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**Melton Valley Liquid Low-Level
Radioactive Waste Storage Tanks
Evaluation**

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**Melton Valley Liquid Low-Level Radioactive
Waste Storage Tanks Evaluation**

June 1995

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ACRONYMS

DNFSB	Defense Nuclear Facilities Safety Board
EDE	effective dose equivalent
EPA	U.S. Environmental Protection Agency
LLLW	liquid low-level radioactive waste
MVST	Melton Valley Liquid Low-Level Radioactive Waste Storage Tanks
ORNL	Oak Ridge National Laboratory
ORR	Oak Ridge Reservation
TDEC	Tennessee Department of Environment and Conservation
WOCC	Waste Operations Control Center

EXECUTIVE SUMMARY

The Melton Valley Liquid Low-Level Radioactive Waste Storage Tanks (MVSTs) store the evaporator concentrates from the Liquid Low-Level Radioactive Waste (LLLW) System at the Oak Ridge National Laboratory (ORNL). The eight stainless steel tanks contain approximately 375,000 gallons of liquid and sludge waste. These are some of the newer, better-designed tanks in the LLLW System. They have been evaluated and found by the U.S. Environmental Protection Agency (EPA) and the Tennessee Department of Environment and Conservation to comply with all Federal Facility Agreement requirements for double containment. The operations and maintenance aspects of the tanks were also reviewed by the Defense Nuclear Facilities Safety Board (DNFSB) in September 1994.

This document also contains an assessment of the risk to the public and ORNL workers from a leak in one of the MVSTs. Two primary scenarios were investigated: (1) exposure of the public to radiation from drinking Clinch River water contaminated by leaked LLLW, and (2) exposure of on-site workers to radiation by inhaling air contaminated by leaked LLLW.

In the evaluation of radiation exposure from a postulated tank rupture and leak through the vault liner, models and techniques generally accepted by the regulators, U.S. Department of Energy (DOE), and industry were used. These models were used to estimate (1) the likelihood that a leak would occur, and (2) the radiation dose that could result from human exposure to the leaked material. Sensitivity and uncertainty analyses were used to estimate distributions of values and confidence levels for unknown quantities.

The estimated frequency of a leak from one of the MVSTs is about 8×10^{-4} events per year, or about once in 1200 years (with a 95% confidence level). If a leak were to occur, the dose to a worker from inhalation would be about 2.3×10^{-1} mrem (with a 95% confidence level). The dose to a member of the public through the drinking water pathway is estimated to be about 7×10^{-1} mrem (with a 95% confidence level). By comparison with EPA Safe Drinking Water regulations, the allowable lifetime radiation dose is about 300 mrem. Thus, a postulated LLLW leak from the MVSTs would not add appreciably to an individual's lifetime radiation dose.

In conclusion, this evaluation suggests the following:

- the likelihood of a leak from an MVST and through its vault liner is very small,
- the health effects of a postulated leak would not be significant, and
- the risk presented to the public and workers from the MVSTs is very low.

1. INTRODUCTION

1.1 PURPOSE

A 1992 screening-level assessment (Dyer March 1992) of the likelihood of failure of a Melton Valley Liquid Low-Level Radioactive Waste Storage Tank (MVST) gave rise to a concern that the probability of premature tank failure is high. The purpose of this report is to summarize the status of the MVSTs as reflected in recent reviews by the U.S. Environmental Protection Agency (EPA), the Tennessee Department of Environment and Conservation (TDEC) and the Defense Nuclear Facilities Safety Board (DNFSB) and to reevaluate the risk to human health and the environment from the tanks by using more realistic assumptions than those used in the screening-level assessments.

1.2 SCOPE

The MVSTs are eight stainless steel, doubly contained tanks designed to store evaporator concentrate from the Liquid Low-Level Waste (LLLW) System. The tanks, which are designated W-24 through W-31, each have a 50,000-gal nominal capacity. The tanks, their secondary containment vaults, and auxiliary systems associated with the tanks and vaults are addressed in this report.

1.3 BACKGROUND

Various facilities at Oak Ridge National Laboratory (ORNL) generate low-level radioactive waste as a byproduct of operating processes. This waste is collected, treated, and concentrated to reduce its volume in the LLLW System. The concentrate from the LLLW evaporator is being stored in the MVSTs until a DOE facility that can accept the waste becomes operational. The volume of waste in the MVSTs varies due to Laboratory operations and normal evaporation from the tanks. In addition, a series of volume reduction and solidification campaigns have been conducted since 1989 to provide enough free-board in the tanks to allow continued operation of the LLLW system. The current volume is approximately 375,000 gal, which includes some transuranic constituents. The waste in the MVSTs represents a majority of the curie content in the active LLLW System. Because of the importance of the MVSTs to the continued operation of ORNL and the significant hazard represented by the tank contents, the design, operation, and current status of these tanks has been thoroughly reviewed by several agencies, including the EPA, the TDEC, and the DNFSB. The results of these reviews together with an assessment of the potential risk to human health and the environment in the event of a tank rupture and subsequent leak of its contents to the environment are summarized in this report.

Among the previous studies of the MVSTs were a screening-level evaluation of the likelihood of a tank failure, *Probabilistic Risk Assessment of Transuranic Waste Storage* (Dyer March 1992) and a screening-level assessment of the health effects of a leak to the environment of the contents of one full tank, *Catastrophic Failure of Melton Valley Storage Tanks* (Chidambariah March 1992). These were screening-level studies that used highly conservative assumptions to estimate levels of concern for the tanks.

In this report, substantially different conclusions are documented regarding the likelihood of a tank failure than are documented in the Dyer report (1992). The primary reason for this difference

is that the data and assumptions used for this evaluation more closely reflect actual tank design and operating parameters. Estimates of public exposure to radionuclides from drinking water are also lower than predicted by Chidambariah (1992) for much the same reason: the earlier screening studies did not attempt to take into account such well-accepted mitigating factors as documented industrial pipe and tank failure data and the liquid leak rate through concrete.

1.4 DESIGN CONFIGURATION

The MVSTs, which became operational in 1980, are arranged in groups of four tanks in each of two stainless steel-lined, underground concrete vaults adjacent to the hydrofracture site in Melton Valley (Fig. 1.1). Constructed of 1/2-in. ASME SA-240, Type 304L austenitic stainless steel, each tank has a nominal capacity of 50,000 gal. Installed in a horizontal configuration, the tanks are 12 ft in diameter with an overall length of 61 ft 4 7/8 in. (Fig. 1.2). The tanks are of fully welded construction with ASME dished heads. Although the tanks operate at or slightly below atmospheric pressure (-1 in. wg), they are designed and stamped as ASME Section VIII pressure vessels with a design pressure and temperature of 15 psig and 150°F, respectively.

The vaults are constructed of reinforced concrete with the base slab set on a limestone rock formation. The vault floors are 20 ft 6 in. below finished grade. Each vault is lined with 304L stainless steel that extends 7 ft up the walls. The floor liner is constructed of 11-gauge material and the side walls are of 16-gauge material. The floor of each vault is sloped to channel any liquids to a 304L stainless steel-lined sump.

The 3 × 3 × 1 ft deep sump in each vault is equipped with a pneumatic level indicator that is set to alarm in the local Evaporator Service Tanks Control Room and the Waste Operations Control Center (Bldg. 3130) in the event that liquids are detected. The piping system can be aligned to empty each sump through the use of the LLLW transfer pumps. The vault design also includes a foundation drainage system to reduce infiltration and relieve hydraulic pressure on the vault walls.

The MVSTs as well as the vaults are equipped with off-gas systems that exhaust locally through HEPA filters. Although explosive materials are not used in the facilities that generate LLLW and are not found in the system, organic solvents and evolved gases may be present. The tank and the vault ventilation systems are designed to prevent the accumulation of any vapors.

1.5 REPORT ORGANIZATION

Historical evaluations of the MVSTs are summarized in Sect. 2. The reevaluation of the risks associated in the MVSTs is documented in Sect. 3. References are in Sect. 4, and detailed discussions of the frequency and exposure assessments are presented in the appendixes.

Fig. 1.1. MVST location in relation to White Oak Creek.

Fig. 1.2. Plan view of MVST and vault configuration.

2. HISTORICAL EVALUATIONS

2.1 FEDERAL FACILITY AGREEMENT

The secondary containment system for the MVSTs was assessed in accordance with the requirements of the Federal Facility Agreement (FFA) for the Oak Ridge Reservation. Appendix F, Section C, of the FFA contains specific requirements for tank-vault containment systems. These requirements address materials of construction, structural capabilities, leak detection system design and operation, liquid removal capabilities, type of containment, vault capacity, infiltration control, chemical-resistant waterstops, compatibility of coatings and liners with the contained material, protection against the formation and ignition of vapors within the vault, and protection against leakage of moisture.

A report, *Design Demonstrations—Category B LLLW Tank Systems*, was prepared in 1992 (DOE/OR/01-1047). The conclusion presented in that document is that the tank-vault containment system meets all requirements of the FFA for double containment:

- The tank and the vault liner are constructed of materials that are appropriate for the contained waste.
- Structurally, the vault foundation was found to be properly designed for its intended service.
- The sloping vault floors that channel liquids to the instrumented sump meet the FFA requirement for a leak detection system.
- The Waste Operations Control Center is continuously manned, and the operating procedures for responding to a sump level alarm from the MVSTs comply with FFA requirements.
- The vault sump can be emptied by using the main tank transfer pumps. Any liquid in the vault could be removed in 24 h, as required by the FFA.
- The gross volume of the lined portion of each vault is approximately 151,000 gal. This is far more than the volume of the largest contained tank and thus exceeds FFA requirements.
- Site and storm drainage design around the perimeter of the vault prevents surface-water buildup during heavy rains.
- An external asphalt membrane prevents infiltration into the vault walls. The walls also incorporate chemically resistant PVC water stops and keyways in all below-grade construction joints, as specified by the FFA.

Other important conclusions resulting from the FFA report are that the 400-cfm tank ventilation system and the 1000-cfm vault ventilation system are sufficient to meet FFA requirements for control of explosive vapors and that ancillary equipment, including piping, is properly designed for leak detection and secondary containment.

The design and operation of the MVST containment system was approved by TDEC and EPA in November 1993 and April 1994, respectively.

2.2 DEFENSE NUCLEAR FACILITIES SAFETY BOARD

The DNFSB visited Oak Ridge Reservation in September 1994 to review the operations and maintenance programs for tanks at ORNL and K-25 that store radioactive material. The DNFSB focused on the risk to health and the environment posed by the contents of these tanks. The ORNL portion of the review dealt primarily with the LLLW system. Because the MVSTs contain the majority of curies in the active LLLW system, the design and operation of these tanks were reviewed in detail. Areas of review included the sources of the waste in the MVSTs, seismic status of the tanks, materials of construction, and system operation.

The DNFSB was particularly interested in the specific processes that generate the waste as well as the details regarding chemicals discharged into the LLLW tanks throughout their history. An extensive chronology of reprocessing and TRU experience for each process facility at ORNL was presented to the DNFSB. In addition, a listing of organic chemicals used in reprocessing operations was presented. The DNFSB questioned how the materials discharged into the LLLW system over the years corresponded to the waste acceptance criteria for the system. The waste acceptance criteria have been modified over the years. Originally, the criteria addressed only the radiological content of the waste. Later versions of the criteria were modified to include limits for organic chemicals and toxic materials.

The MVSTs meet the seismic and structural requirements of PC-3/1020 (Beavers 1994). The seismic design basis for the tanks is 0.15 g, horizontal and 0.10 g vertical. As reported to the DNFSB (Beavers 1994), the tanks meet DOE Order 5480.23 criteria for moderate hazard facilities, and the seismic design is consistent with the applicable requirements for that hazard level. This determination is still valid under the latest DOE guidelines. Under DOE STD-1020-94, the MVSTs meet the criteria for Hazard Category 2. The performance goal for Category 2 facilities is 10^{-5} , which requires a horizontal seismic design basis of 0.15 g.

As reported to the DNFSB (Pawel 1994) the MVSTs are constructed of 304L stainless steel, which was selected because of its inherent resistance to the LLLW composition, its ease of field fabrication, and its resistance to carbon precipitation and resulting intergranular stress corrosion. Since 1952, the LLLW streams have been controlled by the addition of caustic so that the pH remains ≥ 8 . Thus, the pH has been controlled during the entire service life of the MVSTs. This is important because the corrosion rate due to the tank contents in a basic environment such as this is virtually zero. Because the tanks are in vaults, corrosion due to galvanic interaction with acidic soil is not a factor. Considering the design and operation of the MVSTs and their secondary containment systems and the corrosion mechanisms that can affect stainless steel, ORNL reported to the DNFSB (Pawel 1994) that the design life of the tanks should conservatively be 50–60 years, and their effective operating life should well exceed that.

3. RISK ASSESSMENT

3.1 METHODOLOGY

In this section, the risk to workers and members of the public that could result from a leak of LLLW from an MVST is considered. Risks are expressed in terms of the likelihood that a leak could occur and the radiation dose that could result from exposure to the leaked material. Risk assessment techniques used to conduct this assessment are widely accepted in the nuclear and chemical industries.

The public and workers are considered separately because of their differing potentials for exposure. Exposure of the public resulting from a leak in an MVST would require a breach in the secondary containment system vault liner and the migration of waste material off the ORNL site. Worker exposure is assumed to occur as a result of a small fraction of volatile leaked material reaching the soil surface and becoming airborne. This could lead to exposure through inhalation. This is not a contact hazard because only volatile radionuclides will reach the surface.

Exposure from airborne contamination was initially considered for on-site worker receptors to determine whether on-site exposure to workers is of concern. If the on-site exposure is less than the off-site exposure limit, then off-site exposure will be of no concern. The analysis (see Appendix C) indicated that the risk of on-site exposure to airborne contamination was much lower than allowed in the EPA long-term emission standards for off-site exposure as cited in the *Code of Federal Regulations* (40 CFR 61, Subpart H); therefore, off-site airborne contamination, which would be even lower, was not considered.

A leak of LLLW from the rupture of an MVST would flow into a stainless steel-lined vault with a sump equipped with leak detection. The leak would be handled with existing procedures and transferring the leaked LLLW from the liner would present no exposure consequences to workers or the public. The frequency assessment determines the likelihood of a rupture from an MVST and a leak from the vault liner. The uncertainty associated with the size of the leak out of the vault and how much of the leak would be transported to the receptors is developed in the exposure assessment (see Appendix B).

It should be noted that a crack in the concrete vault structure is not necessary for LLLW to migrate through the concrete. A crack, however, could direct the postulated leaked LLLW and increase the transport rate (Castro et al. 1993, pp. 2-13 and 2-19). No credit is taken for adsorption of leaked LLLW in the concrete in its pathway to the surface water.

3.1.1 General Assumptions

The assumptions used in this risk assessment are as follows:

- A leak of concern from the rupture of an MVST into its vault must pass through the vault liner and the concrete floor into the aggregate beneath the vault (see Fig. 3.1). This assumption is basic to both the frequency and exposure assessments.

Fig. 3.1. Typical cross-section of Melton Valley Storage Tanks.

- The Clinch River downstream from the ORR is the first place that surface water or groundwater from ORNL is used for drinking water. Users of the Clinch River for drinking water are the off-site population group with the potential for exposure to the highest level of contaminants. (See Appendix B for additional discussion.)
- A release of concern in one of the MVSTs is a rupture, which is defined as the smallest hole considered to result in the postulated release of LLLW reaching the vault liner hole. Drip size leaks are not considered because the leaked LLLW will most likely evaporate before reaching the hole in the vault liner.
- The quantity of LLLW leaked from the vault is that which leaks for 24 h through the concrete vault floor. The vault floor is assumed to be fractured with cracks of width that are described by a lognormal distribution with a mean of 0.01 cm, a standard deviation of 0.001 cm, and a maximum of 0.05 mm. Crack spacing is assumed to be described by a lognormal distribution with a mean of 30 cm and standard deviation of 3 cm. Length and number of cracks assumed to be exposed to the postulated leaked LLLW are approximated by lognormal distributions equivalent to means of 90 cm and standard deviations of 9 cm. The lognormal distributions are used because cracks in reinforced concrete are expected to occur at a close spacing with very small widths. Values used in the lognormal distributions for crack spacing and widths are based on empirical studies (Walton and Seitz 1992). The values for the area exposed to leaks from the postulated liner hole are based on engineering judgment.
- Leaked material that reaches the aggregate under the vault concrete floor may migrate to surface water or upward to the soil surface. Most of the material that migrates will flow to the surface water via the foundation drainage system. A very small amount (see Appendix C) migrates to the soil surface as vapor.

3.1.2 General Approach

Frequency assessment. For the frequency assessment, an empirical model developed by H. M. Thomas (1981) was used. Thomas used a large data base of industry experience to develop his model for use by the nuclear industry to predict piping and vessel failures resulting from internal mechanisms. The model—which uses information on component size, shape, thickness, and construction details (such as the length of welds)—yields a predicted failure frequency for a pipe or typical storage vessel. The Thomas model has been used by the U.S. and foreign commercial nuclear power industries as well as by the chemical industry.

Uncertainty and sensitivity analyses were used to account for uncertainties in quantifying some factors in the Thomas model as well as other factors in the scenarios. Parameters for each variable that were known or could be estimated with some reasonable basis were fitted into one of several general distribution curves in a commercially available computer software package (Crystal Ball®).

Once distribution curves had been assigned to all applicable variables, a simulation, or series of iterations, was run. A commercially available spreadsheet (Excel®) was used for the calculations. Using the output from each of these iterations, the computer software produced a distribution of

results that estimated the number of outcomes and the probability of occurrence for each. For example, one factor in the Thomas model, a weld penalty factor, accounts for the fact that leaks occur more often in welds than in the base metal of pipes or vessels. Industry literature (see Appendix A) contains a distribution of weld penalty factors that includes the probability of occurrence for each. Rather than selecting a single weld penalty factor, a distribution curve for the range of factors that appears in the literature was used in this evaluation.

A simulation was then run in which the Thomas model equation was solved 10,000 times. During each run through the equation, the computer selected a value for the weld penalty factor from the distribution. Over the course of many runs, the weld penalty factors selected by the computer approached the distribution curve for the factor. This same technique was used for several uncertain quantities as discussed in the appendixes. The results of the simulation are shown as a distribution of frequency vs probability and, to give the reader a sense of the data spread, the results distribution is also shown as a function of confidence level.

For this frequency assessment, contributions from seismic events were not considered. The MVSTs are seismically qualified to the requirements of UCL 15910 (Kincaid, Oak Ridge National Laboratory August 4, 1992).

Assessment of public exposure. The techniques for assessing public exposure are based on the guidance and methods in the EPA report *Limiting Values of Radionuclides in Air, Water and Soil* (Eckerman et al. September 1988) and the ORNL report *Performance Assessment for Continuing and Future Operations at Solid Waste Storage Area 6* (Martin Marietta Energy Systems, Inc. February 1994).

The concentrations and dose conversion factors for ^{90}Sr , ^{244}Cm , and ^{241}Am dominate the postulated public dose. Together, these three radionuclides account for approximately 99% of the postulated dose (Martin Marietta Energy Systems, Inc. February 1994).

An individual residing outside the boundary of ORNL is assumed to use contaminated water for domestic and recreational purposes. The following pathways are postulated:

- direct ingestion of contaminated drinking water,
- ingestion of fish from contaminated water,
- ingestion of milk and meat from cattle that drink contaminated water, and
- external exposure while swimming in contaminated water.

Exposure through ingestion of foodstuffs that have been irrigated with contaminated water is considered insignificant because rainfall is usually abundant in Oak Ridge and irrigation is not widely practiced. Furthermore, ingestion of foodstuffs that have been irrigated by contaminated water would result in lower doses than direct ingestion of radionuclides in drinking water. For most radionuclides, the dose from all pathways is determined almost entirely by the dose from the drinking water pathway alone. Because the public dose from swimming in contaminated water, ingesting fish, and ingesting milk and meat from cattle that drink contaminated water would be less than 4% of the total dose (Martin Marietta Energy Systems, Inc. February 1994, Tables G.7 and G.13), these pathways are not considered in this evaluation.

To assess the potential dose to persons whose drinking water source is the Clinch River, an appropriate comparative index must be selected. EPA has established general regulations for radionuclides, in terms of allowable doses, under the Clean Water Act (40 CFR 141.16), "Maximum Contaminant Levels for Beta Particle and Photon Radioactivity from Man-Made Radionuclides in Community Water Systems." The regulation limits the average annual concentration of beta and photon radioactivity from man-made radionuclides in drinking water to that which will not produce an annual dose equivalent to the total body or any internal organ greater than 4 mrem/year. EPA has not developed concentration limits for every radioisotope. However, if an effective dose equivalent of 4 mrem/year from all radionuclides is assumed, established dose conversion factors (Eckerman et al. September 1988) can be used to determine the dose for the isotopes of interest.

DOE Order 5400.5 requires that liquid effluents from activities on the Oak Ridge Reservation (ORR) shall not cause levels of radioactivity in private or public drinking water systems downstream of facilities to exceed limits specified in 40 CFR 141. Under the current conditions at ORNL, surface water is not a source for the public water supply until it reaches the Clinch River. Surface water that leaves ORNL, as well as groundwater, is continuously monitored for radioactive contamination. The document titled *Federal Facilities Agreement Plans and Schedules for Liquid Low-Level Radioactive Waste Tank Systems at Oak Ridge National Laboratory* states that "no significant public doses (concentrations are far below MCLs) have resulted from total annual releases from the Oak Ridge Y-12 Plant, the Oak Ridge K-25 Site, and ORNL combined" (Martin Marietta Energy Systems, Inc. March 1992). [This report interprets results presented in *Oak Ridge Reservation Environmental Report for 1991* (Martin Marietta Energy Systems, Inc. 1991)]

The assumptions used in this assessment of public exposure are as follows:

- Radionuclides from the postulated MVST failure followed by a leak in the vault liner migrate beyond the facility boundary by way of the surface water pathway.
- Off-site transport via atmospheric dispersion following suspension of radionuclides into the air is considered insignificant (see Appendix C for an assessment of exposure to radionuclides dispersed in the atmosphere).
- Leaking contaminants are directed via the MVST foundation drainage system to the Clinch River tributaries that drain the ORNL site. A hypothetical distribution of contaminant releases to the public water supply was used. The distribution accounts for the uncertainties in leak size and the amount of a leak from the tank failure that would actually reach the water supply. The complete exposure assessment for the public is presented in Appendix B.

Assessment of worker exposure. The techniques for assessing worker exposure are based on the guidance and methods presented in EPA's *Superfund Exposure Assessment Manual* (EPA 1988) and on EPA reports titled *Limiting Values of Radionuclides in Air, Water, and Soil* (Eckerman et al. 1988) and *External Exposure to Radionuclides in Air, Water, and Soil* (Eckerman and Ryman 1993).

A quantitative hazard screening assessment was performed through the use of the technique recommended by EPA (1988) to determine the emission rate to the atmosphere. The emission rate was then converted to a dose through the use of EPA's inhalation dose conversion factors (Eckerman et al. September 1988). Resuspension of particles was not considered because the vaporization

scenario would cause similar or greater exposure. The complete analysis of this scenario is presented in Appendix C.

Accidental releases during vault cleanup after the leak are another potential source of worker exposure. Evaluation of the likelihood and consequences of such an accidental release is outside the scope of this analysis. If vault cleanup is pursued, the likelihood and consequences associated with postulated accidents would be reviewed during a safety analysis of the vault cleanup task.

The assumption used in this assessment of worker exposure is that a leak occurs in one of the MVSTs, followed by a leak from the vault liner, and the migration of LLLW to the aggregate underneath the vault. A very small amount diffuses to the surface and vaporizes into the atmosphere, thus presenting an inhalation exposure to workers.

3.2 RESULTS

3.2.1 Risks Associated with Rupture of an MVST

3.2.1.1 Frequency Assessment

The distribution of forecasted tank rupture frequencies is presented in Fig. 3.2. The predicted frequency of a tank rupture is shown on the x-axis, and its probability of occurrence during the simulation is shown on the y-axis. Distributions for the scaling factor, weld factor, quality factor, number of tanks, and tank wall thickness were the assumptions used in the simulation.

Fig. 3.2. Frequency of MVST ruptures.

The forecasted frequency of a rupture in the MVSTs at the 95% confidence level is 8×10^{-4} events/year. A complete presentation of the results is in Appendix A.

3.2.1.2 Assessment of Public Exposure

Potential exposure of the public is presented as a distribution in Fig. 3.3. The only pathway of significance is the drinking water route. Distributions of the leak size, concentration reduction factor, flow rate of the Clinch River, and potential concentrations of radionuclides in the Melton Valley Storage Tanks were the assumptions used in the simulation to generate the distribution of postulated doses. The distribution of postulated doses to the public from a tank rupture at the 95% confidence level is 7×10^{-1} mrem effective dose equivalent (EDE). The allowable dose through drinking water, as established by EPA in the Safe Drinking Water Act, is 4 mrem/year EDE. The complete presentation of the results is in Appendix B.

3.2.1.3 Assessment of Worker Exposure

The postulated dose to a worker who may unknowingly be exposed to radionuclides from an underground leak of LLLW is presented in Fig. 3.4 as a distribution of values. The distribution for the concentration of the radionuclides in the tanks, the distance up through the ground that any vapor and radionuclides would travel, the exposed subsurface area of the leak, and the hours of worker exposure were the assumptions used in the simulation of results. The distribution of dose to workers at the 95% confidence level is 2×10^{-1} mrem. The complete presentation of the results is in Appendix C.

Fig. 3.3. Distribution of drinking water dose to the public.

Fig. 3.4. Distribution of inhalation dose to workers.

3.2.2 Conclusions

The conclusions of the risk assessment are as follows:

- the likelihood of a leak from an MVST and its vault liner is very small,
- the health effects of a postulated leak would not be significant, and
- the risk presented to the public and workers from the MVSTs is very low.

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Appendix A
FREQUENCY ASSESSMENT

Appendix A FREQUENCY ASSESSMENT

H.M. Thomas used a large data base of industrial data to develop an empirical model for piping and vessel failures resulting from internal mechanisms (Thomas 1981). The Thomas model was developed for the nuclear power industry and is also used in other industries to estimate the failure frequency for leakage, λ_L , when historical or plant maintenance data are lacking. For each vessel in question, the model uses known information on the material size, shape, thickness, and construction details (such as number of welds). The model allows the analyst to make judgements as to the range of values used in some of the factors so that inspection and maintenance activities can be accounted for, thereby enabling the analyst to estimate the failure likelihood on the basis of actual service failure statistics.

Some practitioners (Medhekar et al. April 1993) have modified the model to incorporate scaling factors for fatigue, corrosion, and inspection. The model has been used in estimating the frequency of pipe breaks in loss of coolant accidents for commercial boiling water reactors. The model has also been used in several applications recently, including the chemical industry, and has been found to provide results that are quite reasonable when compared with available industry data sources (Medhekar et al. April 1993).

The leak failure frequency is given by

$$\lambda_L = C \times Q_L \times F \times B \quad (1)$$

where

C	=	an empirical scaling factor,
Q_L	=	a shape and quality factor,
F	=	the equipment age factor,
B	=	the learning factor.

The assumption used in this calculation is that slow leaks will most likely evaporate before reaching the vault sump. Particles that become airborne from evaporation of these slow leaks will be captured in the vault HEPA filters. Therefore, only ruptures of the Melton Valley Liquid Low-Level Radioactive Storage Tanks (MVSTs) will be considered. The frequency of expected pipe and vessel ruptures used in the Thomas model is about 6% that of slow leaks (Medhekar et al. April 1993).

The shape and quality factor Q_L is defined as

$$Q_L = (Q_T + A \times Q_W) \times E \quad (2)$$

where

Q_p = the parent material geometry factor,
 A = the weld penalty factor,
 Q_w = the weld material factor,
 E = the quality factor.

Q_p is represented by the equation

$$Q_p = \frac{D \times L}{T^2} \quad (3)$$

where

D = 144 in., the outside diameter of the tanks,
 L = 684 in., the length of the tanks (not including the heads),
 T = 0.5 in., the wall thickness of the tanks.

By substituting Q_p in Eq. (2), the expression for the shape and quality factor can be rewritten as follows:

$$Q_L = \left[\frac{D \times L}{T^2} + (A \times Q_w) \right] \times E \quad (4)$$

The assumption used in this calculation is that wall thickness ranges from 0.45 to 0.50. This variation is based on ASTM tables for type 304L stainless steel plate tolerances. No allowance for corrosion was used because of the material of construction used in the tanks (type 304L stainless steel) and the conditions under which the tanks have been operated (that is, caustic solutions were added to adjust the waste solutions pH from 9.5 to 13.9) (Oak Ridge National Laboratory March 10, 1994).

The weld material factor equation for the tanks is as follows:

$$Q_w = \frac{1.75 \times NC \times D}{T} + \frac{1.75 \times NL \times L}{3.14 \times T} \quad (5)$$

where

NC = 2, the number of circumferential welds,
 NL = 1, the number of longitudinal welds.

The weld penalty factor, A , is an empirical coefficient that accounts for the fact that failures are more likely to occur in weld material than in parent material. The distribution of weld penalty factors is shown in Table A.1.

Table A.1. Distribution of weld penalty factor *A*

<i>A</i>	Probability
40	0.35
50	0.5
65	0.1
90	0.05

Source: Medhekar, S. R., et al. April 1993. "Prediction of Vessel and Piping Failure Rates in Chemical Process Plants Using the Thomas Model," *Process Safety Progress*, 12 2 , 124-125.

A quality factor, *E*, accounts for periodic testing or inspections. Because the tanks are not routinely tested or inspected but were initially tested and inspected as ASME Section VIII pressure vessels and have been operated as atmospheric storage tanks, the quality factor *E* used in the risk assessment is 0.1 (Medhekar et al. April 1993).

The scaling factor, *C*, is an empirical coefficient that converts the previously calculated factors to actual leakage frequencies.

Table A.2. Distribution of scaling factor *C*

<i>C</i>	Probability
1×10^{-9}	0.2
3.0×10^{-9}	0.25
1.0×10^{-8}	0.25
3.0×10^{-8}	0.3

Source: Medhekar, S. R., et al. April 1993. "Prediction of Vessel and Piping Failure Rates in Chemical Process Plants Using the Thomas Model," *Process Safety Progress*, 12 2 , 124-125.

The equipment age factor, *F*, accounts for the fact that older plants tend to have more failures than newer plants. The learning factor, *B*, accounts for the fact that new designs have more failures than proven designs. The product of the equipment age factor and the learning factor is 1.0 for equipment that has been in service for a number of years (Medhekar et al. 1993).

In this frequency assessment, a single tank was analyzed and the results were multiplied by the total number of tanks times 6%, the Thomas model correction for frequency of ruptures. Substituting Eqs. (2), (3), and (5) into Eq. (1) and then multiplying by 0.06 and by the number of tanks yields

$$\lambda_L = 8 \times 0.06 \times C \times F \times B \times E \times \left(\frac{D \times L}{T^2} + \frac{1.75 \times A \times NC \times D}{T} + \frac{1.75 \times A \times NL \times L}{3.14 \times T} \right) \cdot (6)$$

This equation was used in the Crystal Ball® software to develop the frequency of an MVST rupture, which is presented in the form of a distribution in Fig. A.1.

Fig. A.1. Frequency of MVST ruptures.

The range of frequencies and the confidence levels are shown in Table A.3.

Table A.3. Confidence levels of MVST ruptures

Percentile %	Events/year approx.
5	2×10^{-5}
25	7×10^{-5}
50	2×10^{-4}
75	7×10^{-4}
95	8×10^{-4}

The statistical results from the distribution of frequencies of a rupture are shown in Table A.4.

Table A.4. Statistical results of the distribution of an MVST rupture

Statistics	Value
Trials	10,000
Median approx.	2×10^{-4} /year
Mode approx.	3×10^{-5} /year
Mean	3×10^{-4} /year
Standard deviation	3×10^{-4} /year
Skewness	0.7
Kurtosis	1.8
Range minimum	2×10^{-5} /year
Range maximum	1×10^{-3} /year

The distribution of expected values from the Thomas model predictions are generally in agreement with the data contained in several widely used experiential data bases (see Table A.5). The range of values predicted by Thomas covers the range shown in Table A.5.

Table A.5. Selected vessel failure rate data

Failure mode	Data source				
	C & W ^a	AEC ^b	AICHE ^c	NRC ^d	OREDA ^e
Rupture	$1 \times 10^{-5}/\text{year}$	$10^{-5} - 10^{-6}/\text{year}$	$9.6 \times 10^{-5}/\text{year}$	$5.3 \times 10^{-7}/\text{year}$	$5.2 \times 10^{-3}/\text{year}$

^aSource: Cremer and Warner, Ltd. 1982. "Assessment of Industrial Risks in the Rijnmond Area, Final Report," in *Risk Analysis of Six Potentially Hazardous Industrial Objects in the Rijnmond Area: A Pilot Study*, D. Reidel Publishing Company, Dordrecht, Holland.

^bSource: AEC Advisory Committee on Reactor Safeguards July-August 1974. "Pressure-Vessel-Failure Statistics and Probabilities," *Nuclear Safety* 15-4.

^cSource: American Institute of Chemical Engineers, Center for Chemical Process Safety 1989. *Guidelines for Process Equipment Reliability Data with Data Tables*, New York.

^dSource: U.S. Nuclear Regulatory Commission October 1975. *Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400.

^eSource: OREDA Participants, 1984, *OREDA Offshore Reliability Data Handbook*, publisher OREDA Participants, distributor VERITEC and PennWell Publishing Company, Norway.

The unreliability of the vault liner was calculated to estimate the number of vault liner leaks. The estimate is based on the assumptions of the number of field welds made on the liner. The liners were assumed to be welded from 4-ft-wide, 11-gage 304L stainless steel plates 8 ft long. The sump was assumed to contain eight welds. The total number of welds was estimated at 143. The frequency for weld failures was assumed to be $3 \times 10^{-9}/\text{h}$ (U.S. Nuclear Regulatory Commission October 1975, Table III 2-1) and the vault liner was assumed to have a service life of at least 50 years. The unreliability per vault is estimated by the American Institute for Chemical Engineers, Center for Chemical Process Safety (1989, Table E.3, p. 523) as follows:

$$\bar{r} = 1 - e^{-\lambda t} \quad (6)$$

where

$$\begin{aligned} \lambda &= \text{frequency (1/year),} \\ t &= \text{service life (year)} \end{aligned}$$

Solving yields

$$\bar{r} = 1 - e^{-(3 \times 10^{-9})(143)(8760)(50)}$$

$$\bar{r} = 0.17$$

Therefore, the probability of a leak in each vault liner is 17%. The prediction of the size of the leak is documented in Appendix B.

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Appendix B
ASSESSMENT OF PUBLIC EXPOSURE

Appendix B

ASSESSMENT OF PUBLIC EXPOSURE

In this assessment of the public exposure to radiation from a postulated liquid low-level waste (LLLW) leak from the Melton Valley LLLW Storage Tanks (MVST), radionuclides are assumed to be transported beyond the facility boundary only by way of the surface water or groundwater pathways. Groundwater is not a pathway because the vaults have a foundation drainage system which provides a direct pathway of the postulated release of LLLW to a tributary of the Clinch River. Therefore, the only pathway of concern is surface water. Off-site transport by way of the atmospheric pathway following suspension into the air of radionuclides was found to be unimportant (see Appendix C for assessment of on-site exposure to radionuclides dispersed into the atmosphere). The assessment indicated that both on-site and off-site exposure via the atmospheric pathway would be inconsequential.

B.1 EXPOSURE PATHWAYS

An individual residing outside the boundary of Oak Ridge National Laboratory (ORNL) could be exposed by using contaminated water for domestic and recreational purposes. The following exposure pathways are possible:

- direct ingestion of contaminated drinking water,
- ingestion of fish from contaminated water,
- ingestion of milk and meat from cattle that drink contaminated water, and
- external exposure while swimming in contaminated water.

The use of contaminated water for irrigation by individuals off-site is not considered because rainfall is usually abundant in Oak Ridge and irrigation is not widely practiced. Furthermore, ingestion of foodstuffs that have been irrigated by contaminated water would result in lower doses than direct ingestion of radionuclides in drinking water. For all radionuclides, the dose from direct ingestion of drinking water is greater than the dose from all of the other exposure pathways; therefore, for most radionuclides, the dose from all pathways is determined almost entirely by the dose from the drinking water pathway alone (Martin Marietta Energy Systems, Inc. February 1994, pp. 4-30).

If a leak were to occur, the sensitivity analysis shows that three radionuclides, ^{90}Sr , ^{244}Cm , and ^{241}Am , would provide approximately 99% of the total dose to the public. The performance assessment for Solid Waste Storage Area 6 shows that over 96% of the total dose is via the drinking water pathway when compared with swimming and ingestion of milk and meat (Martin Marietta Energy Systems, Inc. February 1994, Tables G.7 and G.13). Therefore, inputs for the public dose caused by swimming in contaminated water, ingesting fish from contaminated water, and by ingesting milk and meat from cattle that drink contaminated water will be ignored.

B.2 COMPARATIVE INDEX FOR USE IN ASSESSING POTENTIAL DOSE VIA DRINKING WATER

To assess the potential radionuclide dose to persons whose drinking water source is the Clinch River, an appropriate comparative index must be determined. Under the Clean Water Act, the U.S. Environmental Protection Agency (EPA) has established general regulations for radionuclides in terms of allowable doses. EPA has not developed concentration limits for every radionuclide. However, given EPA's 4 mrem/year long-term dose limit and factors for converting from exposures to doses, the doses from the radionuclides of concern can be calculated. To convert from intakes to doses, EPA has adopted the dose conversion factors (DCFs) for individual radionuclides recommended by the International Commission of Radiological Protection and the National Council on Radiation Protection. Derived air concentrations and annual limits on intake for workers have also been developed (Eckerman et al. September 1988). EPA regulations allow a dose limit of 4 mrem/year effective dose equivalent (EDE) for all radionuclides (including alpha emitters).

DOE Order 5400.5 requires that liquid effluents from activities on the ORR shall not cause levels of radioactivity in private or public drinking water systems downstream of facilities to exceed limits specified in EPA's *National Primary Drinking Water Regulations*, 40 CFR 141. Under the current conditions at ORNL, surface water is not a public water supply source before reaching the Clinch River, where the effluents are greatly diluted before any use of the water for the public water supply. The document titled *Federal Facilities Agreement Plans and Schedules for Liquid Low-Level Radioactive Waste Tank Systems at Oak Ridge National Laboratory* states that "Doses to the public have been monitored and assessed in the past and no significant public doses (concentrations are far below MCLs) have resulted from annual releases from the Y-12 Plant, the K-25 Site, and ORNL" (Martin Marietta Energy Systems, Inc. March 1992, page 4-2). [This report interprets results presented in *Oak Ridge Reservation Environmental Report for 1991* (Martin Marietta Energy Systems, Inc. 1991)].

B.3 ASSUMPTIONS

To assess the public exposure to a postulated LLLW leak, the following assumptions were made:

- Surface water is the only pathway of concern.
- The quantity of LLLW is that which leaks through a vault liner hole to the concrete vault slab. The quantity of LLLW postulated to leak from one of the MVSTs and subsequently through the vault liner and the vault is that which leaks through the vault concrete floor for 24 h because standard operating procedures for operation of the MVST facility require that detected liquids in the vault sumps be removed within that time. The vault floor is assumed to be fractured with cracks of width that are described by a lognormal distribution with a mean of 0.01 cm, a standard deviation of 0.001 cm, and a maximum of 0.05 mm. Crack spacing is assumed to be described by a lognormal distribution with a mean of 30 cm and standard deviation of 3 cm. Cracks are an expected part of any massive concrete structure or slab. Causes of cracking include swelling and shrinkage related to temperature changes from heat of hydration, drying shrinkage, settling, and

physical loading. Crack width and spacing are controlled by proper placement of steel reinforcing. Length and number of cracks assumed to be exposed to the postulated leaked LLLW are estimated by lognormal distributions with means of 90 cm and standard deviations of 9 cm. This is the area assumed to be exposed to the leaked LLLW through the hole in the liner and is based on engineering judgment. The lognormal distributions and the assumed values in these distributions were based on empirical studies by Walton and Seitz (1992). On the basis of this study, cracks in reinforced concrete are expected to occur at a close spacing with very small widths.

- The radionuclides are available in drinking water for 1 d. This assumption is made for computational convenience. Shorter release times would be offset by shorter use periods for potential consumers. Conversely, longer release times would be offset by longer use periods. As long as the release time is no longer than 1 year, the annual dose calculation is unaffected.
- Postulated leaked LLLW radionuclides are assumed to be transported to a tributary of the Clinch River by the MVST foundation drainage system. No adsorption of the radionuclides is assumed.
- The radionuclides mix uniformly by the time the plume reaches the Clinch River. Surface-water flow conditions are as follows: average (White Oak Creek flow = 0.03×10^9 L/d, Clinch River flow = 11×10^9 L/d), maximum (White Oak Creek flow = 0.08×10^9 L/d, Clinch River flow = 24×10^9 L/d), and minimum (White Oak Creek flow = 0.02×10^9 L/d, Clinch River flow = 1.7×10^9 L/d). The Tennessee River flow was taken to be its 81-year average at the Chickamauga Dam, 66×10^9 L/d (Martin Marietta Energy Systems, Inc. March 1992, p. 4-5). A triangular distribution consisting of the maximum, minimum, and average values for the Clinch River was used in the analysis.
- Each person drinks 2.0 L of water per day. This is a standard reference intake used in a variety of guidelines, e.g., EPA regulation 40 CFR 141.16.
- A uniform distribution was used for the concentration of radionuclides. The minimum activity is that of the MVST with the lowest activity levels, W-24, and the maximum activity is that of the MVST with the highest activity levels, W-25. The range of activities for each radionuclide is listed in Table B.1.

The committed effective dose equivalent (EDE) or the organ dose equivalent is calculated as follows:

$$DE_i = \frac{Q \times C_i \times DCF_i \times I \times CF}{F_{CR} \times CRF} \quad (1)$$

where

DE_i	=	the organ or effective dose equivalent of the radionuclide (mrem),
Q	=	the quantity released in the leak (L),
C_i	=	the activity concentration of the radionuclide in the mixture that is released ($\mu\text{Ci/L}$),

DCF_i	=	the dose conversion factor for the radionuclide (Eckerman et al. 1988, Table 2.2) (Sv/Bq),
I	=	the daily water intake for a person (2 L),
CRF	=	the concentration reduction factor for environmental transport of the radionuclides through clay and concrete to surface water,
CF	=	the conversion factor to convert DCF, which is in Sv/Bq, to mrem/ μ Ci, 3.7×10^9 mrem-Bq/ μ Ci-Sv,
F_{CR}	=	the flow of the Clinch River.

Table B.1. Distribution of radionuclide activities

Radionuclide	Activity μ Ci/L	
	Minimum	Maximum
^{241}Am	0.16	100
^{14}C	1.0	8.9
^{244}Cm	15	1100
^{60}Co	178	1590
^{134}Cs	1.5	48
^{137}Cs	1680	6400
^{152}Eu	132	2240
^{154}Eu	13	1270
^{155}Eu	11	599
^3H	14	21
^{238}Pu	2.3	10
$^{239/240}\text{Pu}$	1.4	6.9
^{106}Ru	3	98
^{90}Sr	657	55,700
^{233}U	1	120
^{95}Zr	41	629
^{144}Ce	2×10^{-5}	35

Source: Oak Ridge National Laboratory March 10, 1994. LLLW Data Base.

The quantity released in the leak can be calculated by the following formula for flow through fractured concrete (Walton and Seitz 1992)

$$K = \frac{\rho g b^3}{24\mu s} \quad (2)$$

where

- K = hydraulic conductivity (cm/s),
- ρ = density of liquid (gm/cm³),
- g = acceleration of gravity (cm/s²),
- b = crack width (cm),
- μ = viscosity (g/cm-s)
- s = crack spacing (cm).

Average crack spacing can be estimated by the following formula (Walton and Seitz 1992)

$$S_{avg} = 2t \quad (3)$$

where

- t = thickness of concrete over steel reinforcement

The flow rate of LLLW through the concrete was calculated by using the following assumptions

- The concrete crack width, b , is described by a lognormal distribution with a mean of 0.01 cm and standard deviation of 0.001 cm. No cracks exceed a maximum of 0.05 cm.
- Concrete crack length exposed to LLLW under the liner hole is described by a lognormal distribution with a mean of 90 cm and standard deviation of 9 cm.
- Concrete crack spacing is described by a lognormal distribution with a mean of 30 cm and standard deviation of 3 cm.
- The number of cracks exposed to leaked LLLW is described by a lognormal distribution with a mean of 3 and standard deviation of 0.3.

Crystal Ball® software was used to calculate the leak size and substituted in Eq. (1) to calculate the distribution of public dose.

B.4 RESULTS OF EXPOSURE ASSESSMENT

If this exposure is assumed to occur once in an average individual's lifetime of 70 years, the individual could receive a median EDE of 8×10^{-3} mrem from the leak without approaching the allowable lifetime dose from the Safe Drinking Water regulations (2.8×10^2 mrem).

The dose to the public is presented as a distribution. The distributions for the leak size, concentration reduction factor, flow rate of the Clinch River, and concentrations of radionuclides in

the MVST were the assumptions used in a simulation to calculate the drinking water dose distribution. The distribution of doses to the public is presented along with the confidence levels associated with these doses in Table B.2. The distribution of results associated with the calculation of these doses is shown graphically in Fig. B.1.

Fig. B.1. Distribution of drinking water dose to the public.

Table B.2. Confidence levels of forecasted values of the public exposure

Percentile %	mrem approx.
5	3×10^{-3}
25	17×10^{-3}
50	34×10^{-3}
75	69×10^{-3}
95	7×10^{-1}

Statistical results of the public exposure distribution are presented in Table B.3.

**Table B.3. Statistical results of the
forecast of public exposure**

Statistics	Value
Trials	10,000
Median approx.	3×10^{-2} mrem
Mode approx.	7×10^{-2} mrem
Mean	2×10^{-1} mrem
Standard deviation	6×10^{-1} mrem
Skewness	10.5
Kurtosis	162
Range minimum	7×10^{-7} mrem
Range maximum	1.5 mrem

The variance in the distribution of the public dose was most influenced by the distributions of the vault concrete crack width, Clinch River water flow, and concentration of ⁹⁰Sr. Other distributions that contributed in lesser amounts to the variance in the dose distribution were the distributions of the concentration of ²⁴⁴Cm, concrete crack length, and number of concrete cracks exposed to leaked LLLW.

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Appendix C

ASSESSMENT OF WORKER EXPOSURE

Appendix C

ASSESSMENT OF WORKER EXPOSURE

In the scenario evaluated for this exposure assessment, workers are exposed when a Melton Valley Storage Tank (MVST) ruptures and liquid low-level waste (LLLW) leaks through a hole in the vault liner and moves through the concrete to the soil underneath the building. A very small amount of the LLLW could migrate to the surface and diffuse into the atmosphere, thus presenting an inhalation hazard to workers. Therefore, the exposure assessment considered exposure of workers to airborne radionuclides. A quantitative hazard screening assessment was performed by using a technique recommended in the *Superfund Exposure Assessment Manual* (U.S. Environmental Protection Agency April 1988, pp. 16-8) for determining the emission rate of a material to the atmosphere through a soil cover.

C.1 ASSUMPTIONS

The assumptions used in estimating the inhalation exposure to workers are as follows:

- The pathway of exposure is by airborne contamination.
- LLLW has a vapor pressure similar to water at ambient conditions and thus is volatile.
- The radionuclides evaporate at the same rate as the water and exist in the water vapor at the same concentration as in the liquid leaked from the piping. This is a conservative assumption because the radionuclides in the water will evaporate at a rate much less than that of the water and exist in lower quantities in the vapor.
- The radionuclides that are present as cations in the LLLW will not migrate through the aggregate around the building at the same rate as the water because the much larger molecular weight of the salts as compared with the water will reduce the diffusion coefficient in the equations that follow (U.S. Environmental Protection Agency April 1988, p. 17).

C.2 METHODOLOGY

The emission rate is given by the following equation:

$$E = \frac{D \times C_p \times A \times P_i^{4/3}}{d_m} \quad (1)$$

where

E	=	the emission rate (g/s),
D	=	the diffusion coefficient of the material in air (cm^2/s),
C_{vp}	=	the saturation vapor concentration of the component in air (g/cm^3),
A	=	the exposed area (cm^2),
d_{sc}	=	the effective depth of the soil cover (cm),
P_i	=	the soil porosity.

Metric units are used in the source document (U.S. Environmental Protection Agency April 1988); therefore, metric units are used in this equation and, for consistency, throughout this appendix.

The assumptions are as follows:

- to be conservative, possible transport of some of the liquid away from the leak site by groundwater can be ignored; and
- the concentration of the contaminants is undepleted over time.

The diffusion coefficient, D , can be calculated by using Fuller's method (Perry and Chilton 1973, p. 3-233) as follows:

$$D_i = \frac{0.001 T^{1.75} \sqrt{\frac{1}{MW_i} + \frac{1}{MW_{air}}}}{P(V_i^{1/3} + V_{air}^{1/3})^2} \quad (2)$$

where

T	=	the absolute temperature (K),
$MW_i; MW_{air}$	=	the molecular weights of the substance i and air (g/g-mole),
P	=	the absolute pressure (1 atm),
$V_i; V_{air}$	=	the molecular diffusion volume of the substance i and air (cm^3/mole).

The assumptions are as follows.

- The leak occurs on a hot day when the temperature is 311°K (100°F). The absolute pressure is 1 atm, the molecular weight of air is 29 g/g-mole, and the molecular weight of water is 18 g/g-mole. The molecular diffusion volumes for air and water are 20.1 and 12.7 $\text{cm}^3/\text{g-mole}$, respectively (Perry and Chilton 1973, Table 3-310). The value for D_i is calculated to be 0.271 cm^2/s .

The saturation vapor concentration of the component in air, C_{vp} , is determined from a psychrometric chart by finding the humidity and then multiplying that humidity by the density of air. The value for C_{vp} is $4.90 \times 10^{-5} \text{ g}/\text{cm}^3$.

- The exposed area, A , is a lognormal distribution with a mean of 6000 cm² and a standard deviation of 2400 cm². This assumption is based on engineering judgment.
- The average effective porosity, P_p , of the aggregate is assumed to be one. The layer next to the storage tanks and under the building is expected to have many connected voids providing a pathway for volatile radionuclides to the atmosphere. The value of 1 for the soil porosity, P_p , was selected to represent a realistic assessment of the emission rate from aggregate.

The effective distance the contamination would have to travel to the surface was estimated to be about 8.5 m for the eight MVSTs. Figure C.1 shows the elevations and pertinent dimensions for the tank vaults.

After the emission rate was calculated for water vapor from the soil, the emission rate for each radionuclide was derived by using concentration data from the LLLW electronic data base (Oak Ridge National Laboratory March 10, 1994). The concentration of each radionuclide was entered as a uniform distribution by using the maximum and minimum data from Table B.1. The emission rate for water vapor was multiplied by the concentration of the radionuclide and then divided by the density of the aqueous solution:

$$Q = \frac{E \times C_a}{\rho} \quad (3)$$

where

Q	=	the activity emission rate of the respective radionuclide ($\mu\text{Ci/s}$),
E	=	the emission rate calculated in Eq. (2) (g/s),
C_a	=	the activity concentration of the radionuclide ($\mu\text{Ci/L}$),
ρ	=	the density of the aqueous solution (1100 g/L).

An equation for the activity emission rate, Q , of each radionuclide, can be written by substituting Eq. (1) for E into Eq. (3). After writing the equation in terms of the variables, the variables already discussed were substituted into the equation as constants. A new, simplified equation to solve for Q with only three variables was then written as follows:

$$Q = \frac{C_a \times D \times C_p \times A \times P_i^{4/3}}{\rho \times d_m} \quad (4)$$

$$Q = \frac{(C_a) \times (0.271) \times (4.90 \times 10^{-5}) \times (A) \times (1)^{4/3}}{1100 \times d_m}$$

$$Q = (1.21 \times 10^{-4}) \times \frac{C_a \times A}{d_m}$$

Fig. C.1. Typical cross-section of Melton Valley Storage Tanks.

Prediction of the radiation dose for individuals from a rate of emission to the atmosphere is not straightforward because at less than about 100 m, Gaussian dispersion models are not valid. Therefore, the χ/Q was predicted by assuming that the released material is diluted in a 10-m-high cylindrical volume that has a 30-m radius. The 30-m radius is taken from the criteria in DOE-STD-1027-92. To determine a concentration, this volume was assumed to undergo four complete air changes every minute. This assumption is based on the dispersion model used by DOE for hazards that could result in significant on-site consequences (DOE-STD-1027-92). That model assumes airflow to be 4.5 m/s. The volume is estimated as follows:

$$V = \pi r^2 h = \pi (30)^2 (10) = 28,274 \text{ m}^3. \quad (5)$$

The time-varying concentration in this volume is modeled by the following differential equation:

$$\frac{d\chi(t)}{dt} = \frac{Q}{V} - \frac{F}{V}\chi(t) \quad (6)$$

where

$\chi(t)$	=	concentration at time t after the continuous release begins ($\mu\text{Ci}/\text{m}^3$),
Q	=	activity emission rate ($\mu\text{Ci}/\text{s}$),
V	=	mixing volume (m^3),
F	=	air discharge rate from mixing volume (m^3/s).

Where t is very large, an asymptotic concentration equal to Q/F is reached. Thus, a parameter analogous to χ/Q for this case is $1/F$, in units of s/m^3 . For the assumed air changeover rate of 4/min, F is estimated to be $4 \times (28,274 \text{ m}^3/\text{min})$, or $1885 \text{ m}^3/\text{s}$. To calculate the dose, the standard breathing rate of $3.3 \times 10^{-4} \text{ m}^3/\text{s}$ (Eckerman et al. September 1988, p. 10) is used. The inhalation dose conversion factors (DCF) for the effective dose are from Eckerman et al. (September 1988, Table 2.1). No workers are permanently stationed at these facilities. The general equation for calculating the EDE is as follows:

$$\text{EDE} = \chi(t) \times \text{BR} \times \text{DCF} \times \text{BT} \times \text{CF} \quad (7)$$

where

$\chi(t)$	=	the concentration at time t after the release ($\mu\text{Ci}/\text{m}^3$),
BR	=	the breathing rate ($3.3 \times 10^{-4} \text{ m}^3/\text{s}$),
DCF	=	the dose conversion factor for the radionuclide (Sv/Bq),
BT	=	the time that an individual would breathe the radionuclide in air (s),
CF	=	the conversion factor to convert DCF, which is in Sv/Bq , to $\text{mrem}/\mu\text{Ci}$, $3.7 \times 10^9 \text{ mrem-Bq}/\mu\text{Ci-Sv}$.

The annual exposure hours are estimates based upon conversations with ORNL Waste Operations Center personnel (Peterson, Oak Ridge National Laboratory March 1994). The duration of exposure of an operator at the MVST is based on the following information: an operator will stop to read instruments once every 4 h (three times per shift) and will be in the area for approximately 20 min. An operator will be exposed for a total of 1 h in a 12-h shift; the operator works 185 12-h shifts per year; thus, an operator will be exposed for up to a total of 185 h in a year. Therefore, the duration of exposure was entered into the simulation as 185 h.

C.3 RESULTS OF INHALATION EXPOSURE ASSESSMENT

The inhalation dose to the workers is presented as a distribution of potential results in Fig. C.2. The distributions for the activities of the radionuclides in the tanks, the distance to the surface from the leak point, the annual exposure time, and the exposed area of the leak were the assumptions used in the simulation of potential results. Statistical results of the forecast of the dose and confidence levels are presented in Tables C.1. and C.2.

Fig. C.2. Distribution of inhalation dose to workers.

**Table C.1. Statistical results of the forecast
of worker inhalation exposure**

Statistics	Value
Trials	10,000
Median approx.	30×10^{-3} mrem
Mode approx.	24×10^{-3} mrem
Mean	61×10^{-3} mrem
Standard deviation	99×10^{-3} mrem
Skewness	5.2
Kurtosis	42
Range minimum	2×10^{-3} mrem
Range maximum	1.5 mrem

**Table C.2. Confidence levels of forecasted
values of the worker inhalation exposure**

Percentile %	mrem approx.
5	7×10^{-3}
25	17×10^{-3}
50	30×10^{-3}
75	60×10^{-3}
95	230×10^{-3}

The variance of the distribution of the worker inhalation dose was influenced primarily by the distance from leaked LLLW to the surface, concentration distribution of ^{90}Sr , and leaked LLLW exposed area and to a much lesser extent by the concentration distribution of ^{233}U .

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