40-8989

ENVIROCARE OF UTAH, INC.

THE SAFE ALTERNATIVE

March 1, 1994

Mr. Joseph J. Holonich, Acting Chief
Uranium Recovery Branch
United States Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Request For License Amendment
License No. SMC-1559

Dear Mr. Holonich:

Envirocare of Utah, Inc. ("Envirocare") hereby requests that the United States Nuclear Regulatory Commission ("NRC") amend Envirocare's License No. SMC-1559. Envirocare submits the following information to support its request for license amendment.

1. Envirocare requests that License Condition 11.1 be amended to allow the licensee 72 hours, excluding weekends or holidays, to take both the confirmatory sample and the third sample from POC wells. Groundwater sampling requires the use of prepared sampling bottles. Since Envirocare obtains these bottles from the analytical laboratory, complying with the 24 and 48 hour sampling requirements is not practical. Increasing the time to 72 hours, excluding weekends or holidays, will allow Envirocare to resample the POC well in a timely manner, without exceeding the time limitations. Envirocare submits revision pages to Section 7 to address this requested amendment.

2. Envirocare requests that Condition 9.6(e) be amended as follows:

9.6(e) modify the Quality Control/Quality Assurance Plan to provide quality controls for waste sampling and characterization. The plan must also be modified to provide controls for the quality of protective equipment (e.g., anticontamination clothing and equipment that meets the ANSI Z-88.2 guidance (ANSI, 1989)) and respiratory protection equipment.

Justification: The need to identify "higher or lower specific activity waste" is eliminated by the application of the sensitivity analysis which considers all aspects of waste concentration and volume for each radionuclide and the associated handling procedures. Section 16.2.2 has been revised to apply a more standard treatment of wastes, from a radiological safety standpoint, which contain higher concentrations of gamma-emitting radionuclides.

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3. Envirocare requests that Condition 10.2(b) be deleted. The purpose behind this Condition has been eliminated by the sensitivity analysis of Appendix A-2 which now uses the default parameters for porosity, emanation factor and diffusion coefficients for waste and radon barrier considered by NRC to be the most conservative. The sensitivity analysis performed for this amendment request used values of 40 percent for porosity; fraction of saturated moisture of 24 percent; bulk density of 1.59 g/cm^3; radon emanation coefficient of 0.35; thoron emanation coefficient of 0.1 and diffusion coefficient of 1.2×10^{-2} cm^2 s^{-1}. This resulted in a calculated radon emanation of 1.4 $pCi/m^2/s/pCi/g$ ^{226}Ra and a thoron emanation of 29.9 $pCi/m^2/s/pCi/g$ ^{224}Ra over an infinite thickness of waste for waste with radon and thoron in secular equilibrium with the parent radium isotopes and a radon transport fraction through the clay radon barrier of 1.37 percent. In addition, Envirocare is still committed to demonstrating compliance with effluent concentration limits and workplace exposure through monitoring. Therefore, Condition 10.2(b) is no longer needed and should be deleted.

4. Envirocare requests that Condition 10.7 be amended as follows:

10.7 The licensee shall, upon arrival of the waste, perform external exposure rate measurements of the waste conveyances. Any shipment with exposure rates greater than 5 mrem per hour at a distance of 30 cm from any surface, and which cannot be disposed of within 24 hours, shall be posted as a Radiation Area in compliance with 10 CFR 20.1902(a) until disposed.

Justification: Under the previous concept of "lower" and "higher" concentration shipments described in Section 16 of the Application, "higher " concentration shipments were to be so identified. However, not all "higher" concentration shipments would require such handling and certain combinations of "lower" concentration wastes will require posting. Section 16, revised, and this condition provide a greater assurance of proper radiation safety which does not depend on the shipper's manifest for determination.

5. Envirocare requests that Condition 10.7 be amended as follows:

10.8 The licensee shall operate the facility in compliance with the following specifications:

a) **The maximum bulk mass of waste disposed of annually will not exceed 4.536×10^5 tonnes (5×10^5 tons).**

b) **The maximum annual disposal area will not exceed 38,472 m^2.**

c) **The total embankment capacity will not exceed 2.52×10^6 m^3 (3.3×10^6 yd^3).**

d) **The maximum volume of waste that may be stored on site prior to disposal will not exceed 2.743×10^4 m^3 (9.687×10^5 ft^3) at any one time.**

e) **Waste with an average concentration above 2,000 pCi/g of ^{226}Ra or above 6,000 pCi/g of ^{232}Th in any truck load or railcar will not be accepted.**

f) **The licensee shall manage waste receipt, storage and disposal operations in such manner as to assure compliance with the effluent concentration limits of Table 2, Appendix B to 10 CFR 20.1001 - 20.2401 and population dose limits of 10 CFR 20.1301.**

Justification: Conditions (a) through (e), above, are the same as Conditions 10.8 (a), (b), (d), (e) and (f) in the existing license, but have been reordered.

The purpose behind Conditions 10.8(c), (g) and (h) are satisfied by the application of the results of the sensitivity analysis on the model of Appendix A. This proposed new Condition, 10.8(f), requires Envirocare to meet the same goals achieved by the model, but provides managerial flexibility in meeting those goals. The use of new Appendices A-2 and A-3 provides Envirocare with a tool to determine the potential impact of unloading, storing or disposing of wastes with varying concentrations of individual radionuclides. Identifying wastes only as lower or higher activity does not allow for determination of the different environmental impacts of different wastes having the same concentrations of different radionuclides. (One thousand tons of waste containing 1,000 pCi/g of ^{226}Ra has a very different environmental impact than 1,000 tons of waste containing 1,000 pCi/g of ^{232}Th or 1,000 tons of waste containing 1,000 pCi/g of natural uranium.) The appendices permit Envirocare to investigate a range of management techniques which might be necessary to handle wastes while still meeting effluent concentration limits.

The model of Appendix A is based on disposing of 500,000 tons per year of waste containing 500 pCi/g of each of the radionuclides which might be present in 11e.(2) waste. It is unlikely that such a set of conditions could occur, but the sensitivity analysis of Appendix A-2 keeps the annual waste disposal within the envelope of impacts which would result from disposing of the modelled waste.

Envirocare has reviewed the effects of these requested amendments and finds that they have no significant environmental impacts that have not been address by the Final Environmental Impact Statement, therefore, an environmental assessment is not required prior to approval of these amendments.

Envirocare is submitting with this request for license amendment 5 copies of the revised pages to Sections 7, 16 and 17 of the License Application, plus Appendices A-2 and A-3. Two computer discs which contains the computer spreadsheet described in Appendix A-3 have been enclosed. A list of affected pages is also enclosed.

Your review and approval of this request for license amendment is requested. If you have any questions regarding this matter please contact George Hellstrom at 801-532-0920 or Vern Andrews at 801-532-1330.

Sincerely,

Vernon E. Andrews

for George W. Hellstrom
Envirocare of Utah, Inc.

Enclosures

List of Pages Affected
February 16, 1994

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Section 16

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Section 17

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Appendix A-2

New

Appendix A-3

New

performed on Envirocare property, but surrounding the material handling and disposal areas. This pattern is planned to intercept airborne particulates carried off the site in any direction. Samples will be collected continually and filters will be changed weekly or more often, depending on dust loading. Individual samples will be counted for gross alpha and beta activity after allowing at least seven days for decay of thoron (^{220}Rn) daughters, and quarterly composites of filters from each station will be analyzed for ^{226}Ra, ^{230}Th, ^{232}Th, ^{210}Pb, and total U. Measured concentrations will be compared to the limits in 10 CFR [20.1001-2401], Appendix B, Table II.

Radon-222 and Radon-220 in air will be measured at all air sampling stations, plus at two locations one mile north and east which serve as environmental background stations. Passive monitors operate continuously and will be changed quarterly.

Groundwater samples will be collected as described in the Groundwater Monitoring Quality Assurance Plan. Samples will be collected quarterly from each site and analyzed as shown in Table 7.2. A confirmatory sample will be taken within 72 hours, excluding weekends and holidays, after obtaining results that the sample exceeds concentration limits. A third sample will be taken within 72 hours, excluding weekends and holidays, if the second sample does not exceed the concentration limits. Groundwater flow, as described in Section 5 and Appendix D, is toward the quadrant northwest to northeast of the proposed site. Downgradient detection wells are located immediately north, east and west of the proposed disposal areas. These wells are located to give early warning of any movement of radioactivity from the site through groundwater and the direction of travel.

Soil sampling locations, as described in Section 7.3.4, are closely spaced around the disposal areas and

(2) the Commission has approved acceptance of the shipment, as is.

Pending such corrective action, the shipment will remain on Envirocare property in order to eliminate the potential risk associated with transporting the waste, but will not be admitted to the restricted area; however, if such a shipment is in violation of DOT regulations due to leakage of radioactive materials, it will be placed over an approved surface in the restricted area until the situation is resolved in order to prevent contamination of the environment.

16.2 WASTE HANDLING, INTERIM STORAGE, AND DISPOSAL OPERATIONS

This section outlines the procedure for on-site handling of accepted shipments of disposal material:

16.2.1 On-site Material Management

Once the Incoming Shipment form has been completed to indicate that a shipment has been accepted for on-site management, the shipment may be managed in one of the following ways:

a. A rail car of material may be emptied (dumped) using the on-site railcar dumper and the dumped material may be hauled to the disposal embankment or authorized bulk storage areas or may be collected and placed in containers for storage.

b. A rail car may be unloaded by backhoe at the unloading ramp. The unloaded material may be loaded into trucks and hauled to the disposal embankment or authorized bulk storage areas or may be placed in containers for storage.

c. A dump truck or seavan may dump the material directly into the cell or the seavan may be stored.

d. A rail car with containers may be off-loaded.

e. A trailer with containers may be off-loaded.

f. Unloaded bulk shipments may be stored in bulk or in containers.

g. Off-loaded containers may be stored or emptied into the cell where the material will be compacted in place and the container may be cleaned and released or smashed and compacted in the disposal cell

16.2.2 High-Activity Shipments

For purposes of reducing radiation doses to workers handling shipments containing higher-than-usual concentrations of radioactivity, the following procedures will be followed. Rail cars or trucks which are found on the arrival survey to have external gamma exposure rates of greater than 5 mR/h at 30 cm from any surface will be identified as Radiation Areas by placing a "Caution, Radiation Area" sign, as described in 10 CFR 20.1902, on each of the two sides. If there will be a delay of more than 24 hours in emptying the load, it will be roped off to control access to the radiation area around the shipment.

All workers, other than truck drivers in closed trucks, involved in unloading or disposal of waste from shipments labelled as Radiation Areas will be required to wear full-face respirators, providing a respiratory intake protection factor of 100 to reduce inhalation doses below those which would result from wearing of the standard half-face respirators. Truck drivers hauling these wastes will be required to wear half face respirators.

If wastes from shipments identified as Radiation Areas are to be placed in storage, they must be covered with an additional six inches of low activity waste or clean fill material as soon as practical to reduce gamma exposures and radon emissions.

Wastes from shipments identified as Radiation Areas placed in the disposal cell will be covered with low activity wastes or six inches of clean fill after final compaction in order to reduce gamma exposures to workers on subsequent lifts, unless the subsequent lifts are also from such shipments. Wastes containing more than 500 pCi/g of ^{226}Ra or ^{230}Th will not be placed closer than ten feet from the top or any side of a disposal cell in order to prevent radon emissions at the surface of the radon barrier from exceeding 20 $pCi/m^2/s$, as demonstrated by Figure 3.6 of Appendix A. Gamma exposures at the surface of the clay cover are not affected by burial of wastes at ten feet below the top of the waste placed in the cell. The Health Physics and Radiological Health Handbook (Schleien, Bernard; The Health Physics and Radiological Health Handbook, 1989) reports a h-lf-value layer for attenuation of gamma rays from fallout radioactivity at eight inches. Ten feet of waste and seven feet of clay provide approximately 20 half-value layers, or a reduction of 10^6 in gamma exposure rates.

16.2.3 On-Site Storage

16.2.3.1 Bulk Storage (Stockpiling)

When bulk materials are to be stockpiled or unloaded and stored prior to emplacement, they will be placed at a location on the unexcavated portion of the disposal area, near the working area or on the LARW Bulk Storage Pad. Prior to the first placement, the disposal area storage site must be scarified and recompacted to the specifications for the disposal cell bottom and a liner must be placed over the recompacted clay. A polymer dust suppressant will be used on the storage piles as outlined in Appendix Z, Groundwater Protection Plan.

16.2.3.2 Container Storage

Materials in storage containers will be visually inspected on a monthly basis to ensure that the containers have structural integrity. Drums and barrels of material will be stored on pallets and stacked no higher than two (2) high. The containers in storage

SECTION 17. SAFETY ASSESSMENT

17.1 RELEASE OF RADIOACTIVITY

The calculations and results in this Section are primarily based on the reports prepared by Momeni and Associates (M&A), Analysis of Radiological Pathways of Exposure: Disposal of 11e.(2) Materials at Clive, Utah (Appendix A) and Analysis of Pathways of Exposure (Appendix A-2). The waste characteristics, environmental and operating parameters, and local demographic features needed to project the radioactive exposures to the workers and the environment are defined in that analysis and are consistent with those presented in this Chapter. Releases to the ground water are discussed in Section 5.

17.1.1 Characterization of Waste

17.1.1.1 Radionuclides

The 11e.(2) material encompasses a broad spectrum of byproduct wastes including uranium mill tailings, thorium tailings, and other process residues. The concentrations in the original ores and the extraction processes normally limit the concentrations to less than 12,000 pCi/g for any radionuclide, with the average concentration at any large site ranging from a few hundred pCi/g to approximately 1,000 pCi/g. In order to arrive at a reasonable estimate of the characteristics of 11e.(2) waste, Envirocare has considered available data on operating and non-operating uranium mill sites and three sites where uranium and thorium processing has occurred.

The EPA (1989) compiled data on uranium mills for which statistical descriptions of 11e.(2) wastes can be derived. Table 17.1 provides volume and Ra-226 estimates for the 18 UMTRA inactive mill tailings sites where the volume-weighted mean Ra-226 concentration is 421 pCi/g. Probably a better indicator of the type of waste which might be received at the Envirocare site is the site mean concentration and standard deviation for the UMTRA sites, which is 421 ± 508 pCi/g, with a range of 45 to 2315 pCi/g. The highest concentration was reported for the Canonsburg site, which was a radium processing site rather than a mill site. If the Canonsburg site is excluded, the tailings range from 45 to 745 pCi/g.

Ref: EPA, 1989. Environmental Impact Statement, NESHAPS for Radionuclides, Background Information Document. EPA/520/1-89-006-1, U. S. Environmental Protection Agency, Office of Radiation Programs, Washington, D.C. 20460, September 1989.

Characterization data for the UMTRA sites generally show that in acid extraction processes, Th-230 follows the liquid effluent to a greater degree than Ra-226. Therefore, concentrations of Th-230 of up to 10,000 pCi/g are not uncommon in tailings slimes, raffinate pits, and evaporation ponds. However the site-wide average concentration of Th-230, Ra-226, and decay products should be approximately equal. The U-238 concentration averages approximately 8 percent of the Ra-226 concentration in uranium mill tailings.

The EPA also compiled data for the 11 mills that were operating in 1989. Table 17.2 provides the average Ra-226 concentration for the mill tailings where the site Ra-226 concentrations averaged 319 pCi/g with a standard deviation of 230 pCi/g. The Ra-226 concentration range was 87 to 981 pCi/g. No information was provided on tailings volume.

The UMTRA Disposal Site at Clive, Utah was created from relocating the uranium mill tailings from the Vitro Chemical Company Site. There are various reported average Ra-226 concentration values for this material, ranging from 460 pCi/g to 900 pCi/g, with individual sample analyses ranging from 100 to 2,000 pCi/g (DOE, 1983). The DOE used an average of 670 pCi/g as the basis for their environmental impact assessment.

Ref: DOE, 1983. Draft Environmental Impact Statement, Remedial Actions at the Former Vitro Chemical Company Site, South Salt Lake, Salt Lake County, Utah. February 1983. U. S. Department of Energy, Albuquerque Operations Office, Albuquerque, New Mexico.

Other potential sources of 11e.(2) material are similar to those at the Weldon Spring Site, owned by the federal government and managed by the Department of Energy. Four raffinate pits exist at that site with a total volume of 167,194 m^3 The EPA (1987) summarized the waste characteristics for the pits which are provided in Table 17.3. The volume-weighted average concentration of most radionuclides is below 600 pCi/g, with the exception of Th-230 which is greater than 12 thousand pCi/g.

In addition to the material presented in Table 17.3, the Weldon Spring Site reports (EPA, 1989) the storage of various wastes including 140.1 m^3 of 3.8 percent thorium residues in drums, 42,000 m^3 of contaminated plant and demolition rubble, and 422 m^3 of drummed 3 percent thorium residues. Assuming that the Th-232 is in equilibrium with the daughter products, then approximately 562 m^3 of drummed higher activity waste exists at the site with Th-232 and daughter product activities in the range of 9,000 to 12,000 pCi/g.

Another large site where 11e.(2) materials are stored is the Kerr-McGee Rare Earths Facility in West Chicago, Illinois. The material stored at the production facility consists of sludge piles, four ponds, and contaminated soil and debris. Several off-site properties will be decontaminated creating large volumes of slightly contaminated soils. Total volume is estimated at approximately 500,000 cubic yards.

NRC (1987) reports that the thorium and rare earth ore processing tailings for the Rare Earth Facility, West Chicago, averages 82.7 pCi/g U-238, 78.4 pCi/g Ra-226, 323 pCi/g Th-232, 37.8 pCi/g Th-230, and 548.6 pCi/g Ra-228.

Approximately 12 percent of the waste can be classified as higher activity and is associated with the processing waste stream. Unpublished data (Source: Kerr McGee) provide a better understanding of the character of these process wastes which are summarized in Table 17.4. One can see that of the 4 waste types, two are most elevated in Th-232, one is highest in Ra-226, and one is highest in U-238. Samples for three of the waste types ranged up to several thousand pCi/g.

Reference: NRC, 1987 Supplement to the Final Environmental Statement Related to the Decommissioning of the Rare Earths Facility, West Chicago, Illinois, NUREG-0904, 1987, U.S. Nuclear Regulatory Commission, Washington, D.C.

Momeni estimates that the weighted average radium-226 activity for all waste at the West Chicago site is about 300 pCi/g. However, approximately 86 percent of the waste has a radium activity below 200 pCi/g, with an average value of 40 pCi/g. A similar range of concentrations is expected for Th-232, resulting in a weighted average concentration of about 900 pCi/g, but with most of the waste at about 50 pCi/g.

Another large cleanup of 11e.(2) wastes is being planned for properties in Maywood, New Jersey, estimated to create 395,000 cubic yards of contaminated soil and building debris (DOE, 1992). Characterization data available to Envirocare do not provide adequate information on which to base estimates of average radionuclide concentrations. However, individual sample results indicate that thorium concentrations range up to 6,000 pCi/g or more, which is similar to those at other thorium processing plants (e.g. West Chicago Rare Earths Facility). Radionuclides from the U-238 decay chain are present in lesser concentrations. While the maximum concentrations are high, a large portion of the wastes appear to be from the dispersal of process waste and, therefore, may be highly diluted.

Ref: DOE, 1992. Work Plan - Implementation Plan for the Remedial Investigation/Feasibility Study - Environmental Impact Statement for the Maywood site, Maywood, New Jersey Prepared by Argonne National Laboratory and Bechtel National, Inc., 1992.

The waste sites described above all have similar characteristics. Process waste concentrates such as the sludges, slimes, and raffinates usually are segregated and constitute significantly large volumes (1,000 m^3 or more) of higher activity wastes with average Ra-226 concentrations up to 2,000 pCi/g and average Th-232 concentrations up to 6,000 pCi/g.

Building debris, contaminated soils, and mill tailings will make up approximately 80 percent of the waste. The average activity of this material will be below 1,000 pCi/g for any site with most probable averages closer to 400 pCi/g.

Summarizing the data presented above, the following radiological waste character is anticipated for the Envirocare 11e.(2) disposal site. Considering the relative proportions of lower and higher activity waste at the site, Envirocare estimates that the overall average concentration for any radionuclide will be approximately 500 pCi/g; however, individual sites may vary widely around that average, as described above. Because of this, individual shipments of wastes may contain higher average concentrations of Ra-226 and Th-232. In the context of waste deliveries to the disposal site a shipment is taken to mean a single waste-hauling truck or rail car from a single generator. Weighted average concentrations in a shipment must not exceed 2,000 pCi/g for Ra-226 or 6,000 pCi/g for Th-232, although these two radionuclides may both be present in the shipment at those concentrations.

A conservatively-high estimate of the volume of material to be handled and disposed of at the site would be one-half million (500,000) tons/year. Assuming an average Ra-226 and Th-232 concentration of 500 pCi/g, the estimated annual average total activity disposed of would be 337 Curies for each of the radionuclides. Since the daughter products may be assumed to be in secular equilibrium, there would be approximately 337 Curies of each of the other important radionuclides, such as Ra-228 and Ra-224. The amount of Uranium would be expected to be less than 25 percent that of Ra-226. The average Th-230 concentration is expected to be similar to that of Ra-226 and will depend upon the disequilibrium of the radionuclides in that decay series. The actual amount of radioactivity disposed of in a given year will vary around the estimated 337 curies per radionuclide as actual concentrations and disposal amounts vary.

17.1.1.2 Chemical Constituents in the Waste

In addition to the radiological constituents, these wastes would be expected to include those constituents found in mill tailings in general, regardless of the source. The Environmental Protection Agency has reported the upper ranges of elements in mill tailings from several sources which are presented in Table 17.5. In some cases these are not significantly different from "normal" soils but due to the limited number of sources, concentrations of any of these constituents could be several times higher than reported.

Table 17.5 Concentrations of Stable Elements in Uranium Mill Tailings Compared to the Average Earth's Crustal Abundance

Element	Concentration (ppm)	Average Crustal Concentration (ppm)
Aluminum	72,000	81,000
Arsenic	600*†	5
Barium	4,000*†	250
Bromine	6	1.5
Calcium	87,000	36,000
Chlorine	6,800*	310
Chromium	7,300*†	200
Cobalt	140*	23
Copper	1,200*	70
Iron	320,000*	50,000
Lead	3,100*†	16
Magnesium	17,000	21,000
Manganese	2,100*	1,000
Mercury	34*†	0.5
Molybdenum	550*	15
Nickel	1,100*	80
Potassium	25,000	26,000
Rubidium	560	310
Selenium	230*†	0.1
Silver	10*†	0.1
Sodium	47,000	28,000
Strontium	4,100*	300
Terbium	5	0.9
Thallium	10*	0.6
Tin	6,200*	40
Titanium	5,700	4,400
Tungsten	570*	69
Vanadium	4,400*	150
Zinc	2,200*	132

* Maximum observed concentrations substantially greater than average.
† Hazardous constituents from 10 CFR 40, App. A, Criterion 5C.

At these concentrations it is expected that arsenic, barium and lead would fail TCLP and that those wastes would be classified as exempt wastes.

For most of those elements listed as hazardous constituents, the very high concentrations were found at only one mill site; therefore, the average concentrations are expected to be much lower. Rough averages, based on the observed range of concentrations of the hazardous constituents, were less than half of the maximum observed concentrations.

The NRC's Uranium Recovery Field Office in Denver, Colorado conducted an extensive characterization of uranium mill tailing impoundments located in Wyoming, New Mexico and South Dakota over a five-year period to determine what hazardous constituents would likely be found in uranium mill tailings. Based on the findings of the investigation, and verified in a telephone conversation with Gary Konwinski (Uranium Recovery Field Office) on March 3, 1993, the following hazardous constituents were identified:

METALS	VOLATILE ORGANICS	RADIONUCLIDES
Arsenic	Acetone	Radium-226
Barium	2-Butanone	Radium-228
Beryllium	Chloroform	Thorium-230
Cadmium	Carbon disulfide	Thorium-232
Chromium	1,2-Dichloroethane	Uranium
Cyanide	Methylene chloride	
Fluorine	Naphtha	
Lead		
Mercury	SEMI-VOLATILE ORGANICS	
Molybdenum	Diethylphthalate	
Nickel	2-Methylnaphthalene	
Selenium		
Silver		

The hydrogeologic report by Bingham Environmental (Appendix GG) concluded that it would take 400 to 600 years for leachate to travel through the unsaturated zone and then another 800 years to reach the nearest off-site well. No non-radiological constituent would reach the ground water in less than 700 years.

17.1.2 Infiltration

Section 4.1.1 discusses principal design features to minimize water infiltration into the embankment and disposed materials. As indicated in that section, calculations in Appendix M demonstrate that the amount of precipitation that infiltrates into the embankment and percolates to the shallow groundwater under proposed conditions is negligible.

17.1.3 Radionuclide Release - Normal Conditions

Release of radionuclides under normal conditions during operation of the site is limited to the following mechanisms:

1. Release of interstitially trapped radon and thoron gas when handling bulk wastes.

2. Exhalation of radon gas from embankment area(s) that have not been covered with the compacted clay radon barrier.

3. Exhalation of radon gas from embankment area(s) that have been covered with the compacted clay radon barrier.

4. Exhalation of thoron gas from the top layer of embankment areas which have not been covered with a layer of non-thorium-containing waste or clean clay.

5. Localized resuspension of dust from waste handling operations.

6. Windblown materials from the embankment and unloading area.

These release mechanisms, along with the exposure to direct radiation (gamma radiation), result in a radiation dose to the workers and off-site population.

Other release mechanisms have been determined to be insignificant at the Clive site. There exist no surface water systems at the site that could transport waste from the site. In addition, the lack of significant biota within the region reduces the potential for embankment or waste penetration and ultimate release to the environment. The local climate and the principal design features of the embankment create conditions for minimizing infiltration of radionuclides into the groundwater. Because of the negligible impact, these potential release mechanisms will not be discussed further in this section.

After closure, the principal design features of the embankment cover system will eliminate windblown particles from the embankment, reduce the radon emission to 20 pCi/m^2 s , and reduce direct gamma ray exposure rates near the disposal cells to background levels (approximately 10-15 μR/hr).

17.1.3.1 Off-site Impacts from Normal Operations

M&A (Appendices A and A-1) provided estimates of projected radionuclide release rates and radiological impacts during site operations, assuming waste which exhibits the radiological characteristics estimated for the overall 11e.(2) profile (500,000 tons per year of waste containing 500 pCi/g of each of the radionuclides in the uranium and thorium series). While these Appendices demonstrate compliance with 10 CFR 20.1301 and 10 CFR 20.1302 under the assumed conditions, they do not completely serve the purpose of evaluating the variable characteristics of waste quantities and radionuclide concentrations which are expected to occur annually, or over shorter periods of time. M&A performed a sensitivity analysis of Envirocare's waste management procedures and waste characteristics (Appendix A-2). This analysis permits each waste handling procedure, from receipt to final closure, to be evaluated for its environmental impact while handling any quantity of wastes at any specified radioactivity concentration. Output from the analysis of Appendix A-2 will be used as input to the calculational spreadsheet described in Appendix A-3 to provide guidance to Envirocare planners in scheduling waste shipments and planning waste handling operations to meet the effluent concentration limits of Table 2, Appendix B to 10 CFR 20.1001 - 20.2401. The application of Appendices A-2 and A-3 to waste management will allow Envirocare to manage wastes within an envelope of quantities and radioactivity characteristics during the year while meeting the overall environmental results of Appendices A and A-1.

Table 3.20, revised, of Appendix A-1 provides a projection of Total Effective Dose Equivalent (TEDE) to eight receptors. This projection assumed that the waste was made up of both the thorium series and the uranium series with all radionuclide concentrations equal to 500 pCi/g, a conservative and improbable situation chosen to represent the expected long-term average concentrations of waste which might be received. A maximum off-site TEDE of 116.1 mrem/y at the south boundary was projected, if the radon and thoron impacts are included. The maximum TEDE for the nearest members of the public occurs for workers at USPCI of 5.2 mrem/y.

Also reported in Table 3.20, revised, are TEDE for occupants in the controlled area (outside of the restricted area, but within Envirocare's controlled area). The TEDE's for occupants of the Administration Building was calculated to be 76.3 mrem/y.

The regional collective population TEDE was calculated (see Appendix A, Table 3.21) to be approximately 0.016 person rem/year after 16 years of operation. This small value reflects the very limited population in the area and is considered insignificant.

The dose calculations above, from Appendices A and A-1, were based on a single assumed average concentration in waste with an annual total of 500,000 tons of waste disposed, or an annual disposal of 337 Ci of each of the radionuclides in the uranium and thorium series. Occupational and environmental doses are shown to be almost completely dependent upon the total amount of radioactivity managed. While the use of Appendices A-2 and A-3 provide considerable flexibility in waste management, the reliance upon the modelling of Appendices A and A-1 will assure that occupational and environmental impacts are as described in those appendices. With this option, Envirocare can safely dispose of any combination of radioactivity concentrations up the shipment limits of 2,000 pCi/g for ^{226}Ra and 6,000 pCi/g ^{232}Th. Application of this approach would automatically restrict the amount of waste which could be received at higher concentrations.

Included in the modelled receptor locations of Appendix A-2 are the environmental monitoring stations, making it possible to make a direct comparison between model results and measured airborne concentrations. The model and calculational spreadsheet will be used for operational planning purposes, only. Envirocare will use environmental monitoring results to modify operations, if necessary, and to demonstrate compliance with dose and effluent concentration limits.

17.1.3.2 Occupational Radiation Exposures

Projections of annual occupational TEDE were made by M&A for workers performing various operations at the site. It was assumed that the incoming wastes consisted of the uranium and thorium series with each radionuclide present at an average concentration of 500 pCi/g. Using other very conservative assumptions, a maximum TEDE of approximately 1 rem/year for any worker was calculated, meeting the criteria of 10 CFR 20.1201. Projections for each of the six types of waste handing operations are given in Table 3.22 of Appendix A.

The potential for beta doses to the skin and lens of the eye was estimated from the equation :

$$_{\beta}D = 0.23\ E_{\beta}\ \chi$$

where: $_{\beta}D$ = Dose rate from an infinite cloud (rad/s)
E_{β} = Average beta energy per disintegration

(MeV/dis)

χ = Concentration of the beta emitting isotope in the cloud (Ci/m^3)

(ref: Schleien, Bernard; Health Physics and Radiological Health Handbook, 1989)

With 500 pCi/g of each of the nuclides of the thorium and uranium series in waste there are 5,000 pCi/g of beta emitters with an average beta energy of approximately 0.205 MeV. With an airborne particulate concentration of 1 mg/m^3, the beta dose rate to the skin or lens of the eye is calculated to be approximately 2.36E-13 rad/s or 7.4 μrem/y. Therefore, external beta doses are not considered to be significant.

The model of Appendix A, based on an assumption of handling the maximum quantity of waste permitted under this Application (500,000 tons per year) with an average concentration of each nuclide at 500 pCi/g, is believed to be conservative. It is not possible to model each potential situation, such as a shorter waste disposal period while handling wastes at higher concentrations, but as discussed in 17.1.3.1, occupational doses are primarily a function of the total radioactivity disposed of during the year. For those cases where waste containing radioactivity concentrations significantly greater than 500 pCi/g for each radionuclide are handled for extended periods, Envirocare will closely monitor internal and external exposures to maintain TEDE as low as reasonably achievable and, in all cases, below the standards of 10 CFR 20.1201.

17.1.4 Radionuclide Release - Accidents or Unusual Operation Conditions

The U.S. Nuclear Regulatory Commission in its Final Generic Environmental Impact Statement on Uranium Milling (NUREG-0706) categorizes incidents involving releases of radioactivity as trivial incidents, small releases, and large releases. Trivial releases for a model mill all involve plumbing releases up to and including a breach of a tailings disposal line carrying 70 tons per hour of tailings. Small releases include failure of the yellowcake air-cleaning system, fire or explosion in the solvent extraction circuit, and gas explosion in the yellowcake drying operation. Large releases could occur from tornadoes or breaches in the tailings dam caused by flooding, earthquakes, or structural failure. Obviously the types of releases which could occur at the Clive site are more limited than those which could occur at a mill site and would largely be classed as trivial in that the potential for either significant on-site or significant off-site doses would be expected to be small.

Since we have no movement of radioactive materials through piping or other plumbing we would have no releases of radioactivity from piping breaks. Flammable or explosive fuels are not stored in close proximity to the wastes and the principal flammable material is in the fuel tanks of the individual work vehicles. A vehicle fire, even on a loaded haul truck, would not be expected to release any significant quantity of the load as airborne dust.

The possible release scenarios, all of low probability but ranged in order of increasing improbability, are:

1. on-site truck turnover or collision

2. train derailment
3. flooding
4. tornado.

The above scenarios all result in the exposure of wastes to the natural elements and forces of nature. The Department of Energy evaluated the impacts of accidental releases of material associated with the disposal of mill tailings at Clive. (**ref: <u>Draft Environmental Impact Statement, Remedial Actions at the Former Vitro Chemical Company Site, South Salt Lake, Salt Lake County, Utah.</u> U. S. Department of Energy, Albuquerque Operations Office, Albuquerque, NM., February, 1983**) They concluded that the worst accident would result in the spillage of the equivalent of a train car of bulk waste material in transit to the site. A second case was evaluated where a similar size spill occurred but the spillage occurred into the Great Salt Lake. Impacts of these events were found to be negligible compared to the impacts from normal operations.

The average bulk 11e.(2) waste brought to the Envirocare site will be similar in physical and chemical form to the Vitro mill tailings and, therefore, no additional assessments of accidental releases off site will be made.

The following accidental on-site releases have been evaluated:

<u>On-site truck turnover or collision</u>
From NUREG-00706 the probability of a truck accident is in the range of 1.0 to 1.6 x 10^{-6}/km. There are two kinds of truck movements to be considered at the Clive site.

These are arriving waste shipments and haul trucks moving material from the rollover or storage to the trench. Assuming that there are 3 incoming trucks per day and 50 loaded trucks per day from the rollover or storage to the trench and assuming that the on-site distance travelled by any loaded truck is one kilometer, the probability of accident in any one year is:

1.3×10^{-6}/km x 53 loads/day x 260 days/year x 1 km/load

$= 1.8 \times 10^{-2}$ or about 1.8%.

Most of the material from the truck would be deposited on the ground in the immediate vicinity of the truck. Based on NUREG-0706, for a wind speed of 10 mph, about 0.1% of the material would become airborne immediately (for dry material). Obviously if the material is moist, the release fraction would be less. For a 20 ton (40,000 pounds) truck, about 40 pounds or less might become airborne. This compares with about 24 pounds of dust which becomes airborne daily per hectare of a mill tailings pile surface. If the spill were not cleaned up or dust controlled rapidly, the release fraction over a 24 hour period might increase to as much as 0.9% or 360 pounds. This is highly unlikely because of the presence on-site of crews and equipment which are there for the express purpose of managing bulk wastes. Because of moisture differences and differences in waste composition from the model mill assumptions, we would expect to have lower release fractions for the Envirocare wastes.

For a theoretical truck accident involving a yellowcake shipment, a 24-hour release period, all particles in the respirable range, and a population density of 7.5 persons per square mile, NRC estimated 50 year dose commitments to the lungs of the general public in the range of 0.7 to 9 person-rem. The yellowcake specific activity is about 6.77×10^5 pCi/g while the average uranium or thorium concentrations expected at Envirocare would be 500 pCi/g, or a factor of 1300 lower. Individual shipments to Envirocare might have ^{226}Ra concentrations as high as 2,000 pCi/g, or similar to those found in uranium mill tailings. Concentrations of ^{232}Th in a small fraction of shipments could be as high as 6,000 pCi/g. The dose per unit intake via inhalation is higher for Th-232 wastes than for yellowcake by up to a factor of 1000, depending upon the chemical form and radionuclide mix. Therefore, the postulated off-site public doses could be approximately an order of magnitude higher than for a yellowcake spill under the same circumstances. However, the population distribution around the Clive site is insignificant compared to that in the NUREG calculation and, therefore, the off-site population dose would be inconsequential.

For on-site workers, there would be a very short exposure time since there would be no reason to stand downwind for 24 hours (or even one hour). Assuming an accident involving the spill of a load of waste with a concentration of 15,000 pCi/g; a period of three hours for cleanup with no use of respiratory protection; an airborne concentration of 1 mg/m^3, and a respiratory rate of 1.2 m^3/h a total of 54 pCi of each nuclide would be inhaled. Comparing these to the ALI's from Appendix B of 10 CFR 20.1.001 - 4201, the sum of fractions is 0.022. The external gamma dose, using the relationship of 3.1

μrem/h/pCi/g for Ra-226 from Appendix A Section 3.7.3 and doubling for the contribution from Ra-228, would be less than 140 mrem. Such a dose added to the projected maximum TEDE of 1,032 mrem/y would still be well within the permissible annual exposures for radiation workers. In actual fact, no workers would be present under such conditions without respiratory protection and would not be standing on the spilled waste for more than a few minutes.

Radiation doses to non-radiation workers would be limited by promptly evacuating such persons from the vicinity of such an accident. Non-radiation workers who might respond as part of an emergency team would be monitored and would spend a limited amount of time in proximity to the waste. It is believed that no person who is not a radiation worker would remain in the vicinity for more than 30 minutes. Therefore, comparing inhalation exposures and external doses to those for radiation workers, it is obvious that no non-radiation worker would receive in excess of 100 mrem.

"Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Mills Will Be As Low As Reasonably Achievable" and draft Guide DG-8013, "ALARA Levels for Effluents from Materials Facilities." The FRSO will document ALARA activities including:

a. Reviews of new proposed disposal contracts to assure that Envirocare's procedures, facilities, and equipment are appropriate and sufficient to limit exposures to workers and the environment;

b. Monthly reviews of work area, perimeter, and environmental air monitoring results noting trends and adjusting work procedures when practical to further reduce potential exposures; and

c. Monthly reviews of work area gamma-ray exposure rates and advising the Site Manager(SM) on operational changes that will reduce exposures to ALARA levels.

An audit of ALARA activities will be conducted and documented by the CRSO at least annually as a part of the ES&H Internal Audit.

17.4.2 Restricted and Controlled Areas

The Envirocare Site consists of an adjacent controlled and restricted areas with an administration building, which also serves as the access control point to the restricted area, located on the boundary between the two. The restricted area is a fenced area consisting of the materials handling facilities and disposal areas. All licensed waste handling and disposal activities will be conducted within the fenced restricted areas. Other activities such as off-site environmental monitoring and laboratory analysis of environmental samples are conducted in the controlled area which includes a portion of the Administration Building and areas outside the fenced restricted area.

In keeping with 10 CFR 20.1301, Envirocare will limit the exposure to employees restricted to the controlled (but unrestricted) areas of the site to the limits for individual members of the public.

A residence trailer is provided for Envirocare's security guard north of the controlled area on Envirocare-owned property outside of Section 32. The rate of exposures at this residence location will be maintained to that allowed for an individual member of the public.

17.4.3 Radiation Dose Limits

17.4.3.1 Occupational Dose Limits for Adults

Occupational doses to individual adults will be controlled to levels consistent with 10 CFR 20.1201. Except for planned special exposures, the exposures are limited as following:

a. Annual limit will be the more limiting of:

site boundaries. The projected concentrations are in the range of ambient background concentrations and are well below the concentration limits of Appendix B to 10 CFR 20.1001-20.2041. Airborne particulate monitoring will be performed to confirm those predictions.

Envirocare admits members of the public to the site for the purpose of brief site visits and site inspections. All visitors, except those qualified by training or experience as radiation workers, are accompanied by an Envirocare employee who carefully limits the areas in which the visitors may enter. Visitors are issued a pocket ion chamber or digital radiation monitor to monitor external radiation. Visitors are not allowed in areas where respiratory protection is normally required.

17.4.4 Internal Radiation Dose Assessment

17.4.4.1 Calculation of Internal Radiation Exposure from Inhalation

The internal radiation exposure is represented as the product of the Derived Air Concentration (DAC) and time of exposure. An exposure of 2,000 DAC-hours results in a committed effective dose equivalent of 5 rems for nuclides that have their DAC's based on the committed effective dose equivalent. It is calculated for each radionuclide as follows:

$$\text{DAC-hours} = (C/DAC) \times t$$

exposure to airborne tailings where the quarterly intake is equal to ten percent of the TEDE. A similar derivation for other radiological mixes may be required and a different action level used when large quantities of other 11e.(2) materials are being handled.

Based on experience at Envirocare's NORM disposal facility, it is unlikely that any employee's bioassay results will be above the action level. If any result does exceed the action level, the causes for such a level will be investigated and steps will be taken to reduce the employee's future exposure to inhaled or ingested radioactive materials.

A special bioassay sampling will be done for all personnel involved in an incident determined by the CRSO as having a potential for a significant intake of radionuclides. Twenty-four hour fecal and urine samples will be collected on a periodic basis until activities are below the minimum detectable levels or a determination is made that continued monitoring is not necessary. If the waste contained high Th-232 concentrations, lung or whole-body counting techniques may be employed to measure deposition in the body.

Excretion models will be used along with waste characterization data, bioassay data, and operational data to estimate the radionuclide intake and the resultant dose to the organs. Methods recommended in NCRP Report No. 87, "Use of of 0 - 200 mR. Used to record exposures to visitors and temporary employees while in the controlled area.

g. Ludlum Model 1000 Scaler-Timer with Model 43-10 Alpha Scintillation Detector - 1 each. Selected as a reliable, easy-to-use instrument for the counting of gross alpha activity on air samples and swipes.

h. Ludlum Model 2200 Scaler/Ratemeter with Model 43-10 Alpha Scintillation Detector - 1 each. Selected as a reliable, easy-to-use instrument for the counting of gross alpha activity on air samples and swipes.

i. Ludlum Model 2200 Scaler-Timer with Model 120 Gas Proportional Detector - 1 each. Selected as a reliable, easy-to-use instrument for the counting of gross alpha or gross beta activity on air samples and swipes.

j. Technical Associates Model MGS-5AB gas flow counter with Model 5S5T analyzing scaler ratemeter.

The calibration and management of monitoring equipment is based on applicable guidance in NRC Regulatory Guides, 4.14, 8.25, and DG-80030.

All equipment used in measurement of radiation is periodically calibrated by persons licensed to perform such calibrations. The calibration facilities currently used by Envirocare calibrate exposure rate survey meters and dosimeters against Cs-137 standards. All survey equipment will be calibrated at least semiannually or after each repair. All personal dosimeters will be calibrated annually.

Calibrations will be performed by persons who are qualified for the specific calibration.

Beckman. Instant working level meters will be used if adequate sensitivity can be achieved.

17.4.6.6 Environmental Monitoring Program

The environmental monitoring program is presented in Section 7.

17.4.7 Personnel Protection and Contamination Control

17.4.7.1 Access Control

All personnel working in the restricted area(s) are required to enter and exit through an access control gate. All persons entering the area will be required to enter their name in the access control log. (See Figures 17.2 and 17.3).

All personnel working in the restricted area will be monitored by one of three methods described below:

1. Permanent employees will be issued a thermoluminescent dosimeter (TLD) badge provided by Envirocare. These dosimeters will be exchanged and returned to the vendor on a quarterly basis. Permanent employees will pick up and turn in their dosimeters at the beginning and end of their working day at the manned access control point.

2. Individuals who are visiting the site on a limited basis will be issued a pocket dosimeter to record exposure. Visitors will pick up and turn in their pocket dosimeters at the manned access control point when they enter and exit the site. The

dosimeters will be read as the individual leaves the site and recorded in the Access Log.

3. A group of visitors may all use the exposure from either one TLD or one pocket dosimeter in a situation where the entire group is to stay in the same vicinity while in the restricted area.

Persons who do not conform to one of these options will be denied access to the restricted area of the site. Access to the site without prior training and deviation of dosimeter policy must have prior approval from the Corporate or Field Radiation Safety Officer (RSO).

Each person entering the restricted area who will or may receive in one year a radiation exposure in excess of 10 percent of the limits in 10 CFR 20.1201, 10 CFR 20.1207, or 10 CFR 20.1208 will be required to disclose in a written, signed statement, either: (1) that the individual had no prior occupational dose during the current calendar quarter, or (2) the nature and amount of any occupational dose that the individual may have received during that specifically-identified current calendar year from sources of radiation possessed or controlled by other persons.

Records of prior radiation exposure will be obtained from all employees and will be used to update their individual exposure records.

The quarterly dosimeter results from the quarterly exchange of dosimeters will be promptly recorded by the

Table 17.6 SURFACE CONTAMINATION LEVELS ON EQUIPMENT, CLOTHING AND PERSONNEL TO BE RELEASED WITHOUT RESTRICTIONS FROM RESTRICTED AREA

	Column I	Column II	Column III
Nuclide[a]	Average[b,d,f]	Maximum[b,d,f]	Removable[b,e,f]
U-nat,U-235,U-238, and associated decay products	5,000 dpm alpha/100cm²	15,000 dpm alpha/100cm²	1,000 dpm alpha/100cm²
Transuranics, Ra-226, Ra-228,Th-230,Th-228, Pa-231,Ac-227,I-125, I-129	100 dpm/ 100 cm²	300 dpm/ 100 cm²	20 dpm/ 100 cm²
Th-nat,Th-232,Sr-90 Ra-223,Ra-224,U-232 I-126,I-131, I-133	1,000 dpm/ 100 cm²	3,000 dpm/ 100 cm²	200 dpm/ 100 cm²
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except SR-90 and others noted above	5,000 dpm beta-gamma/100 cm²	15,000 dpm beta-gamma/100 cm²	1,000 dpm beta-gamma/100 cm²

a. Where surface contamination by both alpha- and beta-gamma emitting nuclides exist, the limits established for alpha-and beta-gamma emitting nuclides should apply independently.

b. As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

c. Measurements of average contaminant should not be averaged over more than one square meter. For objects of less surface area, the average should be derived for each such object.

d. The maximum contamination level applies to an area of not more than 100 cm2.

e. The amount of removable radioactive material per 100 cm² of surface area should be determined by wiping the area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of know efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

f. The average and maximum radiation levels associated with surface contamination resulting from beta-gamma emitters shall not exceed 0.2 mrad/hr at 1 cm and 1.0 mrad/hr at 1 cm, respectively, measured through not more than 7 milligrams per square centimeter of total absorber.

Records of time spent in the restricted area will be obtained from the Access Control Log kept in the administration building.

There will be no high or very high radiation areas on site due to the concentration limitations in the waste acceptance criteria. As shown in Section 17.1.4, even with wastes as high as 15,000 pCi/g of each radionuclide the external gamma exposure rate would not exceed 50 mR/h. Therefore, no special access control procedures as required in 10 CFR [20.1601-20.1602] will be developed.

17.4.7.2 Protective Clothing and Change Facilities

The administration building includes a locker room where employees change shoes and outer clothing and decontaminate, when necessary. The locker room is equipped with showers and a wash basin. A washer and dryer are used by Envirocare for washing of work wear. Figure 17.1 shows the proposed new layout of the change facilities.

Either cloth or disposable coveralls will be provided for all employees working in the contaminated areas. It is required that this protective clothing be worn at all times by employees while working in the restricted area except for those performing limited duties not involving radioactive waste or contaminated materials while in the immediate vicinity of the administration building.

Supervisors and other visitors to the site who are not operating equipment or working on the embankment are not required to wear protective clothing or wash exposed skin upon exiting. However, they must wear dedicated boots or boot covers and must use the hand and foot monitor(s) and follow all other established criteria when exiting the site.

Permanent employees at the site will be issued dedicated work boots that are to be worn in the controlled area. These boots are not to leave the controlled area. Temporary workers will be issued boots or will be required to wear shoe covers.

Each employee shall be responsible to keep contaminated material inside restricted area(s).

17.4.7.3 Respiratory Protection Program

All personnel working in contaminated areas are required to routinely wear respirators. Half-face respirators have been selected by Envirocare and are provided to each worker. The selection of half-face respirators was based on the need to have better visibility for machine operations than full-face respirators afford, while providing adequate protection against the relatively low concentrations of airborne radioactive particulates.

A respiratory protection program, based on the guidance in ANSI Z88.2-1980, "Practices for Respiratory Protection", will be implemented. The program elements include, employee training, qualitative fit testing, cleaning and maintenance, written standard operating procedure covering the program, medical surveillance, and recordkeeping. The FRSO is responsible for administering the respiratory protection program.

17.4.7.4 Dust Control Measures

Engineering controls and dust suppression techniques will be used to minimize levels of airborne particulates. This will include methods such as vehicle speed control, and use of water and other surface fixatives. Because of the importance of dust control in the minimization of occupational exposure to radioactive particulates, the following engineering controls will be implemented inside the restricted area during periods of site operation:

1. A water truck will be on site all days of operation.
2. Wherever practical, magnesium chloride solution (MgCl[aq]) will be applied to surfaces twice per year. One application will be in the spring and the other in the summer.
3. If any other areas within the restricted area are being used in addition to those which have received MgCl(Aq), these areas will be watered at a minimum of every two hours unless rainfall has exceeded 0.10 inch during the previous 24 hours.
4. Each day of operation a daily record will be kept of water application and/or MgCl(Aq) application. The records will include the following items:

   a) Date of application
   b) Number of treatments

c) Rainfall received

d) Time of day treatments were made

17.4.7.5 Envirocare Site Regulations

Envirocare has established Site Regulations for Envirocare employees (SR-1), contractor employees (SR-2), truck drivers (SR-3), and visitors (SR-4). Basic health and safety requirements are specified including access requirements and limitations, personnel protection equipment, dosimetry requirements, work and work area rules and restrictions, and penalties assessed for violation of site regulations. These regulations are included in the Procedures Manual (Application, Appendix B).

17.4.8 Health and Safety Training

The radiation training program is operated under the direction of the Corporate Radiation Safety Officer. Radiation safety training will be provided to all persons before they are allowed to enter the restricted area. The amount of radiation safety training required for persons to enter the restricted area is related to the activities for which the person will enter the restricted area.

There are three categories of restricted-area functions:

(1) Permanent Employee

(2) Temporary Worker

(3) Visitor

A "Permanent Employee" is an employee of Envirocare hired for a period longer than 20 days, or a long-term employee of a contractor to Envirocare.

A "Temporary Worker" is a service contractor (electrician, welder, consultant, surveyor, driller, sampler, engineer, fence installer, forklift operator, laborer, mechanic, liner installer, excavator, etc.) who works inside the restricted area under a contract or service order but who is not an employee on the payroll of Envirocare or Envirocare's radioactive material contractor.

A "Visitor" is a person whose main interest inside the restricted area is to communicate with personnel in the restricted area, to observe and/or inspect the operations, facilities, programs, location and compliance at the site. Examples of visitors are compliance inspectors, visiting dignitaries, representatives of organizations and corporations, tour groups, and associates of the above and of permanent employees and temporary workers. Most visitors will be required to be in the presence of a qualified escort while in the controlled area. Certain visitors, such as compliance inspectors or auditors will not require escorts.

Training requirements have been established for each of the categories listed above. Refresher training is offered to review and update training information.

The 3-hour Training Session will be directed by the field or corporate Radiation Safety Officer or by a contractor whose training has been approved by the CRSO. The training will include the following items and topics:

- radioactive nature of the material being handled
- fundamentals of handling radioactive materials
- ionizing radiation and biological effects

CATEGORY	Restricted Area Safety Training 1-hr	Read/Sign Site Regs	3-hour Rad-Safe Training	Refresher or Repeat After
Permanent Employee	Yes	Yes	Yes	6 months* Refresher
Temporary Worker	Yes	Yes	No	1 week Repeat **
Visitor	No	Yes	No	3 months Repeat

* Refresher course for permanent employees is one-hour review course.

** After a temporary worker has received training for three weeks of restricted-area work within any one-year period, the temporary worker must receive the permanent employee training prior to performing additional work within the one-year period.

- radiation safety standards, principles and procedures
- emergency procedures
- methods of radiation protection
- presentation to each trainee of a personal copy of the training manual
- question and answer session
- a written examination

Records of training attendance and a copy of the examination provided will be maintained by the Health Physics office. See Appendix C for "Training Manual for Radiation Workers at Envirocare's Low Activity Radioactive Waste Disposal Site in Clive, Utah"; and exams.

APPENDIX A-2

ANALYSIS OF PATHWAYS OF EXPOSURE