I and was probably in the form of NaI.

Of the eight trenches examined by MacKenzie et al. (1985), all of the ^{90}Sr, ^{137}Cs, and ^{60}Co were in solid forms. The amounts of ^{90}Sr from fuel-cycle waste was quite low compared to the non-fuel-cycle sources. Trenches 1 and 2 contain unusually high concentrations of ^{90}Sr, which came from a single shipper. Past estimates of non-fuel-cycle ^{90}Sr made by the NUS Corporation (in MacKenzie et al. 1985) and by the Interagency Task Force (Dragonette et al. 1979) were obviously too low.

3.0 PREVIOUS INVESTIGATIONS

3.1 HYDROGEOLOGIC STUDIES

The most recent and significant hydrogeologic investigations at the Sheffield site were done by the USGS beginning in 1976. Results from these studies have been summarized by Foster and Erickson (1980); Foster et al. (1984a,b); and Garklavs and Healy (1986). Synopses of these investigations are presented below.

Test drilling indicates that the Sheffield site is underlain by a complex series of interbedded and interfingering glacial sediments. The site is located in the glaciated till plain section of the Central Lowlands physiographic province. The disposal trenches cut through a complex series of quaternary deposits, composed primarily of silts, clays, and sands, ranging in thickness from 3 to 20 m. The stratigraphy at the site includes, in ascending order beginning with the bedrock: the weathered shale of the Carbondale Formation; the Duncan Mills Member, the Hulick Till Member, the Toulon Member, the Radnor Till Member, and the Berry Clay Member of the Glasford Formation; the Roxana Silt; the Peoria Loess; and the Cahokia Alluvium (Figure 3.1). More recent units on the site include a modern soil and coal-mine spoils.

The relative position of the water table and the vertical and horizontal relationships of these sediments are illustrated along three geologic section lines: two west-east sections (A-A′ and B-B′) along the northern part of the site and a north-south section (C-C′) along the eastern edge of the site (Figure 3.2). The corresponding geologic sections are shown in Figures 3.3, 3.4, and 3.5.

Depth to ground water beneath the site ranges from 1.8 to 6 m. The configuration of the water table is a subdued version of the surface topography, and ground-water movement is generally in two predominant directions (Figure 3.6): northeast to east toward a strip-mine lake and south to southeast toward small tributary channels belonging to Lawson Creek, which eventually drains into the strip-mine lake southeast of the site. Ground-water flow divides are found west of the disposal facility, north of the facility in strip-mine spoil materials, and on a topographic high found in the southeast corner of the site.

The ground-water system beneath the Sheffield site can be thought of as containing two separate aquifer systems: a regional confined aquifer system contained in deep sandstone and carbonate bedrock and a local unconfined aquifer system contained in the shallow sequence of unconsolidated quaternary-aged sediments. The shallow unconfined aquifer is overlain by a relatively thin unsaturated zone that ranges from about 1.5 to as much as 14 m thick.

	Time/Rock Stratigraphy	Thickness (m)
	PLEISTOCENE: Unconsolidated glacial deposits, loess and alluvium	0-150
	PENNSYLVANIAN: Carbondale Formation, shale with thin sandstone, limestone and coal beds	0-150
	MISSISSIPPIAN: Shale, grey, green	0-8
	DEVONIAN: Limestone	15-30
	SILURIAN: Dolomite, lower part cherty	90-150
ORDOVICIAN	MAQUOKETA: Shale with limestone and dolomite beds	30-60
ORDOVICIAN	GALENA-PLATTEVILLE: Largely dolomite with shale and limestone zones	90-105
ORDOVICIAN	GLENWOOD-ST. PETER: Sandstone, some dolomite and shale beds, some clay at base	15-120
ORDOVICIAN	PRAIRIE DuCHIEN: Dolomite with some sandstone beds	0-120
CAMBRIAN	TREMPEALEAU: Dolomite, thin sandstone beds may occur	0-60
CAMBRIAN	FRANCONIA: Green sandstone, shale and dolomite	25-35
CAMBRIAN	IRONTON-GALESVILLE: Sandstone	40-75
CAMBRIAN	EAU CLAIRE: Predominantly shale and dolomite in upper portion grading to sandstone	120-140
CAMBRIAN	MT. SIMON: Sandstone, thin shale beds	305-610
	PRE-CAMBRIAN: Granite	

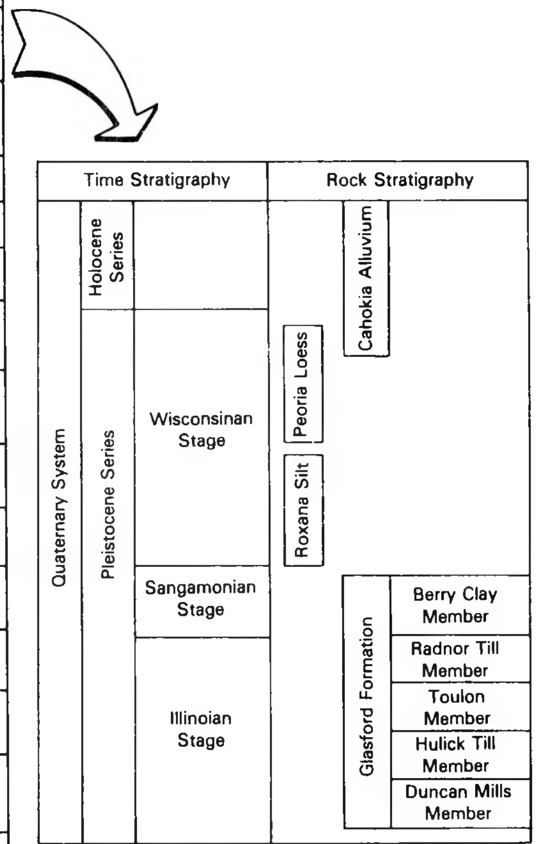


FIGURE 3.1. Time and Rock Stratigraphy of Northwestern Illinois and the Sheffield LLW Site (Garklavs and Healy 1986)

At the site, the shallow aquifer system is isolated from the deep bedrock aquifers by the thick sequence of shales and mudstones belonging to the Carbondale Formation. The underlying shale bedrock roughly parallels the surface topography and forms the core of the topographic high found in the southeast corner of the site. Thus, the deeper bedrock aquifers, which are important regional supplies of water for major municipalities and for agricultural uses, have little relevance to the local ground-water flow system in the vicinity of the Sheffield site. Consequently, past investigations have been more concentrated on understanding water movement within the shallow unsaturated zone and underlying unconfined aquifer system. The relatively impermeable shale forms a barrier between ground water found in the shallow sediments and in deeper regional, confined aquifers found in Silurian, Ordovician, and Cambrian carbonate rocks (Figure 3.1). The shale and coal seams

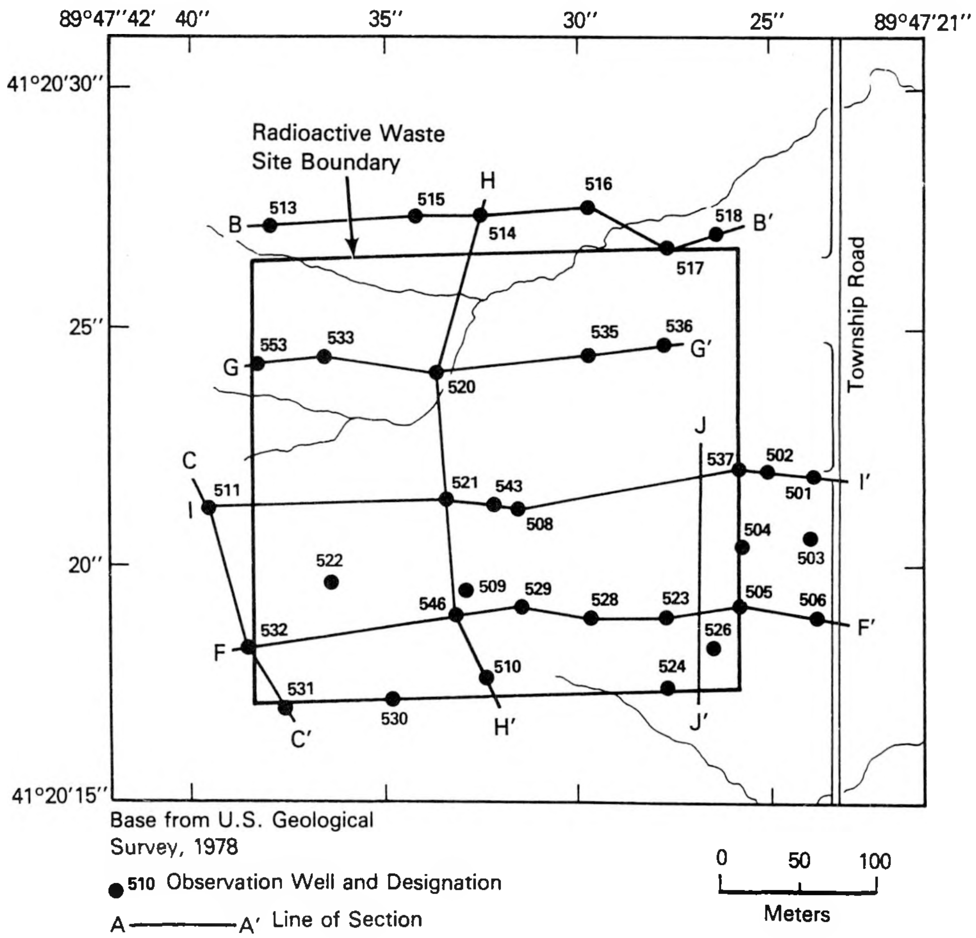


FIGURE 3.2. Location of Geologic Cross Sections Shown in Figures 3.3, 3.4, and 3.5 (Foster et al. 1984a)

near the contact of bedrock and overlying units may, however, provide a pathway for ground-water movement. Fractures, fissures, and bedding-plane separations in the shale, and the usually permeable nature of coal found near the surface, may transmit water in a way so as to be considered part of the shallow aquifer system.

The most significant hydrogeologic unit on the site is a pebbly-sand unit in the Toulon Member of the Glasford Formation. This pebbly-sand unit grades into a coarse gravel with sand and pebbles east of the disposal site. A summary of hydraulic conductivities for various major units found on the site (Table 3.1) illustrates the hydraulic significance of this particular unit. The pebbly-sand unit is significantly more permeable than all other units. It lies unconformably over the Hulick Till Member of the Glasford

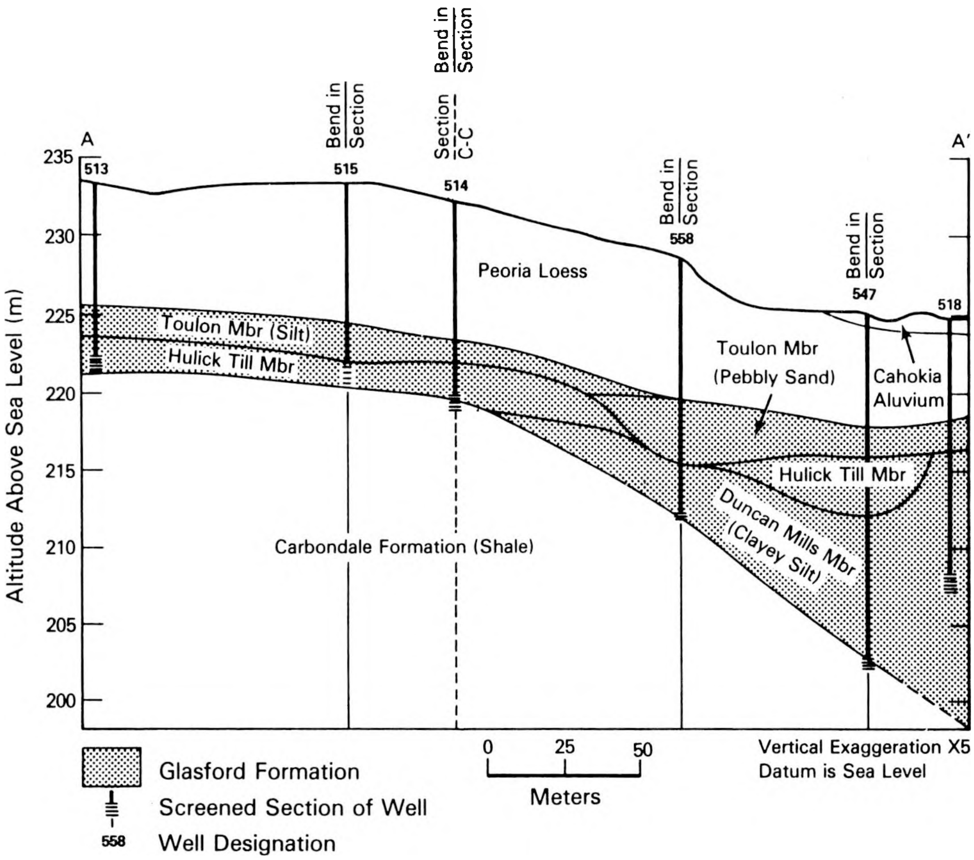


FIGURE 3.3. Geologic Cross Section A-A′ (Foster et al. 1984a)

Formation and is found over approximately 67% of the waste-disposal site. On the site, the water table can be found above, below, and within the pebbly-sand unit. The unit is generally fully saturated along its northern extent and unsaturated along its southern extent. Examples of the areal extent and the degree of saturation within the pebbly-sand unit of the Toulon Member are illustrated in Figures 3.7 and 3.8 for some relatively low (February 1978) and high (June 1979) water-table conditions.

East of the site, a narrow, channel-like depression in the Hulick Till Member is filled with coarse gravelly sand of the pebbly-sand unit of the Toulon Member. This depression extends eastward from the northeast corner of the disposal site to the strip-mine lake, hydraulically connecting the site and the lake. As a result, ground water contaminated with tritium has been detected along the entire length of this channel and in seeps located along the banks of the lake.

Recharge to the unconfined aquifer is derived locally from the direct infiltration of precipitation falling on and immediately west of the site. Prior to the unsaturated zone investigations, recharge was estimated by

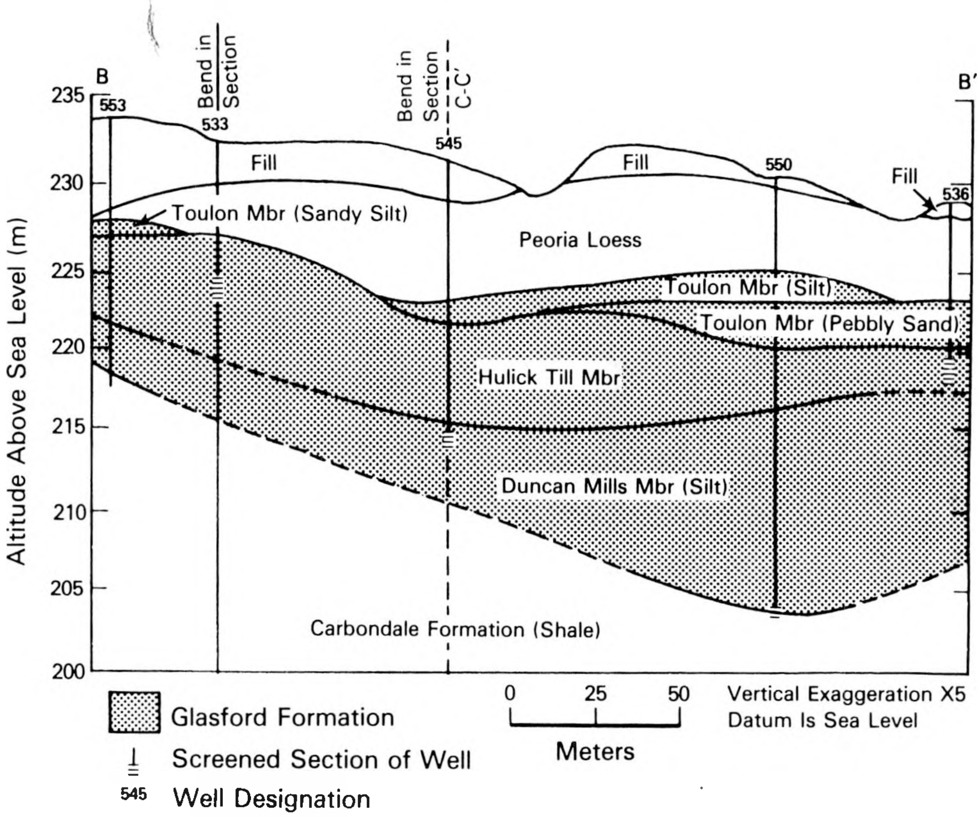


FIGURE 3.4. Geologic Cross Section B-B' (Foster et al. 1984a)

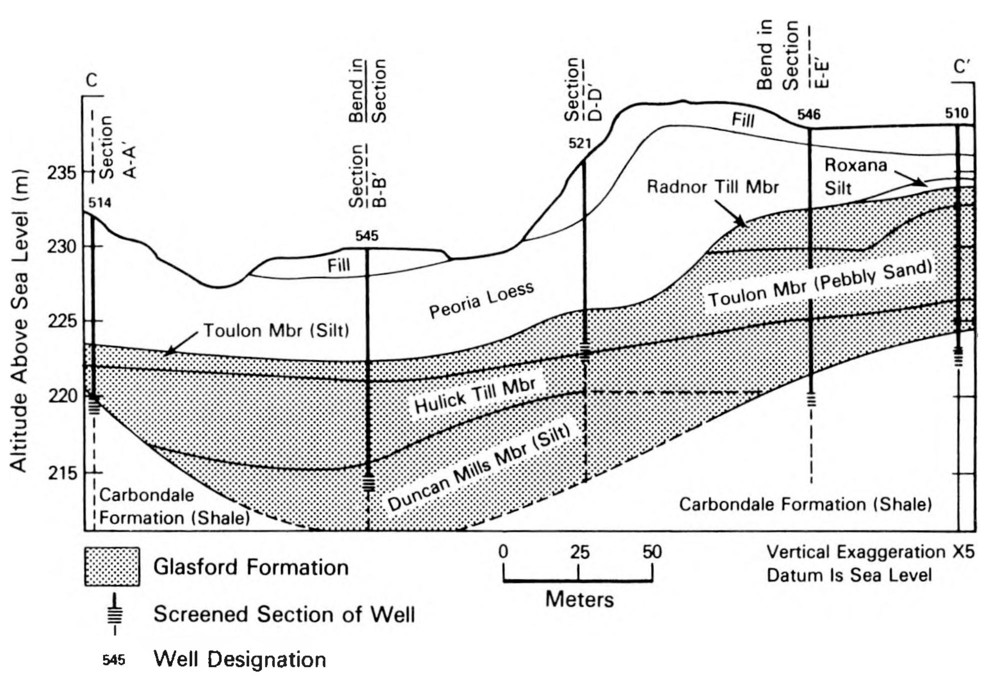


FIGURE 3.5. Geologic Cross Section C-C' (Foster et al. 1984a)

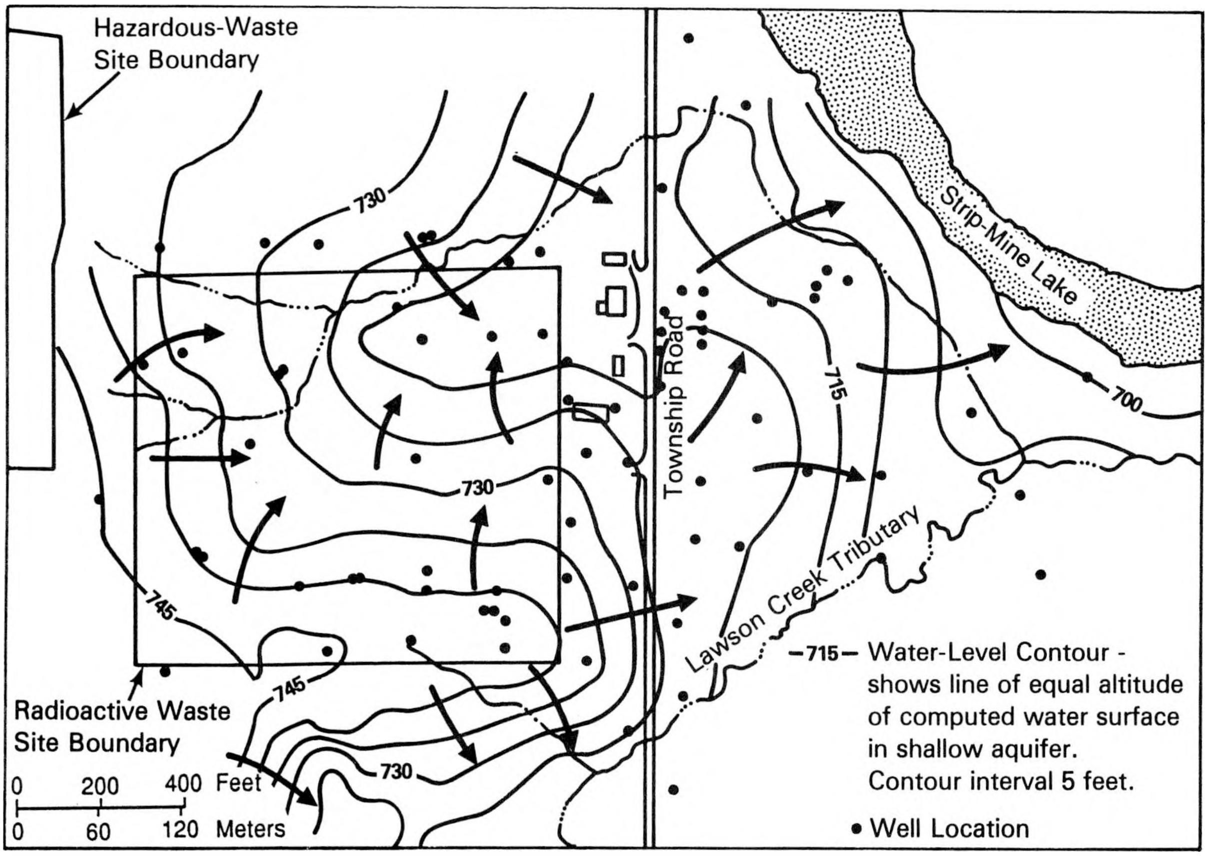


FIGURE 3.6. Direction of Ground-Water Movement at the Sheffield LLW Site (June 1984)

TABLE 3.1. Summary of Hydraulic Conductivities, in m/s, of Selected Hydrostratigraphic Units at the Sheffield LLW Site

Lithology	Vertical	Horizontal
Peoria Loess (silt)	1.1E-7	8.8E-8 to 2.3E-7
Toulon Member		
Pebbly Silty Sand	---	2.3E-7 to 1.7E-7
Sand	---	2.5E-5
Pebbly Sand	---	2.0E-5 to 5.5E-4
Hulick Till (Pebbly Clayey Silt)	1.5E-8	5.4E-9
Duncan Mills Till (Clayey Silt)	6.7E-9	---
Weathered Shale (Clay)	6.4E-10	3.6E-10

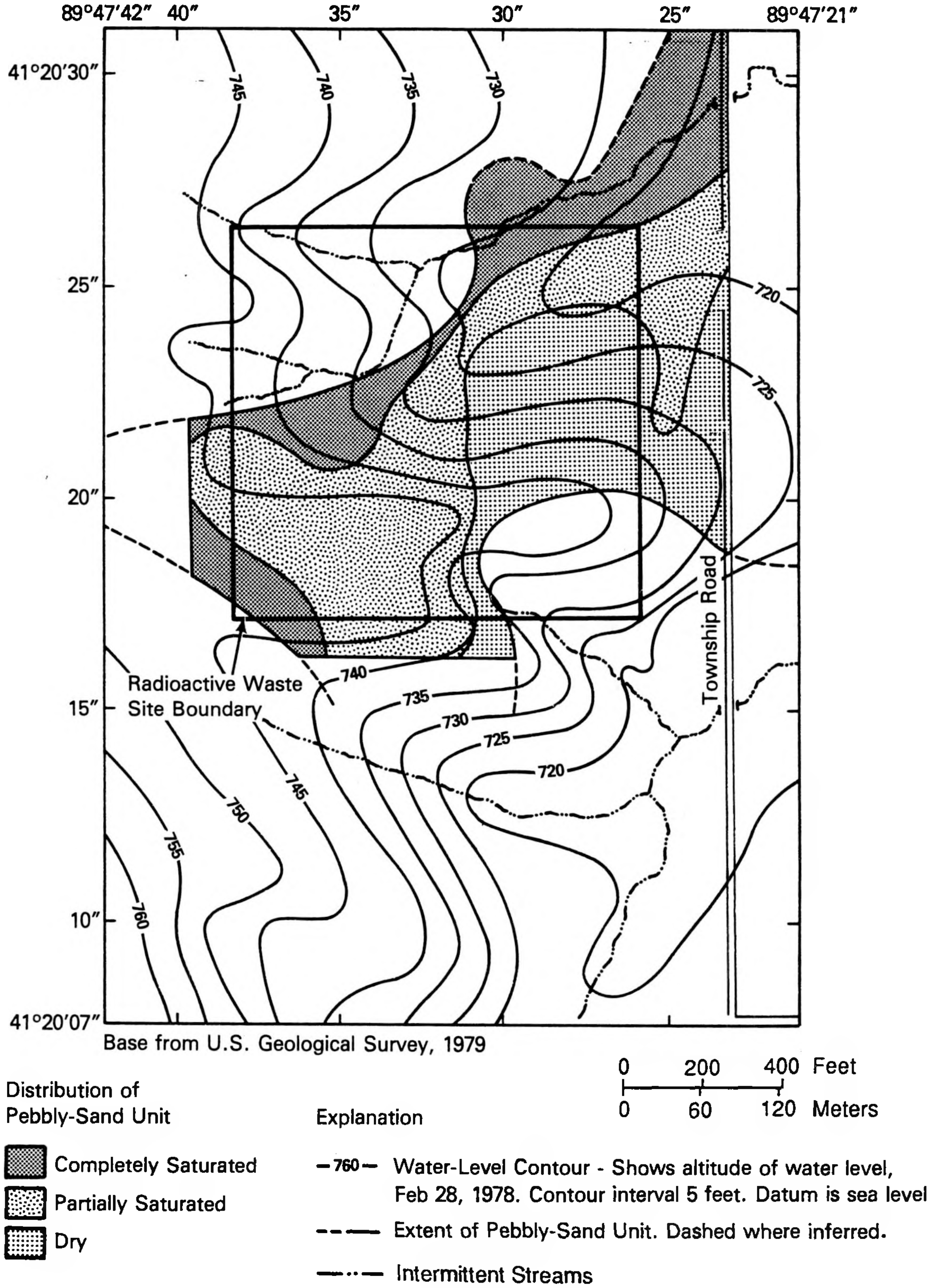


FIGURE 3.7. Water-Surface Contours and Degree of Saturation of Pebbly-Sand Unit of Toulon Member on February 28, 1978

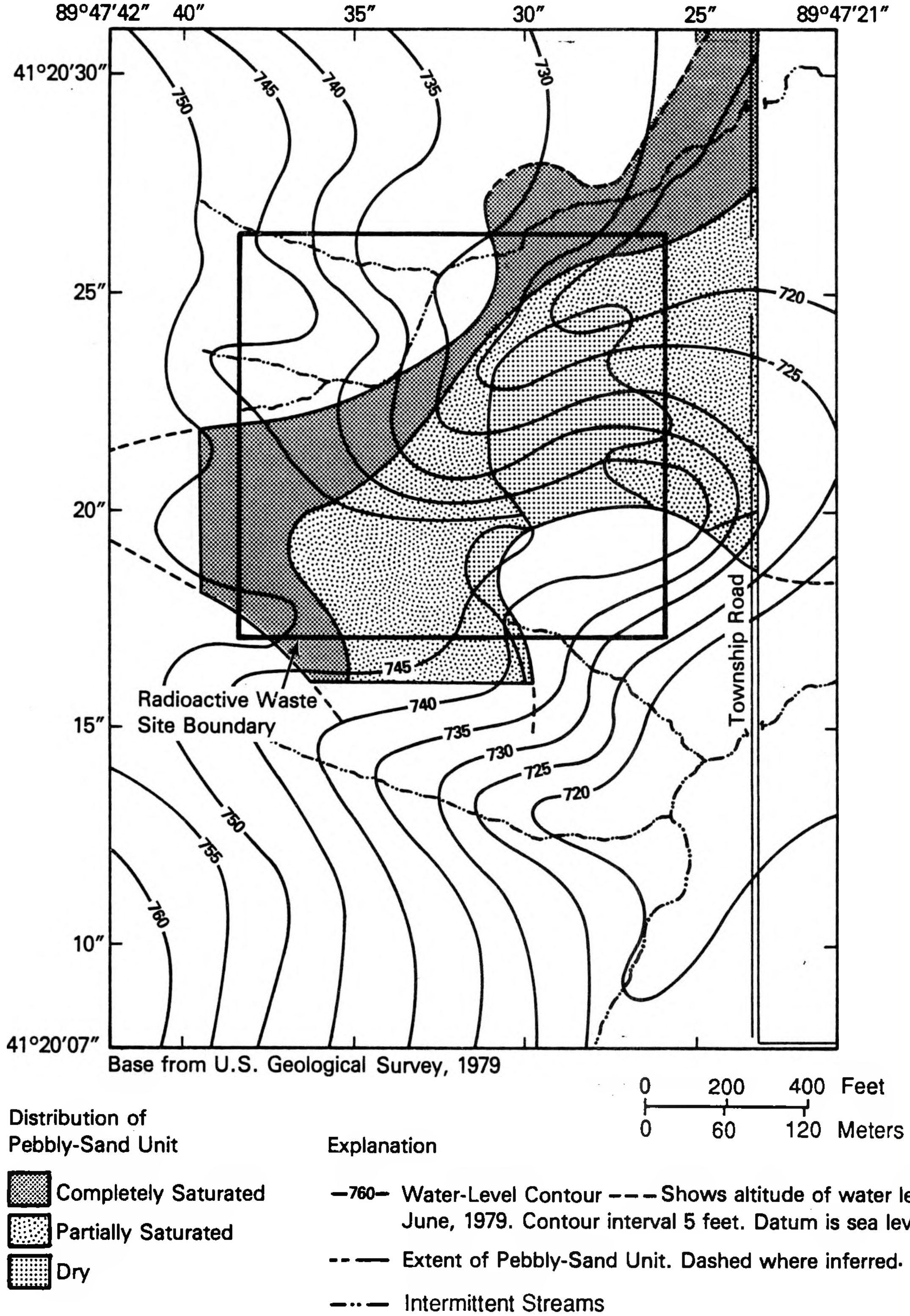


FIGURE 3.8. Water-Surface Contours and Degree of Saturation of Pebbly-Sand Unit of Toulon Member in June 1979

Foster et al. (1984a) to be between 2.5 to 5 cm/yr (1 to 2 in./yr). The amount of local recharge varies from one location to another depending on the vegetation cover, texture of surficial materials, and surface slope.

3.2 STUDIES OF RADIONUCLIDE MIGRATION IN GROUND WATER

In 1967, the Illinois Department of Nuclear Safety (IDNS) began collecting ground-water samples on the site. Since 1976, the USGS has collected additional data. Sampling results indicate that only tritium has migrated on and off the site. Tritium was first observed in 1982, but it has since been detected along the entire length of the buried channel leading east of the site to the strip-mine lake. Tritium has also been detected in seeps along the banks of the lake. Tritium concentrations do not vary significantly from one location to another along the narrow channel, which suggests that migration in the channel is rapid.

Tritium has been found near some trenches along the south end of the burial area; however, with the exception of one well, offsite migration of tritium to the south has not been as extensive as northeast of the site. Tritium concentrations in ground water have ranged from below detection level (about 20 pCi/L) to more than 300,000 pCi/L. Maximum concentrations are below regulatory limits. The areas where tritium has been detected in the vicinity of the site are shown in Figure 3.9.

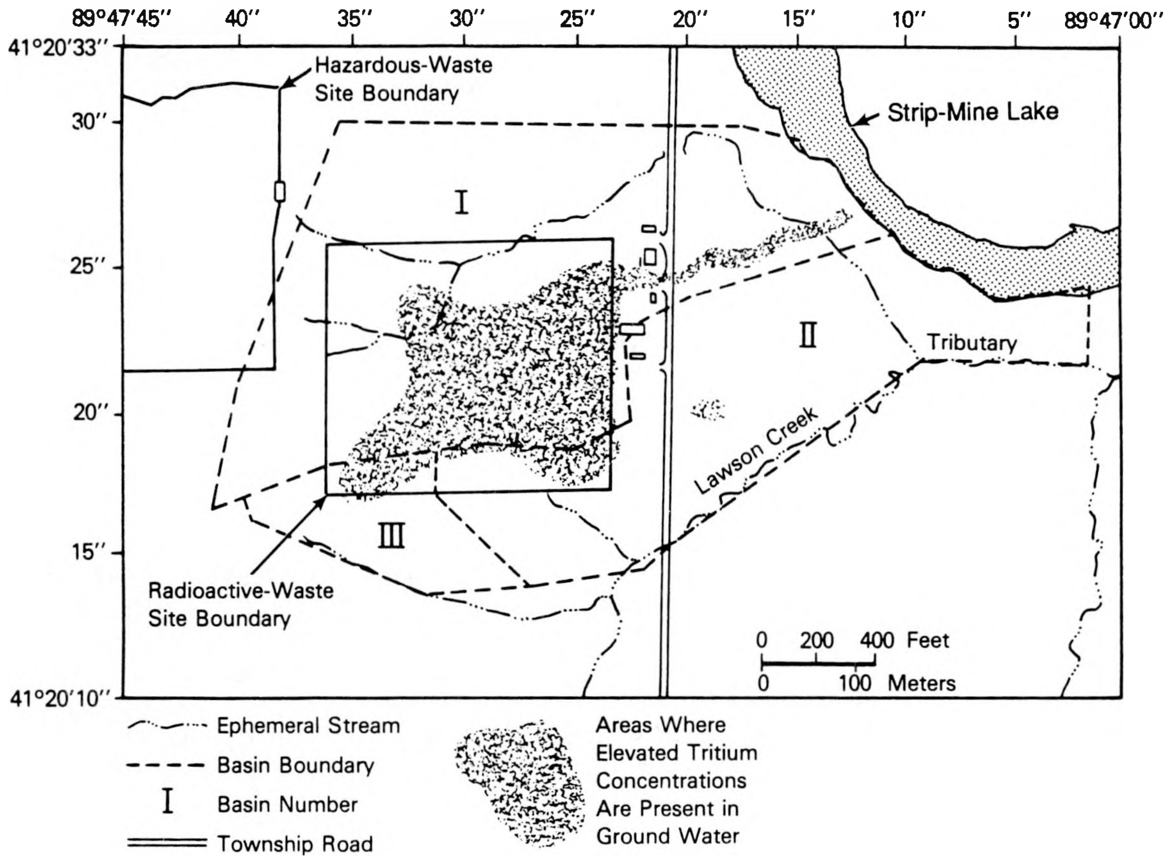


FIGURE 3.9. Areas of Elevated Tritium Concentrations in Sheffield LLW Site Ground Water

4.0 HYDROLOGIC AND RADIONUCLIDE TRANSPORT ANALYSIS OF THE GROUND-WATER PATHWAY

Ground-water flow and transport models were used in the ground-water pathway analysis to simulate the subsurface release of radionuclides from the burial site. Of primary importance to this analysis are the forms of the buried wastes and the mechanisms that control the release of the radionuclides from the Sheffield site. Where reliable data were available to provide realistic values of parameters, these values were used. Where uncertainties exist, values were selected that would produce conservative results. A conservative value of a parameter tends to overestimate rather than underestimate the impact.

4.1 PERFORMANCE OBJECTIVES

The key performance objective as required by the LLW facility radioactive license in 10 CFR 61, "Licensing Requirements for Land Disposal of Radioactive Waste," states that:

> "...during operation and after site closure that concentrations of radioactive material which may be released to the general environment in ground water, surface water, air, soil, plants, or animals will not result in any member of the public receiving an annual dose equivalent of 25 millirems (2.5×10^{-4} Sv) to the whole body, 75 millirems (7.5×10^{-4} Sv) to the thyroid, and 25 millirems (2.5×10^{-4} Sv) to any organ of any member of the public."

In this study, the performance assessment considered only the ground-water pathway. This performance assessment is not intended to provide a determination of Sheffield site compliance with the licensing regulations put forth in 10 CFR 61. For sake of comparison, however, final model results of key radionuclide peak concentrations are compared to dose equivalent concentrations consistent with the regulatory limit of 25-mrem/yr whole body dose.

4.2 GROUND-WATER FLOW MODELING ANALYSIS

To determine the level of modeling complexity needed to characterize the ground-water flow at the site, the results of a two-dimensional and a three-dimensional ground-water flow model are compared. The two-dimensional model was developed by the USGS, and the results documented by Foster et al. (1984a) and Garklavs and Healy (1986). The development, calibration, and results of this model are summarized in Section 4.2.2. The three-dimensional model was developed at PNL, and the development, calibration, and results are documented in Section 4.2.3.

4.2.1 Conceptual Model of Flow System

A conceptual model of the Sheffield site was developed in previous studies by the USGS. In their conceptual model of the site, Garklavs and Healy (1986) divided the area in the vicinity of the site into three different ground-water basins and identified two principal ground-water flow paths within the shallow aquifer system. These basins were divided by topographic highs of the bedrock, which Garklavs and Healy assumed isolated the hydrologic system. The boundaries of the basins are shown in Figure 4.1, and are described as follows: the contact between undisturbed Quaternary materials and strip-mine spoils defines a ground-water divide to the north (Basin I); a bedrock high forms a ground-water divide to the west (Basin II); and a tributary to Lawson Creek forms a ground-water sink to the south (Basin III). This sink directs ground-water flow from about the southern

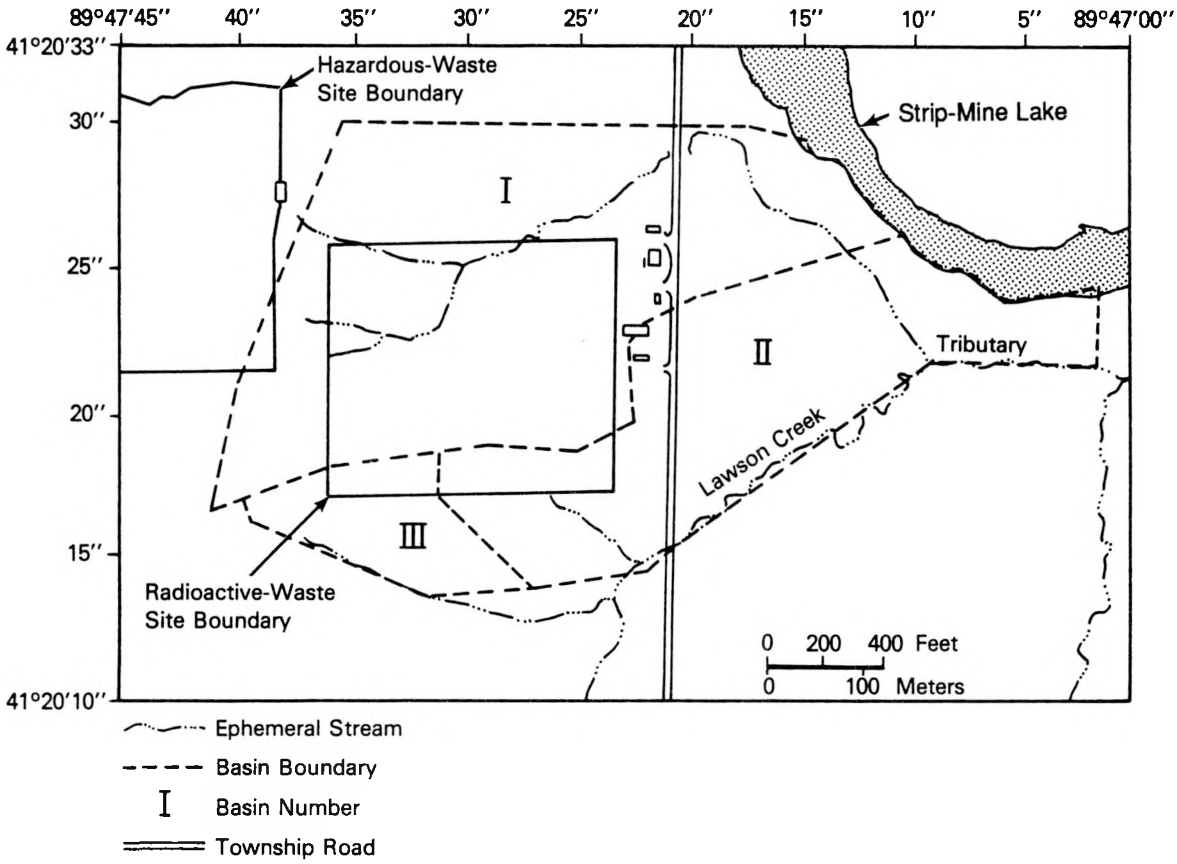


FIGURE 4.1. Ground-Water Basins Designated by Garklavs and Healy (1986)

one-third of the study area southward and eastward toward the strip-mine lake. In the delineation of these basins, the shallow aquifer was assumed to extend eastward to the strip-mine lake.

In Basin I, which covers the northern three-fourths of the site, ground water drains into the pebbly-sand unit of the Toulon Member. This unit has been found to extend continuously from the site to the strip-mine lake (Foster et al. 1984a). Foster et al. concluded that virtually all of the flow through Basin I converges on the channel-like structure in the Hulick Till Member to the east of the site. Ground-water velocities in this flow path may attain a maximum velocity of about 2 m/d. A second flow path, found in Basins II and III, drains ground water from the southern one-fourth of the site toward the channel of the tributary to Lawson Creek. Foster et al. (1984a) estimated ground-water velocities of about 0.02 m/d in this flow path.

The pebbly sand within the Toulon Member is partially to completely saturated through about 80% of its extent. The pebbly sand is unsaturated over the ridge-like structure in the Hulick Till Member on the eastern edge of the site, and it is not found at the bedrock high near the southeastern corner of the site. These areas are coincident with the ground-water divide between Basins I and II. Garklavs and Healy (1986) concluded that the surface configuration of the contact between the pebbly sand and Hulick Till Member defines both a ground-water divide and a structural control of ground-water flow.

4.2.2 Previous USGS Ground-Water Flow Modeling

To gain a better understanding of ground-water movement and waste migration at the Sheffield site, Garklavs and Healy (1986) developed a two-dimensional ground-water flow model of the site. The discussion that follows, which is derived from Foster et al. (1984a) and Garklavs and Healy (1986), provides a brief summary of this modeling effort. The discussion includes the development and design of the model, model calibration, and important modeling conclusions.

4.2.2.1 Development and Design of Model

Garklavs and Healy (1986) used a two-dimensional numerical model to simulate ground-water flow within Basins I and II. Insufficient data were available for modeling Basin III. This modeling effort was an extension of the work previously done by Foster et al. (1984a). Foster et al. listed a number of problems related to modeling the hydrogeologic system at Sheffield. One problem is the irregular shape of the base of the pebbly-sand unit that allows changing water levels to saturate or dewater parts of the unit and significantly alters the rate of flow in these areas. Another relates to the likelihood of lateral water movement within the unsaturated zone, which makes it difficult to determine the distribution of recharge rates at contacts between sedimentary units. The model by Foster et al. (1984a) also covers areas adjacent to the site within the basin of the Lawson Creek tributary where subsurface data were not available. Additional hydrologic information on the area east of the site toward the strip-mine lake became available when Garklavs and Healy (1986) modeled the site.

Several assumptions were made by Garklavs and Healy (1986) prior to the modeling. Bedrock was considered to be an impermeable lower boundary for the shallow, unconfined, aquifer system. There was no lateral inflow to the system. All inflow was assumed to be recharge from precipitation falling directly on the basins. The ground-water system was assumed to be at steady state. All model boundaries were assumed to be no-flow except for the constant head boundary at the strip-mine lake.

Garklavs and Healy (1986) used the partial differential equation describing two-dimensional steady-state ground-water flow in an isotropic aquifer:

$$\frac{\partial}{\partial x}\left[T\frac{\partial h}{\partial x}\right] + \frac{\partial}{\partial y}\left[T\frac{\partial h}{\partial y}\right] + W = 0 \qquad (1)$$

where T = transmissivity (length squared/time)
h = mean hydraulic head in the vertical (length)
x, y = cartesian coordinates (length)
W = recharge rate (length/time).

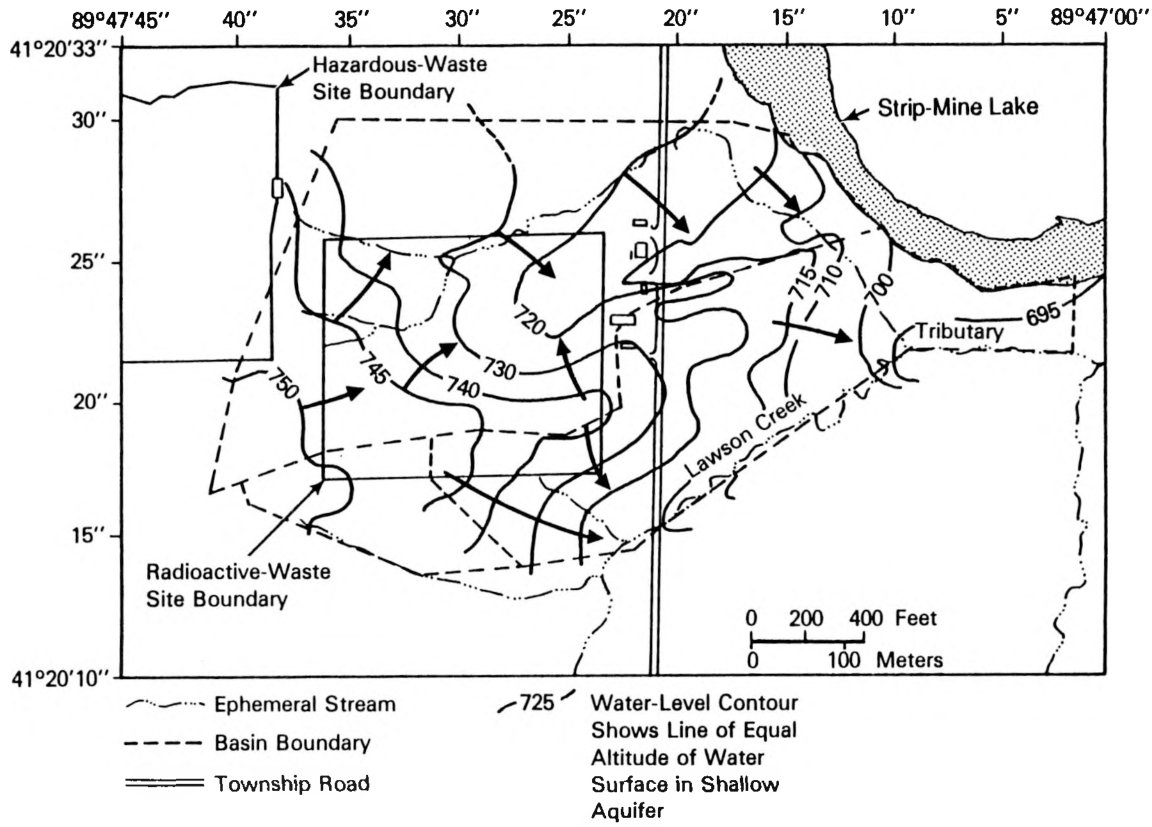
Using a grid system, each basin was divided into different zones where values of hydraulic conductivity and recharge were assumed to be uniform. The model approximates the above equation at each node, using the finite-element technique. Initial estimates of hydraulic conductivity, saturated thickness, and recharge from past field investigations were input to the model. The model computed estimated heads at all of the nodes.

4.2.2.2 Model Calibration

Garklavs and Healy (1986) compared the computed heads to heads measured on June 6, 1982 (Figure 4.2), using a mean squared-error (MSE) term defined as follows:

$$MSE = \frac{1}{N}\sum_{i=1}^{N}(\hat{h}_i - H_i)^2 \qquad (2)$$

where H_i = observed heads
$\hat{h}_i$ = predicted heads
N = number of observations.



<u>FIGURE 4.2</u>. Hydraulic Heads Measured on June 6, 1982 (Garklavs and Healy 1986)

Values of hydraulic conductivity and recharge were changed on a zone-by-zone basis, recomputing the heads, until a best fit was obtained. A best fit was obtained when 1) a minimum value of the MSE term was reached and 2) the difference between observed and computed heads, termed residuals, was normally distributed with a mean of 0. Sensitivity tests were performed on the best-fit model by changing parameter values and boundary conditions and by assessing the resulting changes in computed head values.

During the modeling of Basin I, no-flow boundaries were imposed along the periphery of the basin except along the strip-mine lake, where constant heads equal to lake surface elevation were imposed (Garklavs and Healy 1986). They divided Basin I into six zones (Figure 4.3):

- Zone 1 represents all areas where the saturated zone consists of any geologic material except the Toulon Member. An average value of hydraulic conductivity was assumed.

- Zone 2 corresponds to the pebbly-sand unit.

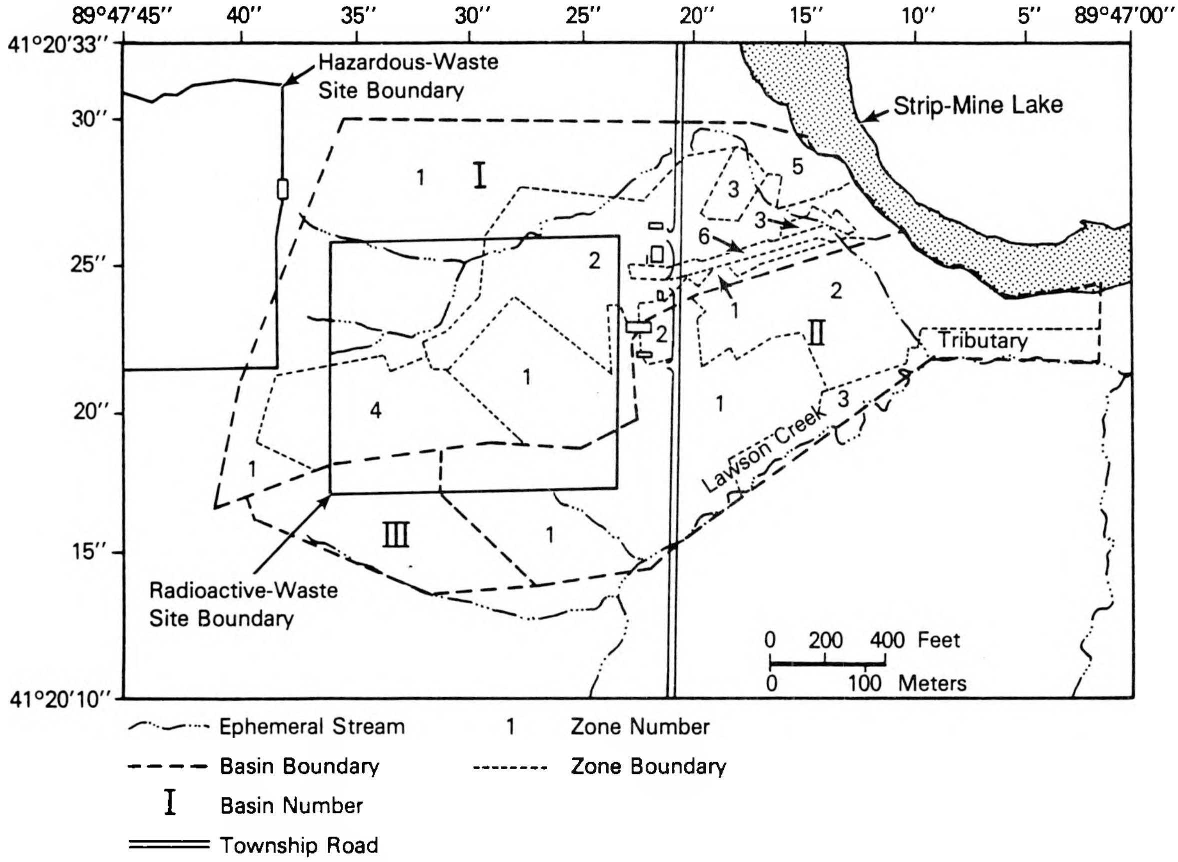


FIGURE 4.3. Hydraulic Conductivity and Recharge Zones Used in the Two-Dimensional Ground-Water Flow Model of Garklavs and Healy (1986)

- Zone 3 represents the pebbly-sand unit in areas where it crops out. Infiltration and recharge would be greater in Zone 3 than Zone 2.

- Zone 4 corresponds to the silty-sand portion of the Toulon Member. The hydraulic conductivity of Zone 4 is less than that of Zone 2, and the depth to the top of the sand unit is greater in this area, potentially reducing the amount of recharge to the sand.

- Zone 5 was specified to have lower hydraulic conductivity than Zone 2, although limited information exists for Zone 5.

- Zone 6 represents the channel-like depression filled with pebbly sand. Zone 6 has the highest hydraulic conductivity.

Table 4.1, from Garklavs and Healy (1986), shows the values of hydraulic conductivity and recharge that produced the best model fit. The MSE had a value of 4.2 m (13.88 ft), and the average of the residuals was 0.21 m (0.39 ft). Field and laboratory values of hydraulic conductivity showed good agreement with the model values (Table 4.1). Water levels computed by the model were compared with water levels measured in June 1982. The model produced a fairly good replication of the observed heads.

The high recharge rate used for Zone 3 reflects the fact that the pebbly-sand unit crops out in this area. According to Healy, there is virtually no runoff in Zone 3, and runoff from adjacent areas tends to collect in this zone. A mass balance of the recharge used in the model for just the 83,610 m^2 (900,000 ft^2) LLW area yields 4.80 cm/yr (1.9 in./yr). This is substantially lower than the recharge estimate of 22.8 cm/yr (9 in./yr) made by Healy et al. (1983).

A grid of 408 nodes and 403 elements was used to model Basin II. All of the boundaries in Basin II were specified as no-flow boundaries, with the exception of the constant-head boundary at the lake. The boundary along Lawson Creek Tributary was considered a no-flow boundary because the stream is ephemeral, and flow in the stream is negligible (Garklavs and Healy 1986). As shown in Figure 4.3, Basin II was divided into three zones:

TABLE 4.1. Values of Modeled Recharge and of Modeled and Measured Hydraulic Conductivity, in m/sec (from Garklavs and Healy 1986)

Zone	Recharge Used in Model (cm/yr)	Hydraulic Conductivity Used in Model	Range in Measured Hydraulic Conductivity, Field and Laboratory Methods
		Basin I	
1	0.99	3.8E-7	2.4E-4 to 1.4E-7
2	9.90	4.20E-5	5.4E-4 to 1.5E-7
3	72.14	4.20E-5	None
4	7.87	1.52E-5	2.3E-7[(a)]
5	9.65	6.09E-6	4.0E-6 to 2.9E-6[(a)]
6	4.82	4.20E-4	1.7E-4 to 1.1E-5
		Basin II	
1	4.82	3.35E-7	1.9E-6 to 2.7E-7
2	1.55	4.11E-5	7.0E-4 to 5.2E-9
3	12.45	1.22E-4	2.7E-7 to 1.9E-6[(a)]

(a) Indicates value(s) from one well.

- Zone 1 represents all areas where the saturated zone consists of any geologic material except the Toulon Member. An average value of hydraulic conductivity was assumed.

- Zone 2 corresponds to the pebbly-sand unit.

- Zone 3 represents the alluvial deposits within the stream channel along the southern border of the model area.

The distribution of hydraulic heads resulting from simulation of June 6, 1982, conditions is given in Figure 4.4. The values of hydraulic conductivity and recharge that produced the best model fit are shown in Table 4.1. The MSE for the best fit was 1.85 m^2 (19.98 ft^2) while the average residual was 0.029 m (+0.096).

Measured heads were available at 29 nodes. The areal distribution of the residuals are plotted in Figure 4.5 to help evaluate the model results. In terms of sign and magnitude, the residuals appear to be evenly distributed

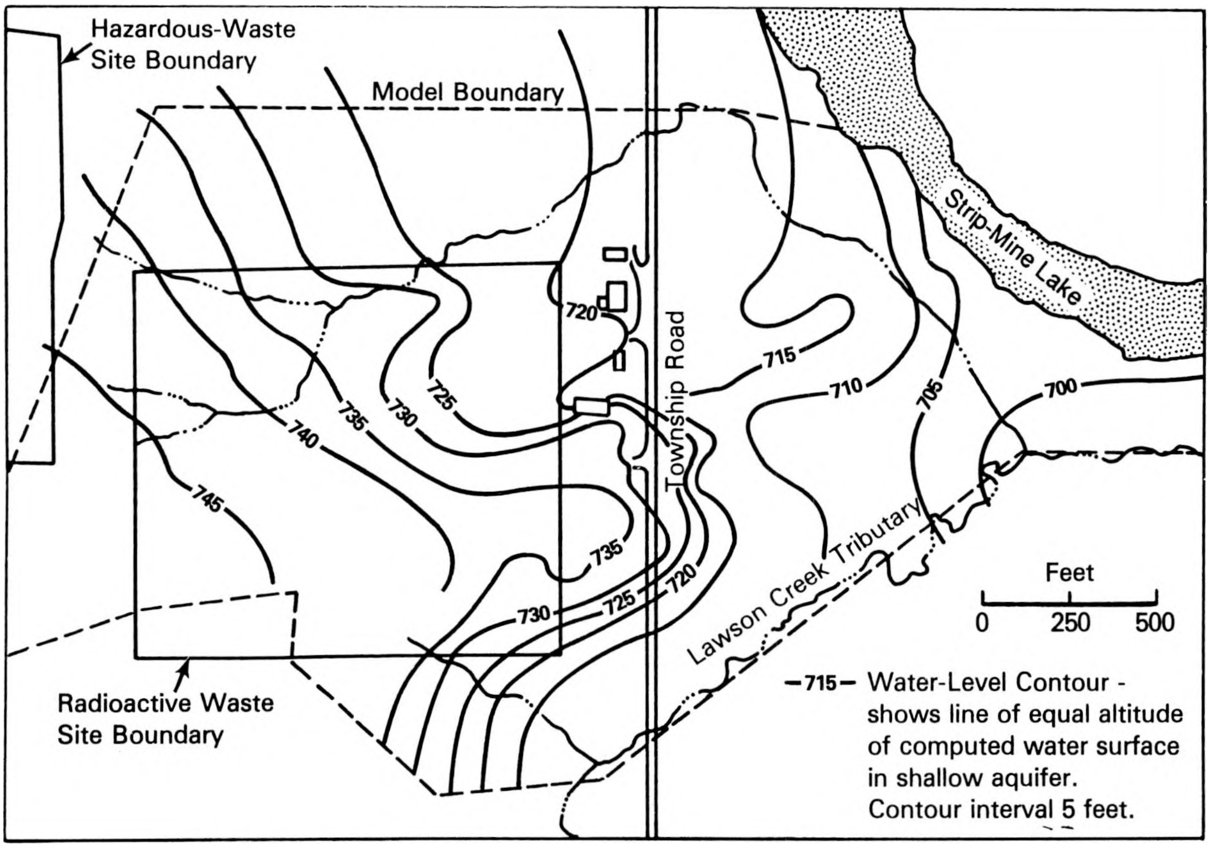


FIGURE 4.4. Hydraulic Heads Predicted by the Two-Dimensional Ground-Water Flow Model of Garklavs and Healy (1986)

throughout the area with the exception of the three shaded areas. In each of the shaded areas, the residuals are uniformly high in magnitude and have the same sign. These areas are adjacent to borders between different zones indicating that the zonal boundaries used in the model may not correspond exactly to the boundaries that exist in the aquifer (Garklavs and Healy 1986).

The water-level contours shown in Figures 4.2 and 4.4 do not match exactly but follow similar patterns. Basin II has a more complex geology and fewer data points than Basin I, making the modeling more difficult.

4.2.2.3 Modeling Conclusions

Garklavs and Healy (1986) calculated ground-water velocities for Basin I using Darcy's equation and the available values for head gradient and horizontal hydraulic conductivity. Volumetric porosities of 0.35 for sand, 0.40 for pebbly sand, and 0.05 for all other units were used. Ground-water

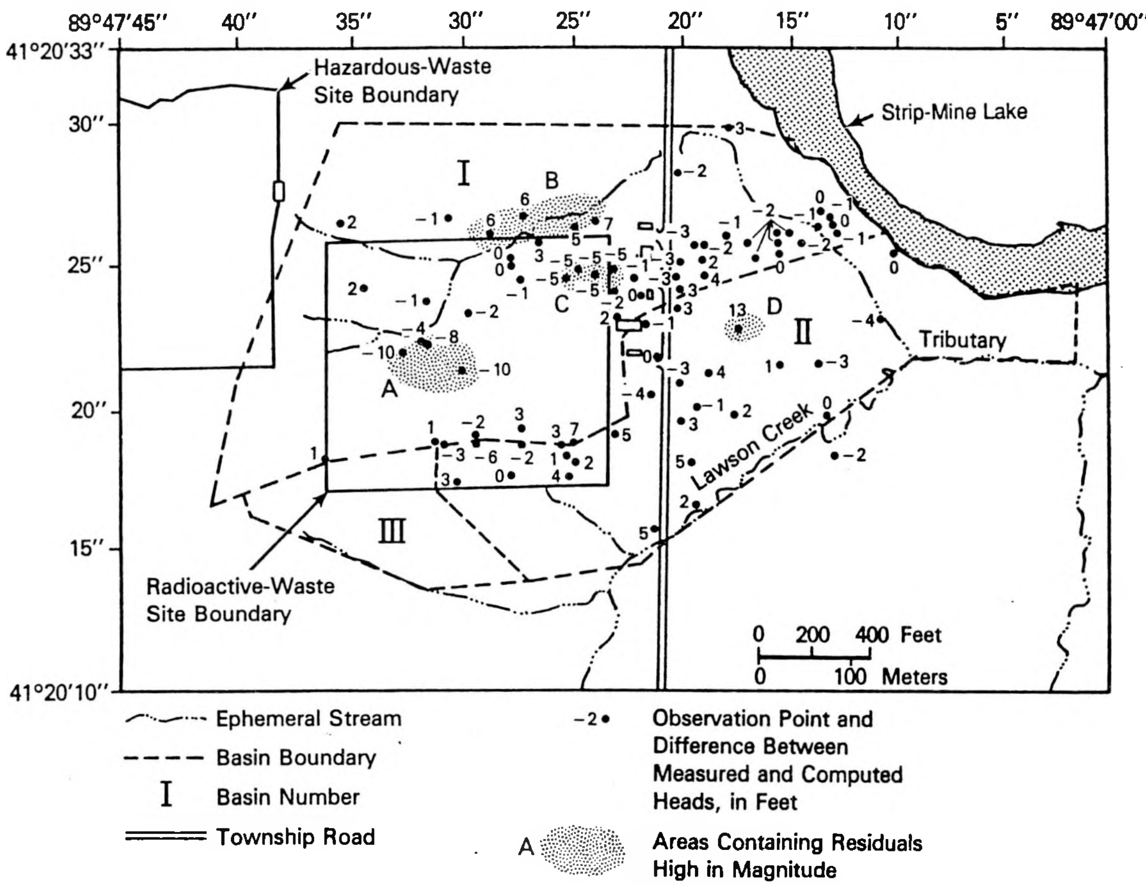


FIGURE 4.5. Difference Between Modeled and Observed Hydraulic Heads from the Two-Dimensional Model of Garklavs and Healy (1986)

velocities ranged from 1.5 m/yr in till to 914 m/yr in the pebbly-sand-filled channel. The calculated value for the pebbly-sand-filled channel agreed well with velocities based on tracer tests (Garklavs and Toler 1985).

Garklavs and Healy (1986) also calculated ground-water velocities for the model-computed heads in Basin II, assuming a porosity of 0.25 for the alluvial deposits, 0.35 for sand, and 0.40 for pebbly sand. Velocities ranged from 12 to 487 m/yr.

From this modeling exercise, Garklavs and Healy (1986) concluded that the pebbly-sand unit could not be in complete hydraulic connection with the lake. By varying the parameters of Zone 5, they found that the pebbly sand may be connected to the strip-mine lake in isolated locations; however, in these locations the saturated thickness is probably quite small. The length of Zone 6 was also varied. Extension to the lake produced a drastic reduction in the computed heads within the basin, while extension onto the site drew down computed heads to the east and south.

Garklavs and Healy (1986) also concluded that, based on a parameter sensitivity analysis, the results are relatively sensitive to changes in the hydraulic conductivity and recharge values assumed. Decreasing the hydraulic conductivity by half or increasing the recharge by two caused a fiftyfold increase in the MSE. They observed the model to be somewhat less sensitive to lake level assumed at the constant head boundary of the model.

4.2.3 Development and Calibration of Three-Dimensional Model

The horizontal and vertical distribution of the high-permeability Toulon pebbly-sand unit proved to be an important factor in the calibration of the USGS two-dimensional ground-water flow model, and also in the distribution of inferred travel paths from the trenches inferred from those modeling results. For this reason, a three-dimensional ground-water flow model was constructed that more accurately represented the distribution of the pebbly-sand unit.

This section presents the results of a three-dimensional ground-water flow model of the site developed at PNL. This modeling effort builds on the previous two-dimensional modeling done by Garklavs and Healy (1986) and Foster et al. (1984a). The discussion includes the development and design of the model, model calibration, travel-path and -time analyses, and important modeling conclusions.

4.2.3.1 Model Selection and Design

For the study, the Coupled Fluid, Energy, and Solute Transport (CFEST) code was selected. The code was developed by Gupta et al. (1987) for the Underground Storage Program managed for the U.S. Department of Energy (DOE) by PNL. CFEST is a finite-element code that is capable of simulating ground-water flow in a three-dimensional region. This code was selected because it has numerous features that simplify constructing and calibrating a three-dimensional ground-water flow model with many nodes. First, CFEST has a preprocessing code that assimilates kriged hydrogeologic unit surfaces and surface node locations into the three-dimensional model grid. Second, a stochastic inverse version of CFEST has been developed that greatly facilitates the definition of hydraulic conductivity and recharge zones for model calibration and automatically calibrates the modeled head distribution to an observed head distribution using available recharge and hydraulic conductivity data. Finally, CFEST has the capability to generate streamlines that make it possible to calculate travel paths and times through complex three-dimensional head and hydraulic conductivity distributions.

The model assumed the site to be enclosed by one large basin consisting of the union of the three ground-water basins delineated by Garklavs and Healy (1986). The boundary of the model is shown in Figure 4.6. The contact between undisturbed Quaternary materials and strip-mine spoils defines a ground-water divide to the north that was treated as a no-flow boundary in the model. A bedrock high forms a ground-water divide to the west that was treated as a no-flow boundary in the model. A tributary to Lawson Creek forms a ground-water sink to the south that was modeled by placing a thin zone of high-permeability material along the southern no-flow boundary. This sink directs ground-water flow from approximately the southern one-third of the study area northeastward toward the strip-mine lake, which is considered to be a prescribed-head boundary. No lateral inflow to the system was assumed. All inflow was assumed to be recharge from precipitation falling directly on the basin. All outflow was assumed to discharge at the strip-mine lake.

The ground-water system was assumed to be at steady state. The top of the ground-water flow model coincided with the water table measured on July 10 and 11, 1984 (Figure 4.7), which is the highest water table on record. The bottom of the model coincides with the surface of the Carbondale (Shale) Formation (Figure 4.8), which is assumed to be impermeable.

The shallow aquifer was assumed to be divided into three distinct layers. Model Layer 1 consists of the Cahokia Alluvium, Peoria Loess, Roxana Silt, Berry Clay, and Radnor Till. Model Layer 2 consists of the pebbly-sand unit of the Toulon. Model Layer 3 consists of the other units of the Toulon Member, the Hulick Till, and the Duncan Mills Silt. The saturated thickness and saturated lateral extent of each model layer are shown in Figures 4.9 through 4.11.

The partial differential equation describing three-dimensional steady-state ground-water flow in an anisotropic aquifer was used:

$$\frac{\partial}{\partial x}\left[Kx\ \frac{\partial h}{\partial x}\right] + \frac{\partial}{\partial y}\left[Ky\ \frac{\partial h}{\partial y}\right] + \frac{\partial}{\partial z}\left[Kz\ \frac{\partial h}{\partial z}\right] + W = 0 \qquad (3)$$

where K = transmissivity (length/time)
h = mean hydraulic head in the vertical (length)
x,y,z = cartesian coordinates (length)
W = recharge rate (length/time).

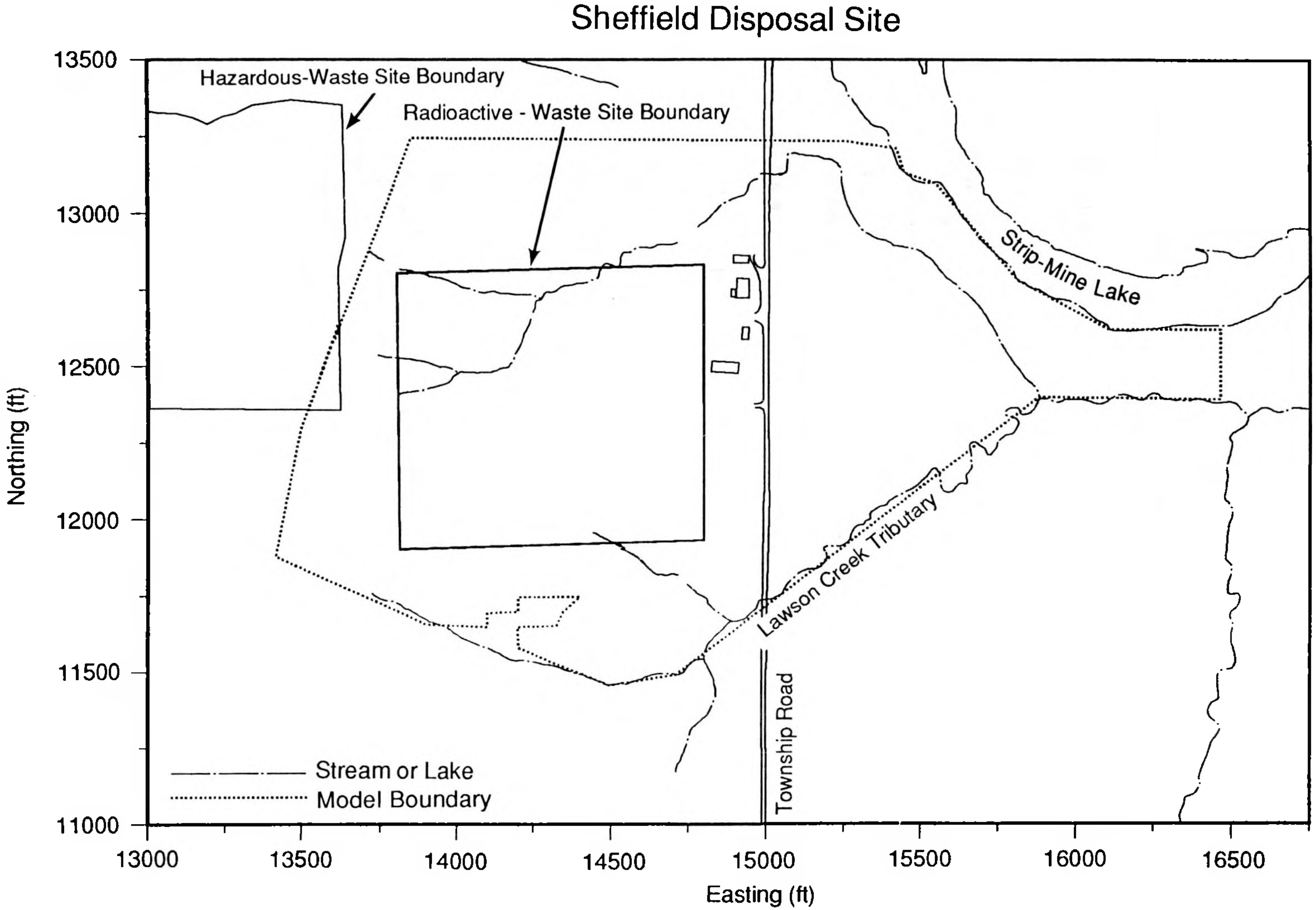


FIGURE 4.6. Horizontal Boundary of the Three-Dimensional Ground-Water Flow Model

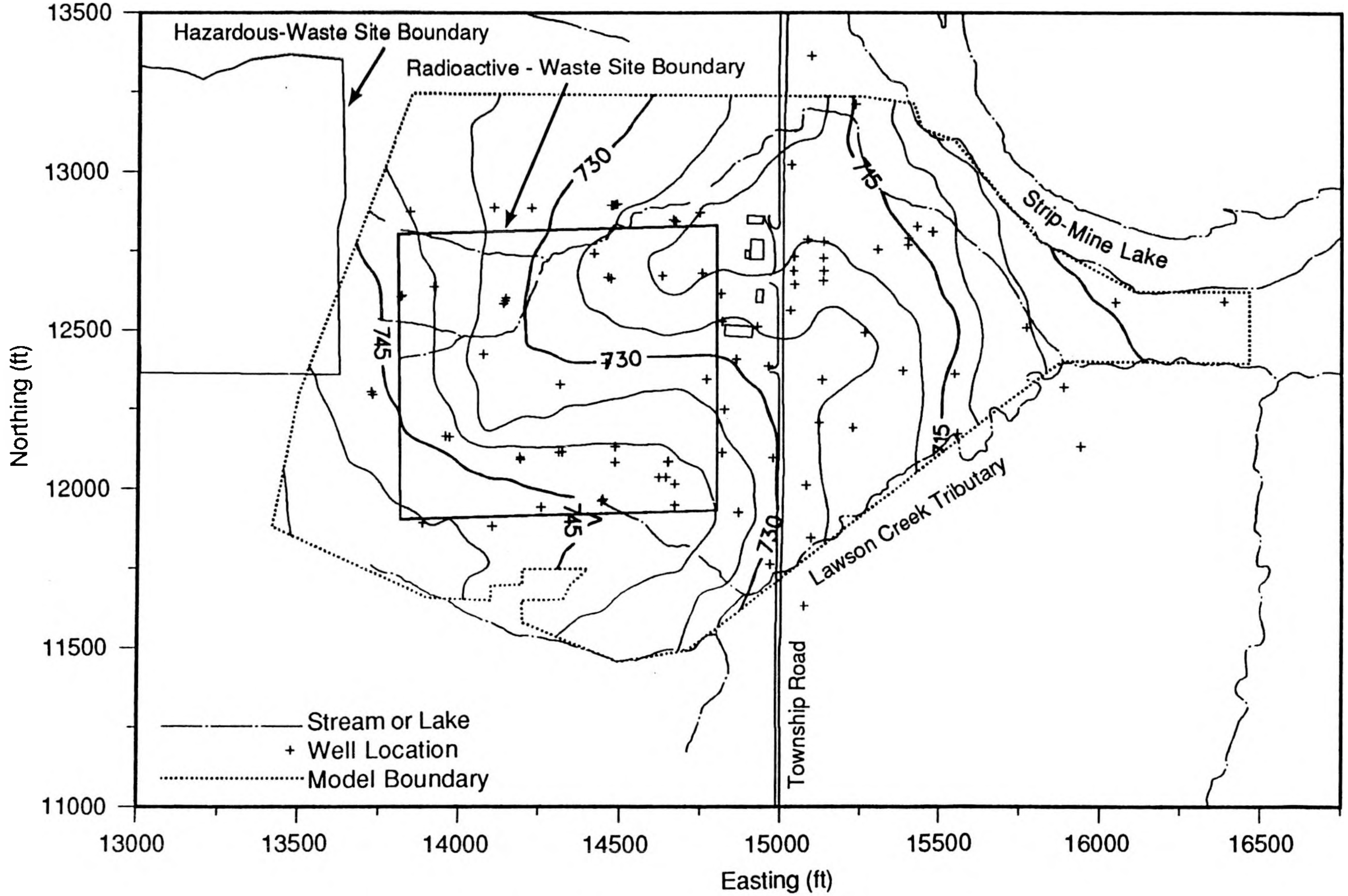


FIGURE 4.7. Water Table Measured on July 10, 1984

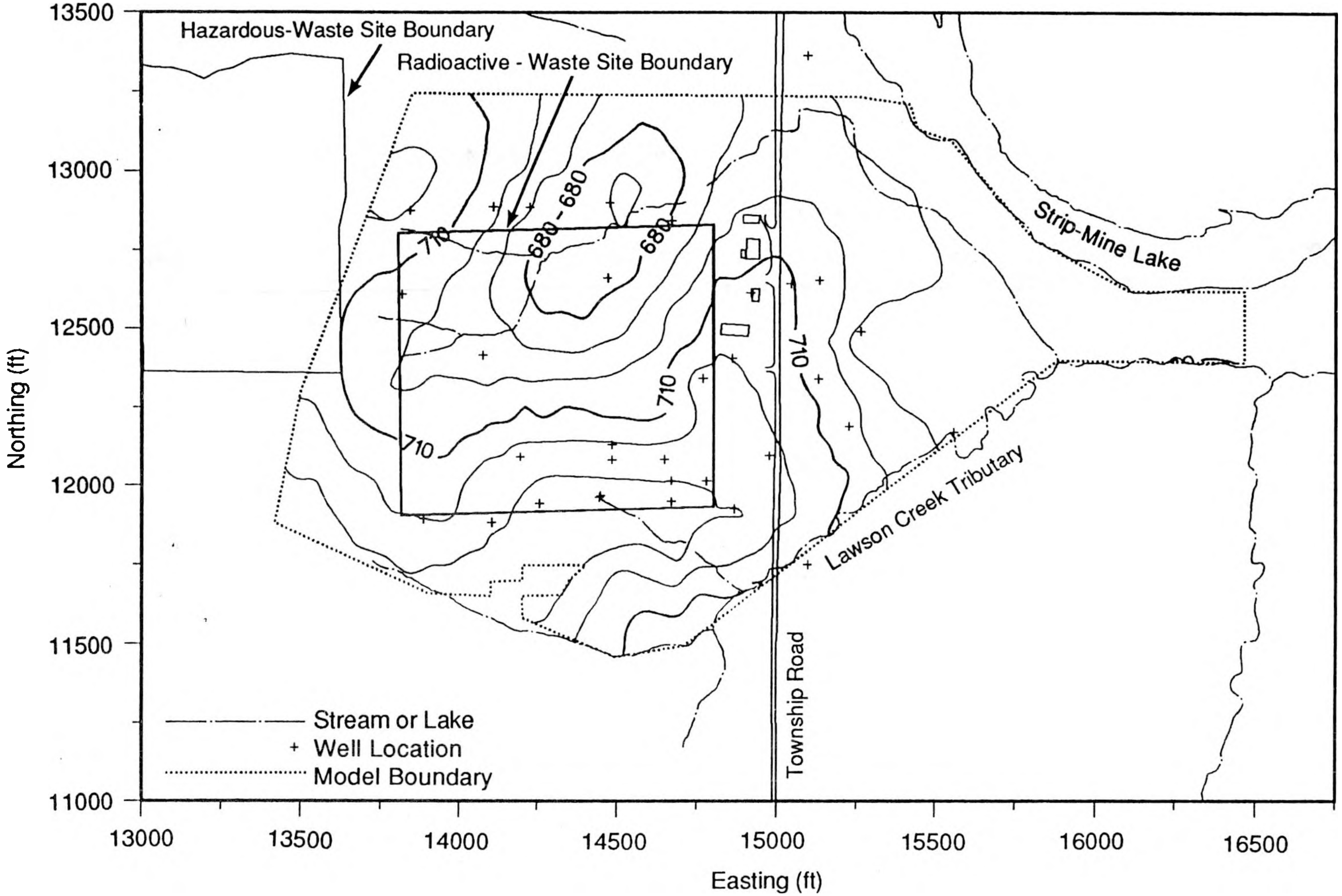


FIGURE 4.8. Upper Surface of the Carbondale Shale Formation (bedrock)

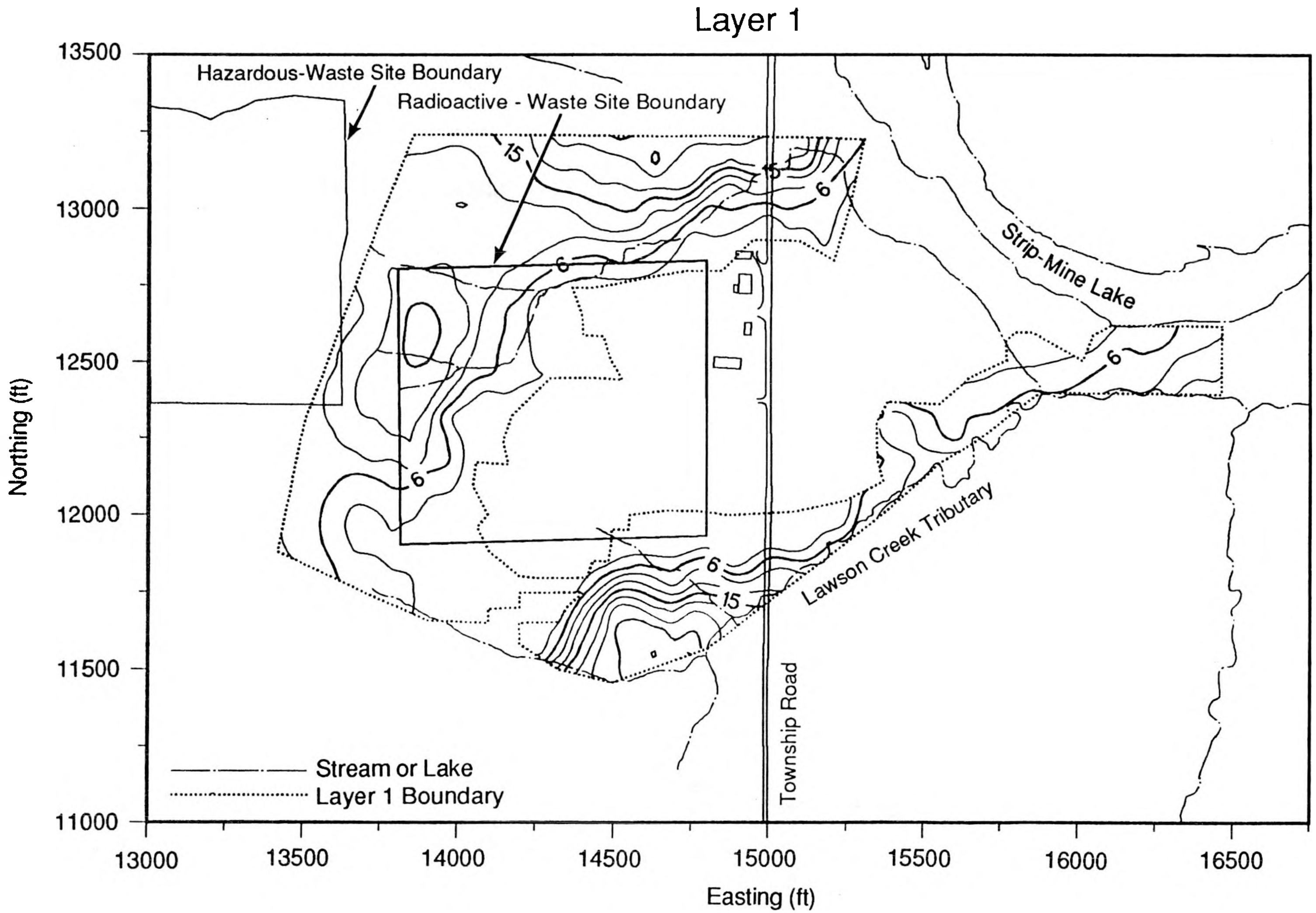


FIGURE 4.9. Saturated Thickness and Lateral Extent of Model Layer 1

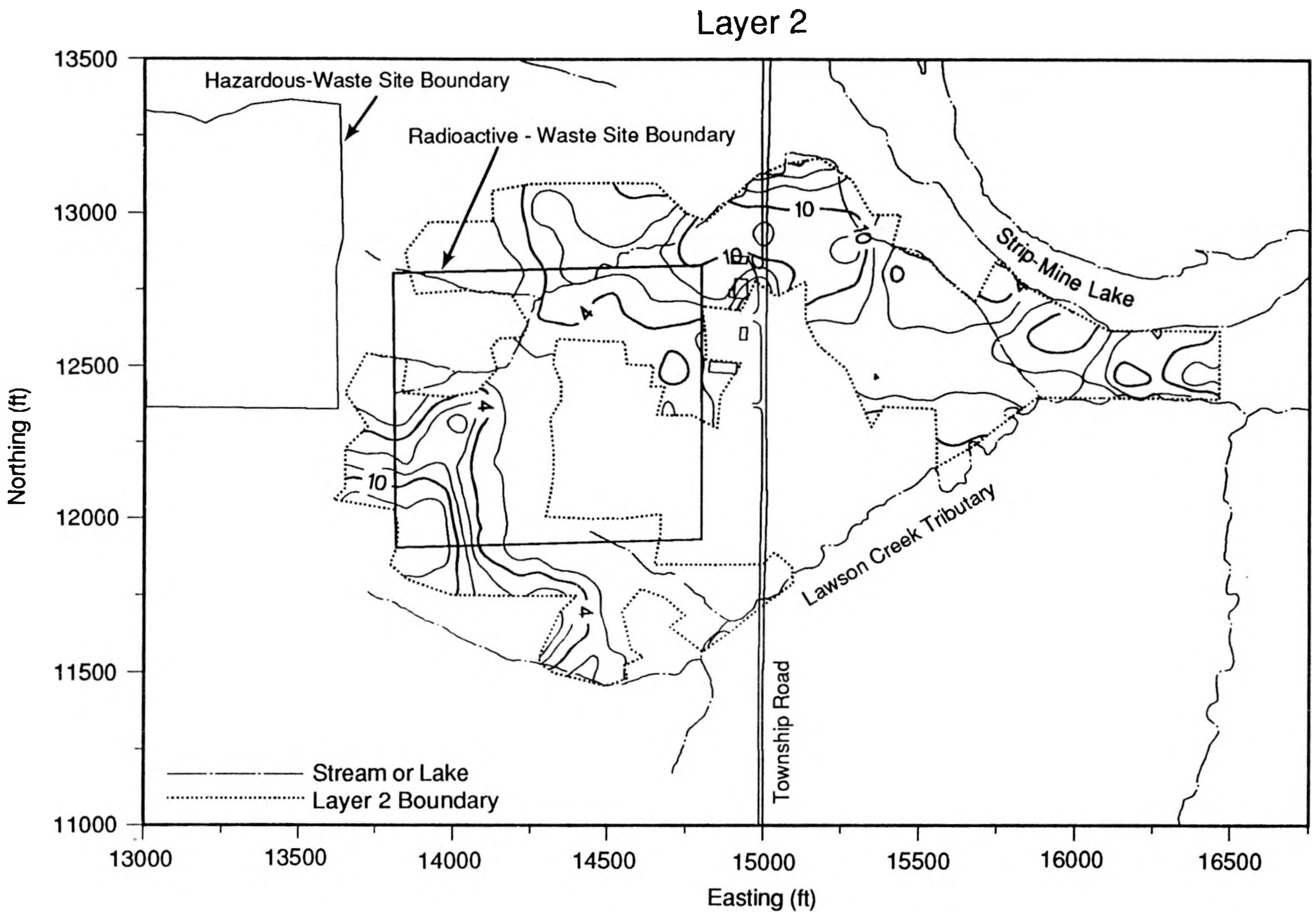


FIGURE 4.10. Saturated Thickness and Lateral Extent of Model Layer 2

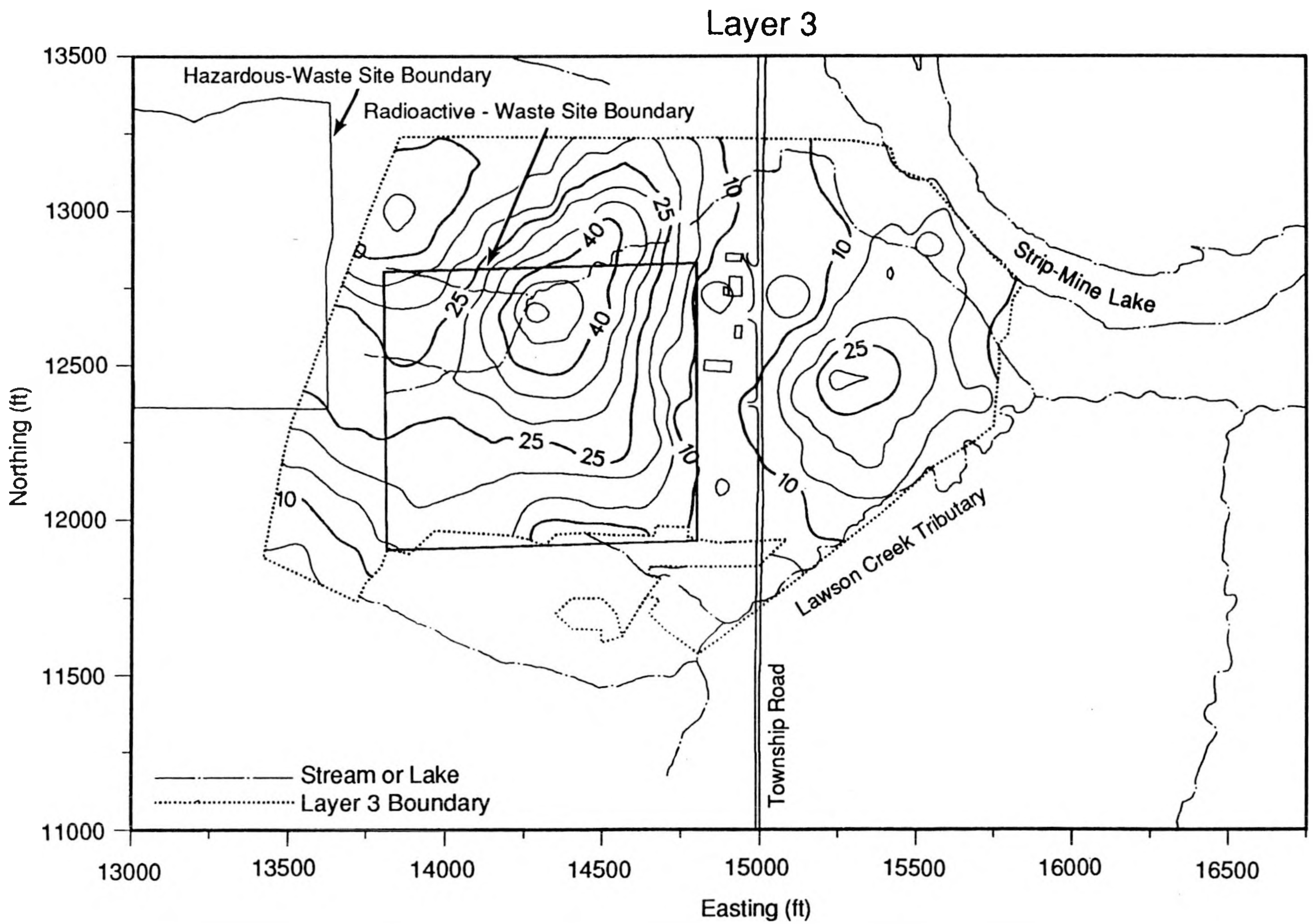


FIGURE 4.11. Saturated Thickness and Lateral Extent of Model Layer 3

The ground-water flow model is based on a bilinear, quadrilateral, finite-element mesh (Figure 4.12) that covers about 0.31 km^2. The mesh was designed so that most borehole locations and important geographic features pertinent to the simulation coincide with a surface node. More nodes exist in the subsurface beneath each surface node at the interface of each model layer. The mesh is composed of 479 surface nodes and 446 surface elements, with 1283 nodes and 864 elements in total.

The model layer surfaces were obtained by two-dimensional kriging of geologic log data. The hydraulic head at each model node was obtained by three-dimensional kriging of observed hydraulic heads. The observed heads were assumed to represent the head in the lowermost hydrogeologic unit penetrated by the screened interval of the well. The location of the water table was determined by iteratively changing the elevation of each surface node until its elevation corresponded to the kriged hydraulic head. The kriged heads at the top of each model layer and at the bottom of the model are shown in Figures 4.13 and 4.14.

4.2.3.2 Model Calibration

Initial estimates of hydraulic conductivity for each model layer were obtained from field and laboratory measurements documented in Garklavs and Healy (1986) and Foster et al. (1984a). The initial estimates of the hydraulic conductivity of each layer are: 9E-8 m/s (0.28 m/yr) for Layer 1, 1.2E-4 m/s (3785 m/yr) for the pebbly-sand unit, and 2.3E-7 m/s (7.6 m/yr) for Layer 3.

The recharge distribution was then calibrated by dividing the model surface into seven recharge zones (Figure 4.15): recharge Zones 1 and 2 delineate where Layer 1 outcrops on the surface, recharge Zones 3, 4, and 5 correspond to where the pebbly-sand unit of the Toulon (Layer 2) outcrops on the surface, and recharge Zones 6 and 7 correspond to where model Layer 3 outcrops on the surface. The optimum recharge values for each zone (based on the initial estimates of hydraulic conductivity) are listed in Table 4.2.

A notable result of the recharge calibration is that the values all lie between 2.5 and 10.2 cm/yr, whereas the range of recharge calibrated in the two-dimensional modeling was from 1 to 71.1 cm/yr.

The hydraulic conductivity distribution was calibrated using these calibrated recharge values by dividing Layer 1 into three zones, Layer 2 into seven zones, and Layer 3 into five zones for a total of 15 hydraulic conductivity zones (Figure 4.16). Zones 1 and 2 represent where Layer 1 intersects the water table. Zone 3 represents the alluvial deposits within the stream channel along the southern border of the model area. Zones 5, 6, and 9 represent where the Toulon pebbly-sand unit intersects the water table. Zones 4, 7, 8, and 10 represent the Toulon pebbly-sand unit where it is overlain by any of the units lumped into Layer 1. Zones 12, 13, and 14 represent where the till and silts of Layer 3 intersect the water table. Zones 11 and 15

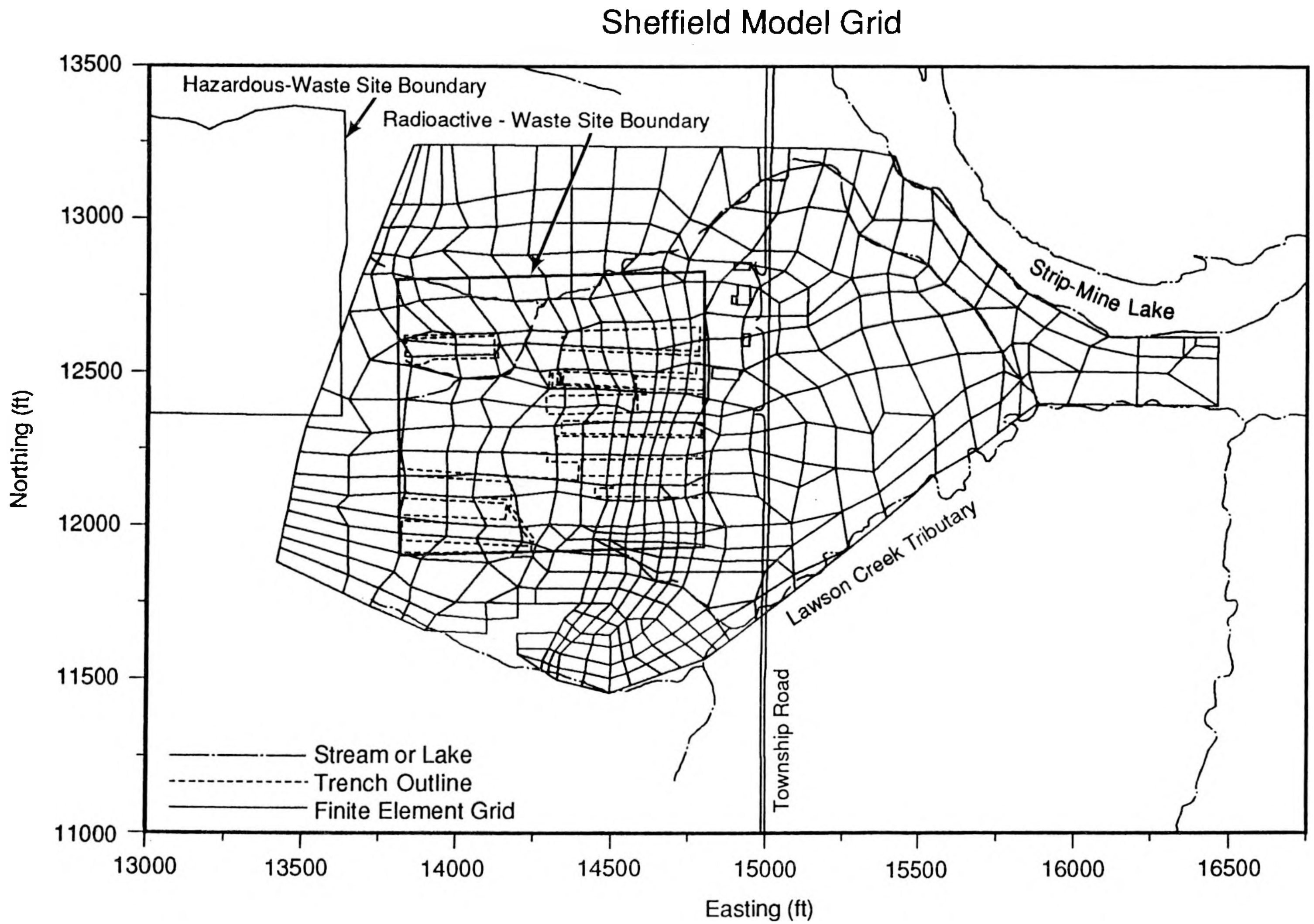
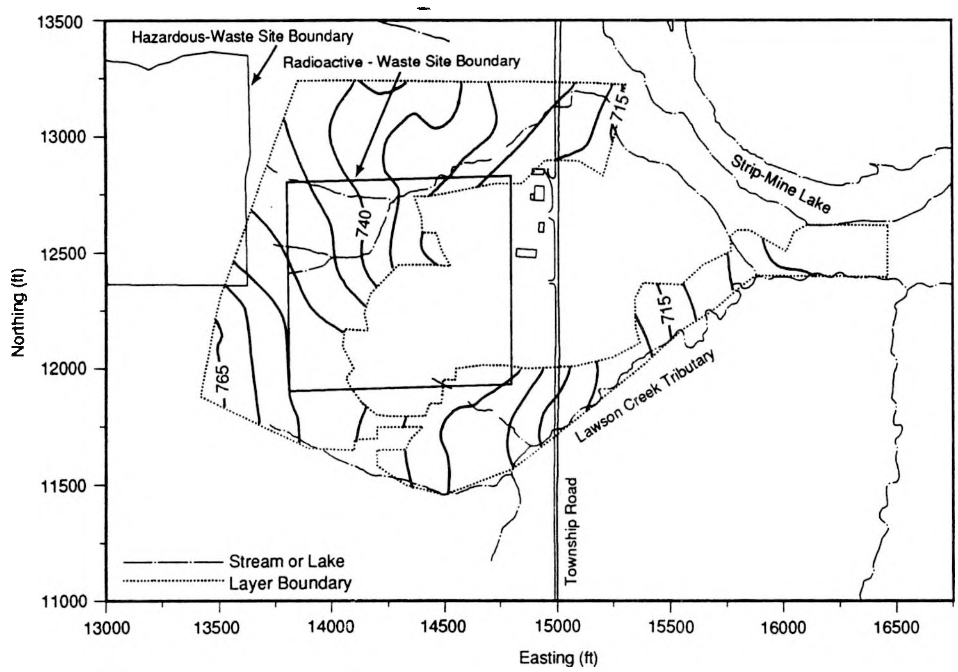
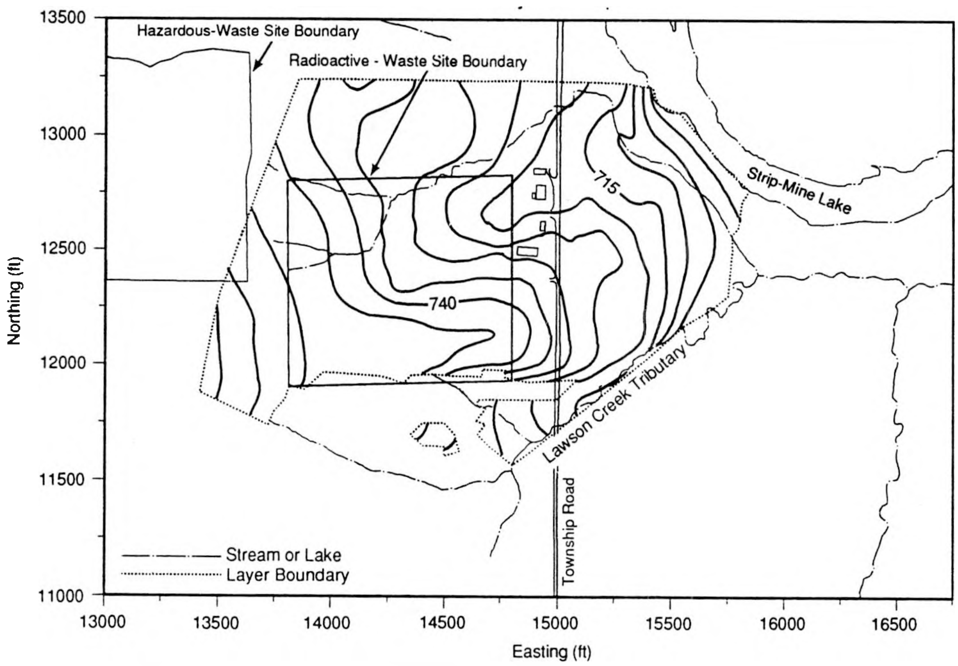


FIGURE 4.12. Finite-Element Grid for Three-Dimensional Ground-Water Flow Model

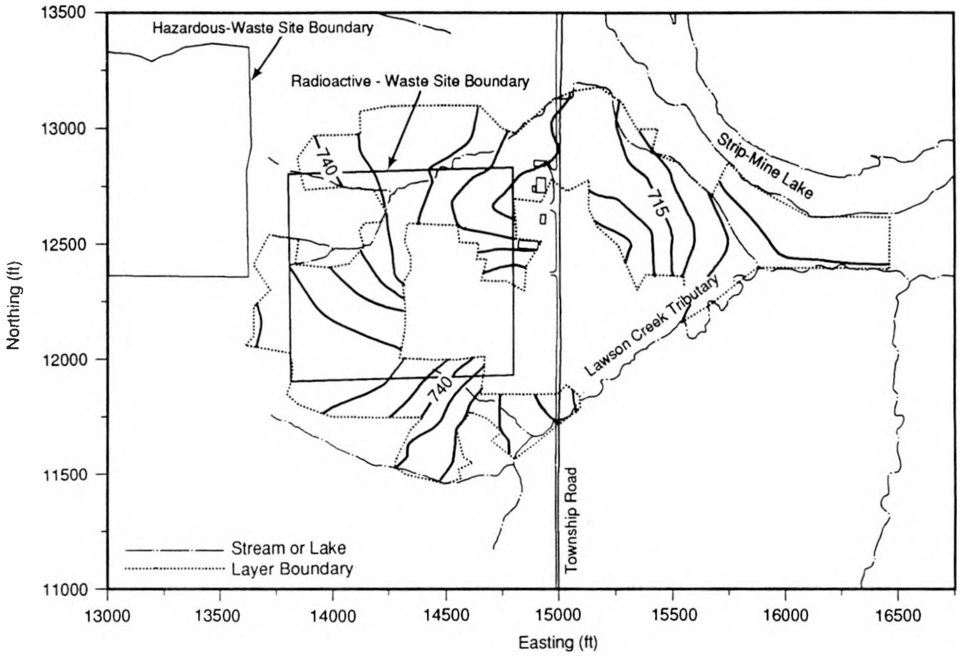


a) Layer 1 Top

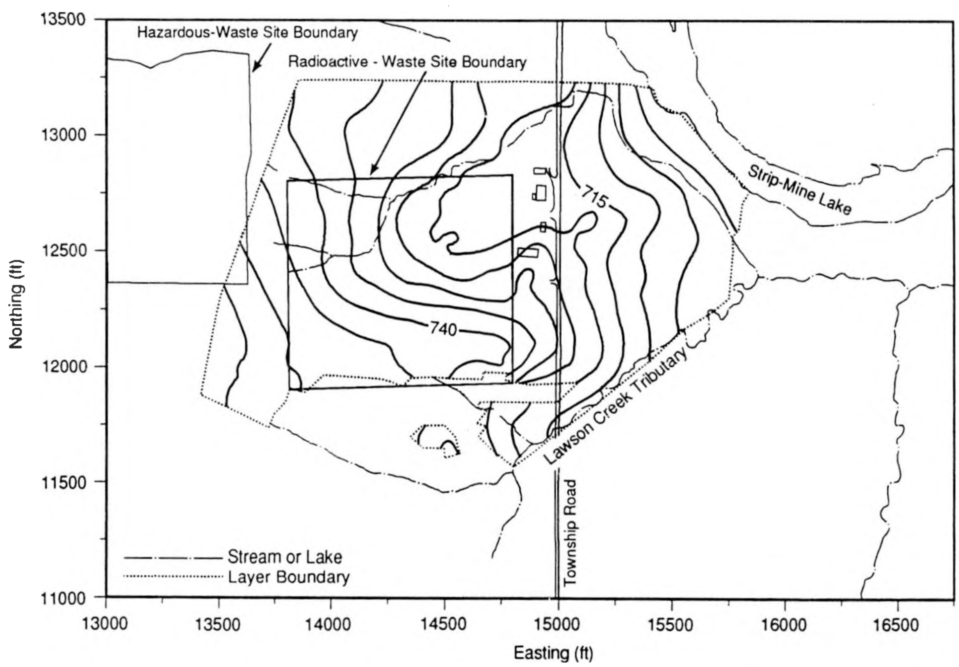


b) Layer 3 Top

FIGURE 4.13. Kriged Hydraulic Heads for the Three-Dimensional Ground-Water Flow Model



a) Layer 2 Top



b) Layer 3 Bottom

FIGURE 4.14. Kriged Hydraulic Heads for the Three-Dimensional Ground-Water Flow Model

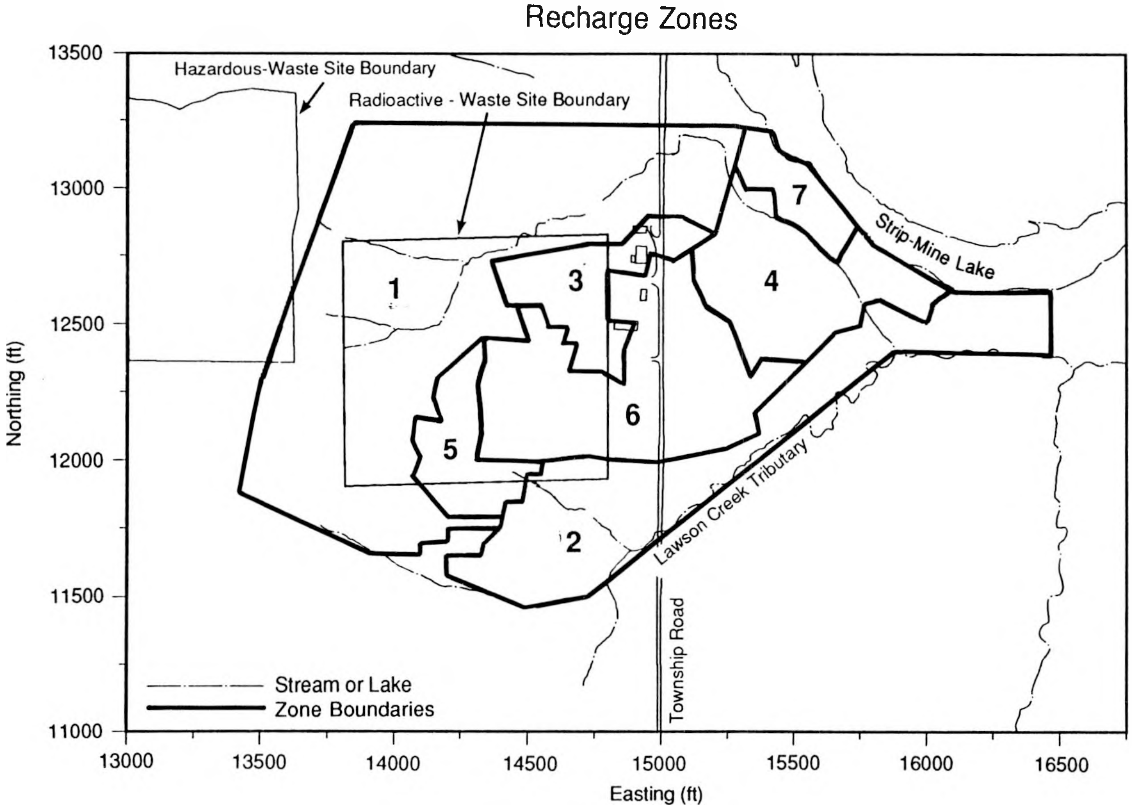


FIGURE 4.15. Surface Recharge Zones for the Three-Dimensional Ground-Water Flow Model

TABLE 4.2. Calibrated Values of Recharge

Recharge Zone	Layer Outcropping	Recharge (m/s)	Recharge (cm/yr)
1	1	3.0E-9	9.7
2	1	1.1E-9	3.3
3	2	2.2E-9	7.1
4	2	1.7E-9	5.6
5	2	1.6E-9	5.1
6	3	1.9E-9	6.1
7	3	1.2E-9	3.6

represent where other units are present between Layer 3 and the water table. The calibrated values of hydraulic conductivity for each of the 15 zones are listed in Table 4.3.

Generally, the calibrated values of hydraulic conductivity for Layer 3 were higher than initially expected. The calibrated conductivity values for Layer 3 could have been lower if a fourth layer had been included in the

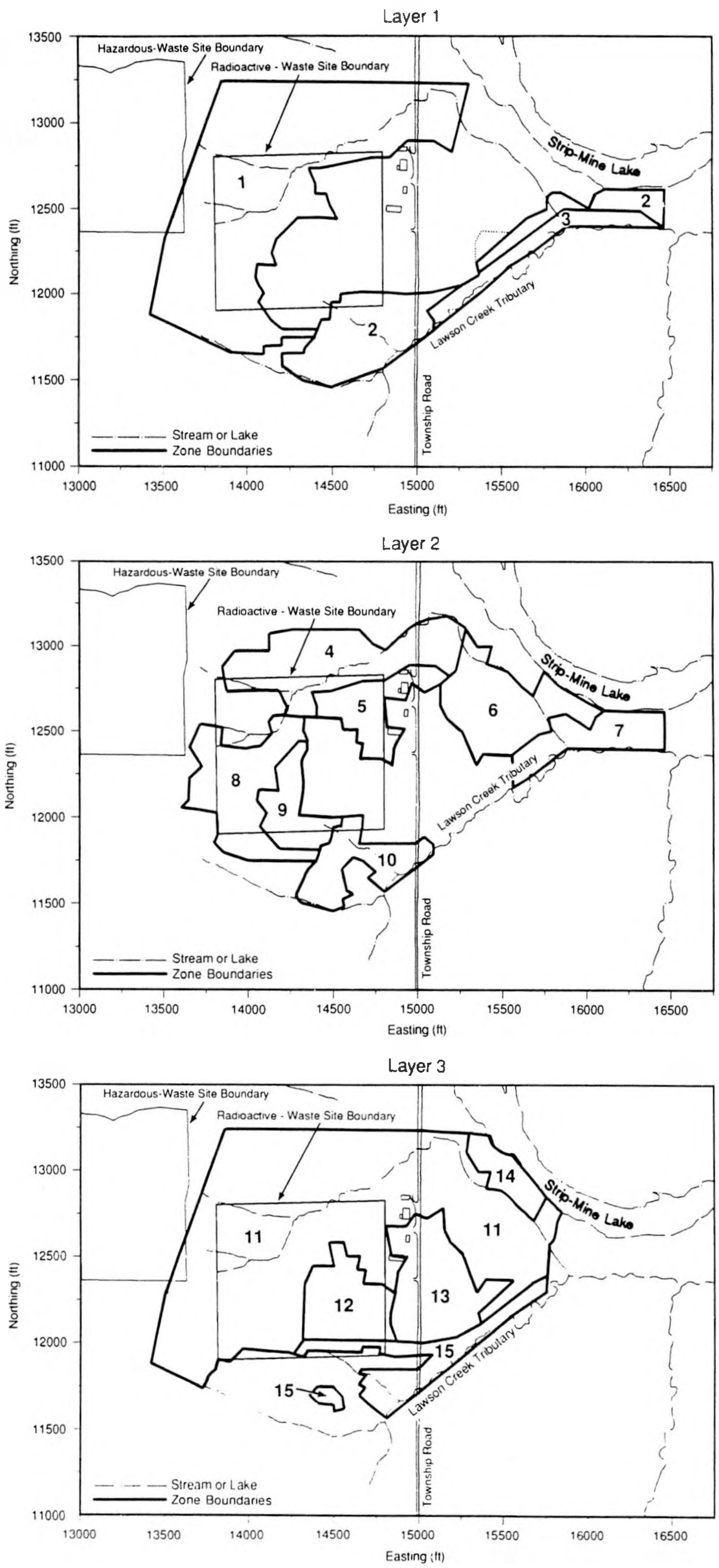


FIGURE 4.16. Hydraulic Conductivity Zones for Each Layer of the Three-Dimensional Ground-Water Flow Model

TABLE 4.3. Calibrated Values of Horizontal (K_x,K_y) and Vertical (K_z) Hydraulic Conductivity

Hydraulic Conductivity Zone	Layer	Hydraulic Conductivity (m/s)		
			Inverse Calibrated Values	
		Initial Value	K_x, K_y	K_z
1	1	9.63E-8	1.26E-6	1.60E-8
2	1	9.63E-8	4.66E-7	3.93E-8
3	1	9.57E-5	7.28E-5	9.42E-5
4	2	9.57E-5	9.63E-5	9.60E-5
5	2	9.57E-5	7.58E-5	9.47E-5
6	2	9.57E-5	9.66E-5	9.44E-5
7	2	9.57E-5	1.18E-5	9.54E-5
8	2	9.57E-5	6.49E-6	9.51E-5
9	2	9.57E-5	6.31E-6	9.51E-5
10	2	9.57E-5	1.19E-5	9.20E-5
11	3	2.37E-7	2.33E-6	1.92E-7
12	3	2.37E-7	2.37E-7	1.75E-7
13	3	2.37E-7	3.47E-7	1.61E-7
14	3	2.37E-7	7.80E-7	2.04E-7
15	3	2.37E-7	1.77E-7	2.29E-7

model, representing the high-permeability fracture and coal-seam area at the top of the bedrock. The model was not sensitive to the vertical hydraulic conductivity of the pebbly-sand unit (Layer 2). The horizontal hydraulic conductivity of the pebbly-sand unit was predicted to be lower in the southwest portion of the site. The model was sensitive to both the horizontal and vertical hydraulic conductivity of Layer 1. Anisotropy ratios of 100 in Zone 1 and 10 in Zone 2 were predicted.

The calibrated water table is shown in Figure 4.17. Water levels computed by the model were compared with water levels measured in July 1984. The model produced a fairly good replication of the observed heads. The water-level contours shown in Figures 4.7 and 4.17 do not match exactly but follow similar patterns. The hydraulic heads predicted at each of the model surfaces and at the bottom boundary of the model are shown in Figures 4.18 and 4.19 for comparison to Figures 4.13 and 4.14. The differences between the observed and modeled heads are shown in Figure 4.20. The MSE for the best fit to the 55 observation points was 3.93 m while the average residual was +0.18 m. The highest residuals appear in the same locations as in the two-dimensional modeling. Many of the highest residuals fall near the Layer 2 (Toulon pebbly-sand unit) boundary, indicating again that the zonal boundaries used in the model may not correspond exactly to the boundaries that exist in the aquifer. Perhaps more conductivity zones are needed in the model to match the hydraulic conductivity distribution that exists in the aquifer.

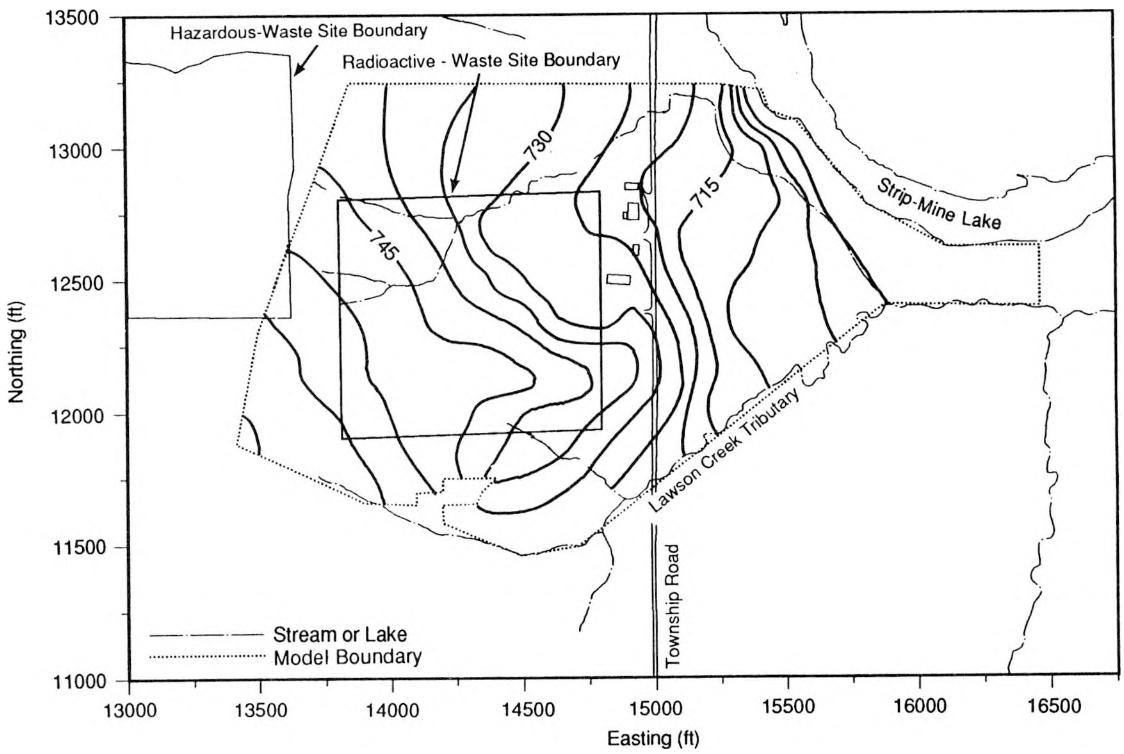
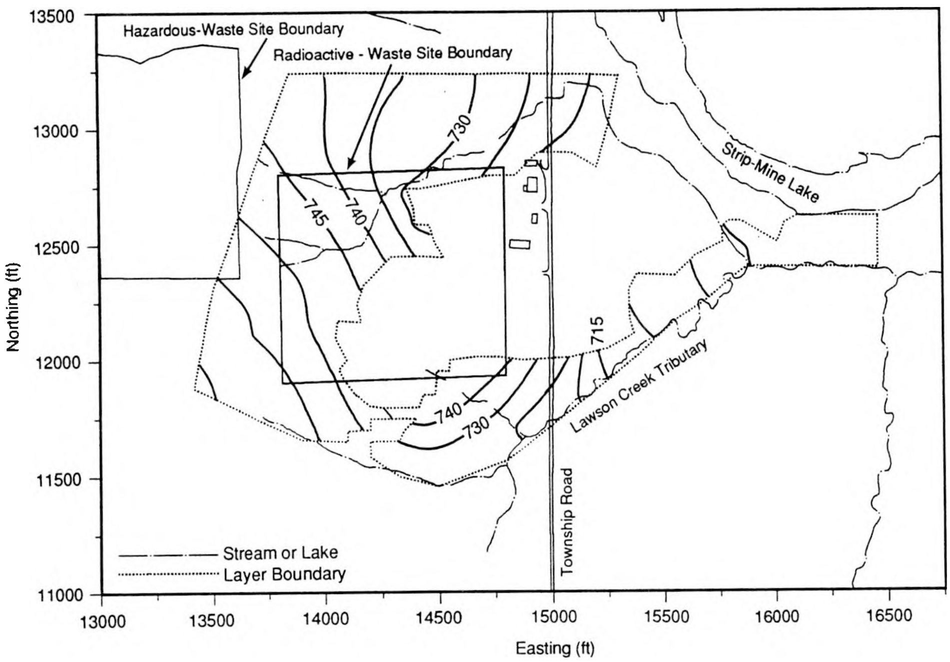


FIGURE 4.17. Calibrated Water Table Predicted by the Three-Dimensional Ground-Water Flow Model

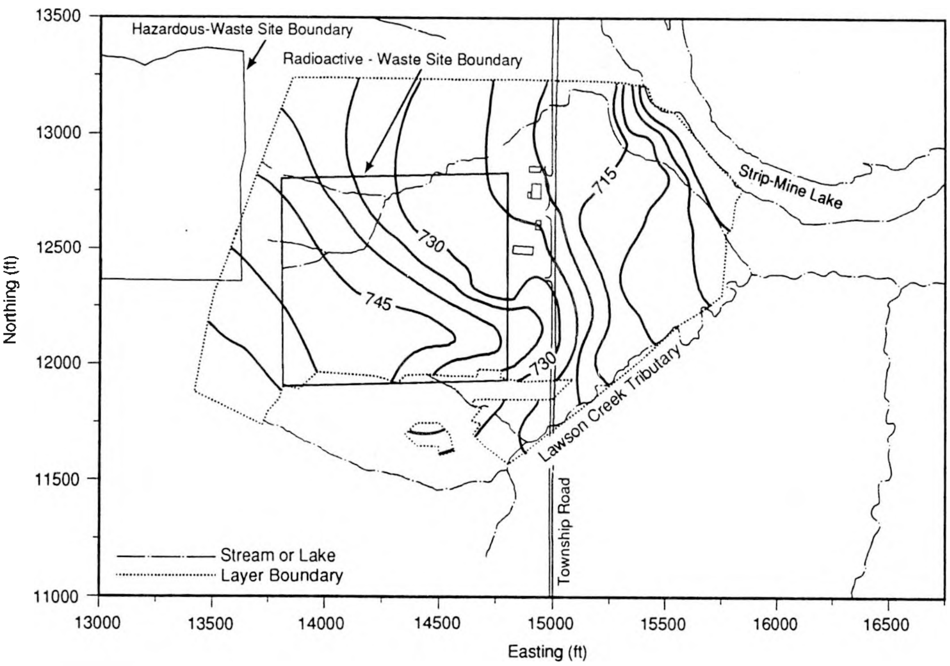
4.2.3.3 Streamline and Travel-Time Analysis

Streamlines originating from the east and west ends of each of the larger trenches were generated using the three-dimensional calibrated head and hydraulic conductivity distribution. The travel distances and times associated with the streamlines are presented in Table 4.4. A plan view of the streamlines is shown in Figure 4.21. Eventually, the streamlines all find their way into the Toulon pebbly-sand unit (Layer 2). Because the Toulon is not in complete hydraulic connection with the strip-mine lake, the streamlines exiting there have to follow a more circuitous route than in the two-dimensional model. Hence, the shortest travel times predicted by the three-dimensional model are about 4 yr for trenches in the northeast portion of the site.

For sake of comparison with the USGS two-dimensional modeling results, the hydraulic conductivity zones (Figure 4.2) and the water-table elevations (Figure 4.3) predicted by Garklavs and Healy (1986) were used with hand-constructed streamlines to calculate travel times at the east and west ends of each of the larger trenches within the disposal area (Figure 4.22). All of the streamlines converge on the high-permeability channel of Toulon pebbly sand (Zone 6, Basin I). The resulting travel times and distances obtained

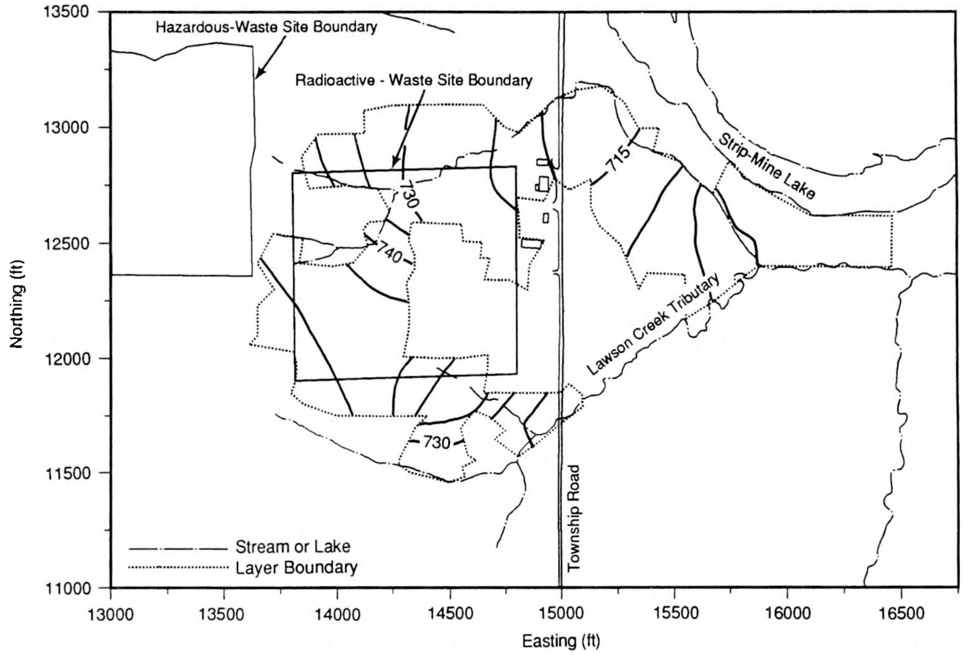


a) Layer 1 Top

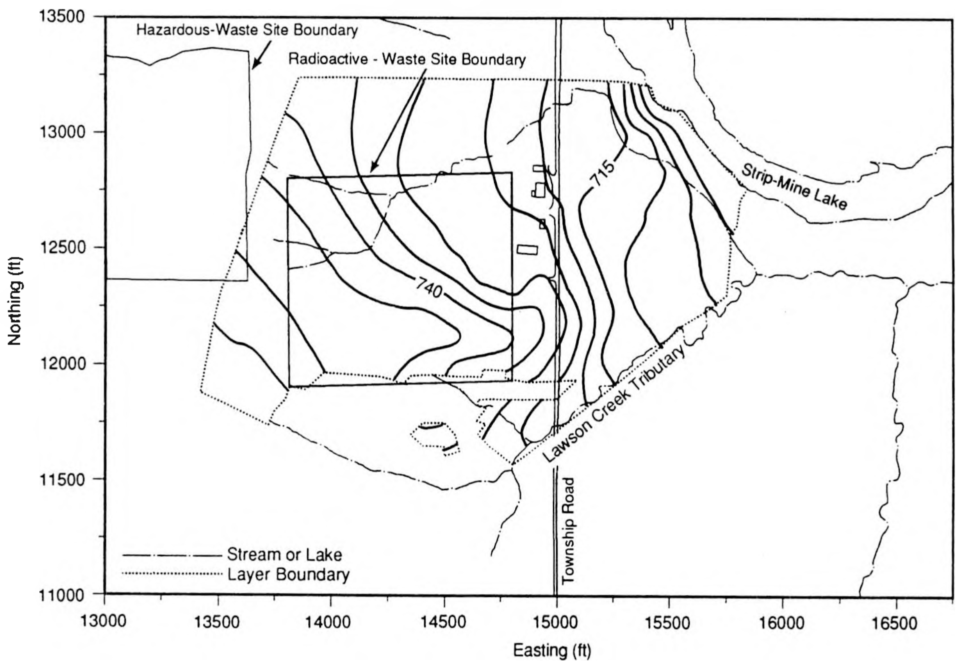


b) Layer 3 Top

FIGURE 4.18. Calibrated Hydraulic Heads for the Three-Dimensional Ground-Water Flow Model



a) Layer 2 Top



b) Layer 3 Bottom

FIGURE 4.19. Calibrated Hydraulic Heads for the Three-Dimensional Ground-Water Flow Model

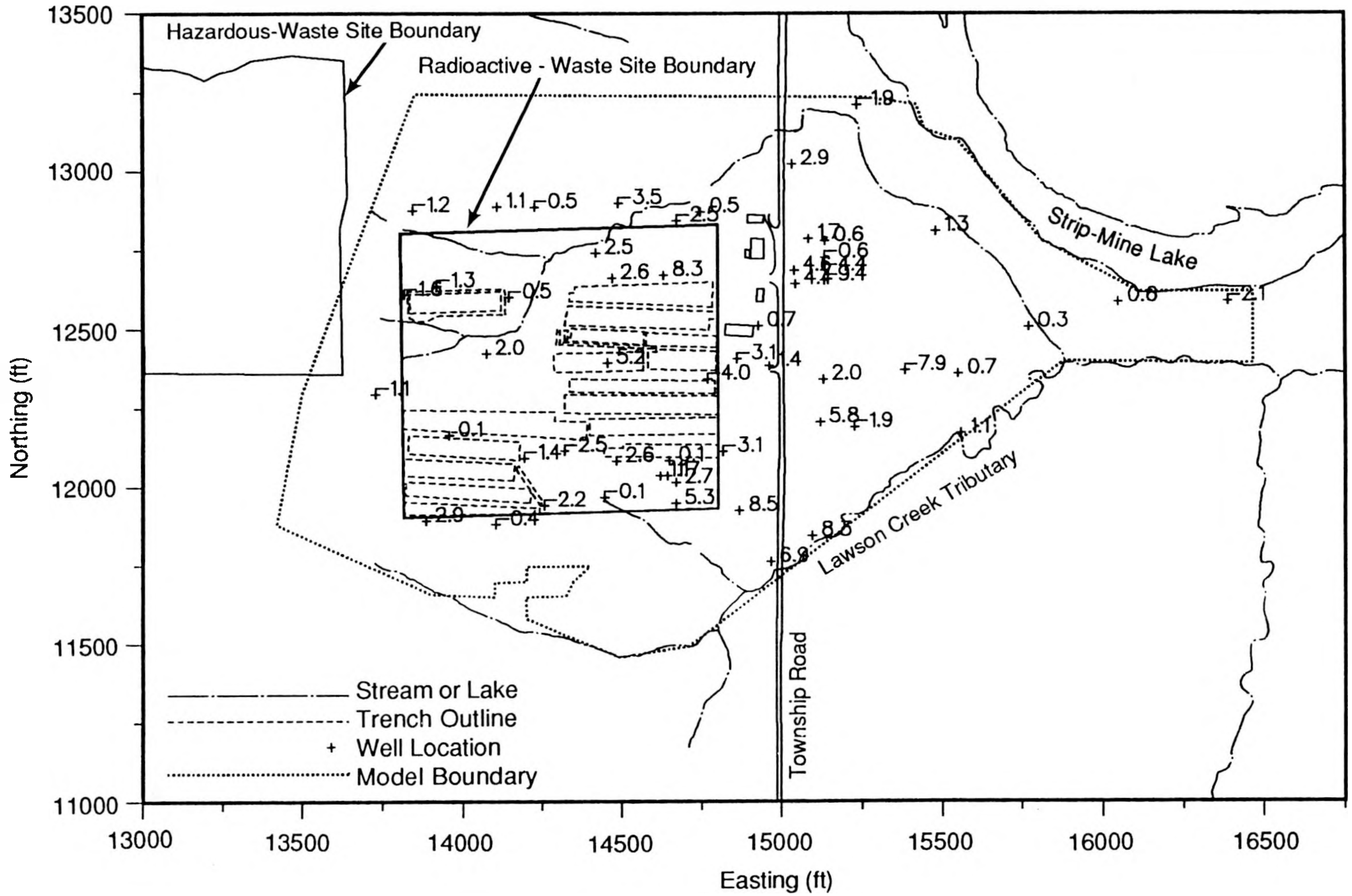


FIGURE 4.20. Difference Between Modeled and Observed Hydraulic Heads for the Three-Dimensional Ground-Water Flow Model

TABLE 4.4. Travel Distances and Times from Trenches to Model Boundary for the Three-Dimensional Model

Trench	Side	Travel Distance (m)	Travel Time (yr)
1	West	598	86.3
	East	561	86.0
2	West	622	115.1
	East	561	86.0
3	West	621	159.7
	East		101.3
4, 26	West	588	14.4
	East	522	6.7
5	West	799	115.3
	East	98	105.4
6	West	781	126.2
	East	193	16.3
7	West	352	179.5
	East	178	69.1
8, 25	West	590	12.1
	East	506	4.4
9	West	747	106.9
	East	703	101.3
11	West	195	83.7
	East	614	112.6
14	West	662	31.4
	East	654	20.7
14a	West	720	35.9
	East	654	228.
23	West	546	4.5
	East	439	6.6
24	West	560	6.7
	East	465	5.2

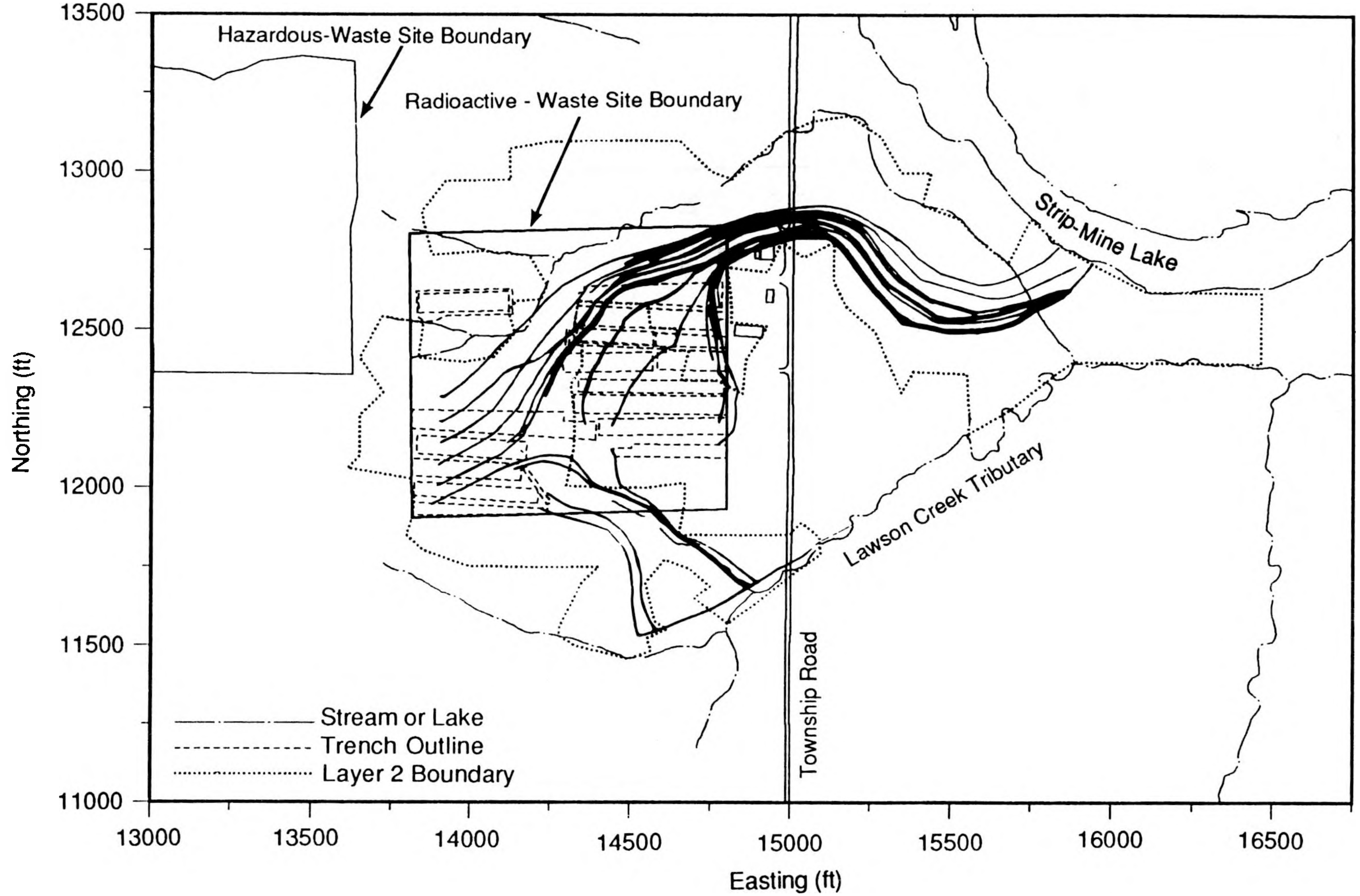


FIGURE 4.21. Travel Paths from Each of the Site Trenches Predicted by the Three-Dimensional Ground-Water Flow Model

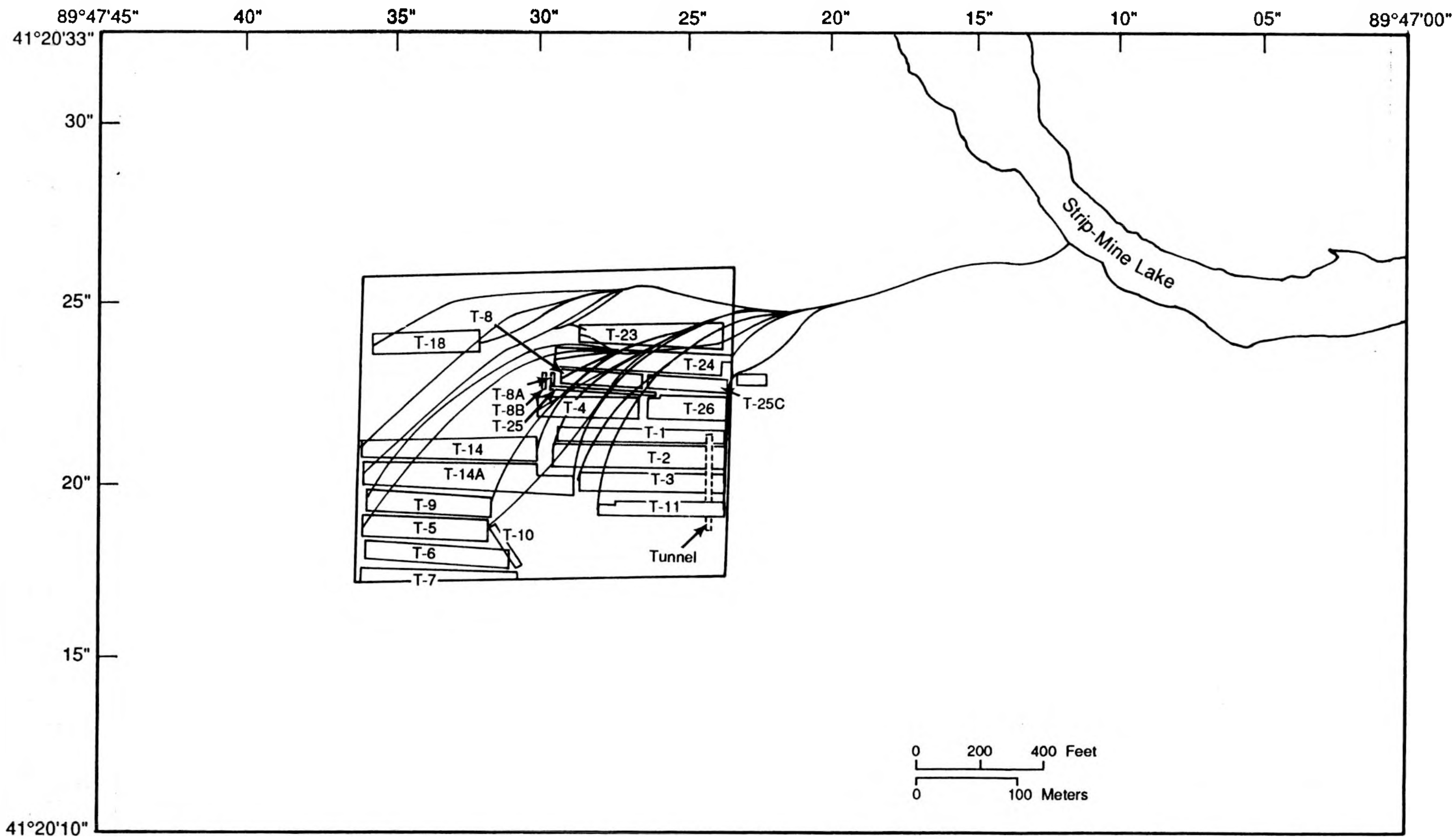


FIGURE 4.22. Travel Paths from Each of the Site Trenches, Hand-Constructed from the Hydraulic Head Distribution Predicted by the Two-Dimensional Ground-Water Flow Model of Garklavs and Healy (1986)

based on an assumed porosity of 0.35 are shown in Table 4.5. Estimated travel times varied from as little as 1 yr from the east end of Trench 23 to as much as 292 yr for the west end of Trench 5. Travel times for Trenches 6 and 7 could not be calculated because they are outside the model boundary (in Basin III). In general, the streamlines originating from Trenches 14, 14a, and 9 have much shorter travel times with the three-dimensional model because they are able to enter the pebbly-sand unit much closer to their point of origin than they were able to with the two-dimensional model. The streamlines originating from Trenches 5 (east side), 6, 7, and 11 (west side) all head south to the Cahokia Alluvium along the Lawson Creek Tributary before exiting the basin. Apparently the flow divide predicted by the three-dimensional model is further north than the flow divide assumed in the two-dimensional modeling.

4.2.3.4 Modeling Conclusions

The calibrated values of recharge for the three-dimensional model seem more reasonable than those of the two-dimensional model, which predicted 71 cm/yr of recharge to the pebbly-sand unit. This seems unreasonable in an area that receives 89 cm/yr (35 in./yr) of rainfall.

Both the two-dimensional and three-dimensional models had high residual differences between the observed and modeled heads in the same locations. These differences occurred in the same locations even though the two models were calibrated to water tables measured at two different times (1980 and 1984). Perhaps more hydraulic conductivity zones are needed in the model, there is some phenomenon occurring at these locations that the model does not take into account, or there is an error in the head measurements being taken at these points.

The travel times predicted by the three-dimensional model had a smaller range (4.4 to 228 yr) than those predicted by the two-dimensional model (1.1 to 292 yr). In both models, travel times were generally longer for flow paths from trenches close to the strip-mine lake and shorter for trenches far from the lake. The three-dimensional travel paths are controlled by the shape of the Toulon pebbly-sand unit across the site.

The two-dimensional modeling results suggested that the Toulon pebbly-sand unit is not completely in hydraulic connection with the lake. This conclusion is borne out by the results of the three-dimensional modeling, which shows the Toulon pebbly sand to have a saturated thickness in connection with only the southern portion of the lake. However, in the three-dimensional model, the travel paths must dip southward before entering the lake, whereas in the two-dimensional model they are able to enter the lake directly from the northeast corner of the site. Perhaps the lake level (211.8 m above msl) assumed in the three-dimensional model was too low. If the lake level had been almost a meter higher, then the Toulon could be in hydraulic connection with the north end of the lake, allowing more direct travel paths. Then, however, calibration of observed heads might have been more difficult.

TABLE 4.5. Travel Distances and Times from Trenches to Model Boundary for the Two-Dimensional Model

Trench	Side	Travel Distance (m)	Travel Time (yr)
1	West	494	52.3
	East	351	27.4
2	West	500	56.9
	East	340	34.9
3	West	500	45.8
	East	340	22.4
11	West	514	57.5
	East	354	31.7
4	West	487	48.1
	East	335	21.9
26	West	487	48.1
	East	335	21.9
5	West	670	292.0
	East	588	221.0
8 & 25c	West	467	38.7
	East	325	21.1
9	West	656	259.0
	East	561	157.0
14	West	664	254.6
	East	561	64.5
14a	West	656	229.2
	East	520	60.9
18	West	602	203.5
	East	521	73.6
23	West	433	2.1
	East	318	1.1
24	West	433	2.1
	East	331	1.7

4.3 RADIONUCLIDE RELEASE AND TRANSPORT MODELING

This section describes the process used in development and application of the radionuclide release and transport analysis, including the conceptualization of waste release and transport, the selection and design of the release model and related modeling assumptions, an initial screening analysis done to identify key radionuclides, and the application of the release and transport model to the Sheffield site inventory.

4.3.1 Conceptual Model of Waste Release and Transport

To facilitate the development of a conceptual model of waste release and transport for the Sheffield site, the waste inventory and site characteristics were evaluated to identify 1) the quantity of radionuclides in the inventory important in the long-term radionuclide release and performance of the Sheffield site, 2) waste form and container characteristics important in controlling this release, 3) relevant release and transport processes, and 4) assumptions that would be necessary to facilitate performance assessment modeling of these radionuclides and processes.

4.3.1.1 Inventory of Radionuclides in Trenches

The major source of information on the trench and inventory characterization for the Sheffield site was extracted from the report by MacKenzie et al. (1985). In this study, radioactive shipping records (RSRs) were examined for eight of the 17 trenches at the site, and an inventory of isotopes and total activity was compiled. Trenches 1, 2, 7, 11, 14a, 23, 24, and 25c were studied as representative of the site as a whole. Where possible, the type of containment structure or waste stream was also indicated. The waste was further categorized into fuel-cycle or non-fuel-cycle waste. From this compilation, a list of relevant radionuclides and their associated waste types was made to develop a conceptual model of the waste release to the trench.

From information summarized by MacKenzie et al. (1985), a compilation of the inventory of relevant isotopes of fuel-cycle and non-fuel-cycle waste in the eight waste streams for each trench was made. The compilations are listed in Tables 4.6 through 4.13. Table 4.14 is a summary table of the eight trenches giving total activity for each waste stream divided among fuel-cycle and non-fuel-cycle waste. An approximation of the activity, waste streams and isotopes in the trenches not covered by the MacKenzie et al. (1985) report was derived by taking the mean activity of the trenches in the database and applying a correction factor for volume. The typical trench inventory and size is listed in Table 4.15 and the trenches not included as part of the MacKenzie et al. (1985) report are shown in Tables 4.16 through 4.24.

Initial analysis of the data compilations indicated that a total of eleven radionuclides had long enough half-lives and/or were shipped to Sheffield in sufficient quantities to warrant further consideration. These

TABLE 4.6. Inventory of Trench 1

Sheffield Site Trench 1 Waste Stream Type	Tritium		^{137}Cs		^{60}Co		^{14}C		^{90}Sr		^{26}Al		Total Activity per waste stream
	fc	nfc	fc	nfc	fc	nfc	fc	nfc	fc	nfc	fc	nfc	
55-gallon drums (cement)	4.7	0	109	0	262	97	0	0.48	2	0	0	0	470.48
55-gallon drums (uncons.)	0	232	46	0	231	0	0	0.48	0	0	0	0	509.48
Lead-lined drums	0	0	0	1010	0	0	0	0	0	406	0	0	1416
Fiberboard boxes	0	78.4	0	0	0	0	0	0.24	0	0	0	6	84.64
Steel liners (cement)	0	0	15	0	14	0	0	0	0	0	0	0	29
Steel liners (uncons.)	0	0	37	0	185	0	0	0	0	0	0	0	222
Total nuclide activity (Ci) per trench	4.7	310.4	207	1010	692	97	0	1.2	2	406	0	6	

Total activity in Trench 1 2736.3

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.7. Inventory of Trench 2

Sheffield Site Trench 2 Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{241}Pu fc	^{241}Pu nfc	^{241}Am fc	^{241}Am nfc	Total Activity per waste stream
55-gallon drums (cement)	3.5	25	40	0	173	75	0	11.4	1.4	0	721.7	0	1.5	0	1049
55-gallon drums (uncons.)	0	288.2	39	0	192	0	0	11.4	0	0	0	0	0	0	530.6
Lead-lined drums	0	0	0	959	0	0	0	0	0	862	0	0	0	0	1821
Fiberboard boxes	0	319.8	0	0	0	0	0	5.7	0	0	0	0	0	0	325.5
Steel liners (cement)	0	0	1.5	0	3	0	0	0	0	0	0	0	0	0	4.5
Steel liners (uncons.)	0	0	31	134	153	0	0	0	0	134	0	0	0	0	452
Total nuclide activity (Ci) per trench	3.5	633	111.5	1093	521	75	0	28.5	1.4	996	721.7	0	1.5	0	

Total activity in Trench 2 4186.1

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.8. Inventory of Trench 7

Sheffield Site Trench 7 Waste Stream Type	Tritium		^{137}Cs		^{60}Co		^{14}C		^{90}Sr		^{241}Pu		Total Activity per waste stream
	fc	nfc	fc	nfc	fc	nfc	fc	nfc	fc	nfc	fc	nfc	
55-gallon drums (cement)	0.5	0	28	0	35	1	0	0.16	0.5	0	791	0	855.66
55-gallon drums (uncons.)	0	13	2.7	0	13	0	0	0.16	0	0	0	0	28.86
55-gallon drums (Urea-F)	0	0	0.2	0	0.2	0	0	0	0	0	0	0	0.4
Lead-lined drums	0	0	0	0	0	0	0	0	0	2.7	0	0	2.7
Fiberboard boxes	0	16.1	0	2.6	0	0	0	0.08	0	0	0	0	18.78
Steel liners (cement)	0	0	6	0	5	0	0	0	0	0	0	0	11
Steel liners (Urea-F)	0	0	7.8	0	7.8	0	0	0	0	0	0	0	15.6
Steel liners (uncons.)	0	0	2	0	11	0	0	0	0	0	0	0	13
Total nuclide activity (Ci) per trench	0.5	29.1	46.7	2.6	72	1	0	0.4	0.5	2.7	791	0	
												Total activity in Trench 7	946.5

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.9. Inventory of Trench 8

Sheffield Site Volume 49364.7 ft3 Trench 8 Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	1.3	31.3	96.5	3.5	103.3	6.3	0	1.4	1.5	0	0	0	63.6	0
55-gallon drums (uncons.)	0	24.5	6.6	1.4	26.1	0	0	1.4	0	0.8	0	0	0	0
55-gallon drums (Urea-F)	0	0	0.7	0	0.7	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	65.6	0	0.1	0	0	0	41.5	0	0	0	0
Fiberboard boxes	0	20.7	0	0.1	0	0	0	0.7	0	0	0	0.2	0	0
Steel liners (cement)	0	0	20.2	0	17.8	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	24.2	0	24.2	0	0	0	0	4.4	0	0	0	0
Steel liners (uncons.)	0	0	4.1	4.4	20.7	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	1.3	76.5	152.3	75	192.8	6.4	0	3.5	1.5	46.7	0	0.2	63.6	0

Sheffield Site Volume 49364.7 ft3 Trench 8 Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	$^{95}Zr,Nb$ fc	$^{95}Zr,Nb$ nfc	Total Activity per waste stream
55-gallon drums (cement)	1.8	0	0.9	0	1.3	0	0.1	0	0.3	0	313.1
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	60.8
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	1.4
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	107.2
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	21.7
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	38
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	52.8
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	29.2
Total nuclide activity (Ci)	1.8	0	0.9	0	1.3	0	0.1	0	0.3	0	

Total activity in Trench 8 624.2 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.10. Inventory of Trench 14A

Sheffield Site Trench 14A Waste Stream Type	Tritium		^{137}Cs		^{60}Co		^{14}C		^{90}Sr		Total Activity per waste stream
	fc	nfc	fc	nfc	fc	nfc	fc	nfc	fc	nfc	
55-gallon drums (cement)	9.7	716.42	962	60	917	15	0	12.16	13.1	0	2695.68
55-gallon drums (uncons.)	0	0	22	0	104	0	0	12.16	0	11.2	149.36
55-gallon drums (Urea-F)	0	0	3	0	3	0	0	0	0	0	6
Fiberboard boxes	0	3.8	0	0	0	0	0	6.08	0	0	9.88
Steel liners (cement)	0	0	208	0	182	0	0	0	0	0	390
Steel liners (Urea-F)	0	0	104	0	104	0	0	0	0	0	208
Steel liners (uncons.)	0	0	16	0	82	0	0	0	0	0	98
Total nuclide activity (Ci) per trench	9.7	720.22	1315	60	1392	15	0	30.4	13.1	11.2	
									Total activity in Trench 14		3566.62

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.11. Inventory of Trench 23

Sheffield Site Trench 23 Waste Stream Type	Tritium fc	 nfc	^{137}Cs fc	 nfc	^{60}Co fc	 nfc	^{14}C fc	 nfc	^{90}Sr fc	 nfc	Total Activity per waste stream
55-gallon drums (cement)	8.6	63	829	43.2	823	5	0	6.52	12	0.2	1781.92
55-gallon drums (uncons.)	0	63	29	43.3	138	0	0	6.52	0	0	279.82
55-gallon drums (Urea-F)	0	0	4	0	4	0	0	0	0	0	8
Lead-lined drums	0	0	0	31.5	0	0	0	0	0	0	31.5
Fiberboard boxes	0	63	0	0	0	0	0	3.26	0	0	66.26
Steel liners (cement)	0	0	177	0	156	0	0	0	0	0	333
Steel liners (Urea-F)	0	0	146	0	146	0	0	0	0	0	292
Steel liners (uncons.)	0	0	21	0	108	0	0	0	0	0	129
Total nuclide activity (Ci) per trench	8.6	189	1206	118	1375	5	0	16.3	12	0.2	
										Total activity in Trench 23	2930.1

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.12. Inventory of Trench 24

Sheffield Site Trench 24 Waste Stream Type	Tritium fc	 nfc	^{137}Cs fc	 nfc	^{60}Co fc	 nfc	^{14}C fc	 nfc	^{90}Sr fc	 nfc	^{241}Pu fc	 nfc
55-gallon drums (cement)	8.8	105	723	0	681	0	0	8.2	12.4	0	3	0
55-gallon drums (uncons.)	0	105	13	0	64	0	0	8.2	0	10.6	0	0
55-gallon drums (Urea-F)	0	0	9	0	9	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	8.8	0	1.6	0	0	0	0	0	0
Fiberboard boxes	0	105	0	0	0	0	0	4.1	0	0	0	0
Steel liners (cement)	0	0	157	0	136	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	328	0	328	0	0	0	0	0	0	0
Steel liners (uncons.)	0	0	9	0	49	0	0	0	0	0	0	0
Total nuclide activity (Ci)	8.8	315	1239	8.8	1267	1.6	0	20.5	12.4	10.6	3	0

Sheffield Site Trench 24 Waste Stream Type	^{241}Am fc	 nfc	^{54}Mn fc	 nfc	^{59}Fe fc	 nfc	^{65}Zn fc	 nfc	$^{95}Zr,Nb$ fc	 nfc	Total Activity per waste stream
55-gallon drums (cement)	48.2	0	28.2	0	39.6	0	3.8	0	8.1	0	1580.8
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	200.8
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	18
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	10.4
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	109.1
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	293
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	656
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	58
Total nuclide activity (Ci)	48.2	0	28.2	0	39.6	0	3.8	0	8.1	0	
							Total activity in Trench 24				3014.6 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.13. Inventory of Trench 25C

Sheffield Site Trench 25C Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	Total Activity per waste stream
55-gallon drums (cement)	1.5	38	124	0	127	0	0	1.56	1.6	0	292.16
55-gallon drums (uncons.)	0	38	46	0	25	0	0	1.56	0	0.1	110.66
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	0
Fiberboard boxes	0	38	0	0	0	0	0	0.78	0	0	38.78
Steel liners (cement)	0	0	26	0	23	0	0	0	0	0	49
Steel liners (Urea-F)	0	0	0.5	0	0.5	0	0	0	0	0	1
Steel liners (uncons.)	0	0	4	0	20	0	0	0	0	0	24
Total nuclide activity (Ci) per trench	1.5	114	200.5	0	195.5	0	0	3.9	1.6	0.1	
							Total activity in Trench 25C				517.1

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.14. Total Inventory for Eight Documented Trenches

Sheffield Site Total trench inventory Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	39.9	958.52	2958	108.2	3166	194	0	41.48	46.4	0.2	0	0	1950	0
55-gallon drums (uncons.)	0	750.3	204.3	43.3	799	0	0	41.48	0	25.6	0	0	0	0
55-gallon drums (Urea-F)	0	0	20.2	0	20.2	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	2009.3	0	1.6	0	0	0	1270.7	0	0	0	0
Fiberboard boxes	0	635.2	0	2.6	0	0	0	20.74	0	0	0	6	0	0
Steel liners (cement)	0	0	620.5	0	545	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	740.3	0	740.3	0	0	0	0	134	0	0	0	0
Steel liners (uncons.)	0	0	125	134	633	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	39.9	2344.02	4668.3	2297.4	5903.5	195.6	0	103.7	46.4	1430.5	0	6	1950	0

Sheffield Site Total trench inventory Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	$^{95}Zr,Nb$ fc	$^{95}Zr,Nb$ nfc	Total Activity per waste stream
55-gallon drums (cement)	54.7	0	28.2	0	39.6	0	3.8	0	8.1	0	9597.1
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	1863.98
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	40.4
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	3281.6
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	664.54
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	1165.5
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	1614.6
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	892
Total nuclide activity (Ci)	54.7	0	28.2	0	39.6	0	3.8	0	8.1	0	
Total activity in 8 trenches											19119.72 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.15. Inventory of Typical Trench

Sheffield Site Volume 188.877 ft3

Typical trench inventory Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	4.9875	119.815	369.75	13.525	395.75	24.25	0	5.185	5.8	0.025	0	0	243.75	0
55-gallon drums (uncons.)	0	93.7875	25.5375	5.4125	99.875	0	0	5.185	0	3.2	0	0	0	0
55-gallon drums (Urea-F)	0	0	2.525	0	2.525	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	251.1625	0	0.2	0	0	0	158.8375	0	0	0	0
Fiberboard boxes	0	79.4	0	0.325	0	0	0	2.5925	0	0	0	0.75	0	0
Steel liners (cement)	0	0	77.5625	0	68.125	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	92.5875	0	92.5875	0	0	0	0	16.75	0	0	0	0
Steel liners (uncons.)	0	0	15.625	16.75	79.125	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	4.9875	293.0025	583.5875	287.175	737.9875	24.45	0	12.9625	5.8	178.8125	0	0.75	243.75	0

Sheffield Site Volume 188.877 ft3

Typical trench inventory Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	$^{95}Zr,Nb$ fc	$^{95}Zr,Nb$ nfc	Total Activity per waste stream
55-gallon drums (cement)	6.8375	0	3.525	0	4.95	0	0.475	0	1.0125	0	1199.637
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	232.9975
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	5.05
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	410.2
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	83.0675
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	145.6875
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	201.925
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	111.5
Total nuclide activity (Ci)	6.8375	0	3.525	0	4.95	0	0.475	0	1.0125	0	
Total activity for typical trench											2390.065 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.16. Inventory of Trench 3

Sheffield Site Volume 191200.9 ft3
Trench 3

Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	5	121.2	374.2	13.7	400.5	24.5	0	5.2	5.9	0.025	0	0	256.7	0
55-gallon drums (uncons.)	0	94.9	25.8	5.5	101	0	0	5.2	0	3.2	0	0	0	0
55-gallon drums (Urea-F)	0	0	2.6	0	2.6	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	254.2	0	0.2	0	0	0	160.7	0	0	0	0
Fiberboard boxes	0	80.4	0	0.3	0	0	0	2.6	0	0	0	0.75	0	0
Steel liners (cement)	0	0	78.5	0	68.9	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	93.7	0	93.7	0	0	0	0	17	0	0	0	0
Steel liners (uncons.)	0	0	15.8	17	80	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	5	296.5	590.6	290.7	746.7	24.7	0	13	5.9	180.925	0	0.75	256.7	0

Sheffield Site Volume 191200.9 ft3
Trench 3

Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	$^{95}Zr,Nb$ fc	$^{95}Zr,Nb$ nfc	Total Activity per waste stream
55-gallon drums (cement)	6.9	0	3.6	0	5	0	0.475	0	1.0125	0	1223.912
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	235.6
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	5.2
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	415.1
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	84.05
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	147.4
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	204.4
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	112.8
Total nuclide activity (Ci)	6.9	0	3.6	0	5	0	0.475	0	1.0125	0	
Total activity in Trench 3											2428.462 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.17. Inventory of Trench 4

Sheffield Site Volume 197896.4 ft3
Trench 4

Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	5.2	125.6	387.5	14.2	414.7	25.4	0	5.4	6.1	0.025	0	0	255.5	0
55-gallon drums (uncons.)	0	98.3	26.8	5.7	104.7	0	0	5.4	0	3.4	0	0	0	0
55-gallon drums (Urea-F)	0	0	2.6	0	2.6	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	263.2	0	0.2	0	0	0	166.5	0	0	0	0
Fiberboard boxes	0	83.2	0	0.34	0	0	0	2.7	0	0	0	0.79	0	0
Steel liners (cement)	0	0	81.3	0	71.4	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	97	0	97	0	0	0	0	17.6	0	0	0	0
Steel liners (uncons.)	0	0	16.4	17.6	82.9	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	5.2	307.1	611.6	301.04	773.3	25.6	0	13.5	6.1	187.525	0	0.79	255.5	0

Sheffield Site Volume 197896.4 ft3
Trench 4

Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	$^{95}Zr,Nb$ fc	$^{95}Zr,Nb$ nfc	Total Activity per waste stream
55-gallon drums (cement)	7.2	0	3.7	0	5.2	0	0.5	0	1.1	0	1257.325
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	244.3
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	5.2
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	429.9
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	87.03
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	152.7
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	211.6
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	116.9
Total nuclide activity (Ci)	7.2	0	3.7	0	5.2	0	0.5	0	1.1	0	
Total activity in Trench 4											2504.955 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.18. Inventory of Trench 5

Sheffield Site Volume 120999.1 ft3 Trench 5 Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	3.2	76.8	237	8.7	253.7	15.5	0	3.3	3.7	0	0	0	156.2	0
55-gallon drums (uncons.)	0	60.1	16.4	3.5	64	0	0	3.3	0	2.1	0	0	0	0
55-gallon drums (Urea-F)	0	0	1.6	0	1.6	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	161	0	0.1	0	0	0	101.5	0	0	0	0
Fiberboard boxes	0	50.9	0	0.2	0	0	0	1.7	0	0	0	0.5	0	0
Steel liners (cement)	0	0	49.7	0	43.7	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	59.3	0	59.3	0	0	0	0	10.7	0	0	0	0
Steel liners (uncons.)	0	0	10	10.7	50.7	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	3.2	187.8	374	184.1	473	15.6	0	8.3	3.7	114.3	0	0.5	156.2	0

Sheffield Site Volume 120999.1 ft3 Trench 5 Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	$^{95}Zr,Nb$ fc	$^{95}Zr,Nb$ nfc	Total Activity per waste stream
55-gallon drums (cement)	4.4	0	2.3	0	3.2	0	0.3	0	0.65	0	768.95
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	149.4
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	3.2
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	262.6
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	53.3
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	93.4
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	129.3
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	71.4
Total nuclide activity (Ci)	4.4	0	2.3	0	3.2	0	0.3	0	0.65	0	
Total activity in Trench 5											1531.55 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.19. Inventory of Trench 6

Sheffield Site Volume 197365.9 ft3 Trench 6 Waste Stream Type	Tritium fc	 nfc	^{137}Cs fc	 nfc	^{60}Co fc	 nfc	^{14}C fc	 nfc	^{90}Sr fc	 nfc	^{26}Al fc	 nfc	^{241}Pu fc	 nfc
55-gallon drums (cement)	5.2	125.2	386.4	14.1	413.6	25.3	0	5.4	6.1	0	0	0	254.7	0
55-gallon drums (uncons.)	0	98.1	26.7	5.7	104.4	0	0	5.4	0	3.3	0	0	0	0
55-gallon drums (Urea-F)	0	0	2.6	0	2.6	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	263	0	0.2	0	0	0	166	0	0	0	0
Fiberboard boxes	0	83	0	0.34	0	0	0	2.7	0	0	0	0.8	0	0
Steel liners (cement)	0	0	81.1	0	71.2	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	96.7	0	96.8	0	0	0	0	17.5	0	0	0	0
Steel liners (uncons.)	0	0	16.3	17.5	82.7	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	5.2	306.3	609.8	300.64	771.3	25.5	0	13.5	6. 1	186.8	0	0.8	254.7	0

Sheffield Site Volume 197365.9 ft3 Trench 6 Waste Stream Type	^{241}Am fc	 nfc	^{54}Mn fc	 nfc	^{59}Fe fc	 nfc	^{65}Zn fc	 nfc	$^{95}Zr,Nb$ fc	 nfc	Total Activity per waste stream
55-gallon drums (cement)	7.1	0	3.7	0	5.2	0	0.5	0	1.1	0	1253.6
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	243.6
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	5.2
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	429.2
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	86.84
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	152.3
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	211
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	116.5
Total nuclide activity (Ci)	7.1	0	3.7	0	5.2	0	0.5	0	1.1	0	
							Total activity in Trench 6				2498.24 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.20. Inventory of Trench 8

Sheffield Site Volume 49364.7 ft3 Trench 8 Waste Stream Type	Tritium		^{137}Cs		^{60}Co		^{14}C		^{90}Sr		^{26}Al		^{241}Pu	
	fc	nfc	fc	nfc	fc	nfc	fc	nfc	fc	nfc	fc	nfc	fc	nfc
55-gallon drums (cement)	1.3	31.3	96.5	3.5	103.3	6.3	0	1.4	1.5	0	0	0	63.6	0
55-gallon drums (uncons.)	0	24.5	6.6	1.4	26.1	0	0	1.4	0	0.8	0	0	0	0
55-gallon drums (Urea-F)	0	0	0.7	0	0.7	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	65.6	0	0.1	0	0	0	41.5	0	0	0	0
Fiberboard boxes	0	20.7	0	0.1	0	0	0	0.7	0	0	0	0.2	0	0
Steel liners (cement)	0	0	20.2	0	17.8	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	24.2	0	24.2	0	0	0	0	4.4	0	0	0	0
Steel liners (uncons.)	0	0	4.1	4.4	20.7	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	1.3	76.5	152.3	75	192.8	6.4	0	3.5	1.5	46.7	0	0.2	63.6	0

Sheffield Site Volume 49364.7 ft3 Trench 8 Waste Stream Type	^{241}Am		^{54}Mn		^{59}Fe		^{65}Zn		$^{95}Zr,Nb$		Total Activity per waste stream
	fc	nfc	fc	nfc	fc	nfc	fc	nfc	fc	nfc	
55-gallon drums (cement)	1.8	0	0.9	0	1.3	0	0.1	0	0.3	0	313.1
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	60.8
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	1.4
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	107.2
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	21.7
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	38
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	52.8
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	29.2
Total nuclide activity (Ci)	1.8	0	0.9	0	1.3	0	0.1	0	0.3	0	
							Total activity in Trench 8				624.2 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.21. Inventory of Trench 9

Sheffield Site Volume 185237.5 ft3
Trench 9

Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	4.9	117.5	362.7	13.3	388.2	23.8	0	5.1	5.7	0	0	0	239.1	0
55-gallon drums (uncons.)	0	92	25	5.3	98	0	0	5.1	0	3.1	0	0	0	0
55-gallon drums (Urea-F)	0	0	2.5	0	2.5	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	246.4	0	0.2	0	0	0	155.8	0	0	0	0
Fiberboard boxes	0	77.9	0	0.3	0	0	0	2.5	0	0	0	0.7	0	0
Steel liners (cement)	0	0	76	0	66.8	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	90.1	0	90.8	0	0	0	0	16.4	0	0	0	0
Steel liners (uncons.)	0	0	15.3	16.4	77.6	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	4.9	287.4	571.6	281.7	723.9	24	0	12.7	5.7	175.3	0	0.7	239.1	0

Sheffield Site Volume 185237.5 ft3
Trench 9

Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	$^{95}Zr,Nb$ fc	$^{95}Zr,Nb$ nfc	Total Activity per waste stream
55-gallon drums (cement)	6.7	0	3.5	0	4.9	0	0.5	0	1	0	1176.9
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	228.5
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	5
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	402.4
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	81.4
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	142.8
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	197.3
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	109.3
Total nuclide activity (Ci)	6.7	0	3.5	0	4.9	0	0.5	0	1	0	
Total activity in Trench 9											2343.6 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.22. Inventory of Trench 14

Sheffield Site Volume 394399.8 ft3
Trench 14

Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	10.4	250.1	772	28.2	826.3	50.6	0	10.8	12.1	0.1	0	0	509	0
55-gallon drums (uncons.)	0	195.8	53.3	11.3	208.5	0	0	10.8	0	6.7	0	0	0	0
55-gallon drums (Urea-F)	0	0	5.3	0	5.3	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	524.4	0	0.4	0	0	0	331.6	0	0	0	0
Fiberboard boxes	0	165.8	0	0.7	0	0	0	5.4	0	0	0	1.6	0	0
Steel liners (cement)	0	0	161.9	0	142.2	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	193.3	0	193.3	0	0	0	0	35	0	0	0	0
Steel liners (uncons.)	0	0	32.6	35	165.2	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	10.4	611.7	1218.4	599.6	1540.8	51	0	27	12.1	373.4	0	1.6	509	0

Sheffield Site Volume 394399.8 ft3
Trench 14

Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	$^{95}Zr,Nb$ fc	$^{95}Zr,Nb$ nfc	Total Activity per waste stream
55-gallon drums (cement)	14.3	0	7.4	0	10.3	0	1	0	2.1	0	2504.7
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	486.4
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	10.6
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	856.4
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	173.5
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	304.1
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	421.6
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	232.8
Total nuclide activity (Ci)	14.3	0	7.4	0	10.3	0	1	0	2.1	0	
Total activity in Trench 14											4990.1 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.23. Inventory of Trench 18

Sheffield Site Volume 120655.7 ft3
Trench 18

Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	3.2	76.6	236.3	8.6	252.9	15.5	0	3.3	3.7	0	0	0	155.7	0
55-gallon drums (uncons.)	0	60	16.3	3.5	63.8	0	0	3.3	0	2	0	0	0	0
55-gallon drums (Urea-F)	0	0	1.6	0	1.6	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	160.5	0	0.1	0	0	0	101.5	0	0	0	0
Fiberboard boxes	0	50.7	0	0.2	0	0	0	1.7	0	0	0	0.5	0	0
Steel liners (cement)	0	0	49.6	0	43.5	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	59.2	0	59.16	0	0	0	0	10.7	0	0	0	0
Steel liners (uncons.)	0	0	10	10.7	50.6	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	3.2	187.3	373	183.5	471.56	15.6	0	8.3	3.7	114.2	0	0.5	155.7	0

Sheffield Site Volume 120655.7 ft3
Trench 18

Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	$^{95}Zr,Nb$ fc	$^{95}Zr,Nb$ nfc	Total Activity per waste stream
55-gallon drums (cement)	4.4	0	2.3	0	3.2	0	0.3	0	0.6	0	766.6
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	148.9
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	3.2
Lead-lined drums	0	0	0	0	0	0	0	0	0	0	262.1
Fiberboard boxes	0	0	0	0	0	0	0	0	0	0	53.1
Steel liners (cement)	0	0	0	0	0	0	0	0	0	0	93.1
Steel liners (Urea-F)	0	0	0	0	0	0	0	0	0	0	129.06
Steel liners (uncons.)	0	0	0	0	0	0	0	0	0	0	71.3
Total nuclide activity (Ci)	4.4	0	2.3	0	3.2	0	0.3	0	0.6	0	
Total activity in Trench 18											1527.36 Ci

fc = fuel cycle
nfc = non-fuel cycle

TABLE 4.24. Inventory of Trench 26

Sheffield Site Volume 166137.9 ft3
Trench 26

Waste Stream Type	Tritium fc	Tritium nfc	^{137}Cs fc	^{137}Cs nfc	^{60}Co fc	^{60}Co nfc	^{14}C fc	^{14}C nfc	^{90}Sr fc	^{90}Sr nfc	^{26}Al fc	^{26}Al nfc	^{241}Pu fc	^{241}Pu nfc
55-gallon drums (cement)	4.4	105.4	325.4	11.9	348.3	21.3	0	4.6	5.1	0	0	0	214.5	0
55-gallon drums (uncons.)	0	82.5	22.5	4.8	87.9	0	0	4.6	0	2.8	0	0	0	0
55-gallon drums (Urea-F)	0	0	2.2	0	2.2	0	0	0	0	0	0	0	0	0
Lead-lined drums	0	0	0	221	0	0.2	0	0	0	139.8	0	0	0	0
Fiberboard boxes	0	69.9	0	0.3	0	0	0	2.3	0	0	0	0.7	0	0
Steel liners (cement)	0	0	68.3	0	60	0	0	0	0	0	0	0	0	0
Steel liners (Urea-F)	0	0	81.5	0	81.5	0	0	0	0	14.7	0	0	0	0
Steel liners (uncons.)	0	0	13.7	14.7	69.6	0	0	0	0	0	0	0	0	0
Total nuclide activity (Ci)	4.4	257.8	513.6	252.7	649.5	21.5	0	11.5	5.1	157.3	0	0.7	214.5	0

Sheffield Site Volume 166137.9 ft3
Trench 26

Waste Stream Type	^{241}Am fc	^{241}Am nfc	^{54}Mn fc	^{54}Mn nfc	^{59}Fe fc	^{59}Fe nfc	^{65}Zn fc	^{65}Zn nfc	^{95}Zr,Nb fc	^{95}Zr,Nb nfc	Total Activity per waste stream
55-gallon drums (cement)	6	0	3.1	0	4.4	0	0.4	0	0.9	0	1055.7
55-gallon drums (uncons.)	0	0	0	0	0	0	0	0	0	0	205.1
55-gallon drums (Urea-F)	0	0	0	0	0	0	0	0	0	0	4.4

Lead-lined	8,653.99					
	25	02-75/05-75	195.89	0	0	411.36
	25C	04-76/08-76	863.86	177.58	282.25	1,857.22
	26	05-75/08-75	991.20	1,087.80	13,431.91	4,705.03
Totals			61,224	54,353	272,594	89,737

radionuclides along with their half-lives and an estimate of their total activity level at Sheffield are presented in Table 4.25.

The following list of radionuclides: ^{26}Al, ^{63}Ni, ^{94}Nb, ^{99}Tc, ^{99m}Tc, ^{36}Cl, ^{51}Cr, ^{110m}Ag, ^{131}I, ^{125}I, ^{32}P, ^{35}S, ^{75}Se, ^{237}Np, ^{252}Cf, and ^{134}Cs, although mentioned by MacKenzie et al. (1985) as part of the Sheffield inventory, were not considered in this analysis because they were insufficiently characterized, had very low activity levels, and/or very short half-lives. Uranium and thorium were also not included because no information on their inventory was available in either the MacKenzie et al. (1985) report or others done at Sheffield.

4.3.1.2 Waste Forms and Containers Disposed at Sheffield

An important part of the conceptual model of the release and transport at the Sheffield site is the determination of availability and release of radionuclide contaminants from the waste forms and containers. For this analysis, a distinction is made between the availability and the release rate of radionuclides in the trenches. Availability is defined in this context as the accessibility of the radionuclide in the trench to a removal mechanism such as leaching by infiltrating water. The release rate is defined as the removal rate by the water, based upon the form of the radionuclide. Availability is affected by the type and competence of the waste form as well as the integrity of the waste container.

The waste forms and containers varied widely for most radionuclides. Plastic bags, boxes, paint cans, and glass vials are some of the secondary containers that held the radionuclides placed in the trenches. There was no standardization of containers or containment methods for these secondary barriers. Eight waste types, consisting of different combinations of waste forms and containers, were identified for consideration in the release analysis. These eight waste types encompass most of the waste and represent the full range of primary waste types at Sheffield. A brief description and likely containment capability for each of these waste types, as described by MacKenzie et al. (1985), are summarized below.

TABLE 4.25. Half-Lives and Total Inventory of Fuel-Cycle and Non-Fuel-Cycle Waste for 11 Radionuclides at the Sheffield LLW Site

Radionuclide	Half-Life (yr)	Fuel-Cycle Waste (Ci)	Non-Fuel-Cycle Waste (Ci)
Tritium	12.7	39.9	2344.0
^{137}Cs	300	4668.3	2297.4
^{60}Co	5.26	5903.5	195.6
^{14}C	5730	0	103.7
^{90}Sr	28.10	46.4	1430.5
^{241}Pu	2.16E7	1950	0
^{241}Am	2.16E7	54.7	0
^{54}Mn	0.82	28.2	0

55-gallon (208.1-L) drum, waste solidified in cement. This is a standard 17H carbon steel 55-gallon (208.1-L) drum with a volume of approximately 0.2 m^3 (7 ft^3). The top is sealed with a press-fit lid. The waste inside is assumed to be solidified in cement. For all steel drums, the MacKenzie et al. (1985) report assumes a 15% breakage rate during emplacement. This assumption is being carried over for conceptual model development. The report also indicates that in soils similar to Sheffield soils, the steel drums are breached by corrosion in 4 to 5 yr. A further conservative assumption is made that upon emplacement, 100% of the drums suffer some fracture of the cement solidifying the waste, and that 50% of the waste loses the benefits of cement solidification, and becomes available upon corrosion or breach of the drum. Thus 8%, or 15% of 50% of the waste in this waste stream becomes immediately available upon emplacement in the trench. The remaining 42% of that which is not contained by the cement becomes available to the environment after 4 yr, when the drum corrodes open. Fifty percent is assumed to remain intact, and is affected only by diffusion. Diffusion as a release mechanism is discussed in the next section.

55-gallon (208.1-L) drum, waste solidified in urea-formaldehyde. This is a steel drum with a press-fit lid seal. The waste inside is assumed to be packed in urea-formaldehyde. These steel drums are subject to the same conditions as above with regard to breaching on emplacement. However, since the waste is not solidified in cement, 100% of the waste is potentially available. Thus, 15% is made available upon emplacement. The MacKenzie et al. (1985) report describes the failure of the steel drum containing urea-formaldehyde from within caused by the interaction between the steel and urea-formaldehyde. The report gives the time to initial breach of the drum in 0.8 yr. The remaining 85% of the waste activity in this waste form then becomes available.

55-gallon (208.1-L) drum, waste unconsolidated. This is a steel drum with a press-fit lid seal. The waste inside is assumed to be laboratory trash and general waste without any stabilizing agent. These steel drums are also subject to the initial breaching rate of 15%. The waste inside is potentially 100% available, because of the lack of a stabilizing agent. Normal corrosion will breach unopened drums in 4 yr. Therefore, 15% of the waste in this waste stream is available immediately, with the remaining 85% becoming available in 4 yr.

A special case is made for tritium. The MacKenzie report characterizes tritium in more detail; 18.8% of the tritium shipped to Sheffield is in the form of luminous paint chips, and they are contained in unconsolidated steel drums and fiberboard boxes. These paint chips are required to be non-leachable for public use, and are therefore considered to be unavailable. For tritium, the proportions then become 12.2% available immediately and 69% available after 4 yr.

Lead-lined steel drums. These 30-gallon (113.55-L) steel drums are encased in a 3-in. lining of lead. The drums are sealed with a lead lid and a steel press-fit lid. The waste inside is assumed to be unconsolidated. These drums are emplaced by a crane instead of the 'kick and roll' method more common to waste emplacement of drums in the trenches. These drums have a lead lid placed over the inner lead waste container, and a steel banded lid covering the outer steel container. MacKenzie et al. (1985) computed a corrosion time of the 3-in. lead lining of 300 yr, and a corrosion time of the steel lid of the drum at 7 yr. No information other than the crane emplacement of the waste was given, so the orientation of the drums is not known. Using the conservative assumption that the drums were placed on their side, the time required to make the waste available is limited only by the time it takes to corrode the steel lid completely. This allows the inside lead lid to fall away and the waste becomes available. Thus, 0% is available immediately, and 100% is available after 7 yr. For drums placed upright, the time to availability would be 307 yr, plus the time required to fill the interior of the drum with water and establish a path to the environment.

Fiberboard containers. These are fiberboard or cardboard boxes of various sizes with a volume of approximately 0.14 m^3 (5 ft^3). The waste inside is assumed to be unconsolidated. Since fiberboard or cardboard containers provide no barrier to water, the removal mechanism used in this conceptual model, 100% of the waste within is available immediately.

Once again, a special case must be made for tritium. Since 18.8% of the tritium is in the form of non-leachable paint chips, only 81.2% of the tritium placed in fiberboard boxes is available for removal.

Steel liners, waste consolidated in cement. This is an approximately 2.8-m^3 (100-ft^3) steel liner with the waste inside solidified in cement. MacKenzie et al. (1985) report that the corrosion of the steel liners takes place in 12 yr. The same assumptions of cement fracture are made for the liners as for the drums. However, it is assumed that the liners, because of their size, were placed with a crane, and no containers were breached on emplacement. Thus, none of the waste is available immediately, 50% becomes available in 12 yr, and 50% would only become available by diffusion to the exterior of the liner.

Steel liners, waste solidified in urea-formaldehyde. This is a steel liner with the waste inside packed in urea-formaldehyde. MacKenzie et al. (1985) report that the corrosion of the steel liners is accelerated by the conjunction of the urea-formaldehyde, and the steel is first breached after 1 yr. Thus, none of the waste is available immediately, but 100% of the waste is available after 1 yr.

Steel liners, waste unconsolidated. This is a steel liner with the waste inside in an unconsolidated state. The corrosion of the steel liner in 12 yr applies for unconsolidated waste, but in this case, 100% of the waste is potentially available. Thus, 0% is available immediately, and 100% of the waste becomes available in 12 yr.

4.3.1.3 Relevant Release and Transport Processes

Once the waste container or waste form have been compromised, radionuclides are made available for release and subsequent transport into the environment. Based on a review of the Sheffield inventory and waste form characteristics, it was determined that waste from the site can be released by a number of processes: a release controlled by diffusion from the cement-based waste forms (diffusion-controlled release), a release controlled by a geochemically determined solubility-limiting concentration (solubility-controlled release), or a release controlled by the affinity for radionuclides to be sorbed to the natural sediments underlying the waste (adsorption-controlled release). Discussions of each of these release processes in the context of the Sheffield inventory and waste types are provided below.

Diffusion-Controlled Release. A determination was made of the effect of diffusion of stabilized radionuclides in cement at Sheffield. Using coefficients of diffusion for various radionuclides in cement (Martin 1988; and Serne et al. 1987), calculations were made to determine the ratio of the original concentration of the waste to the waste leaving the cement waste form. A 10 cm distance was chosen as a representative distance to travel within a cement-stabilized steel drum. The following equation applies:

$$C(x,t)/C_0 = \mathrm{erfc}(x/2\sqrt{D^* \bullet t}) \qquad (4)$$

where $C(x,t)$ = the solute concentration at the distance and time of interest

C_0 = the original concentration

x = the distance through the cement

D^* = the diffusion coefficient

t = the time.

To determine whether the effects of diffusion contribute a significant amount to the radionuclide released to the environment, a time equivalent to 7 half-lives was chosen. This is the time by which the radionuclide is considered effectively decayed to zero. Table 4.26 lists the radionuclide, the half-life, the D^*, and the $C(x,t)/C_0$ for 7 half-lives using the equation above.

The only radionuclide that would diffuse through the cement and emerge with significant concentration levels before decaying out is tritium. The amount of initial inventory in all the eight trenches contained in concrete (499.2 Ci) that would be released at 7 half-lives is 0.012 Ci. Diffusion as a significant release mechanism is therefore neglected.

TABLE 4.26. Half-Lives, Diffusion Coefficients, and Concentrations at a Distance of 10 cm for Seven Radionuclides at Sheffield

Radionuclide	Half-life (yr)	D^*	$C(x,t)/C_0$
Tritium	12.3	1E-9	2.5E-5
^{137}Cs	30.0	5.2E-12	0
^{60}Co	5.26	5.1E-12	0
^{14}C	5760	1E-14	0
^{90}Sr	28.1	4.4E-12	0
^{241}Pu	13.2	1E-16	0
^{241}Am	458	6E-16	0
^{59}Fe	0.12	1E-9[a]	0
^{65}Zn	0.67	1E-9[a]	0
^{95}Zr,Nb	0.27	1E-9[a]	0

(a) Conservatively assumed to move with water. $C(x,t)/C_0$ still 0.

Solubility-Controlled Release. Solubility-controlled release is another possible mechanism controlling release rates. Estimates of solubility-controlling concentrations relevant to the Sheffield site were derived from selected literature that included reports by Baes and Mesmer (1976), Matthess (1982), Linsday (1979), Rai (1981; 1987), Rai et al. (1980; 1981), Early (1986), and Hostetler et al. (1988). For the purposes of these estimates, we assumed that a carbonate-dominated, oxidizing ground water, typical of the Sheffield environment, contacts the waste. We also assumed that water contacting the ground water and the waste has a pH of 7. The pH of the ground waters in the Sheffield area vary from 6.2 to 7.8. A summary of these estimates for radionuclides relevant to Sheffield and related references are given in Table 4.27. The radionuclides are treated separately, and because the waste distribution is unknown, chemical interaction was assumed to be limited to reactions with the ground water. Tritium is not included in this table since it is a form of water and moves as such in the ground-water environment and thus is not limited by solubility controls.

Adsorption-Controlled Release. In an adsorption-controlled release, the release is governed by the retardation factor and the original source strength of the individual radionuclide in the solid phase. The retardation factor of a solute relative (R_F) to the bulk mass of water is defined by the following equation:

TABLE 4.27. Solubility Limits for Relevant Radionuclides at the Sheffield LLW Site

Radionuclide	Solubility (Ci/L)	Reference
^{137}Cs	9.788E-2[a]	Matthess (1982)
^{60}Co	6.718E-3[a]	Matthess (1982)
	5.643E-2[b]	Early (1986)
^{14}C	5.513E-3[a]	Lindsay (1979)
^{90}Sr	1.40E-1[a]	Matthess (1982)
	2.8E-1[b]	Rai (1987)
	1.273[c]	Lindsay (1979)
^{241}Pu	2.759E-3[c]	Rai et al. (1981)
^{241}Am	7.873E-9[c]	Rai et al. (1980)
^{54}Mn	82.077[a]	Matthess (1982)
	2416.722[b]	Rai (1987)
	2553.5[c]	Lindsay (1979)
^{59}Fe	0.274[a]	Matthess (1982)
	0.548[b]	Rai (1987)
	8.483E-3[c]	Lindsay (1979)
^{65}Zn	8.041[a]	Matthess (1982)
	5.360E-2[b]	Hostetler et al. (1988)
	5360[c]	Lindsay (1979)
^{95}Zr	1.934E-4[c]	Baes and Mesmer (1976)

(a) Typical maximum ground water concentrations for conditions similar to Sheffield.
(b) Typical maximum concentration at waste sites.
(c) Calculated solubility limit for conditions at Sheffield.

$$R_F = \frac{V}{Vc} \quad 1 + \frac{\sigma}{n} K_d$$

where V = the average linear velocity of water

Vc = the velocity of the C/Co = 0.5 point of the solute concentration profile (Freeze and Cherry 1979, p. 403)

σ = the bulk density of the solid

n = the porosity of the solid

C = the solute concentration at some time, t

C_o = the initial solute concentration, and

K_d = the distribution coefficient.

If one assumes that the adsorption is fast, reversible, and linear, this distribution coefficient can be used to describe the partitioning of the contaminant between the water and the soil. This coefficient is defined as follows:

$$K_d = Cs/Cw$$

where Cs is the mass of the solute on the solid phase per unit mass of the solid phase, and Cw is the concentration of the solute in solution.

Some data of the affinity of radionuclides to sorb on Sheffield soils were available in Murphy and Bergeron (1991). Representative distribution coefficients for relevant radionuclides at Sheffield along with appropriate references are given in Table 4.28. Values of ^{241}Pu and ^{241}Am were not available for Sheffield soils. Values in the table are reflective of K_d of these radionuclides at the Hanford Site, Washington, where carbonate-rich ground water and soils are geochemically quite similar to conditions at Sheffield. These data indicate that, with the exception of ^{90}Sr, all of these radionuclides have the potential to be highly adsorbed and retarded in Sheffield soils.

TABLE 4.28. Distribution Coefficients for Relevant Radionuclides at the Sheffield LLW Site

Radionuclide	K_d (ml/g)
^{137}Cs	3939 - 10,562[a]
^{60}Co	138 - 847[a]
^{90}Sr	3.4 - 17.3[a]
^{241}Pu	~70[b]
^{241}Am	>1200[c]

(a) Murphy and Bergeron (1991).
(b) Bergeron et al. (1987).
(c) Routson et al. (1977).

4.3.2 Release and Transport Code Selection and Model Design

Based on the review of the Sheffield waste inventory and the results of the previous and present ground-water water flow model analyses, we selected a one-dimensional streamtube transport modeling approach. This approach facilitates the evaluation of the transport of many radionuclides using a variety of release models and hydrologic conditions. The depiction of the streamtube approach to transport in an unconfined aquifer system is schematically illustrated in Figure 4.23. This transport modeling approach makes maximum use of the available field-measured or simulated aquifer information and reduces the number of parameter values subjectively assigned. The method used, which is based on a theory of transport described in Simmons (1982), is incorporated in the transport code referred to as TRANSS. The transport model contained in the TRANSS code is based on the one-dimensional stochastic-convective theory of transport and relates dispersion directly to the variation observed in the travel time from the source.

The TRANSS transport code, which is documented in Simmons, Kincaid, and Reisenauer (1986), includes the following features that offer certain advantages for use on the Sheffield site:

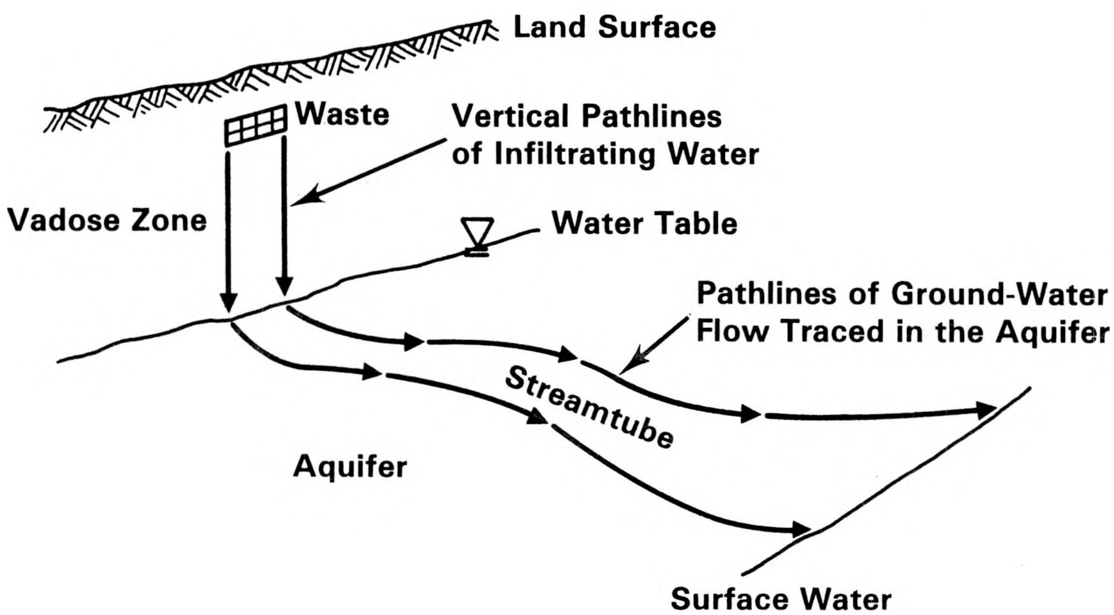


FIGURE 4.23. Depiction of the Streamtube Approach to Transport in Aquifer System at Sheffield

- One-dimensional transport is represented along each streamline by an analytical solution of the convective-dispersion equation. (This approach assumes a constant flow velocity that varies with a local-scale dispersion coefficient.)

- A probability-weighted summation of either the fluxes or concentration is calculated along the streamline with a constant flow velocity determined from the travel time and length of the hydrologic streamline. This approach, schematically shown in Figure 4.24, allowed the effects of a distribution of travel times associated with each trench on radionuclide transport at the site to be incorporated.

- The model contains a general empirical description of contaminant release but also includes the choice of three optional release models: 1) a constant fractional release rate, 2) a concentration-limited release based on chemical solubility, and 3) an adsorption equilibrium release based on the K_d value for the nuclide. If known, the user can provide a variable fractional release rate as input. These various release models provided the capability to evaluate a range of contaminant release mechanisms important at the site.

- The retardation factor used in the adsorption-controlled release and the contaminant transport is based on a fixed distribution coefficient (K_d) for each nuclide.

- Radioactive decay of the contaminant is applied to contaminants in both the waste source and the ground-water system. The capability becomes important in simulating the effects of delayed release caused either by waste form containment or stage releases of radionuclides during the period of disposal.

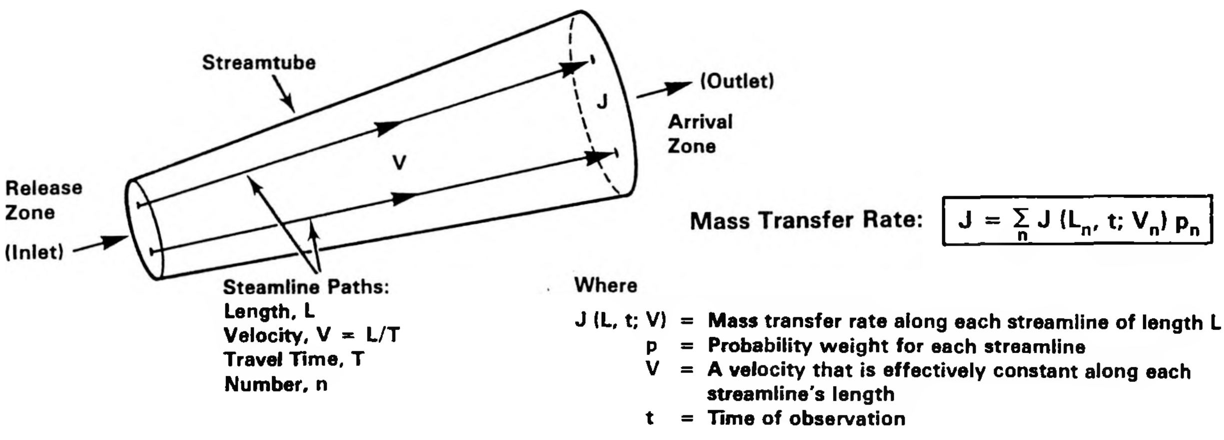


FIGURE 4.24. Streamtube Transport Approach Used in the TRANSS Code

Because the depth to the water table beneath the trenches was relatively small, transport through the vadose zone, as shown in Figure 4.23, was not considered. Given the depth to water and nature of rainfall at the Sheffield site, this assumption is believed to have a negligible effect on simulated transport results.

4.3.3 Modeling Assumptions, Analysis, and Results

The model analysis was performed in two stages. The first stage used a series of simplified one-dimensional streamtube transport using the TRANSS code to perform an initial screening of the list of 11 radionuclides identified from analyses of the inventories. These radionuclides included tritium, ^{137}Cs, ^{60}Co, ^{14}C, ^{90}Sr, ^{241}Pu, ^{241}Am, ^{59}Mn, ^{59}Fe, ^{65}Zn, and ^{95}Zr. The focus of these preliminary analyses was to evaluate the significance of the 1) initial inventory, 2) waste forms or container information, and 3) release or transport processes that would affect radionuclide releases at the site release point. In these analyses, a minimal ground-water travel time from the site to the strip-mine lake was considered. From these preliminary analyses, the list was narrowed to a final short list of radionuclides, which were then evaluated in more detail. In the final analysis, these inventories and more realistic ground-water travel times estimated from the previous USGS modeling and the three-dimensional model developed in this study were used.

4.3.3.1 Initial Screening Analysis

The initial screening analyses were performed in three levels, each of which considered additional levels of information on waste form and container competence and, if available, information on relevant release and transport processes. All of these levels of analysis included a common set of assumptions outlined as follows:

- The entire inventory of the disposal facility for any one radionuclide was released into the underlying ground-water system at the easternmost site boundary. The inventory was released over the disposal period using the step functions illustrated in the Appendix. This release reflected the time-variant disposal of the inventory into the disposal area.

- The area of the release was assumed to include combined areas of all of the trenches at the site. This cumulative area was 21,620 m^2. The inventory of each radionuclide was assumed to be homogeneously mixed within this volume.

- The soil moisture content within the disposal unit was assumed to be 0.10.

- The infiltration rate was assumed to be 5.1 cm/yr, which is the approximate basin-wide average estimated in the three-dimensional model of the site.

- The release point in the analysis was assumed to be at the end of the buried channel containing the pebbly-sand unit of the Toulon Member located east of the site at the strip-mine lake.
- The resulting contaminant mass and volumetric flow derived from the disposal area was introduced and instantaneously mixed with ground water flowing through a streamtube that had a width of 6.1 m and a depth of 3.05 m, which approximated the dimensions of the buried channel east of the site.
- The ground-water velocity was assumed to be 760 m/yr, which is the approximate maximum value of ground-water velocity estimated east of the site at a tracer test site by Garklavs and Toler (1986).
- The effective porosity of the aquifer was assumed to be 0.35.

Further analyses were performed to examine the effect of waste containers and waste form and, if applicable, the effect of solubility-controlling concentrations or adsorption. In the analyses that considered waste containers and waste forms, it was assumed that the dominant waste type derived from the inventory analysis would control the release of the inventory. As a crude approximation of the release, it was assumed that the release of the inventory would occur linearly over the estimated life expectancy of the waste container, as detailed in MacKenzie et al. (1985) and described previously in Section 4.3.1.2. The same assumptions were made about availability of certain amounts of the inventory based on the scenario of breakage of certain waste forms on disposal. For the adsorption-controlled release analyses that require a depth of waste, the waste depth was assumed to be 5 m. Because other controls were used in further analyses, the time-variant addition of the inventory as summarized in the Appendix were not considered.

Results of the screening analyses, detailing the type of source-term release and supplementary information and peak concentrations resulting from the analyses, are presented in Table 4.29. These results indicated that only ^{14}C and ^{90}Sr were released in significant enough concentrations to be considered in further analysis. The remaining radionuclide inventories either decayed or adsorbed onto the Sheffield site soils such that the resulting concentrations released in the vicinity of the strip-mine lake were considered insignificant from a performance assessment standpoint. Tritium, although not particularly significant from a dose standpoint, was carried on for further analysis so that limited model validation could be done with actual field measured concentrations.

4.3.3.2 Transport Analysis of Key Radionuclides

The final analysis of radionuclide transport for the site considered only three radionuclides: tritium, ^{90}Sr, and ^{14}C. Two sets of analyses were performed: one set used estimated ground-water travel times inferred from the two-dimensional analysis done by Garklavs and Healy (1986), the other set was based ground-water travel times predicted with the three-dimensional

TABLE 4.29. Summary of Initial Screening Results of Relevant Radionuclides

Radionuclide	Half-Life (yr)	Type of Release Model: Linear	Type of Release Model: Adsorption-Controlled Distribution-Coefficient (ml/g)	Solubility-Controlled Concentration (pCi/L)	Peak Concentration (pCi/L)
Tritium	12.7	34.7(86)[a]	-	-	1.6E6
^{137}Cs	30.0	-	3,940	-	-
^{60}Co	5.26	-	138	-	-
^{14}C	5,730	-	-	70,000	4.15
^{90}Sr	28.1	-	3.4	-	6,600
^{241}Pu	2.44E4	-	~70	-	-
^{241}Am	2.44E4	-	1,200	-	-
^{54}Mn	0.82	0.4(86)[a]	-	-	0.32
^{59}Fe	0.123	0.4(86)[a]	-	-	-
^{65}Zn	0.67	0.04(86)[a]	-	-	<0.002
^{95}Zr-Nb	0.274	0.08(86)[a]	-	-	<2E-11

(a) Annual Release Rate (years of release).

analysis developed in this study. These analyses were performed on individual trenches or groups of trenches in close proximity to each other so that ground-water travel times were not considered significant. This was done to accommodate the wide variation in ground-water travel predicted by the ground-water models for individual or groups of trenches and to isolate the impact of inventories in each trench.

The majority of the assumptions detailed in the screening analysis were used with a few exceptions. Because each trench was analyzed separately, the surface area and depth of the waste intercepting infiltration water was adjusted to the dimensions provided in Table 2.2. The inventory estimates were based on the information presented in Tables 4.6 through 4.25. The inventory within each trench was assumed to be thoroughly mixed within the assumed waste volume of each trench.

The ground-water travel times used in the transport analyses for individual trenches were derived from the previous two- and three-dimensional modeling (Tables 4.4 and 4.5). The values from the modeling analyses provided two estimates of travel time because the streamlines were initiated at both the east and west end of the trenches. It was assumed that the range defined by the two values represented the range in travel time of water originating from the trenches. The ability of the TRANSS code to account for dispersion in the ground water, by assigning probability weights to different travel times and distances, was used to incorporate the effects of the range of travel times derived from the modeling. Discussion of further important assumptions for the specific release analysis for tritium, ^{90}Sr, and ^{14}C are provided in the following sections.

Tritium Transport Analysis. Waste containment was the only controlling release mechanism considered in the release of tritium. The simulated release of tritium was significantly affected by the underlying assumptions made by MacKenzie et al. (1985) and the estimated tritium inventory by waste type. The distribution of tritium by waste type, as summarized in Table 4.14, illustrates the overall trends in tritium disposed of at the site. Since this distribution was used to estimate the inventories in the remaining undocumented trenches, given in Tables 4.15 through 4.24, this pattern reflects the tritium used in almost half of the tritium inventories released. This documentation shows a significant percentage of the tritium contained in fiberboard boxes. In this release analysis, it was assumed that all of the inventory is available immediately for release. The remainder of the tritium inventory was assumed to be held in steel drums containing unconsolidated waste and cemented waste. Because of assumptions about a certain amount of inventory becoming available on emplacement breakage, an additional significant component of the tritium inventory becomes available immediately. The remainder of the waste is released linearly over the expected lifetime of the steel drums or about 86 years.

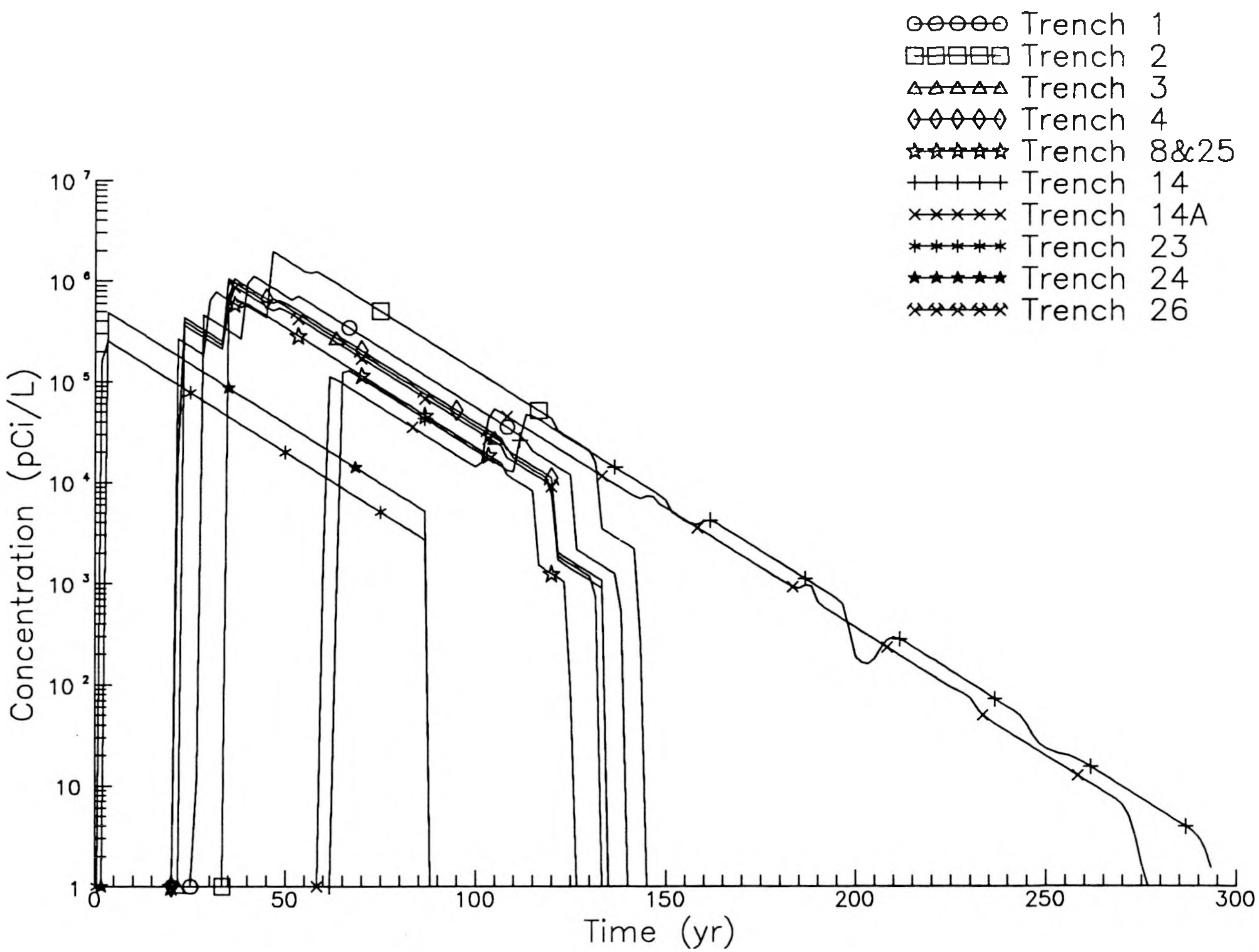


FIGURE 4.25. Summary of Transport Results by Trench for Tritium Using Flow Results from the USGS Two-Dimensional Model

Results of the tritium release for the 14 trenches analyzed, based on the two-dimensional ground-water model travel times provided in Figure 4.25, are consistent with the assumed postulated tritium release. The plots resulting from each trench are adjusted in time for the operational life of the trench relative to the opening of the site to demonstrate the time lag between the individual trench releases. The earliest arrival times of elevated levels of tritium are from Trenches 23 and 24 at about 2 yr, which is consistent with the short travel times derived from the USGS model. The peak tritium concentration from a release from Trench 2 occurs after 47 yr when it reaches a level of 1,911,000 pCi/L. The concentration of tritium in the effluent decreases over time by decay to less than 1 pCi/L after about 290 yr. When a trench inventory becomes depleted, predicted concentrations drop to zero. In Figure 4.25, the effect of inventory depletion for several of the trenches occurs at about 150 yr.

The combined results of the tritium analyses using the three-dimensional model travel times (illustrated in Figure 4.26) show similar results as those resulting from the two-dimensional model. In general, the three-dimensional model results have a later first arrival and an overall shorter release time than the results using the two-dimensional travel times. These results are a reflection of the overall distribution of travel times and distances predicted by the three-dimensional model as compared to those predicted by the two-dimensional model. Differences are a function of the different travel times estimated between the two modeling approaches. The first elevated concentration of tritium at the release point from the trenches is from Trenches 23, 24, and the combination of Trenches 8 and 25C at about 7 yr. This represents the first arrival of ground water beneath these trenches by the three dimensional model. The peak concentration for the entire release is derived from Trench 14 at about 7 yr with a concentration of 1,632,500 pCi/L. Concentrations drop, because of the inventory depletion by decay, below the 1 pCi/L level at about 220 yr.

Strontium-90 Transport Analysis. Two controlling release mechanisms considered in the ^{90}Sr release were adsorption and waste containment. In initial runs, only the effect of adsorption of ^{90}Sr was considered. This delays the arrival of ^{90}Sr at the release point by a factor of 17.5. This factor is the calculated retardation factor based on a minimal K_d of 3.4 ml/g.

The combined results of the ^{90}Sr analysis for the 14 trenches modeled using the above assumptions and the two-dimensional model analysis travel times are shown in Figure 4.27. The earliest arrival times of elevated levels of ^{90}Sr are from Trenches 23 and 24 at about 25 yr. The peak concentrations occur at about 40 yr where they reach a level of about 6940 pCi/L. The concentration of ^{90}Sr at the release point decreases over time by the decay process. Concentrations decrease below the 10 pCi/L level after about 450 yr. Beyond about 600 yr, concentrations drop below 1 pCi/L permanently. The ^{90}Sr inventory from all other trenches is exhausted by decay before any release is realized.

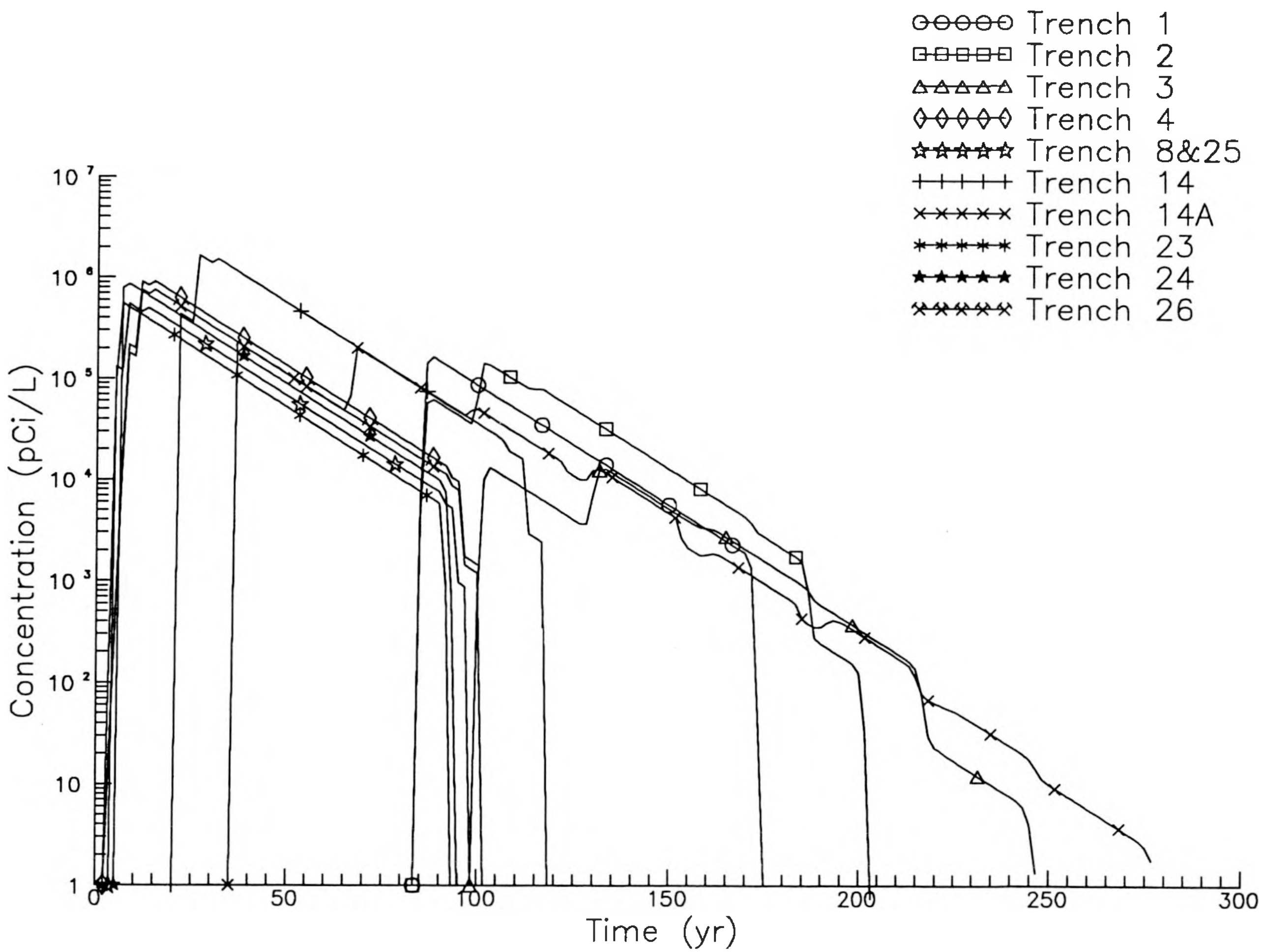


FIGURE 4.26. Summary of Transport Results by Trench for Tritium Using Flow Results from the Three-Dimensional Model

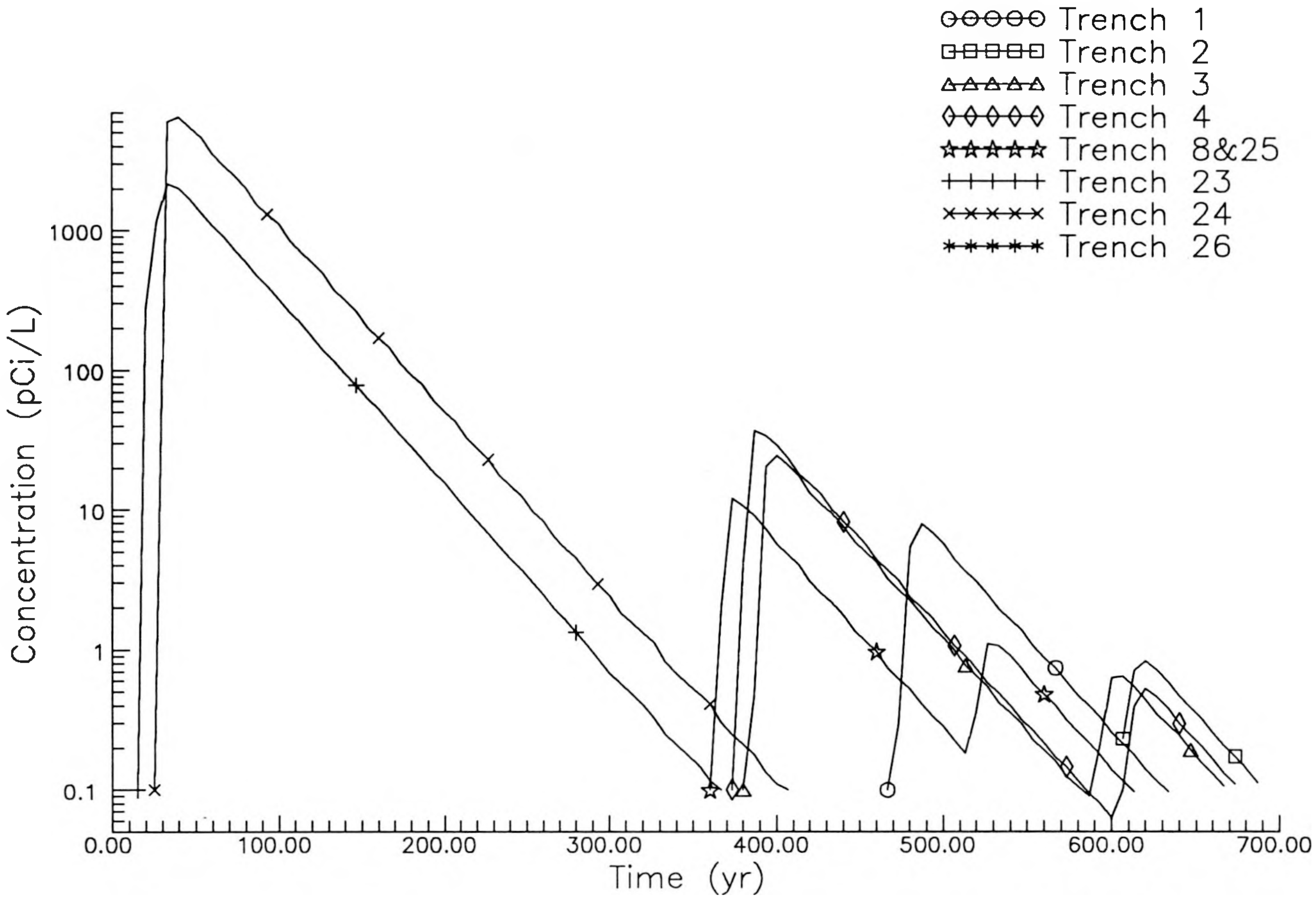


FIGURE 4.27. Summary of Transport Results by Trench for ^{90}Sr Using Flow Results from the Two-Dimensional Model

The combined results of the ^{90}Sr analysis using the three-dimensional ground-water model travel times and the same assumptions about waste release and adsorption are shown in Figure 4.28. The distributions of concentration profiles are quite different than those predicted by the two-dimensional model results. However, the magnitudes of the concentrations exhibit the same range. The earliest arrival times of elevated levels of ^{90}Sr are again from Trenches 8 and 25 at about 75 yr. A release from the combination of Trenches 23 and 24 and Trench 4 are realized after about 100 and 120 yr, respectively. The peak concentration of ^{90}Sr, 6790 pCi/L, is derived from Trench 4 after about 118 yr. The ^{90}Sr inventory from a larger number of trenches (1,2,3,11,14, and 14a) was exhausted by decay before release.

The effect of waste containment was also evaluated. As in the case of the tritium release, the release of strontium was correlated to its estimated inventory by waste type. The distribution by waste type, as summarized in Table 4.14, illustrates that the majority of strontium is contained in lead-lined drums. Because these waste containers have a life expectancy of 300 yr, it was assumed the majority of the inventory would not be released from the trenches until about 300 yr. However, for some trenches (Trenches 23 and 24), the inventory of ^{90}Sr was assumed to be primarily contained in steel drums containing unconsolidated and cemented waste. With the assumptions made on availability caused by emplacement breakage, a part of ^{90}Sr inventory from a number of the trenches, most notably Trenches 23 and 24, is made available immediately. The remainder of the steel drum waste is released linearly over about 86 yr, which is the mean total disintegration time of the steel drums as estimated by MacKenzie et al. (1985).

When waste containment is included with adsorption, predicted peak concentrations are reduced. In this analysis, where the release is delayed by the expected lifetime of the most competent waste, reductions in peak concentrations are linearly related to changes in available inventory. For an 86-yr release, the peak concentration from Trench 24 (6790 pCi/L) would be reduced by a factor of 0.0116 (78.7 pCi/L). For a 300-yr release, the factor would be 0.003 (20.4 pCi/L). Peak concentrations for trenches releasing at later times would also be reduced by the same factors.

<u>Carbon-14 Transport Analysis</u>. For ^{14}C release, a solubility-controlled release process appeared to be most appropriate for use in this analysis. The MacKenzie et al. (1985) estimates of inventory (Table 4.14) indicate that all of the ^{14}C is either in fiberboard boxes or steel drums containing unconsolidated and cemented waste. Given the assumptions on waste availability and waste release for these waste types, a significant amount of ^{14}C would be available for release early on. The remainder would be released linearly over the 86 yr postulated mean total disintegration time of the steel drums (MacKenzie et al. 1985).

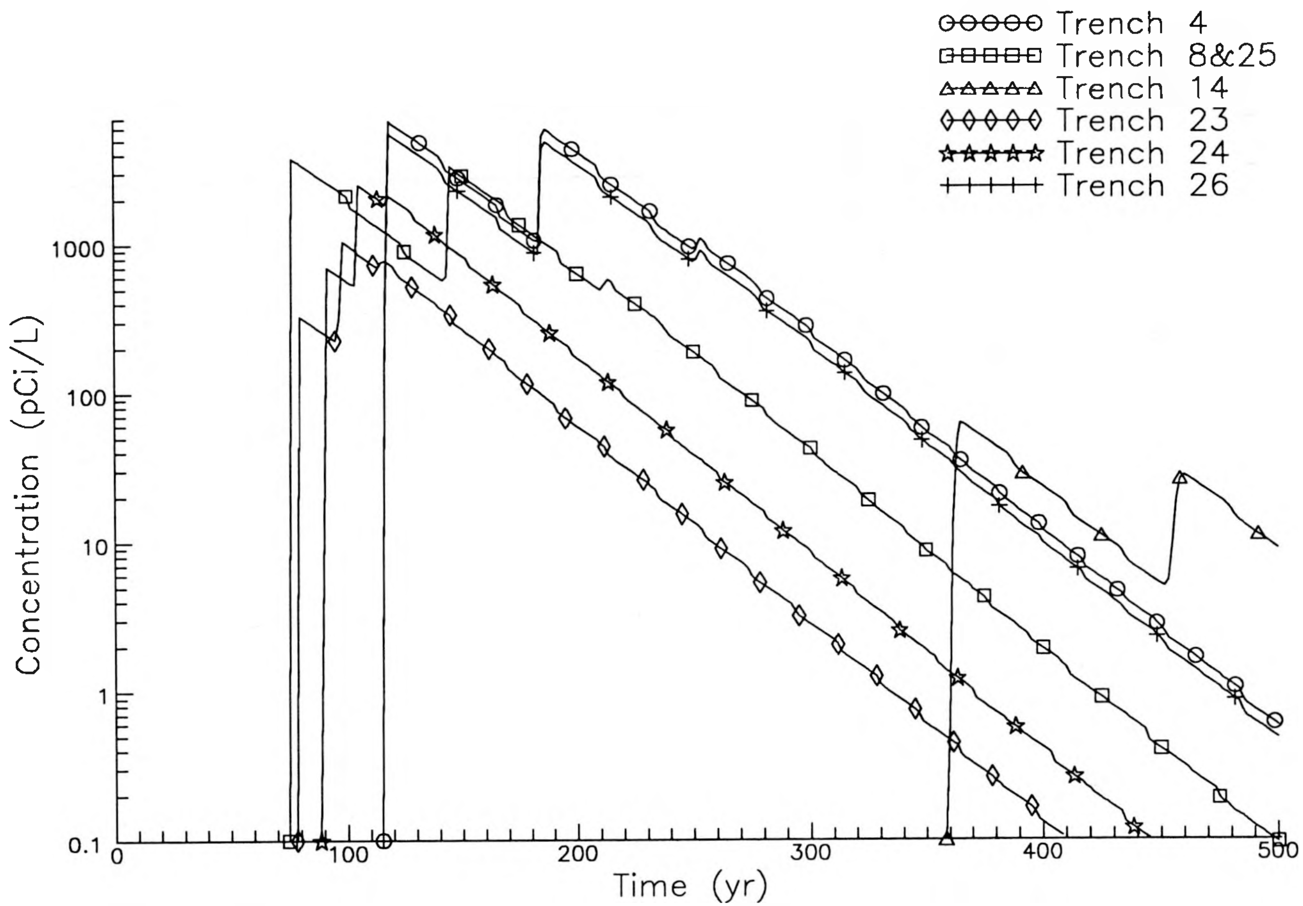
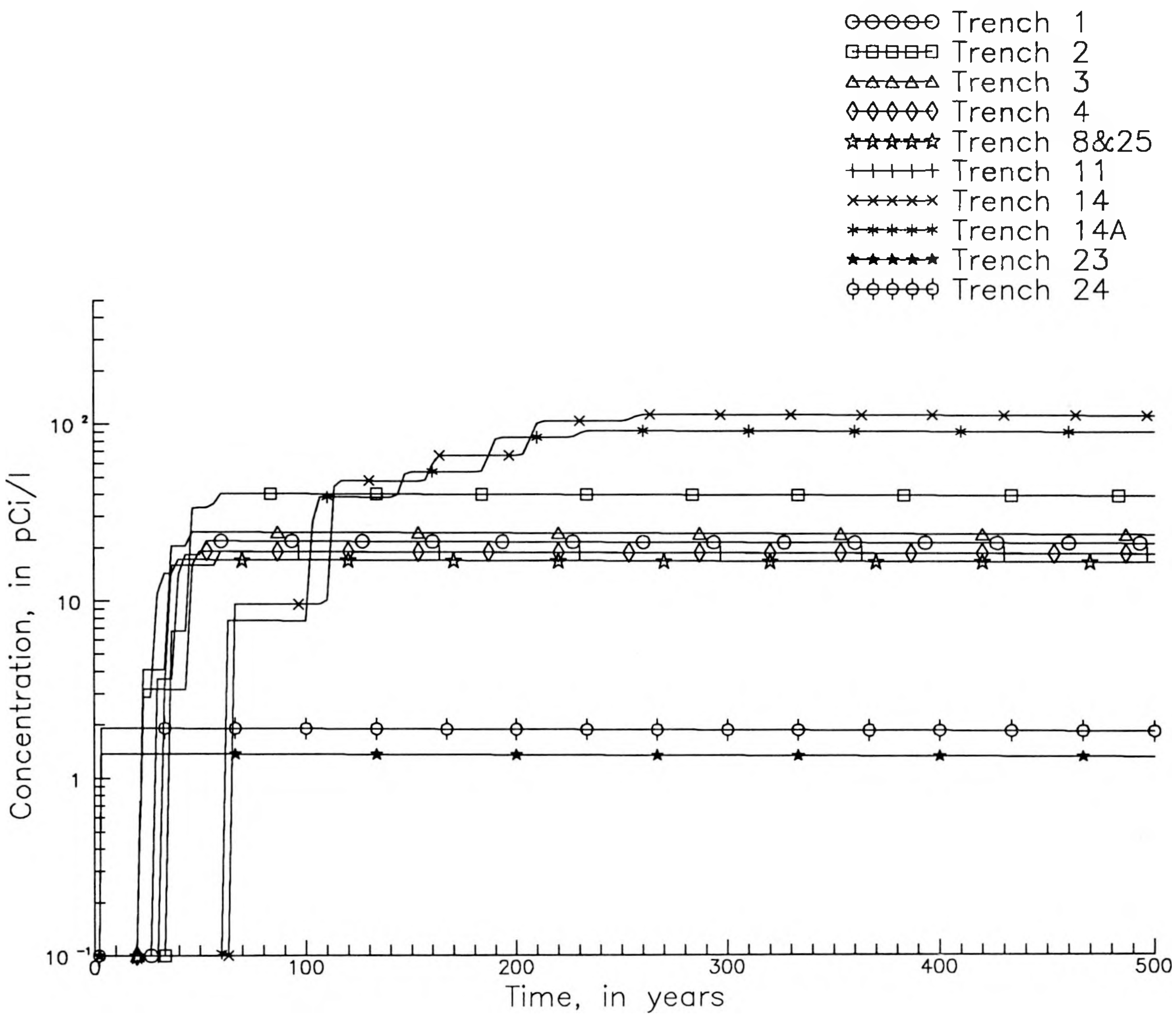


FIGURE 4.28. Summary of Transport Results by Trench for ^{90}Sr Using Flow Results from the Three-Dimensional Model

Chemical solubility would control the release concentration of ^{14}C and perhaps extend the release of the inventory over a longer period of time. The difficulty in using this type of release model is that no data on ^{14}C solubility have been collected at Sheffield. Thus, theoretical limits must be relied on, such as given in Table 4.8, which are derived for fairly narrow geochemical conditions and are not likely to reflect the complex geochemical conditions encountered in a trench filled with low-level waste. For the same reasons given in Bergeron et al. (1987), appropriate solubilities controls may be found by examining trench leachate chemistry from existing commercial LLW sites. At the West Valley LLW site in New York, where ground-water chemistry could be considered similar to Sheffield (i.e., carbonate-dominated, near neutral pHs), numerous analyses of trench leachate have been performed and documented in Davis et al. (1980). These analyses can provide insight into the solubility of ^{14}C in LLW trench environments. To approximate a likely ^{14}C solubility for this analysis, the highest concentration of ^{14}C found at the West Valley site was used (70,000 pCi/L).

The results of the ^{14}C analysis for the 14 trenches based on the two-dimensional flow model results and the solubility-controlled concentration are shown in Figure 4.29. The earliest arrival times of elevated levels of ^{14}C are from Trenches 23 and 24 at 3 yr. The initial concentration is in the 1 to 2 pCi/L range. As the remainder of the inventory is released, concentrations rise to a level of about 110 pCi/L after about 200 yr. The concentration of ^{14}C eventually decreases over time by the decay of the isotope, but the extremely long half-life of the isotope makes the attenuating effect of decay unnoticeable on the time scale depicted in this plot. The primary means of inventory exhaustion is the release of the contaminant at the assumed solubility. For the estimated trench inventories and solubility, this occurs over a period of several hundred to a few thousand years.

Results of the ^{14}C analysis using the three-dimensional model travel times are shown in Figure 4.30. The first elevated ^{14}C concentration occurs from Trenches 23 and 24, and the combined 8/25C trench after 7 yr, with other trenches soon following. The initial elevated concentration of ^{14}C is in the 3 to 5 pCi/L range. The concentration resulting from the release from other trenches rises to a level of about 60 pCi/L (from Trench 2) after about 120 yr. As in the two-dimensional model results, the effects of the long half-life of ^{14}C and decay are not noticeable on this graph. Thus, the release concentration of ^{14}C is sustained until the inventory becomes exhausted.



<u>FIGURE 4.29</u>. Summary of Transport Results by Trench for ^{14}C Using Flow Results from the USGS Two-Dimensional Model

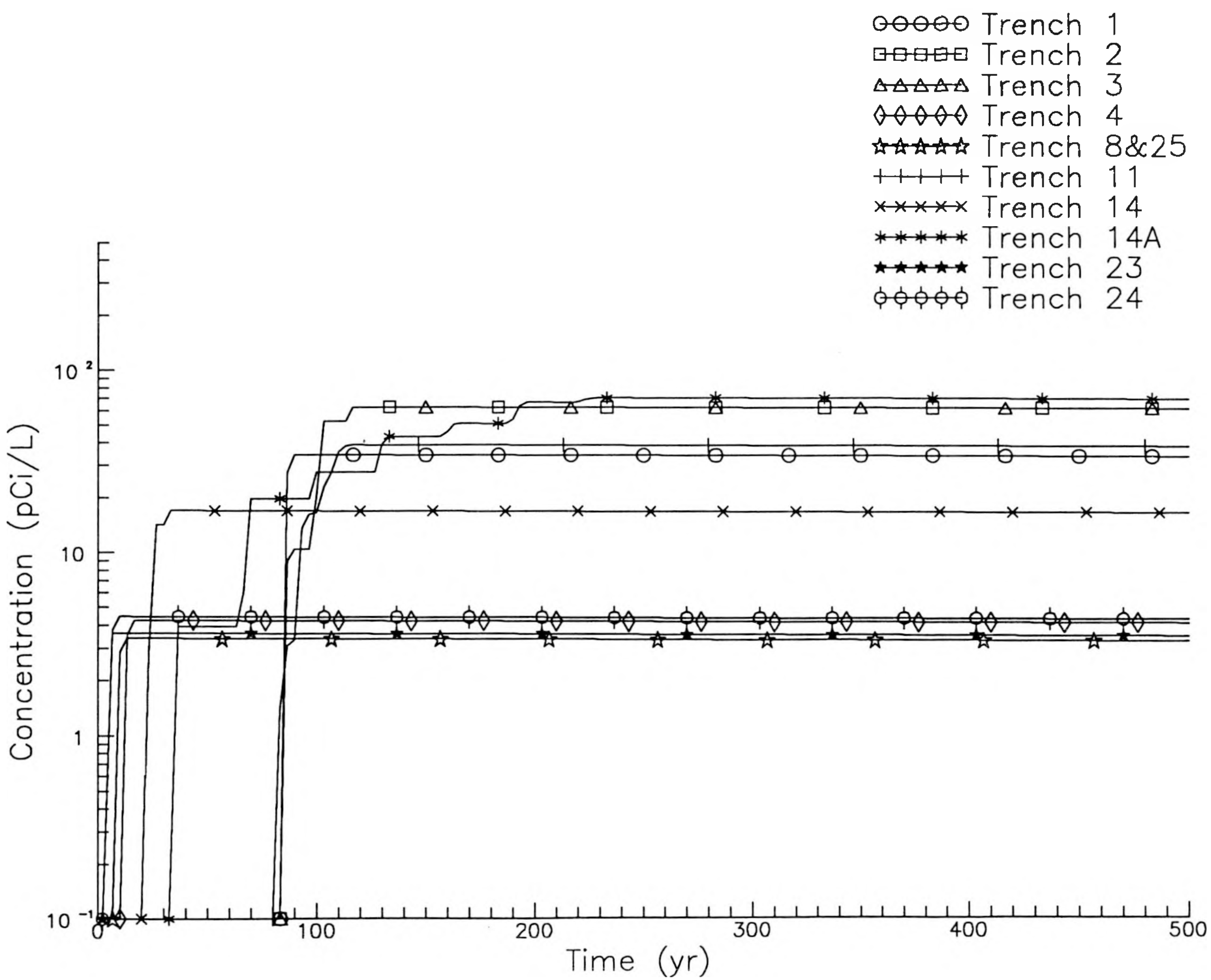


FIGURE 4.30. Summary of Transport Results by Trench for ^{14}C Using Flow Results from the Three-Dimensional Model

5.0 CONCLUSIONS AND DISCUSSION OF PERFORMANCE ASSESSMENT RESULTS

Results of the performance assessment analysis suggest that tritium, ^{90}Sr, and ^{14}C would be the only radionuclides released from the Sheffield site in any significant concentration. The following section discusses the adequacy and reasonability of these performance assessment results and describes the uncertainty and sensitivity of key parameters used in the analysis. As stated earlier, this performance assessment analysis was not intended to provide a definitive judgment of Sheffield site compliance with the licensing regulations put forth in 10 CFR 61. No dose calculations were performed to assess site performance with regard to regulatory limits. For sake of comparison, however, peak concentrations of the key radionuclides from the model results were compared to applicable drinking water standards and dose equivalent concentrations that would be consistent with the regulatory limit of 25 mrem/yr whole body dose.

5.1 GROUND-WATER FLOW MODEL ANALYSIS RESULTS

The complexity of the hydrogeologic setting and the apparent large changes in hydraulic conductivity both vertically and horizontally within the surficial sediments indicated that an appropriate approach to modeling the Sheffield site would be with the use of a three-dimensional model. A three-layer model of the site that treated the relatively highly permeable pebbly-sand unit of the Toulon as a separate unit was deemed appropriate. The travel times predicted by the three-dimensional model had a smaller range (4.4 to 228 yr) than the two-dimensional model (1.1 to 292 yr). In both models, travel times are generally longer for trenches close to the strip-mine lake and shorter for trenches far from the lake. However, while the three-dimensional model may have provided slightly different estimates of the flow direction and the overall ground-water travel time to the lake, the overall results did little to significantly change the transport results derived from the two-dimensional modeling performed by Garklavs and Healy (1986).

Transport results using the two-dimensional flow analysis were similar for all three radionuclides, although results from the three-dimensional model travel times produced slightly lower peak concentrations and the release occurred over a short period of time. Although the magnitudes of concentrations were about the same, there were clearly differences in the contribution of individual trenches. However, for this particular site, changes in the overall performance assessment using the two approaches were not significant for the radionuclides in question.

5.2 TRANSPORT RESULTS

5.2.1 Tritium

A comparison of peak concentrations from the modeling results with the regulatory standards, given in Table 5.1, indicates that predicted

TABLE 5.1. Comparison of Regulatory Limits and Peak Concentration Resulting from Performance Assessment Simulations

Radionuclide	25 mrem Dose Equivalent Concentration (pCi/L)[a]	25 mrem Dose Equivalent Concentration (pCi/L)[b]	Peak Concentration (pCi/L)
Tritium	281,000	595,000	1,911,000[c] 1,632,500[d]
^{14}C	610	17,857	111.7[c] 62.3[d]
^{90}Sr	5	284	
K_d-control			6,938[c] 6,791[d]
300 Year Release			20.8[c] 20.4[d]

(a) Full garden scenario.
(b) Drinking water only.
(c) Based on two-dimensional flow model results.
(d) Based on three-dimensional flow model results.

concentrations of tritium is well above the 25 mrem dose equivalent concentration using the full garden scenario and exceeds the drinking water standard (DWS) of 20,000 pCi/L by about a factor of about 100. This conclusion is not considered significant, however, since concentrations of tritium in excess of the DWS have been measured east of the site for a number of years. Tritium was first observed in 1982 and has since been detected along the entire length of the buried channel leading east of the site to the strip-mine lake. Tritium has also been detected in seeps along the banks of the lake. Tritium concentrations do not vary significantly from one location to another along the narrow channel, ranging from 10,000 to 100,000 pCi/L. This lack of variability in concentration suggests that migration in the channel is rapid. Tritium concentrations in ground-water in the channel east of the site have commonly ranged between 10,000 and 100,000 pCi/L. Tritium in ground water around the site has ranged from below detection level (about 20 pCi/L) to more than 300,000 pCi/L.

A comparison of simulated tritium concentrations east of the site in the time frame of the burial history would suggest that model results are greater than highest measured values by a factor of 2 or 3. Model results within the 20-yr time frame give tritium concentrations on the order of several hundred thousand pCi/L, while measured values range from 10,000 to 100,000 pCi/L. This discrepancy between actual and predicted concentrations likely reflects errors in the assumed tritium inventory estimates, tritium availability in the inventory, and/or the actual tritium release from the multitude of waste forms considered in the performance assessment. Both the arrival of simulated peak concentrations and the magnitude of the peak values from Trench 23 would suggest that the observed tritium migration could reflect leaching of tritium from this trench alone.

The modeled release of tritium indicates that tritium concentrations in the channel will be sustained and will increase as tritium leached from other trenches in the burial area migrate offsite.

A number of parameters have a significant effect on predicted tritium results. One area of obvious uncertainty lies in estimates of the inventory. The error associated with these estimates is difficult to quantify but given the nature of the way the inventories were estimated, errors of $\pm 50\%$ are not unreasonable. To illustrate the sensitivity of changes in inventory estimates, Table 5.2 lists peak concentrations predicted for Trench 14 with different inventory estimates. For the approach used in this study, the relationship is linear.

Another area of uncertainty is the time-dependent nature of tritium release from the waste inventory. This parameter is by the far the most important transport parameter, but also the most difficult to quantify. In this analysis, a number of assumptions were made about initial breakage, inventory availability, waste form corrosion and leaching that would allow simplifying the temporal behavior of tritium release into a linear function following the initial breakage. It was assumed all of the tritium would be eventually released over the anticipated lifetime of the most competent waste form (i.e., steel drums). In reality, the release is very complex because of the wide variety of waste forms and the great uncertainty associated with the breaching, corrosion, and leaching rates of the various waste forms. The intent of using a linear release was to approximate the total release rather than the details of actual release rates from the variety of waste forms in an individual trench. Certainly, actual release rates could be far less or far greater on an annual basis than those depicted in a linear release. Because of the sheer number of permutations involved in assessing the effects of various plausible release models, a detailed sensitivity analysis of all possibilities could not be performed. However, the sensitivity of these release rates can be illustrated by evaluating model results using different linear release periods. A summary of sensitivity analyses over a limited range of linear release periods is presented in Table 5.3.

TABLE 5.2. Predicted Peak Tritium Concentrations Resulting from Changing Inventory Estimates for Trench 14

Available Inventory (Ci)	Predicted Peak Concentration (pCi/L)
246	2,493,700
492[a]	1,632,500
738	830,220

(a) Base case values.

5.2.2 Strontium-90

Final predicted ^{90}Sr concentrations using the adsorption-controlled release approach were far in excess of the 25 mrem/yr dose equivalent concentration. This analysis used a retardation factor representative of the low end of distribution coefficients for ^{90}Sr measured at Sheffield (3.4 ml/g). The highest K_d was on the order of 17.3 ml/g and most values were in the 4 to 7 ml/g range. Table 5.4 and Figures 5.1 and 5.2 illustrate the effect of varying K_d on predicted peak concentrations and arrival times for Trench 24. These results demonstrate that small changes in the K_d can result in significant changes in both the predicted concentrations and arrival times. However, within the range of K_ds measured at the site, predicted peak ^{90}Sr concentrations are still well in excess of regulatory standards. For this particular trench, the K_d would have to be in excess of 10 ml/g before predicted concentrations fall below these standards.

The delayed or slowed release caused by waste form or containers can reduce predicted concentrations by orders of magnitude. If it is assumed that, at the very least, the waste is released linearly over a period of time equivalent to the expected lifetime of the most competent waste form, the amount of inventory available for release is reduced by a factor equal to one over the waste form lifetime. The predicted peak concentration would correspondingly be reduced by this amount. Thus, since the majority of ^{90}Sr is found in lead-lined drums and containers that have an estimated lifetime of 300 yr, the peak concentration of 6790 pCi/L predicted by the three-dimensional model results could potentially be further reduced by a factor of 0.003 to about 20 pCi/L. Any further reductions in the annual availability of ^{90}Sr due to waste containment or increases in the effective K_d would decrease model results well below this level.

Another parameter key to this analysis is the recharge or infiltration rate used to mobilize radionuclides released from the wastes. In this analysis, a recharge rate of 5.1 cm/yr is used, which is believed to be the

TABLE 5.3. Predicted Peak Tritium Concentrations Resulting from Changing Linear Release Periods for Trench 14

Linear Release Period (yr)	Predicted Peak Concentration (pCi/L)
43	3,324,900
86[(a)]	1,632,500
172	831,200

(a) Base case values.

TABLE 5.4. Predicted Peak ^{90}Sr Concentrations Resulting from Changing Retardation Factors for Trench 24

Retardation Factors	Distribution Coefficient (ml/g)	Predicted Peak Concentration (pCi/L)	Peak Arrival Time (yr)
1.0	0.0	485,500	6.6
9.75	1.7	15,680	53.3
17.5	3.4[(a)]	2,414	107
20.91	4.1	1,234	127
27.7	5.5	347	166
35.0	6.8	94.3	210
50.0	10.0	7.2	300
100.0	20.0	0.003	520

(a) Base case values.

approximate average site-wide recharge rate determined from the modeling analysis. Certainly, based on previous studies by the USGS and others, it is reasonable to assume that the variability around this "average" value is perhaps as little as 2.5 cm/yr or as much as 25 cm/yr. Using lower or higher values of infiltration of this magnitude will affect transport results. A comparison of predicted peak concentrations (Table 5.5) for Trench 24 using the three-dimensional model results over a limited range of infiltration rates illustrates this effect.

Another parameter affecting model results is the effective porosity of the aquifer system. For this analysis, 0.35 was used because it is thought to be representative of the pebbly-sand unit of the Toulon member. The net effect of decreased porosity is a proportional decrease in travel time and, because less decay occurs, a net increase in concentration. The effect of porosity, presented in Table 5.6, illustrates this relationship.

Using conservative estimates of the various parameters, the performance assessment done in this study predicts that significant levels of ^{90}Sr will be released. However, if more typical values of Kd are combined with the effects of waste form containment, these predicted ^{90}Sr concentrations can be reduced well below regulatory standards. The sensitivity of recharge and porosity within the range of uncertainty are not considered as important to modeled results as the effects of K_d and waste containment.

5.2.3 Carbon-14

Carbon-14 concentrations on the order of 62,000 pCi/L, as controlled by a prescribed solubility limit in the model analysis, are predicted to occur after a period of about 100 yr. This concentration level is above the 25 mrem dose equivalent concentration given in Table 5.1. The levels of ^{14}C

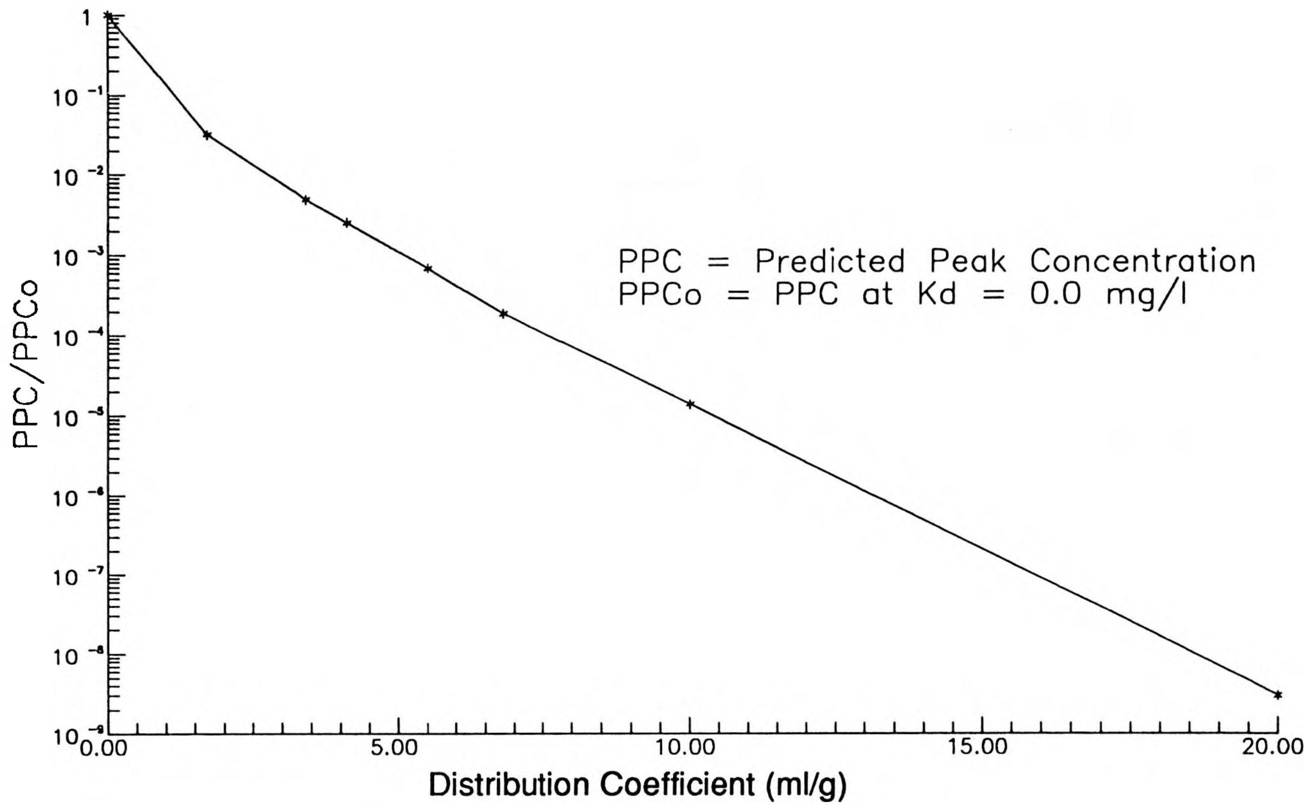


FIGURE 5.1. Sensitivity of Predicted Peak Concentration to Changes in Distribution Coefficients for Trench 24

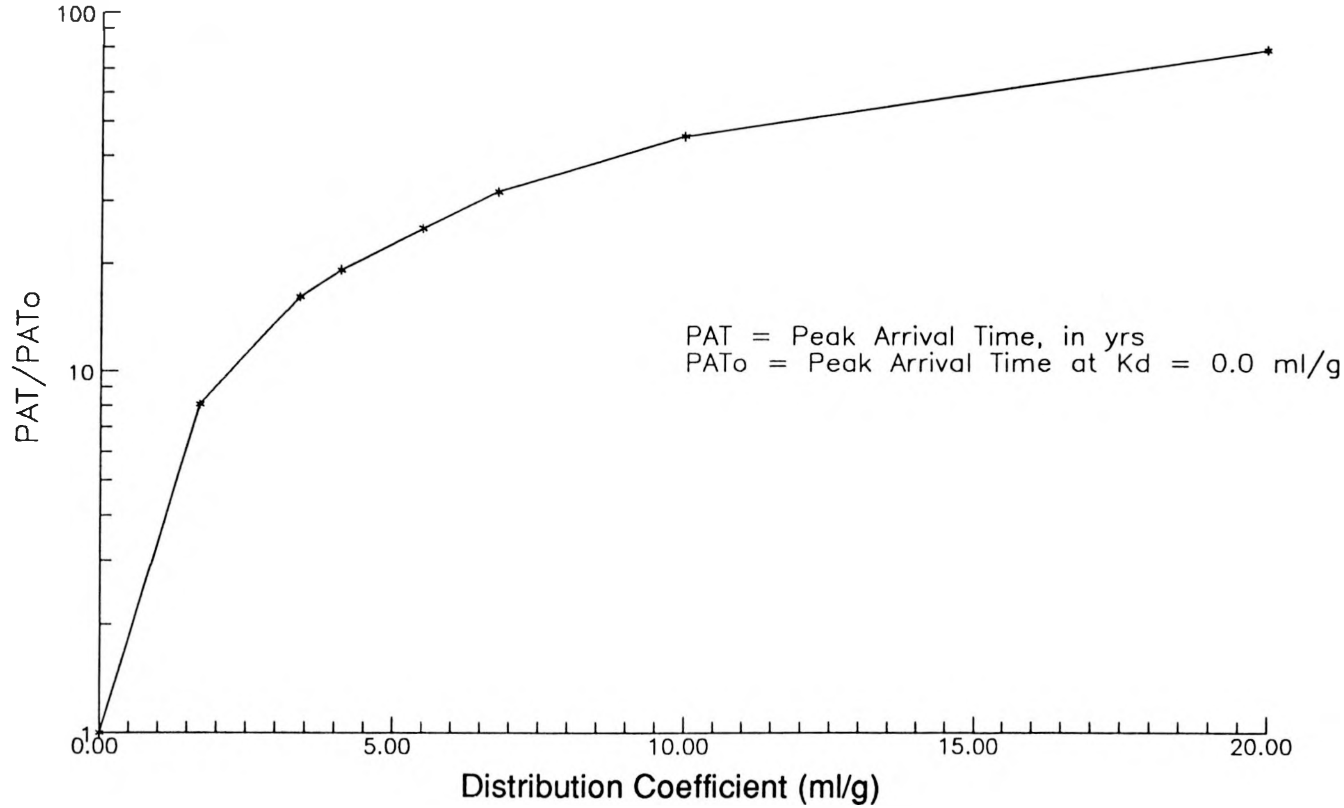


FIGURE 5.2. Sensitivity of Peak Arrival Time to Changes in Distribution Coefficients

TABLE 5.5. Predicted Peak ^{90}Sr Concentration Resulting from Changing Recharge Rates for Trench 24

Recharge Rates (cm/yr)	Predicted Peak Concentration (pCi/L)
2.5	329
5.1[a]	2,414
10.2	4,701
25.0	8,850

(a) Base case values.

TABLE 5.6. Predicted Peak ^{90}Sr Concentrations Resulting from Changing Effective Porosity for Trench 24

Effective Porosity	Predicted Peak Concentration (pCi/L)
0.15	5,633
0.25	3,380
0.35[a]	2,414
0.45	1,878

(a) Base case values.

predicted offsite before 10 yr are inconsistent with the lack of detection of ^{14}C in site ground water. The lack of measured ^{14}C in ground water indicates that actual leaching and the migration of ^{14}C are less than assumed in the model analysis.

When West Valley ^{14}C concentrations are used as solubility limits, results are below regulatory limits. The use of more typical values of leachate concentrations measured at West Valley (i.e., 70 to 7000 pCi/L) could further reduce predicted transport concentrations to negligible levels. The effect of changing solubility limits for the ^{14}C analysis are direct and linear as illustrated in Table 5.7.

TABLE 5.7. Predicted Peak ^{14}C Concentrations Resulting from Changing Solubility-Controlled Concentrations for Trench 24

Solubility Concentration (pCi/L)	Predicted Peak Concentration (pCi/L)
70	0.016
700	0.16
7,000	1.65
70,000[(a)]	16.96
700,000	163.54

(a) Base case values.

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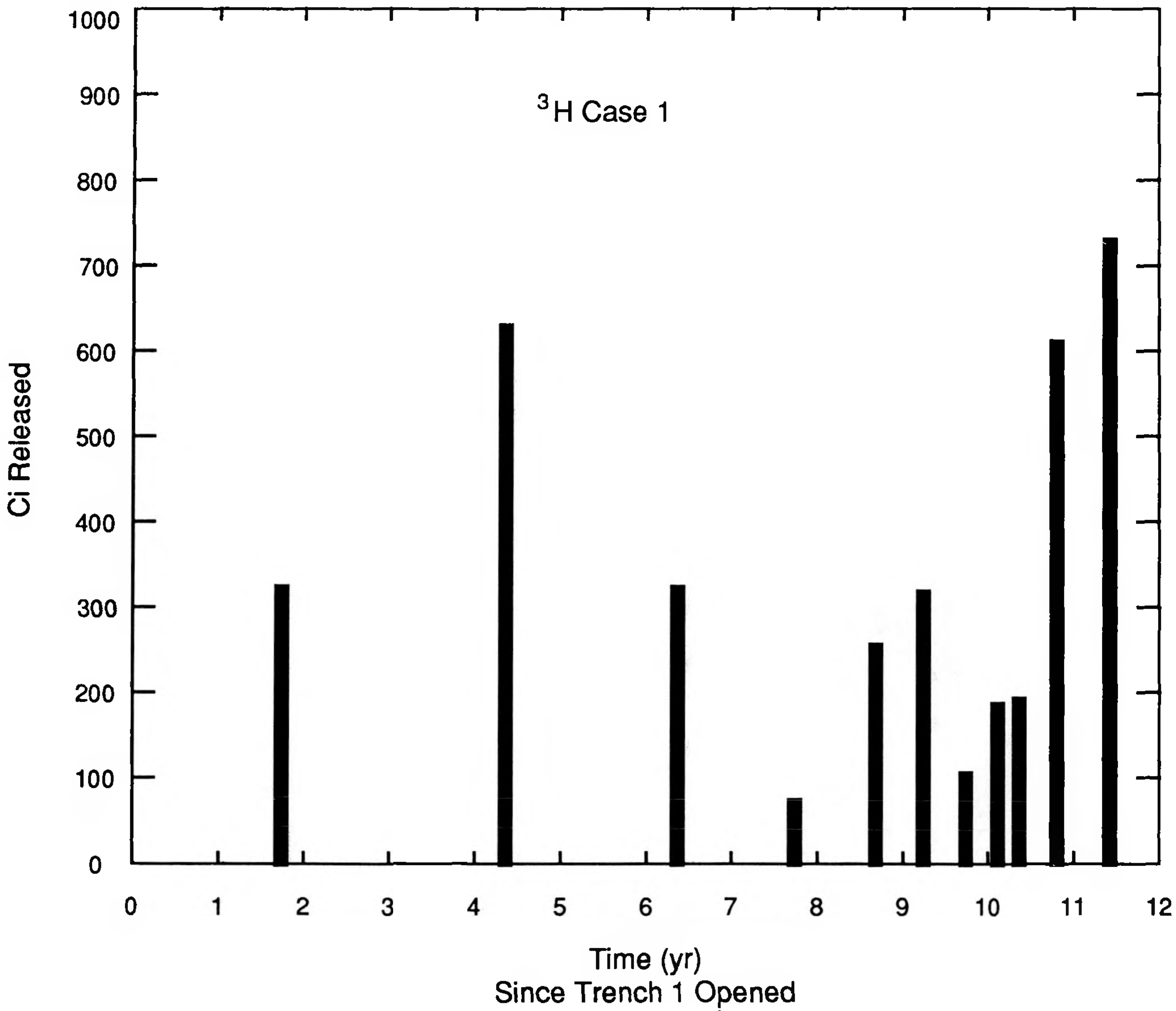
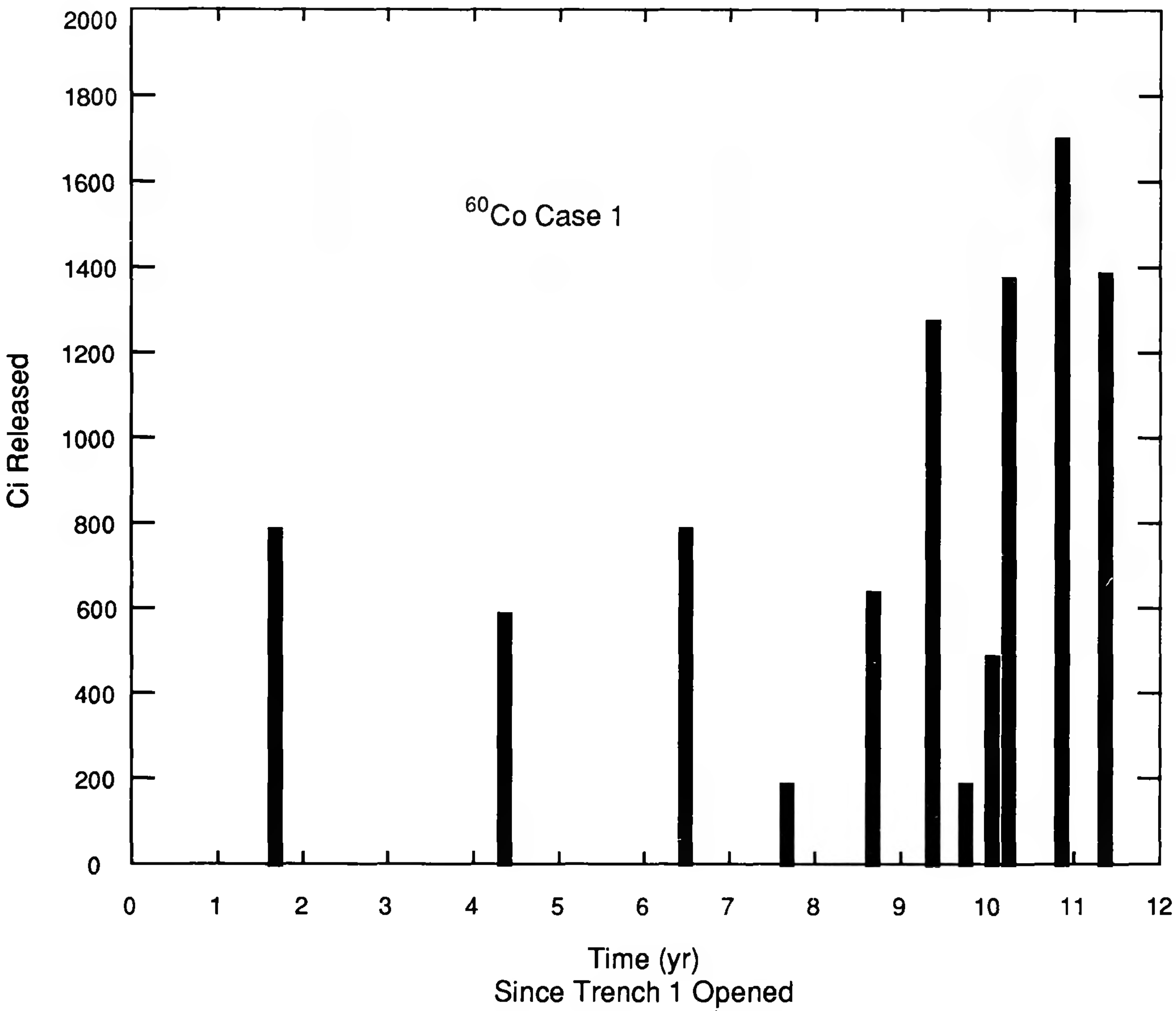
APPENDIX

WASTE RELEASE SCENARIOS USED IN INITIAL SCREENING ANALYSIS FOR SELECTED RADIONUCLIDES

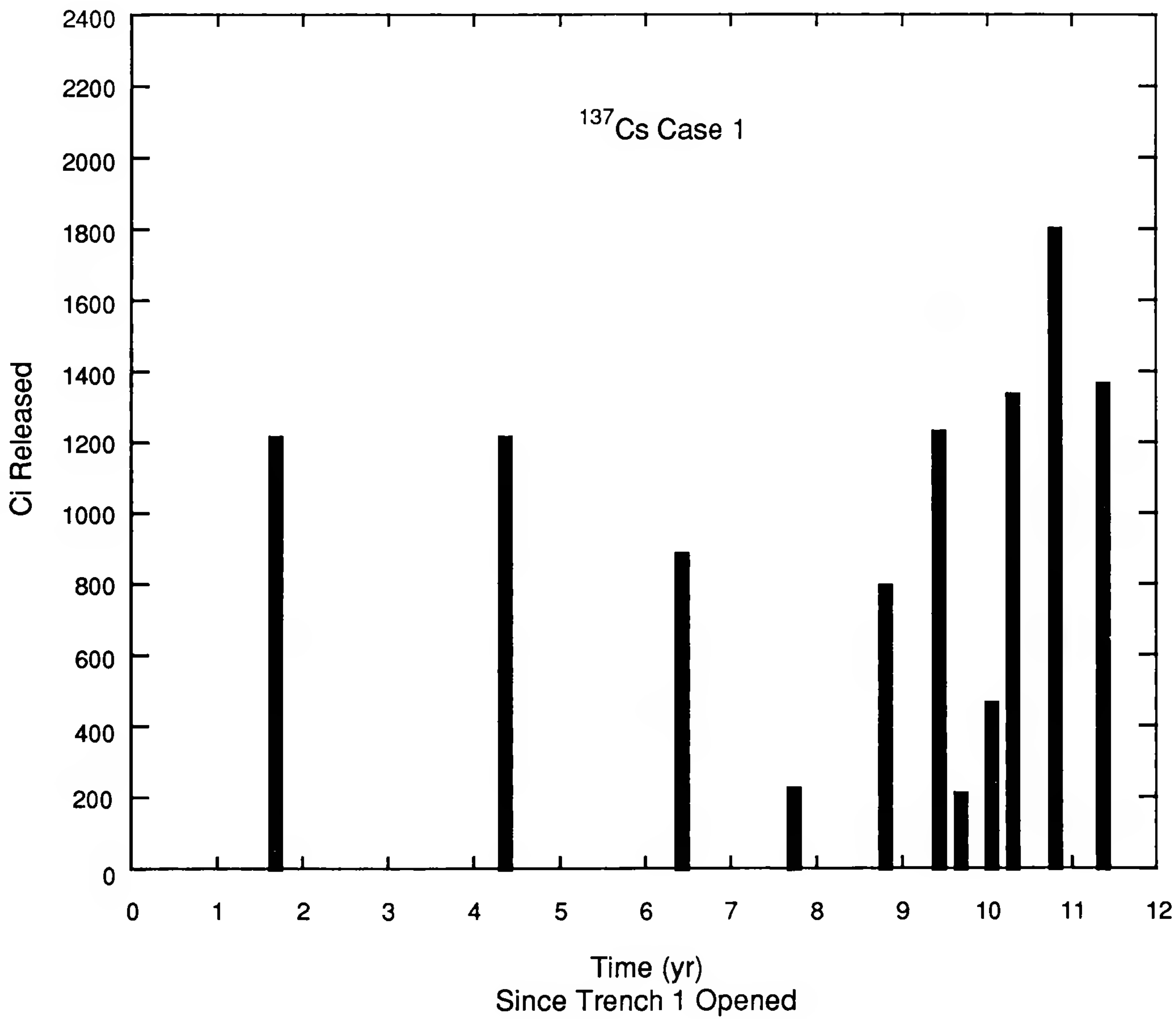

^{3}H Case 1
Ci Released
0
100
200
300
400
500
600
700
800
900
1000
0
1
2
3
4
5
6
7
8
9
10
11
12
Time (yr)
Since Trench 1 Opened

S8901095.1M

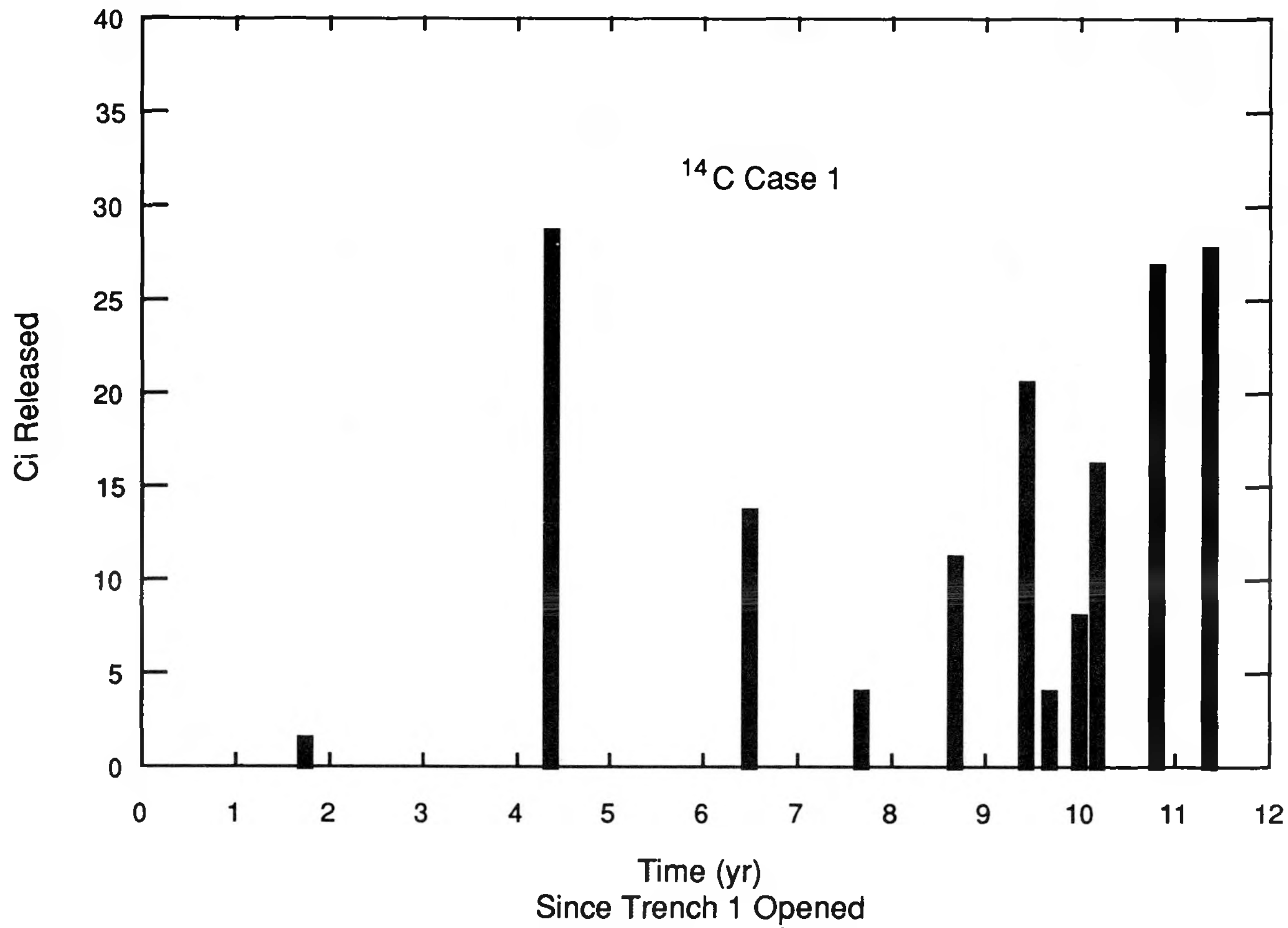


^{60}Co Case 1
Ci Released
0
200
400
600
800
1000
1200
1400
1600
1800
2000
0
1
2
3
4
5
6
7
8
9
10
11
12
Time (yr)
Since Trench 1 Opened

S8901095.3M

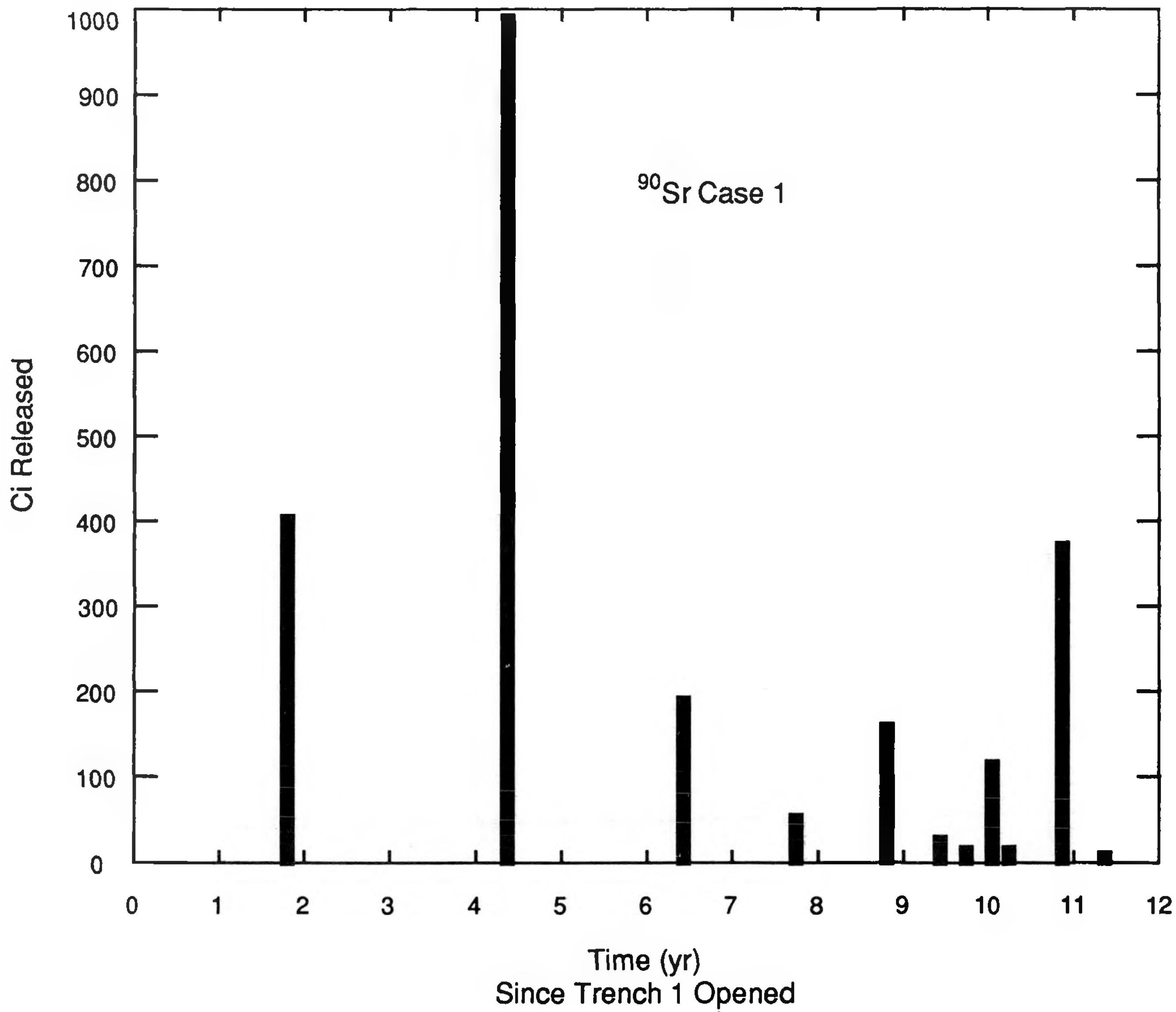

^{137}Cs Case 1
Ci Released
2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200
0
0
1
2
3
4
5
6
7
8
9
10
11
12
Time (yr)
Since Trench 1 Opened

S8901095.2M



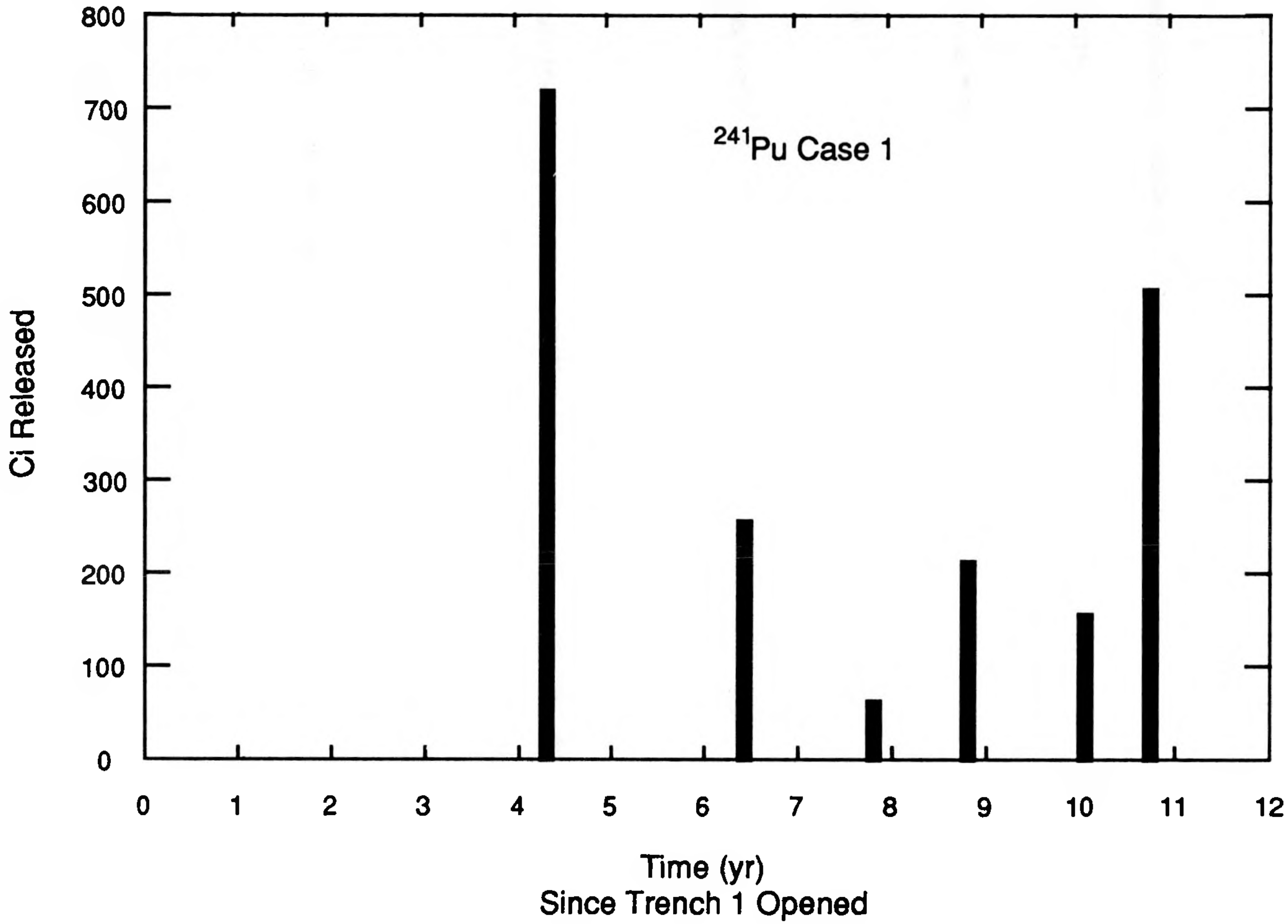
^{14}C Case 1
Ci Released
0
5
10
15
20
25
30
35
40
0
1
2
3
4
5
6
7
8
9
10
11
12
Time (yr)
Since Trench 1 Opened

S8901095.4M



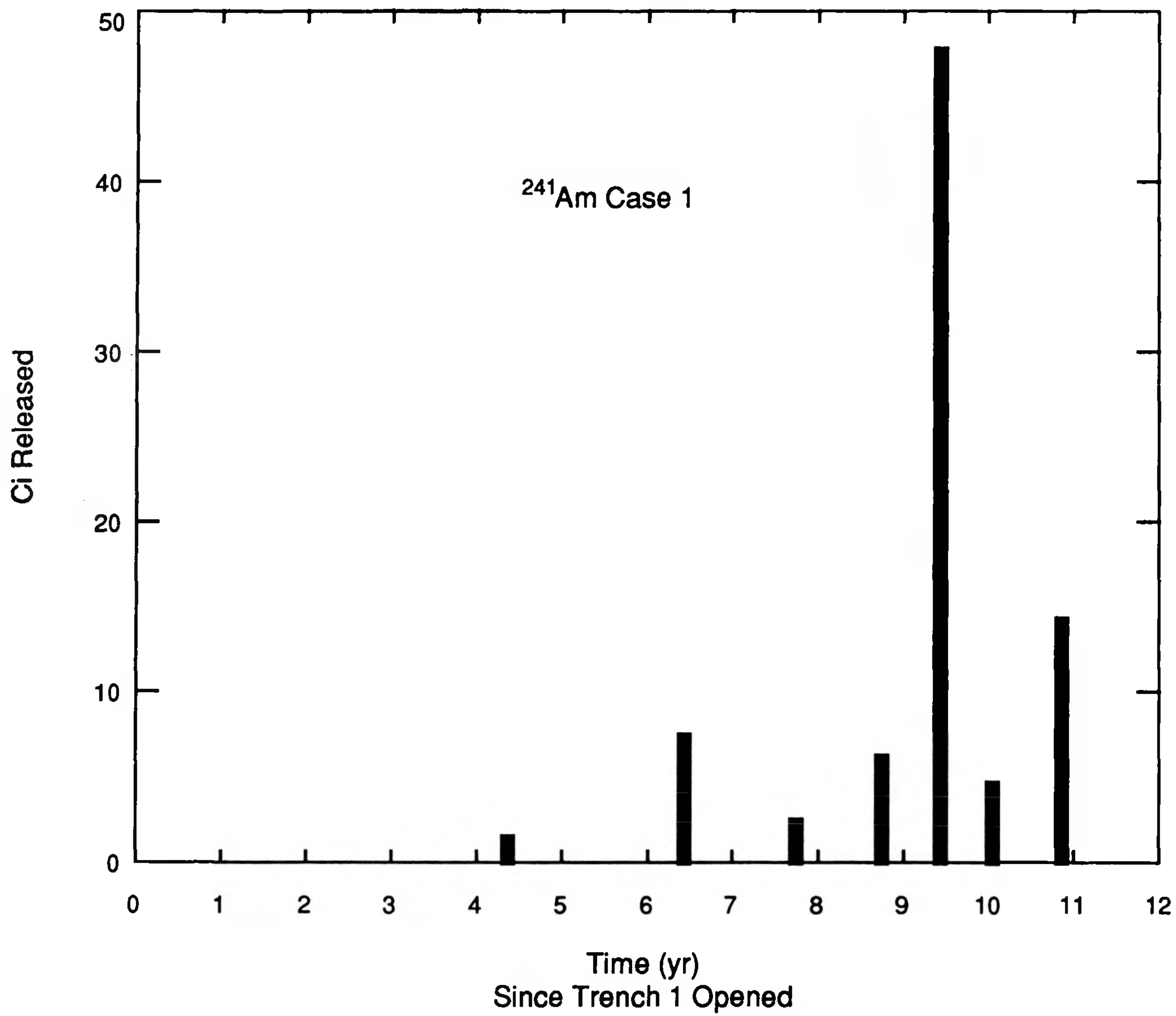
^{90}Sr Case 1
Ci Released
0
100
200
300
400
500
600
700
800
900
1000
0
1
2
3
4
5
6
7
8
9
10
11
12
Time (yr)
Since Trench 1 Opened

S8901095.5M



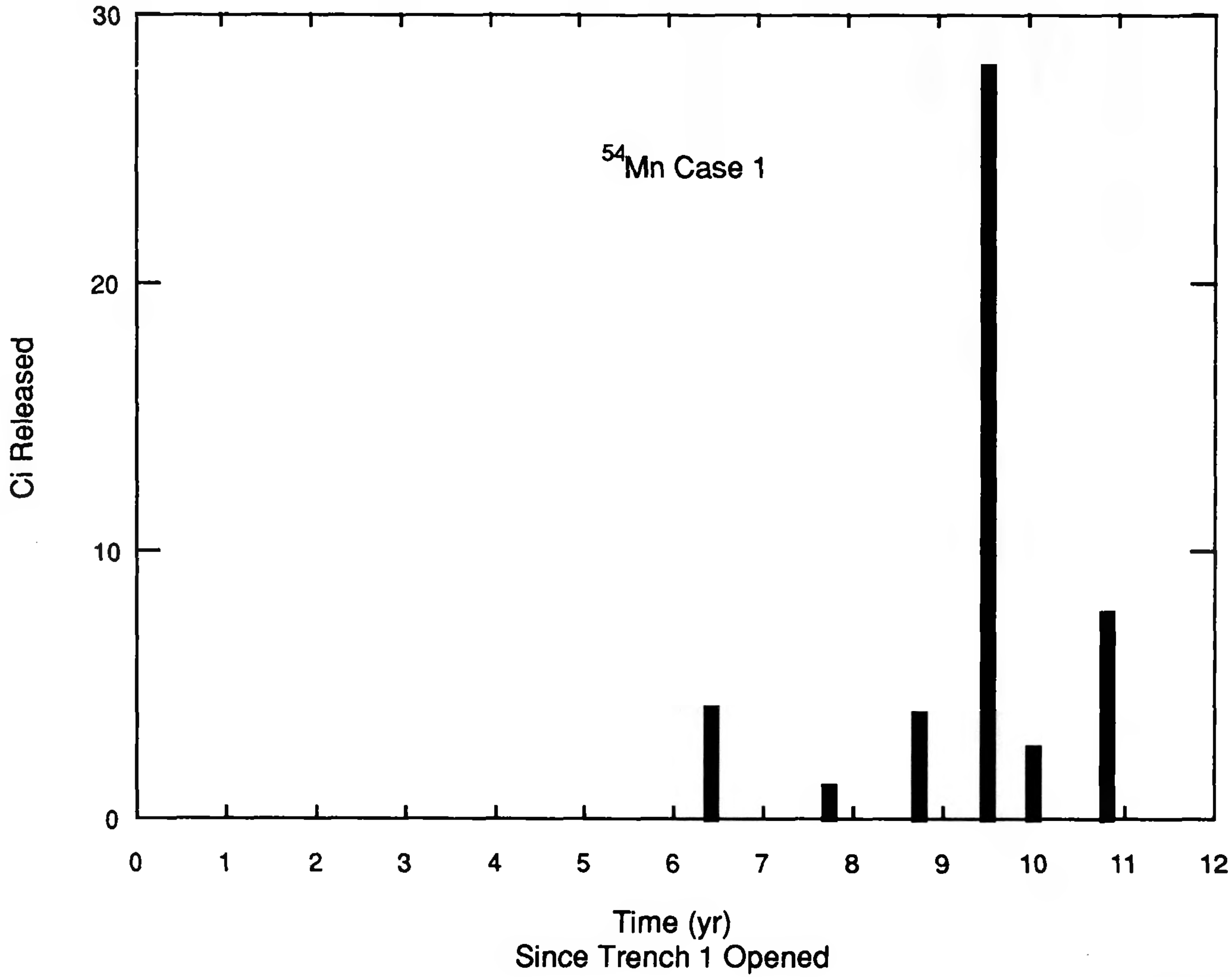
241Pu Case 1
Ci Released
0
100
200
300
400
500
600
700
800
0
1
2
3
4
5
6
7
8
9
10
11
12
Time (yr)
Since Trench 1 Opened

S8901095.7M



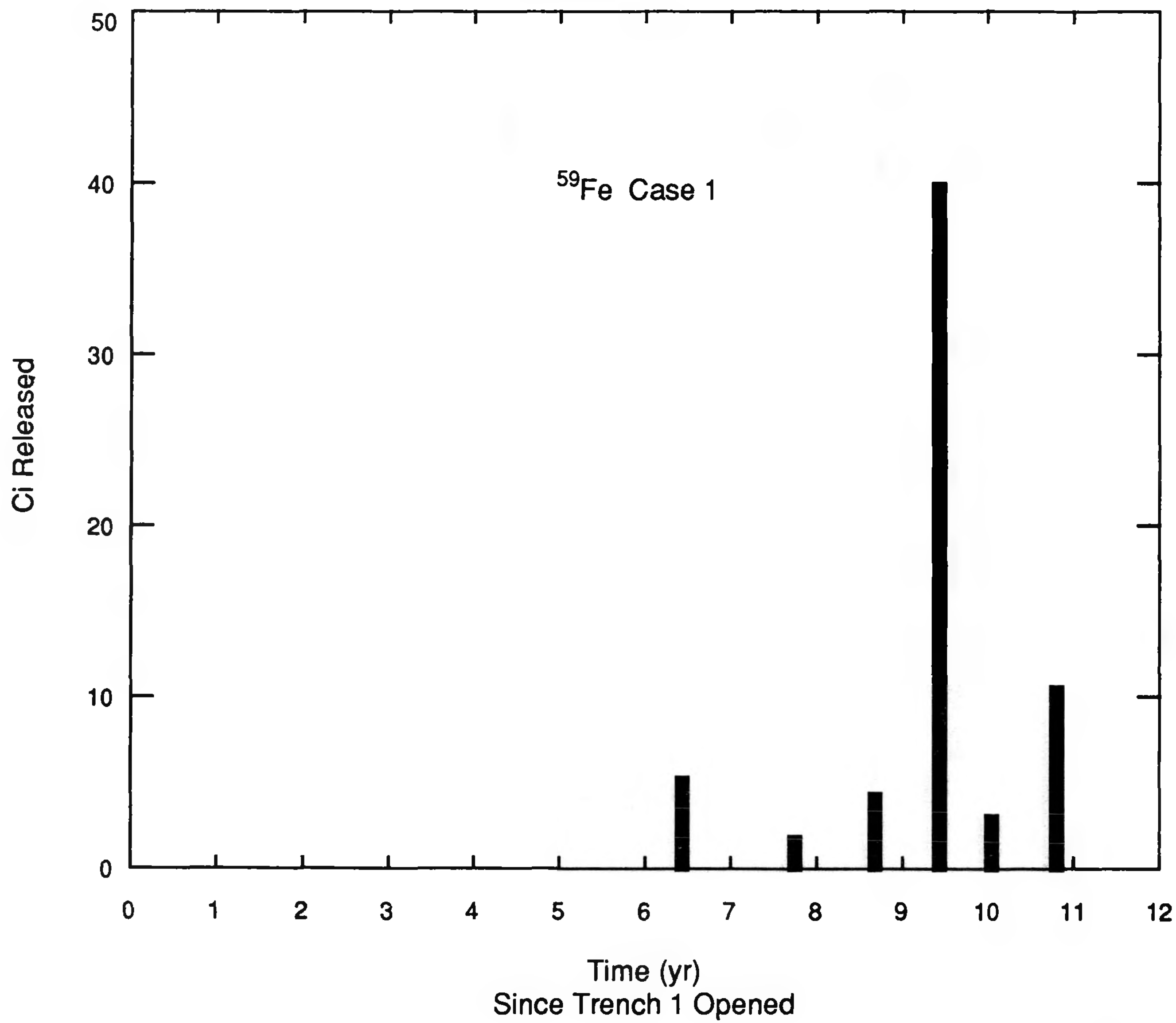
50
40
30
20
10
0
241Am Case 1
Ci Released
0
1
2
3
4
5
6
7
8
9
10
11
12
Time (yr)
Since Trench 1 Opened

S8901095.8M

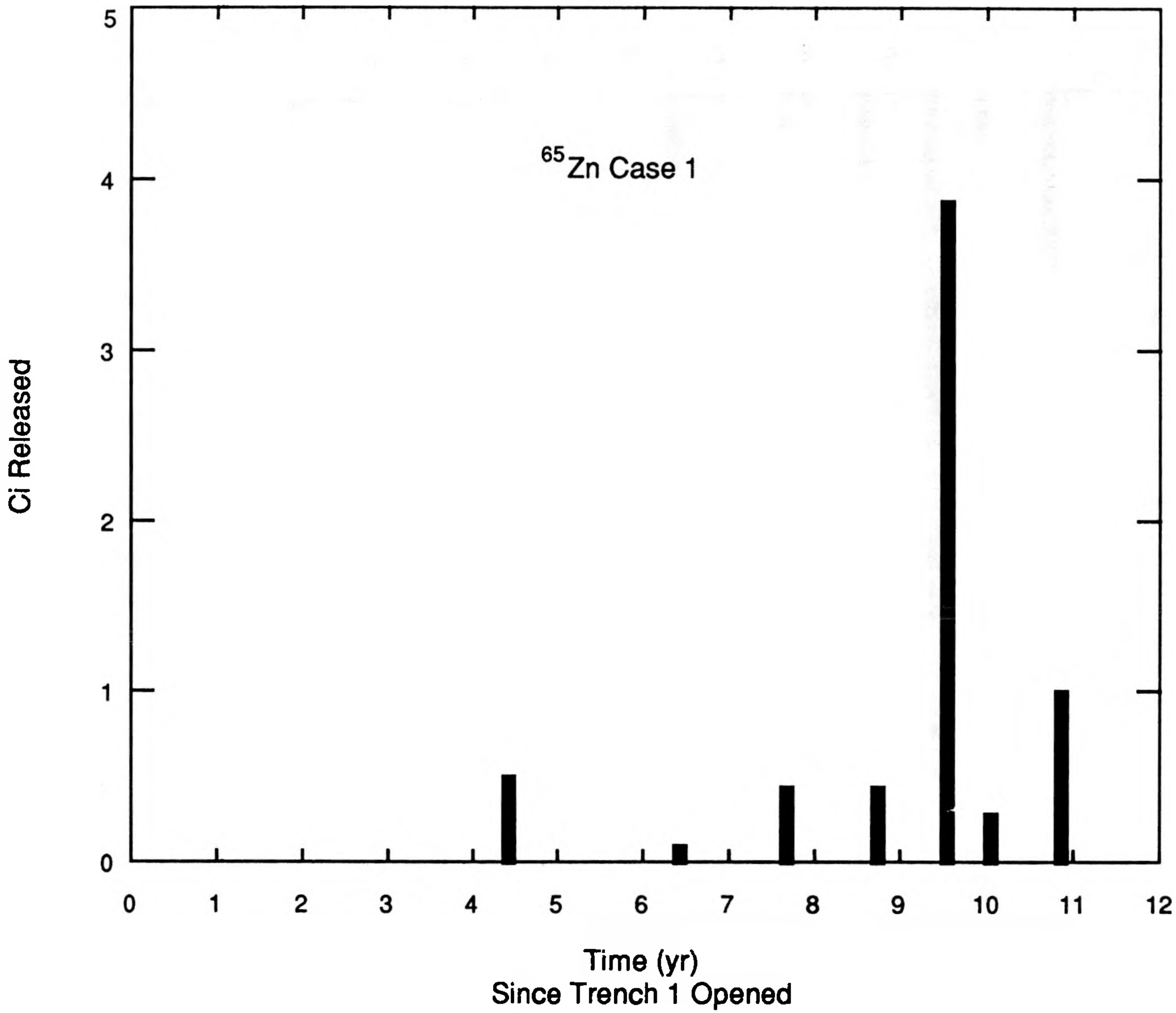


^{54}Mn Case 1
Ci Released
30
20
10
0
0
1
2
3
4
5
6
7
8
9
10
11
12
Time (yr)
Since Trench 1 Opened

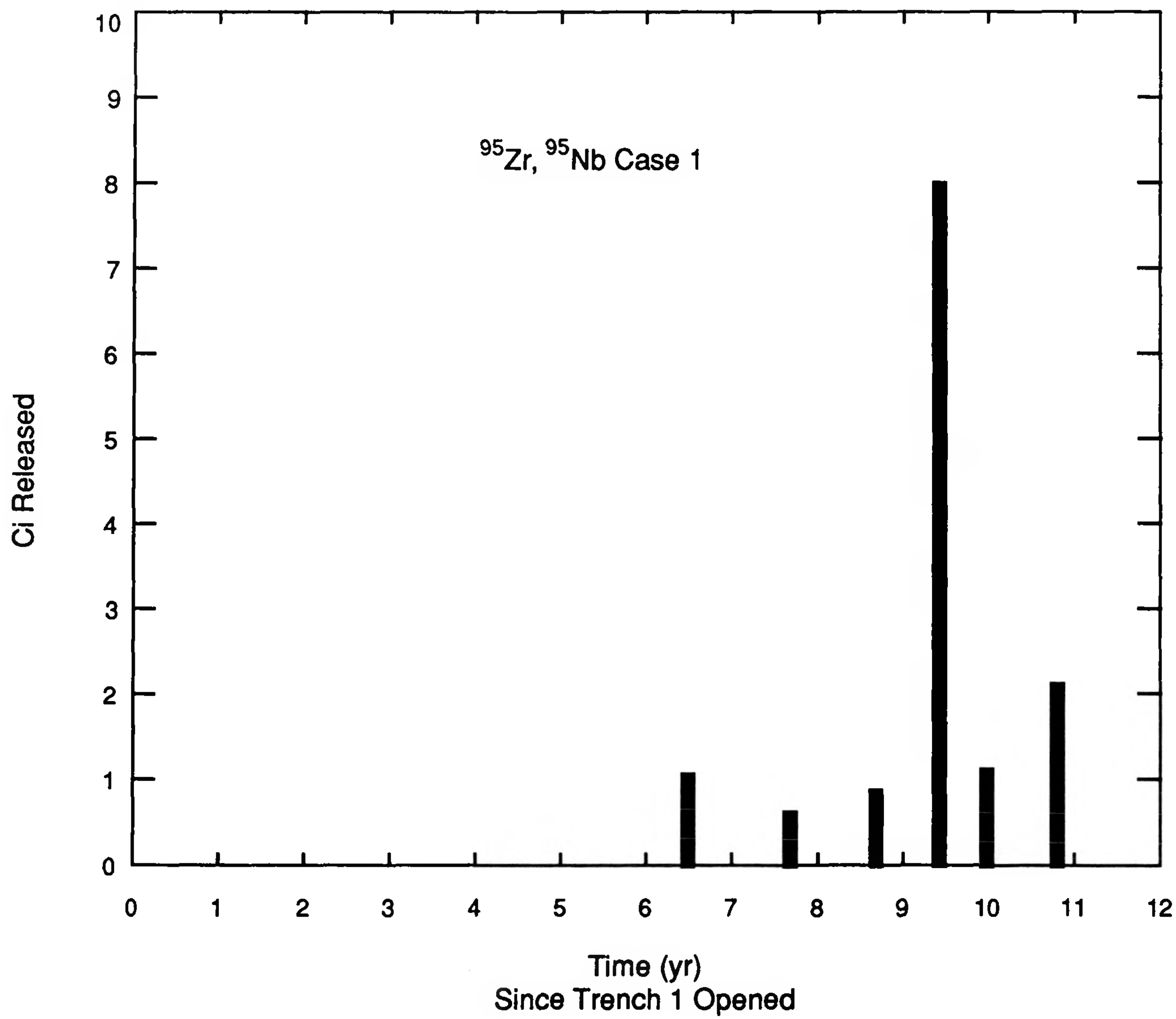
S8901095.9M



S8901095.10M



S8901095.11M



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10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*
A hydrogeologic performance assessment was conducted for the commercial low-level radioactive waste disposal site located about 3 mi southwest of the town of Sheffield, in Bureau County, northwestern Illinois. The site has 21 trenches, which contain about 900,000 m^3, of buried waste and about 60,000 Ci of nuclear by-product material. The disposal trenches cut through a complex series of Quaternary deposits, and are composed primarily of silts, clays, and sands. Ground water beneath the site, which ranges in depth from 1.5 to 14 m, generally moves in two directions: northeast to east toward a strip-mine lake and south to southeast toward small tributary channels belonging to Lawson Creek, which eventually drains into the strip-mine lake southeast of the site.

The results in the performance assessment, which focused on the site ground-water pathway, suggest that tritium, ^{90}Sr, and ^{14}C would be the only radionuclides released from the Sheffield site in any significant concentrations. A comparison of simulated tritium concentrations east of the site in the time frame of the burial history would suggest that model results are greater than the highest measured values by a factor of 2 or 3. The discrepancy between actual and predicted concentrations likely reflects errors in the assumed tritium inventory estimates, availability in the inventory, and/or the actual release from the multitude of waste forms considered in the performance assessment. A comparison of transport results for ^{90}Sr and ^{14}C is not possible since neither has been detected in ground water near the site.

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