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From: Sandra L. Wastler (SLW1)
To: RLB2, EWB, JJS, MHF1 R Bangart, M Fliegel, J Surmeier
Date: Tuesday, June 1, 1993 1:04 pm
Subject: ENVIROCARE FEES
For your information -

I just had a "heads up" from King. Should we hear from the Comptroller's office to suspend the Envirocare review, we should not do so without checking with Taylor's office first. The word is that Mr. Taylor prefers some other alternative - the only alternative mentioned by King was Taylor pulling Semnani in here for a meeting.

I keep you informed should I hear more.

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6.2.1.2 Occupational Exposure

NRC's regulations in 10 CFR 20.1201 require that licensee's control occupational dose to individual adults to the more limiting of the total effective dose equivalent of 5 rems (0.05 Sv) or other dose limits described therein. The regulations in §20.1201(d) allow licensees to demonstrate compliance with the occupational dose limits using the Derived Air Concentrations (DACs) in Table 1 of Appendix B of 10 CFR 20. In addition to the annual dose limits, Part 20 requires licensees to limit the soluble uranium intake by an individual to 10 mg per week to avoid chemically toxic effects on workers (§20.1201(e)). This section describes and assesses the applicant's program for protecting workers against radiological hazards and toxic effects associated with uranium intake.

The principal pathways of occupational exposure at the applicant's proposed disposal facility for 11e.(2) waste include:

- Direct external exposure to waste and ground-deposited dust;
- Internal exposure from inhalation of airborne radioactive particulates and radon and thoron and their decay products released from the waste during receipt, storage, processing, and disposal of the waste.

6.2.1.2.1 Direct Gamma Exposure

The applicant did not employ any model or code to assess direct gamma exposure to workers. The applicant determined that it would not be practical to project potential external exposure for each planned activity. The applicant would prefer to rely on personnel monitoring to demonstrate compliance with the occupational dose limit of 10 CFR Part 20.1201 after facility operations and waste handling commences. The applicant intends to issue dosimeters to measure external radiation exposures (thermoluminescent dosimeters (TLDs)) to all staff working within the boundary of the site of the proposed disposal facility for 11e.(2) waste. Personnel will not be monitored for doses from beta radiation directly. Instead, Envirocare intends to measure beta radiation intensity at the working environment using appropriate radiation survey instruments, such as thin-window Geiger-Mueller probes, ion chambers, and scintillation detectors. Envirocare proposes to measure radiation intensity with open and closed windows and estimate the beta radiation intensity from the difference between the measurements.

The applicant has proposed that personnel monitoring not be segregated into each type of activity. In other words, Envirocare plans to use the same personnel dosimetry for each radiation worker regardless of the type of job activities and will not distinguish sources of worker external exposure. The applicant clearly stated that some of the radiation worker activities will be associated with disposal work at the NORM and Mixed waste disposal facilities located adjacent to the applicant's proposed disposal facility for 11e.(2) waste. However, the applicant also commits to ensure that the dose to any worker will not

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exceed the occupational dose limits in 10 CFR 20.1201, regardless of the alleged source of the exposure.

The applicant analyzed gamma exposure rates to workers based on operational activities at the Envirocare facility. The predicted average occupational exposure rates (mrem/yr) for each of the three activities (unloading, storing, and placing waste) was the same for the projected deep dose equivalent at about 400 mrem/yr. Details of occupational exposure rates based on activity type were provided by the applicant in Table 3.22, Appendix A, page 124.

The applicant estimated gamma exposure rates (at a distance of 1 m) from receiving the waste containing 500 pCi/g of ^{226}Ra and ^{228}Ra at 4.2×10^{-4} and 2.6×10^{-4} $\mu R/hr$, respectively [Boby, check these values; they appear to be extremely low]. The applicant projected gamma exposure rates up to 5 mR/hr. However, the applicant expects that the average exposure rate would be on the order of 0.2 mR/hr. Based on 2000 hours annual working time, the estimated deep dose equivalent from receiving the waste to be about 0.4 rem/yr.

In estimating the dose to the front end loader operator in unloading and transporting the waste, the applicant assumed a distance of 4 m between the operator and the waste. The applicant estimated a gamma exposure rate to the front end loader operator at about 400 mrem/yr.

The applicant estimated an external radiation dose rate to the equipment operator, involved in waste storage, at 100 mrem/yr. The waste disposal process requires spreading and compacting of the waste in the embankment using specialized heavy equipment. The applicant estimated an exposure rate of 400 mrem/yr for an individual operating heavy machinery over the waste surface. The applicant assumed a shielding factor of 10 times due to the steel construction of the equipment.

NRC staff reviewed the applicant's estimates of worker doses from direct gamma exposures during unloading, transporting, storage, and emplacement of the waste. At the average concentrations assumed for the facility (i.e., 500 pCi/g ^{226}Ra and other associated radionuclides), the staff estimates that the direct gamma doses may be greater than projected by the applicant. For example, using the conversion factor provided in (Ref. XX) [Boby, see NCRP report that Wrenn relied on for Molycorp's conversion factor], the exposure rate 1 m above the surface of the embankment where waste was being emplaced would be expected to be on the order of 500 $\mu R/hr$. If an individual were exposed continuously while working at the site at that location for 2000 hours per year, the approximate estimated dose would be 1 rem (0.01 Sv). In addition, NRC staff believes that applicant's assumed shielding factor is non-conservative and leads to underestimated doses from direct gamma exposure. Nevertheless, in accordance with §20.1502(a), the applicant will monitor doses from external gamma exposure to workers and is required to comply with the dose limits in Part 20. Therefore, the applicant will need to implement controls and additional protective measures (e.g., greater shielding, restricted access, time constraints) to limit external doses based on personnel dosimetry and other monitoring.

6.2.1.2.2 Internal Dose from Inhalation

The applicant has estimated annual intakes of radionuclides via inhalation by workers using the following relationship:

$$\text{Intake (pCi/yr)} = \text{Activity Concentration (pCi/m}^3\text{)} \times \text{Inhalation Rate (m}^3\text{/hr)} \times \text{Work Period (hr/yr)}$$

The activity concentration was calculated by the applicant by assuming a dust loading of 1 mg/m^3 (based on (Ref. XX)). The activity of each radionuclide in the waste is assumed to be 500 pCi/g for ^{226}Ra, ^{232}Th, and decay products in secular equilibrium. Thus, the airborne activity in the dust loading would be 0.5 pCi/m^3 for each radionuclide, which corresponds to an airborne concentration of 5.0×10^{-13} µCi/ml. The applicant, however, incorrectly calculated a concentration of 0.05 pCi/m^3 and converted it to an airborne concentration of 5.0×10^{-14} µCi/ml. The inhalation rate of the industrial worker was assumed to be 1.2 m^3/hr; assuming that occupational worker spends 2000 hr/yr on the job, the total amount of air inhaled would be 2.4×10^3 m^3/yr, which corresponds to an intake rate of 1.2×10^{-4} µCi/yr. Using this relationship, the applicant estimated the average worker intake at 120 pCi/yr. However, using the above relationship and proper conversions this intake should be 1200 pCi/yr (1.2×10^{-3} µCi/yr).

The DAC limits in Table 1 of 10CFR Part 20, Appendix A are listed in the following table for comparison with the calculated airborne concentrations:

Table 15. Comparison of the Applicants Estimated DACs with 10 CFR Part 20, Appendix B

Radionuclide	10 CFR Part 20 DACs (µCi/ml)	Applicant's Estimated Airborne Concentrations (µCi/ml)*	Estimated DAC-hours (Assuming 2000 hours per year)
^{220}Rn	1.0 WL	9E-5 WL	0.18
^{222}Rn	0.33 WL	?	?
^{226}Ra	3E-10	5E-13	3.4
^{228}Ra	5E-10	5E-13	2
^{230}Th	3E-12	5E-13	340
^{228}Th	7E-12	5E-13	140
^{232}Th	5E-13	5E-13	2000

* Based on corrected conversion of the dust loading from 0.05 to 0.5 pCi/m^3

Even though the applicant projected air concentrations less than the DAC values, Envirocare proposed standard operational procedures that would require using respiratory protection with a protection factor of at least 10 in the dust-forming areas. In the dose assessment of inhalation exposure to radionuclides in the ^{232}Th and ^{226}Ra decay chains, the applicant assumed use of respiratory protection with a protection factor of at least 10. The applicant concluded that the annual committed effective dose equivalent from inhalation of airborne particulates would be 325.2 mrem/yr. Without respiratory protection, the projected CEDE from inhalation alone would be estimated at 3252 mrem/yr. However, the applicant did not explain how these doses were derived.

The applicant's dose assessment only considered particulate releases from the low-activity waste (500 pCi/g). For handling the high-activity waste, the applicant estimated an additional dose of 130 mrem/yr. Thus the applicant's total estimated CEDE from inhalation of airborne particulates from handling and emplacement of the combined waste would be on the order of 4,553 mrem/yr without respiratory protection and 455 mrem/yr with respiratory protection.

The applicant also estimated doses from radon and thoron releases from the waste by calculating concentration levels as a function of distance and direction from the waste. The applicant employed the same procedure for calculation of exposure in terms of WL and WLM as that detailed in UDAD code manual (pages 15, and 63-64). The applicant calculated the average thoron exposure to workers (assuming continuous occupation) at 0.1 and 0.5 Km from the embankment to be about 6.1×10^{-3} and 3.2×10^{-3} WLM, with an average of 4.6×10^{-3} WLM for the facility. Considering a working period of 2000 hours per year, the applicant estimated the average occupational exposure to each worker would be 1.1×10^{-3} WLM. The applicant assumed each WLM corresponded to a 5 rem dose to the bronchial epithelium to estimate the radiation dose rate to workers [Boby, check this conversion; NCRP 78 gives a conversion factor in the range of 12.6 to 14.2 rem per WLM for radon decay products and 2.3 rem per WLM for thoron decay products]. Thus, the committed effective dose equivalent to the bronchial epithelium of individual worker was estimated to be 13.4 ?mrem/yr. Similarly, the applicant estimated the average dose rate to workers from radon decay products at 45.0 ?mrem/yr. Thus, the applicant's predicted dose rate (CED) to the bronchial epithelium for the average radiation worker from inhalation of radon and thoron at the Envirocare facility was would be about 58.4? mrem/yr. [Boby, ensure that these are all CEDEs]

The applicant committed on Appendix A, page 122 to ensure compliance with the occupational dose limits in 10 CFR Part 20 by estimating the sum of the committed effective dose equivalents from each inhaled radionuclide and show the compliance with:

$$\Sigma_i C_i/(DAC)_i \leq \text{Criteria or } \Sigma_i \text{ intake}/ALI_i \leq 1 \text{ Criteria}$$

Where C_i is the concentration (μCi/ml) of radionuclide *i* and $(DAC)_i$ is the derived annual concentration (μCi/ml) inhalation limit of radionuclide *i* listed in 10 CFR Part 20, Appendix B, Table 1, Column 3. The $(ALI)_i$ is the annual inhalation limit of intake (μCi) for radionuclide *i* as listed in 10 CFR Part 20, Appendix B, Table 1, Column

2. The applicant will also limit the duration of worker exposure during operations that result in heavy airborne dust releases and will use respiratory protection for all workers involved in the operation of dust generating machines. The applicant also reiterated that during receiving, relocating, and disposing of higher specific activity waste, workers involved in such operations will use full-face respirators with a protection factor of 50.

With respect to intake of radioactive material by inhalation, NRC states in §20.1202(b) that the total effective dose equivalent limit will not be exceeded if the sum of the deep-dose equivalent divided by the total effective dose equivalent limit and one of the following does not exceed unity:

- The sum of the fractions of the inhalation ALI for each radionuclide, or

- The total number of DAC-hours for all radionuclides divided by 2000, or

- The sum of the calculated committed effective dose equivalents to all significantly irradiated organs or tissues calculated from bioassay data using appropriate biological models and expressed as a fraction of the annual limit.

NRC staff compared the applicant's estimated internal exposures against the DACs for the different radionuclides. The applicant claimed that the estimated radioactive particulate concentrations are at the most 10 times lower than the DACs in Table 1 of Appendix B of 10 CFR Part 20. Using the corrected airborne activity concentration value of 0.5 pCi/m^3 and using the above relationships, however, the applicant's estimated airborne concentrations would be either at or slightly above the DACs for ^{230}Th and ^{232}Th. By summing the DAC-hours listed in the table above, the total exposure would be 2485 DAC-hours, which exceeds the limits in §20.1202(b). This calculation assumed continuous exposure at the calculated airborne concentrations for a 2000-hour work year, which is unlikely. Nevertheless, the calculations are based on a modeling effort that used non-conservative parameter values and has several significant uncertainties, especially close to the waste handling areas as described above.

Therefore, the use of respiratory protection and other controls (such as restricting the amount of exposure time) will likely be necessary in accordance with the provisions §20.1702 to limit airborne intakes of radioactive material. The applicant committed to measure the dust loading and airborne radioactive particulate concentrations, and radon and thoron and decay products, at the disposal site using portable generators and particulate and gas sampling. The applicant should follow NRC Regulatory Guide 8.25 (1992) for appropriate guidance on the design and implementation of an effective air sampling program in the work place. The applicant also committed to certain measures to mitigate dust suspension. These controls and measures may have to be adjusted based on operational experience, variations in waste concentrations, and personnel monitoring (including bioassay) to ensure that worker doses do not exceed the radiation protection and uranium toxicity limits in 10 CFR Part 20.

In addition, the applicant did not address the issue of the soluble uranium intake by worker, which must be less than 10 mg soluble uranium in a week in accordance with 10 CFR 20.1201(e). The applicant did not assess the solubility of the uranium that may be released from the handling the waste nor show that cumulative uptake of uranium via inhalation will not exceed the 10 mg weekly limit. The applicant needs to address this issue and propose specific actions that will be taken if uranium intakes are projected to exceed the 10 mg limit. NRC staff considers this an open issue.

6.2.1.2.3 Total Dose

The applicant will have to limit the sum of the deep dose equivalent and committed effective dose equivalent to no more than 5 rem per year from all sources under the applicant's control at the site. The analysis of doses from inhalation of radioactive materials above indicates that doses via this pathway may alone be close to the 5 rem per year limit. In addition, direct gamma doses also may be significant depending on the effectiveness of the applicant's control program. Further, operation of adjacent disposal facilities for low-level radioactive waste, mixed waste, and naturally occurring radioactive material may also contribute doses to individuals that work at the proposed site. Consequently, the applicant will need to implement an effective combination of procedural controls, mitigative measures, and personnel and environmental monitoring to ensure that total doses to workers do not exceed the occupational dose and chemical toxicity limits in 10 CFR Part 20.

6.0 RADIATION SAFETY AND HEALTH PHYSICS

6.1 Introduction

This section of the document provides the staff's review and assessments of the overall radiation safety and health physics aspects of the Envirocare license application and their compliance with appropriate regulations (specifically 10 CFR Part 20 and relevant provisions of 10 CFR Part 40, Appendix A). The major issues to be addressed in this review include: sources of radiation exposure; procedures, methods, and instrumentation to be employed; dose assessment and compliance with dose limits; and details of radiation survey and monitoring programs. The overall assessment, in this respect, will be generally concerned with protection of on-site workers and off-site members of the public, and possible contamination of the surrounding natural environment from releases of radioactive materials. Since the site-specific conditions and waste characteristics are crucial information in the overall safety assessment, such information will also be addressed in this review.

6.1.1 Waste Volume and Characteristics

6.1.1.1 Embankment Capacity and Waste Volume

The applicant has provided the following information with regard to embankment capacity, volume, and waste characteristics:

(a) The 11e.(2) waste embankment is composed of two cells. The first is the northern cell; it has a right angle triangular shape with ultimate dimensions of 634 m (2080 ft) x 457 m (1500 ft) (see Figure 1.3 of the license application). The total area for this cell is calculated to be 144,869 m^2 (1,560,000 ft^2). The second disposal cell is the southern cell; it has also a right angle triangular shape with ultimate dimensions of 669 m (2195 ft) x 481 m (1580 ft) (see Figure 1.3 of the license application). The area for the southern cell is calculated to be 160,895 m^2 (1,734,050 ft^2). The maximum depth of the embankment is 2.1 m (7 ft) below ground elevation; the maximum height is 11.3 m (37 ft) above ground elevation (page 4-4 of the license application). The average thickness of the embankment was assumed by the applicant to be 10 m (32.8 ft).

(b) The embankment's total design capacity, as reported by the applicant, is 2,523,031 m^3 (3,300,000 yd^3) (page 3-3 of the license application and Page 10, Appendix A).

(c) The average bulk density of the 11e.(2) waste was assumed by the applicant to be 1.6 g/cm^3 (100 lb/ft^3) (page 41 Appendix A).

(d) The projected annual disposal area will be 229 m (751 ft) x 168 m (551 ft), which is equivalent to 38,472 m^2 (413,800 ft^2) (Appendix A, page 8). The 11e.(2) waste will be

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disposed of in about one-third of the annual area at any one time (Appendix A, page 8).

(e) The maximum volume of waste that can be processed (i.e., received, stored, and disposed) annually is 2.835×10^5 m^3 (this figure was derived from the applicant's figure for the total mass of waste that can be processed annually, which is 4.536×10^5 tonnes (5.0×10^5 tons) (Appendix A, page 10, and assuming a waste density of 1.6 g/cm^3).

(f) The maximum volume of waste that can be disposed of annually within the embankment is 3.658×10^5 m^3 (1.292×10^7 ft^3) (Table 1.1, Appendix A).

(g) The maximum volume of waste accepted for disposal, but kept unloaded in transportation containers is 4.572×10^5 m^3 (1.61×10^7 ft^3) (Appendix A, Table 1.1).

(h) The maximum volume of waste that may be stored on site prior to disposal is 2.743×10^4 m^3 (9.687×10^5 ft^3) (Appendix A, Table 1.1).

NRC staff has reviewed the above specifications provided for embankment and operational capacity. Staff compared the specifications listed above with capacity, area, and volume estimates based on the design drawings of the embankment (Figures 3-1, 3-2, and 3-3 and Figures 4-1 and Figure 4-2 and Sheets 2 through 7 of Appendix O). NRC staff calculated the total surface area of the northern cell at 144,869 m^2 (1,560,000 ft^2) and the total surface area of the southern cell at 160,895 m^2 (1,734,050 ft^2). Thus, the total surface area of the two cells was 305,764 m^2. Considering the embankment's total design capacity provided by the applicant of 2,523,031 m^3 (3,300,000 yd^3), NRC staff calculated the average waste thickness in the embankment at 8.25m[1] (27 ft). The total capacity of the northern cell would be 1,195,169 m^3 (42,207,064 ft^3) and the total capacity of the southern cell would be 1,327,384 m^3 (46,876 192 ft^3). Considering a bulk waste

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[1]This thickness (8.25 m) was calculated based on the embankment design capacity, which is 2,523,031 m^3 (3,300,000 yd^3) for both cells (page 3-3 of the license application). Assuming the volume of waste to be disposed of annually is 4.536×10^5 m^3 (Table 1.1, Appendix A of the license application), the annual disposal area is 38,472 m^2 (page 8 of Appendix A of the license application), and the average bulk is density of the waste (1.6 g/cm^3), the average thickness of waste would be 7.37 m. Thus, the applicant used neither the average thickness as calculated from the embankment design capacity, nor the average thickness as calculated from volume of waste to be disposed annually and the annual disposal area.

density of 1600 Kg/m^3 (100 lb/ft^3)[2], the total bulk waste mass to be disposed of in both cells would be 4,036,027 tonnes (4,448,958 tons).

NRC staff concludes from this review that the applicant's design specifications and design waste capacity are generally consistent with the design drawings for the embankment and facility plans. However, the following discrepancies were noted:

(a) Although the average waste thickness in the embankment is calculated at 8.25 m, the applicant has assumed, for estimation of radon release rates, an average depth of disposed waste in each cell of 10 m. NRC staff does not anticipate that this small difference in thickness will change the specific radon or thoron flux because the applicant has shown in Appendix A (Figures 3.2 and 3.3) that radon flux is nearly independent of the assumed depth for depths in excess of about 3 m (10 ft).

(b) The applicant's conversions of waste volumes from English to Metric units are inaccurate. For example, the applicant provided the maximum volume of waste to be disposed of within the embankment as 3.017×10^6 m^3 (3.3×10^6 cubic yards). It should be noted that 3.3×10^6 yd^3 is equivalent to 2.52×10^6 m^3 based on an accurate conversion factor for yd^3 to m^3.

(c) The applicant indicated on Page 10 of Appendix A that the maximum volume of waste that may be processed (received, stored or disposed) annually is 2.835×10^5 m^3 (4.536×10^5 tonnes, assuming waste density of 1600 Kg/m^3). On the other hand, the applicant reported that the maximum volume of waste to be disposed of annually within the embankment is 3.658×10^5 m^3 (Table 1.1 page 10, Appendix A). These two figures are inconsistent. In actuality, NRC staff anticipates that the maximum volume of waste to be disposed in the cell is less than the volume of waste to be processed because some of the waste will be stored onsite or rejected.

(d) The applicant has conducted a radiological impact analysis based on the assumption that the projected annual disposal area will be about 38,472 m^2 (414,100 ft^2) with average dimensions of 229 m (750 ft) x 168 m (550 ft). The disposal rate of waste material, as estimated by the applicant, would be 453,592 tonnes/yr (500,000 tons/yr). Assuming a

[2]This density value was assumed by the applicant (Appendix A, page 41) and is consistent with NRC Regulatory Guide 3.64 (Ref.1) and NUREG-0706 (Ref. 2).

rectangular area and an average waste density of 1600 Kg/m^3, NRC staff calculated that the total waste volume disposed of annually would be 283,495 m^3. Considering an annual disposal area of 38,472 m^2, the average waste thickness would have to be 7.37 m (24.2 ft) for this rate of disposal. The average waste thickness for annual disposal is apparently less than the average waste thickness in the embankment. However, this discrepancy will not significantly affect worker or public exposures.

(e) The applicant has indicated that the initial 11e.(2) waste embankment will be constructed with a disposal capacity of 3.0 million cubic yards (page 2-5 of the license application). On the other hand, the applicant stated on page 3-3 that the above-grade embankment was designed to hold 3,300,000 cubic yards. These two statements are inconsistent. However, this discrepancy will not significantly affect worker or public exposures.

Since the applicant has based estimates of worker and public exposures on calculations that employed the above parameters, the applicant should not deviate from these design specifications, unless such deviations are supported by radiological dose assessments that consider the potential impacts associated with design and operational changes. In particular, these specifications include the following:

1. The maximum bulk mass of waste disposed of annually will not exceed $4.536X10^5$ tonnes ($5x10^5$ tons).

2. The maximum annual disposal area will not exceed 229 m x 168 m (equivalent to 38,472 m^2).

3. The 11e.(2) waste will be disposed in no more than about one-third of the annual disposal area at any one time.

4. The total embankment capacity will not exceed $2.52x10^6$ m^3 ($3.3x10^6$ yd^3).
5. The maximum volume of waste that may be stored on site prior to disposal will not exceed $2.743x10^4$ m^3 ($9.687x10^5$ ft^3).

6.1.1.2 Waste Characteristics

6.1.1.2.1 Radiological Characteristics

The applicant has described the radiological characteristics of the 11e.(2) waste, in section 17.1.1 of the license application, using available waste characteristics data for operating and non-operating uranium mill sites and three 11e.(2) waste sites where other uranium and thorium processing has taken place. The applicant provided data on the radiological characteristics of uranium mill tailings from Uranium Mill Tailings Remedial Action Program (UMTRAP) sites. In addition, the applicant used radiological data from four sites: the UMTRAP disposal site at Clive, Utah; raffinate pits at Weldon Spring, Missouri; the Kerr-McGee Rare Earths Facility in West Chicago, Illinois; and the Maywood Stepan Chemical site in Maywood, New Jersey. Using data from these sites, Envirocare generically described the physical form of the waste to be disposed disposed of in the proposed disposal facility. For example, the applicant stated that building debris, contaminated soils, and mill tailings will constitute approximately 90% of the waste (Page 17-6). The average total activity of such waste was estimated by the applicant to be below 1,000 pCi/g for any waste generator site, with the most probable average activity close to 400 pCi/g (page 17-7).

The applicant has also generically described another waste category designated as high specific activity waste. Such waste is generated from tailing waste concentrates such as sludges, slimes, and raffinate concentrates. Envirocare indicated that the weighted average ^{226}Ra concentrations in such waste must not exceed 2,000 pCi/g and average ^{232}Th concentration must not exceed 6,000 pCi/g (page 17-7). The applicant, however, assumed in the dose assessments for the proposed facility that the average concentration for any radionuclide in the high specific activity waste is 1000 pCi/g (See Appendix A, summary section, and Attachments C and D). The applicant did not provide a rationale for why the high specific activity waste will have an average concentration of 1,000 pCi/g for all radionuclides in the uranium and thorium decay series. The applicant has assumed in the dose assessments that the high specific activity waste represents 10% of the 11e.(2) waste. The applicant did not specify a maximum concentration of radionuclides in the high specific-activity waste.

Based on such radiological data, the applicant proposed the following radiological characteristics of the 11e.(2) waste:

(a) The average bulk concentration (for the disposal site) for any radionuclide in the U and Th decay chains will be 500 pCi/g or less (Page 17-7). It should be noted that the applicant has analyzed occupational and public doses based on the assumption that 90% of the waste will have an average bulk-specific activity of 500 pCi/g for each radionuclide in the uranium and thorium decay series (Appendix A, page 47).

(b) Individual shipments of waste may contain higher average concentrations of ^{226}Ra and ^{232}Th (page 17-7). The applicant emphasized, however, that the weighted average concentrations (by weight) in a shipment would not exceed 2000 pCi/g for ^{226}Ra or 6000 pCi/g for ^{232}Th (Page I-1 and 16-10). Both radionuclides may be present in the shipment at these concentrations (page 17-7).

(c) Assuming that the total bulk mass of waste to be handled annually is 451,500 tonnes (500,000 tons), and considering an average concentration of 500 pCi/g for each ^{226}Ra and ^{232}Th, the total annual activity for each radionuclide was estimated by the applicant at 227 Ci (Page 17-7).

(d) The applicant has assumed that all decay products of ^{226}Ra and ^{232}Th are in secular equilibrium. Thus, there would be 227 Ci of each radionuclide (such as ^{228}Ra and ^{224}Ra) in the two decay chains (page 17-8). However, this assumption was not apparently applied when the applicant calculated radon and thoron particulate releases (see Section 6.2.1.1 of this SER).

(e) The concentration of ^{238}U in the waste was assumed to be less than 25% of the ^{226}Ra concentration and the average ^{230}Th concentration is assumed to be equal to the ^{226}Ra concentration (page 17-8). The applicant based this estimate of the ^{238}U concentration on isotopic ratios observed at UMTRAP sites (as discussed below). The applicant estimated the ^{230}Th concentration based on the assumption that ^{230}Th was in secular equilibrium with ^{226}Ra.

NRC staff reviewed the applicant's analysis of waste characteristics by comparing proposed characteristics with the characteristics of wastes from the facilities at West Chicago, Weldon Spring, Maywood, Clive, and UMTRAP tailings in general. Based on this review, the NRC staff concluded that the proposed radiological characteristics of the waste appear reasonable for ^{226}Ra. However, for the other radionuclides in the U and Th decay series, the applicant's proposed characteristics have not been adequately justified and additional information is necessary to support the proposed characteristics. The concentration of key radionuclides in the waste may significantly affect projected occupational and public doses associated with the proposed facility. Therefore, the staff considers this an open issue.

NRC staff understands the difficulty in estimating waste characteristics for the proposed 11e.(2) waste disposal facility due to lack of prospective information and the large uncertainties associated with the available data on radionuclide concentrations and waste volumes. To a large extent, the characteristics of the waste will be known just before receiving each waste shipment, when the generators send their statements describing the radiological characteristics of the waste to be shipped along with certifications acknowledging that the material is not a hazardous waste as defined by 40 CFR 261.3.

In the absence of known radiological characteristics of the 11e.(2) waste, NRC staff has reviewed the applicant's estimates of the average radionuclide concentrations in the bulk 11e.(2) waste to be processed at the Envirocare facility. NRC staff has evaluated the radiological characteristics provided by the applicant for the UMTRAP tailings, the operating uranium mills, the UMTRAP site at Clive, Utah; the Weldon Spring Site in Missouri; the Kerr-McGee West Chicago site, and the Stepan Chemical, Maywood site in New Jersey. All of these sites are either potential 11e.(2) waste generators or contain waste similar to 11e.(2) waste. NRC staff has summarized the radiological characteristics for these sites in the following table:

Table 9

Summary of the Radiological Characteristics of Mill Tailings for Different UMTRAP sites.

Name of Tailings or Site	Mean ^{226}Ra Concentration (pCi/g)	Standard deviation of ^{226}Ra Concentration (pCi/g)	^{226}Ra Concentration Range (pCi/g)	Other Radionuclide Concentrations
UMTRAP Mill Tailings (Ref. 1)	464	509	15-2315	^{230}Th (Up to 10^4 pCi/g in slime, raffinates, and evaporation ponds). Site wide ^{230}Th is equal to ^{226}Ra ^{238}U is 8% of ^{226}Ra
Tailings at Operating Uranium Mills (Ref. 1)	319	230	87-981	na*
UMTRA Site at Clive, Utah (Ref. 2)	670	na*	460-900 (100-2000)**	^{230}Th is equal to ^{226}Ra, ^{238}U is 8% of ^{226}Ra
Weldon Spring Site, Missouri (Ref. 1)	343	216	11-460	^{230}Th: 12448; ^{232}Th: 118 ^{238}U: 556; ^{234}U: 598; ^{228}Ra: 157
Kerr-McGee W. Chicago, IL (Ref. 3)	300 (47)	na*	1.8-7526	^{238}U:43-135(83) ^{230}Th: equal to ^{226}Ra ^{232}Th: 6.6-5284 (900) ^{228}Ra: 549
Stepan Chemical Maywood, NJ (Ref. 4)	na*	na*	na*	^{232}Th: 6000 ^{238}U chain less than Th.

* na: not available **(...): Concentration ranges reported in the reference.

Considering the radiological characteristics presented in Table 9 above and the data presented by the applicant, the NRC staff concluded the following:

(a) The data presented by the applicant and the data shown in Table 9 indicate that the ^{226}Ra mean concentration is in most cases less than 500 pCi/g. Thus, the average ^{226}Ra concentration in the waste should not exceed 500 pCi/g. Because radiological impacts will depend largely on this average concentration, the applicant is required to maintain the ^{226}Ra concentration, averaged over the waste disposed of in any one year, below 500 pCi/g. This is a license condition.

(b) NRC staff realizes that ^{226}Ra concentrations may vary over a wide range. For example concentration levels of 2000 pCi/g or higher may be encountered for certain categories of 11e.(2) waste. The applicant has indicated that average concentration of ^{226}Ra above 2,000 pCi/g in any shipment will not be accepted. The applicant did not specify, however, an upper limit on the average radionuclide concentration in the high activity waste. The applicant also did not establish lower/upper concentration limits to segregate the waste from the low activity waste. Further, the applicant did not explicitly identify the maximum concentration limits for the high activity waste. Nevertheless, the applicant has assumed in the radiological assessment that the average radionuclide concentration in the high activity waste will be 1,000 pCi/g. Furthermore, the applicant has assumed that 10% of the bulk waste will be high activity waste. The applicant did not provide any data or information to support the assumption that 10% of the waste is high activity waste. Therefore, the applicant needs to resolve the following issues to provide confidence that projected doses to workers or the public will not exceed applicable limits:

(1) Provide upper and lower concentration limits for the high activity waste and provide volume and concentration data and rationale in support of the selected limits.

(2) Provide data to support the assumption that no more than 10% of the waste will be high activity waste.

(3) Establish upper concentration limits for high activity waste.

The above sub-issues associated with high activity waste are considered an open issue to be addressed and resolved by the applicant.

(c) The applicant has not shown that average concentrations of radionuclides other than ^{226}Ra will not exceed 500 pCi/g. The dose assessments depend upon the average radionuclide concentration; changes in the average concentration may affect compliance with the public and occupational dose limits. NRC staff noted that ^{230}Th concentration, for example, could be either equal to the concentration of ^{226}Ra (e.g., Clive site, Utah) or much higher by a factor of 10 to 20 times the concentration of ^{226}Ra (e.g., Weldon Spring site). Certain 11e(2) waste sources may also contain much higher levels of ^{232}Th (e.g., Kerr-McGee site in West Chicago) that could significantly increase projected doses (see Table 9 above). Such large difference could affect project doses through changes in direct gamma exposure, airborne particulate inhalation, or radon or thoron inhalation. Consequently, the applicant needs to demonstrate that average concentrations will not exceed 500 pCi/g. The applicant should discuss plans, approaches, and methodologies to be adopted to ensure the maximum average concentrations for radionuclides other than ^{226}Ra will not exceed an annual average concentration of 500 pCi/g. The applicant is required to submit plans and approaches to ensure that the maximum average concentration of low activity waste will not exceed 500 pCi/g for any radionuclide in the U/Th series. **NRC staff considers this an open issue.**

6.1.1.2.2 Non-Radiological Constituents

Envirocare stated that 11.e(2) waste would be expected to contain similar hazardous constituents to those found in uranium mill tailings, in general, regardless of the source. The applicant provided in Appendix Y of the application (Table 3) information to show typical chemical and radiological properties of tailings wastes (dry solids and liquids) generated by a model uranium mill. Additional information on elements present in tailings from acid-leach uranium mills and typical soils were also provided in Table 4 of Appendix Y.

Envirocare also provided upper ranges of elemental concentrations in mill tailings based on EPA reports. The upper ranges were provided in a table and compared with average elemental crustal concentrations (see Table 17.5 on page 17-9 of the license application). The applicant assumes that 90% of the waste arriving at the proposed 11e.(2) waste disposal facility will less than or equal to average concentrations of the constituents in the earth's crust. Examples of such concentrations given in the license application are provided in Table 10.

Table 10 - Weighted Average Concentrations of Non-Radiological Constituents

ELEMENT	CONCENTRATION (ppm)	AVERAGE CRUSTAL CONCENTRATION (ppm)
Arsenic	600	5
Barium	4000	250
Chromium	7300	200
Lead	3100	16
Mercury	34	0.5
Selenium	230	0.1
Silver	10	0.1

Envirocare pointed out that arsenic, barium, and lead, in concentrations shown in Table 9, would cause the waste to exhibit the toxicity characteristic under EPA's hazardous waste regulations under 40 CFR 261.24 (i.e., fail the Toxicity Characteristic Leaching Procedure (TCLP)). Envirocare indicated that most of the highest concentrations were found only at one mill site. Therefore, the applicant expects that the average concentrations of non-radiological constituents will be much lower. Envirocare anticipated (based on rough estimates) that actual concentrations of non-radiological constituents may be less than half of the maximum observed concentrations.

Envirocare indicated in its application that low concentrations of hazardous volatile and semi-volatile organic compounds could also be present in thorium 11e.(2) waste materials (page 17-10 of the license application). Examples of organic constituents include: Acetone; 2-Butanone; Chloroform; Carbon disulfide; 1,2-Dichloroethane; Methylene chloride; Naptha; Diethylphthalate; and 2-Methylnaphthaline.

The NRC staff reviewed Envirocare's characterization of the non-radiological constituents in the waste that would be disposed of at the proposed facility. Similar to deficiencies in its characterization of radiological constituents, Envirocare has not sufficiently estimated the non-radiological characteristics of the waste in terms of constituent concentrations. Specifically, the applicant did not present assessments or references to justify the assertion that 90% of the waste arriving at the proposed 11e.(2) waste disposal facility would be at concentrations that approximated average concentrations of the constituents in the earth's crust.

Envirocare needs to justify this assertion by presenting assessments and data as a basis for estimating the non-radiological characteristics of the waste. The staff considers this an open issue.

In the event that hazardous constituents are detected in groundwater in the uppermost aquifer after the facility begins operations, it will be necessary for Envirocare to characterize concentrations of the non-radiological constituents in the wastes. Retroactive characterization of wastes after they have already been placed in the embankment may be extremely difficult, costly, and fraught with uncertainties. Therefore, Envirocare should commit to characterizing the concentrations of hazardous constituents listed in Criterion 13 of Appendix A of 10 CFR Part 40 upon receipt of the waste, or justify, with a high degree of confidence, that the 11e.(2) waste does not contain any of the hazardous constituents listed in Criterion 13 other than the 15 constituents proposed by the applicant. The staff considers this an open issue.

NRC staff also observed that Envirocare listed in Table 17.5 an arsenic concentration value of 600 ppm. This concentration is considerably larger than would be expected for most uranium mill tailings based on NUREG-0706. Therefore, Envirocare may have considerably overestimated the average concentration of arsenic in the waste expected to be received at the 11e.(2) disposal facility.

More importantly, Envirocare assumes that the incoming waste would not need to be disposed of and regulated as hazardous waste if, when averaged over all waste received, the bulk of the waste would not exhibit the Toxicity Characteristic. Hazardous waste that contains 11e.(2) byproduct material would have to be disposed of in accordance with both NRC requirements and EPA or authorized State requirements under the Resource Conservation and Recovery Act (RCRA). Such waste may be generated inadvertently by mixing 11e.(2) waste with hazardous waste during remediation and decommissioning. Envirocare needs to assess the regulatory status of these potential mixed wastes and ensure that such wastes are disposed of in compliance with all applicable requirements.

In addition, if the applicant proposes to accept only 11e.(2) waste that is exempt under RCRA, then Envirocare should propose a mechanism for ensuring that the source of any non-radiological constituents (e.g., arsenic) disposed of at the site is only from 11e.(2) byproduct material. For example, the applicant could propose that waste generators certify that waste shipped to the disposal facility only contains 11e.(2) waste and does not contain hazardous waste regulated under RCRA. This approach would need to be supported by some mechanism for enforcing against false certifications and verifying such certifications through onsite, as-received inspections of the waste to ensure conformance with the certification description. The staff considers this an open issue.

Further, the applicant requested in the license application (Appendix Y) to reduce the number of constituents to be determined in the waste to:

- 13 inorganic constituents (Arsenic, Barium, Beryllium, Cadmium, Chromium, Cyanide, Fluorine, Lead, Mercury, Molybdenum, Nickel, Selenium, and Silver),
- 7 volatile organic compounds (Acetone, 2-Butanone, Chloroform, Carbon disulfide, 1,2-Dichloroethane, Methylene Chloride, and Naphtha),
- 2 semi-volatile organic compounds (Diethylohthalate, and 2-Methylnaphthalene), and
- 5 radionuclides (^{226}Ra, ^{228}Ra, ^{230}Th, ^{232}Th, and U).

These constituents were the only hazardous constituents identified in the tailings solutions at active uranium mills in the western United States from the list of constituents presented in Criterion 13 of Appendix A of 10 CFR Part 40. They were selected based on a survey of the composition of tailings solutions at active uranium mills. Envirocare attempted to justify its request to reduce the number of constituents to be tested in the waste based on an internal NRC memorandum that documented the results of the survey (Ref. 5) and a telephone conversation with an NRC staff member.

Criterion 13 of 10 CFR Part 40, Appendix A, provides for the determination of a comprehensive list of hazardous constituents in 11e.(2) waste. The survey results, however, are only representative of the wastes present at active uranium mills. The composition of the tailings solutions may not be representative of the chemical and radiological content of 11e.(2) waste that may be received at the proposed facility because the sources of the waste may include facilities other than the existing uranium mills (e.g., thorium processing mill, remediation projects, in-situ leach operations). In fact, Envirocare presented data on hazardous constituents present in thorium tailings (see Table 17.5 and page 17-10) that included constituents other than the constituents listed above. In addition, wastes received at the proposed facility may be expected to be somewhat more variable in composition because of the diversity of sources of the waste.

Therefore, in accordance with Criterion 5B of Appendix A of 10 CFR Part 40, the applicant will need to identify hazardous constituents present in the 11e.(2) waste, including any of the constituents listed in Criterion 13 of Appendix A. Information on the non-radiological characteristics of the 11e.(2) waste will be essential at the time hazardous constituents are identified in determining whether the constituents are reasonably expected to be in or derived from the waste.

6.1.2 Facility Operations

6.1.2.1 Generic Description of Waste Handling Operation

The procedure for accepting, inspecting, receiving, handling, storing, and disposing of incoming waste at the facility has been described fully in Chapter 16 of the license application. This section provides an overview of waste handling operations; a more detailed description and review of the waste handling operations follows.

The waste will be received either by directly unloading of the content of rail cars at the "rollover" facility, removal from rail cars at the rollover facility using a specially designed front-end loader, or directly unloaded with a front-end loader at the storage or disposal facility. Envirocare will use a dust abatement technique, consisting of a water spray application, when receiving the waste at the rollover facility. In addition, Envirocare will mitigate dusting prior to rollover and after relocation of the waste to the disposal area by applying water to dry waste.

Envirocare will also use respiratory protection for individuals unloading waste in the rollover facility (Appendix A, page 128). Thus, all individuals directly involved with the rollover procedure will use half-face masks with a protection factor of 10 (in accordance with Appendix A of 10 CFR Part 20) or greater respiratory protection (i.e., full-face mask) for unloading higher activity waste.

The applicant described the process of 11e.(2) waste handling, which will include unloading the waste into the embankment using a forklift, front-end loader, truck hydraulic lift, or through rollover of the waste onto a below-grade and sheltered concrete pad. The unpackaged waste received either by train or truck will be unloaded using a specialized front-end loader. The waste on a train will be unloaded using a specially designed train rollover facility. A front-end loader will be also used to unload prepackaged waste. Waste unloaded at the rollover facility will be transported to the storage or disposal area using trucks. The waste will be loaded onto the trucks using a front-end loader. Envirocare emphasized that front-end loader operators will wear a half-face dust mask with a minimum protection factor of 10 during removal and loading operations (Appendix A, Section 4.4.3). The truck drivers will also wear a half-face mask during the loading/unloading operations.

The applicant has established certain conditions on waste handling based on the radiologic characteristics of the wastes. These conditions include:

(a) Wastes would be segregated into two categories of specific activities: lower activity and higher activity (Appendix A, page 48). Envirocare intends to dispose the higher activity wastes (those with more than 1,000 pCi/g of any radionuclide) shortly (e.g., no more than 10 days) after acceptance. Envirocare will place such waste in the embankment region at depths more than 3.3 m below the elevation of the radon barrier at the top of the embankment. Consequently, the higher activity waste will be located closer to the bottom of the embankment. The applicant has also indicated that the higher activity waste will be covered in the embankment either with lower specific activity materials or soils (Appendix A, page 48).

(b) The applicant has indicated that in general, low activity waste received at the Envirocare disposal facility would be mostly blended together. Once received, the waste will be mixed together through the process of waste emplacement in the embankment (Appendix A, page 47).

(c) The applicant has assumed that none of the high activity waste will be stored on the site. However, the applicant has indicated that if high activity wastes must be stored, they will be covered with an additional 14 cm (6 in) of low activity waste or clean fill material to reduce gamma exposures and radon emissions (page 16-18).

(d) The applicant has indicated that only solid waste will be received and accepted for disposal. Any containers having more than 1% free-standing liquid will be rejected and returned to the originator (page 16-10). The applicant proposes to determine the amount of free-standing liquid by visual inspection of freight container and packages.

(e) Prior to shipment of the waste to the proposed disposal facility, Envirocare will obtain from the generator a description of the waste to be managed at the Envirocare facility. The description of the waste will include concentrations of radionuclides present in the waste to ensure that these concentrations are within the acceptable limits of the license (page 16-3).

The applicant described waste handling procedures for the proposed disposal facility as follows:

1. **Determination if the radioactive characteristics of the waste are within the license limits.** Envirocare has indicated that the first

step in receiving the waste is to obtain, in advance of the waste shipment, assurance from the waste generator (a form of certification) declaring that the material to be delivered for disposal at the site is within the parameters of the license. Lists of laboratory analytical data for all radionuclides present in the waste will also be provided. The generator will also acknowledge that the material is not a hazardous waste as defined by 40 CFR 261.3.

2. **Sampling of Incoming Shipment**. The next step is independent sampling of incoming waste shipment by Envirocare. For each waste stream, the minimum number of samples to be analyzed are:

 - One sample for each of the first ten shipments (rail or highway cars of waste volume approximately 100 yd^3), or

 - One sample for each 100 yd^3 up to 1,000 yd^3, and

 - Thereafter, one sample for each additional 500 yd^3 following the first ten shipments or following the first 1,000 yd^3.

 Envirocare will analyze the samples to determine concentrations of principle radionuclides in the waste to compare with reported waste concentrations.

3. **Waste Acceptance Criteria**. The third step is the determination by Envirocare whether the incoming shipment meets the waste acceptance criteria. The waste acceptance criteria include:

 (a) Proper manifests in accordance with Appendix F, 10 CFR Part 20

 (b) Each shipment of low specific activity waste will not exceed a maximum average radionuclide concentration in the shipment of 500 pCi/g

 (c) Compliance with DOT requirements for marking, labelling, and placarding

 (d) The generator has certified that the waste is 11e.(2) waste and that it does not contain RCRA hazardous waste as defined by 40 CFR Part 261

 (e) Average concentration in each individual shipment will not exceed 2,000 pCi/g of ^{226}Ra and 6000 pCi/g of ^{232}Th.

 (g) Laboratory results of radiological analysis have been received from the generator within 45 days of the arrival of the shipment

(h) The waste does not contain in excess of 1% of free standing liquid

The applicant has provided (Pages 16-4 to 16-6) a comprehensive description of the procedure to be followed by Envirocare in the event that the waste received or disposed of was found to be (e.g., from laboratory results) "Non-Conforming waste". The applicant has assumed that the average time required for processing a shipment prior to disposal (i.e., the difference between arrival time of shipment and the time when shipment is ready for disposal or storage) will not exceed 10 days.

4. **Transportation**. Waste will be loaded at the generator site and shipped to the proposed dipsosal facility using either rail or highway transportation vehicles. The waste will be contained in either bulk rail, bulk highway shipments, or in boxes of capacity 20 or 3.5 yd^3. Drums, barrels and/or bags may also be used to contain waste during shipment.

5. **Review of the "Radiation Waste Shipment & Disposal Record" (RSR).** The RSR will serve as means of compliance with the requirements outlined in 10 CFR 20.2006. The RSR will include the following information:

   a) Description of container type, volume, and number.

   b) Estimated weights and activities for all materials and isotopes.

   c) Generator's certification of packaging, classification, markings, labels, conditions of container and compliance with the applicable regulations and license terms.

   d) Generators warranty that information provided in the RSR is correct.

   e) Check-list for inspection.

   f) Generator's certification that the waste is 11e.(2) waste and does not contain RCRA hazardous waste as defined by 40 CFR Part 261.

   g) Identification of parameters to be analyzed by an independent third party.

   h) Documentation of Envirocare's acceptance or rejection of the shipment.

6. **Container Survey and Inspection**. The freight container will be externally surveyed for gamma radiation with a micro-R meter to assure compliance with DOT regulations. The external radiation exposure should not exceed 1,000 μR/hr at any point of the surface. ^{226}Ra concentration (pCi/g) in the shipment will be estimated roughly as 2.5 X μR/h. Each container will also be smear tested to ensure that any removable contamination on any portion the surface will not exceed 0.01 μCi/100 cm^2. Visual inspection will be conducted to ensure compliance with DOT regulations in 49 CFR and to verify integrity of the container and absence of any physical damage.

NRC staff has reviewed Envirocare proposed procedures for inspecting and accepting incoming waste shipments by comparing with applicable NRC regulations and provisions in 10 CFR Part 20, Subparts H, I, J, K, and L. The staff also compared the applicant's procedures with practices conducted at other operating, licensed waste disposal facilities. Staff concludes that Envirocare's protocols and procedures are consistent and comply with NRC regulations.

NRC staff noted, however, that the following information was not addressed by the applicant:

1. Waste manifests should include information on the principal chemical form, solidification agents (if any), and chelating agents and their identification, if the waste contains more then 0.1% of such agents (Criterion I, Appendix F, 10 CFR Part 20)

2. Envirocare needs to acknowledge receipt of the waste within a week of receipt by returning a signed copy of the manifest or equivalent documentation. (Criterion III-B.1 of Appendix F).

3. Envirocare needs to comply with the provisions for recordkeeping and tracking in Criterion III (Control and Tracking) of Appendix F of 10 CFR Part 20.

4. Envirocare needs to describe an appropriate method for determining the amount of free liquids in the waste. The applicant has committed to ensure that no more than 1% free liquids will exist in waste accepted for disposal. The applicant's proposed method for determining free liquids is by visual inspection. However, it is unlikely that visual inspection will be sufficient to ensure compliance with this waste specification.

5. Quality control checking to ensure compliance with waste acceptance criteria should also be addressed (Appendix F, Criterion III-A-3). Thus, Envirocare should describe its quality controls for waste sampling, characterization and classification as high specific activity or low specific activity waste. Controls for the quality of the protective equipment (e.g., anticontamination clothing and equipment that meets the ANSI Z-88.2 (Ref. 6)) and the respiratory protective equipment, including a respiratory protection program that satisfies the guidance of Regulatory guide 8.15 and NUREG-0041 (Ref. 7).

These issues need to be resolved prior to waste receipt and processing to ensure compliance with 10 CFR Part 20. Therefore, staff considers these items as open issues.

6.1.2.3 Disposal Site Preparation and Site Services

The applicant has described disposal site preparation activities, which entail excavation of the disposal site, preparation of the clay liner, and construction of the embankment. The site service activities include application of wetting agents, cleaning of trucks and trains after unloading, equipment services, and laboratory services for waste characterization. Other activities included in site services are sampling and engineering control measurements. Individuals performing these activities are not directly involved in waste processing for disposal. However, these individuals would be exposed to radiation from direct exposure to adjacent sources and from inhalation of airborne radioactive materials.

The NRC staff reviewed these proposed activities and determined that they will not significantly affect worker or public exposures to radioactive materials or radiation.

6.1.2.4 Waste Handling, Interim Storage, and Disposal

The applicant has described waste handling activities as the processing of accepted shipments of two different categories of wastes:

- Low-activity waste -- waste with radioactivity less than 1,000 pCi/g of ^{226}Ra, ^{230}Th, or ^{232}Th or which have a gamma exposure rate no greater than background levels when measured at 1 m from any surface of the container, and

- High-activity shipments -- waste with radioactivity above 1,000 pCi/g or that has a gamma exposure rate in excess of background levels measured at 1 m from the surface.

Waste disposal in the embankment involves spreading and compacting of the waste using specialized heavy mechanical equipment. Envirocare has specified the types and use of such equipment for waste disposal in Appendix A, Sections 3.3 and 3.4. Placement and compaction of waste in the embankment will generate dust. The applicant has committed that equipment operators involved in the waste disposal will wear half-face respirators with a protection factor of 10 (in accordance with Appendix A of 10 CFR Part 20). Full-face respirators may be used, as necessary, to provide greater protection during the waste disposal operations, particularly for high-activity waste. After emplacement of the waste in the embankment, the applicant proposes to place a soil cap (i.e., radon barrier) on top of the compacted waste once the embankment is filled to design height. This operation will generate dust during spreading and compaction of the soil to be placed over the waste.

Low-Activity Waste Handling

a) A rail car of waste may be emptied (dumped) using the on-site railcar dumper. The dumped waste may be hauled by truck or backhoe to the disposal embankment or may be collected and placed in containers for storage. The applicant did not specify the type of containers to be used for storage.

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

c) A dump truck or sea-van may dump the material directly into the cell or the sea-van may be stored.

d) A rail car or a trailer with containers may be off-loaded at the rollover facility and transported for either storage or direct disposal.

e) Unloaded bulk shipments may be stored in bulk or in containers.

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

High-Activity Waste Handling

a) The rail cars or trucks will be identified on arrival by placing a "Caution, Radiation Area" sign, as described in 10 CFR 20.1902, on both sides of the transporation container (e.g., rail car, truck).

b) If there will be a delay of more than 24 hours in emptying the load, the container will be roped off to control access to the radiation area associated with the high-activity shipment.

c) All workers involved in unloading or emplacement of the high-activity waste will be required to wear full-face respirator providing a respiratory protection factor of 50.

d) If there is a need for storage of the high-activity waste, the bulk waste will be covered with additional 14 cm (6 in) of low-activity waste or clean fill.

e) When placed in the disposal cell, high-activity waste will be covered with low-activity waste or 14 cm (6 in) of clean fill after the final compaction.

f) High-activity waste will not be placed closer than 3 m (10 ft) below the base elevation of the radon barrier.

The applicant proposed to store waste in bulk form (e.g., waste piles) or in containers like drums and barrels. The bulk material may be stockpiled or unloaded and stored in waste piles prior to emplacement. Storage will be located on the unexcavated portion of the disposal area near the working area, where waste is being actively emplaced. The storage area must be scarified and recompacted and a liner be placed on the compacted clay prior to placement of waste for storage. The applicant has proposed application of a polymer dust suppressant on waste piles in storage to minimize resuspension of radioactive particulates. Containers will be placed on pallets and stacked no higher than 2 high. The applicant has proposed to inspect visually containers in storage on monthly basis. The applicant has committed in Appendix A that the waste will not be stored onsite for longer than 90 days.

The applicant has proposed to locate and mark the embankment using four permanent survey monuments (USGS "brass caps"). These monuments will be surveyed into the USGS and the National Geodetic Survey (NGS) control networks and the State Plane Coordinate system. No site markers will be placed during the operational phase of the embankment. However, upon completion of either disposal cell in the embankment, the site will be marked by one site marker at the entrance to the site and one site marker near the center of the crest of the embankment. Detailed descriptions of the markers are provided in section 16.2.4.3 of the license application.

The applicant has also committed to prepare and retain records on the details of disposal material location using a grid location (N-S, E-W and depth coordinates) and waste description (specifically, radionuclide concentration, name of generator, and transport vehicles used). Envirocare has committed to maintain records in the Engineering office in compliance with the disposal record-keeping requirements of 10 CFR 61.80(f).

The NRC staff has reviewed the applicant's proposed procedures for waste handling, storage, emplacement, and embankment marking. The proposed procedures were compared against the requirements in 10 CFR Part 20 and pertinent sections of 10 CFR Part 61 (with respect to recordkeeping and site markers). Staff compared Envirocare procedures and approaches with NRC regulations promulgated in 10 CFR 20, Subparts G, H, I, J, K. L, and M. NRC staff also compared the proposed procedures with conventional industry practice at other licensed radioactive waste management facilities to ensure protection of occupational workers and the environment. Based on this review, the staff concluded that the proposed procedures are acceptable and comply with NRC requirements in 10 CFR Part 20 and Part 61. However, as discussed below, these procedures need to be accompanied by appropriate monitoring and radiation protection procedures to ensure adequate protection of workers and members of the public. Modifications to these procedures may be necessary depending on the monitoring results.

6.1.2.5 Decontamination and Release of Vehicles, Packages, and Equipment

The applicant has committed that all vehicles and equipment will be visibly clean before leaving the site. All potentially contaminated materials will be removed by a broom, shovel, or other means (like using high pressure water). The Radiation Safety Officer will supervise the release surveys (using alpha-activity and gamma exposure rate measurements) and quality control to ensure that all items to be released are in compliance with the release policy.

Envirocare proposes to classify vehicle and equipment releases into two categories: limited release and unlimited release. The limited release applies to equipment or vehicles that are frequently used on and off the site. For example, such equipment includes groundwater sampling equipment, water trucks, dedicated railcars, and closed boxes used for waste shipment. The unlimited release, however, applies to equipment that will leave the site and will probably not used again on the site. Examples of such equipment are: construction equipment (other than dedicated heavy equipment for waste transportation, unloading, and placement), and other vehicles, packages, and equipment not identified for limited release.

The applicant indicated that any equipment or packages released for limited release will meet DOT standards. The applicant, however, did not specify these standards. In addition, the applicant did not commit to meet specific limits for fixed and removable contamination on the surfaces of the equipment, vehicles, and packages.

For the unlimited release limits (see section 17.4.7.1), the applicant committed (via a corrected page insert) to release limits derived from NRC Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors" (Ref. 7).

NRC staff has reviewed the proposed procedures for decontamination and release of equipment, vehicles, and packages. Based on this review, the NRC staff believes the proposed procedures are generally adequate with respect to radiation protection aspects of the decontamination and release procedures. The applicant has also addressed provisions required by 10 CFR 20.1906 for survey of packages and materials. The applicant's approaches and procedures appear to be in conformance with 10 CFR 20.1906.

With respect to releasing materials from the site, the applicant commits to ensuring that all materials (except for rejected waste shipments) are "visibly clean" prior to release. However, the applicant did not commit to specific limits for surface activity for the so-called "limited release." Release of equipment, vehicles, or packages off the licensed site should only occur after the applicant ensures that contamination on the surfaces of the equipment, vehicles, or packages has been reduced in accordance with appropriate limits (e.g., Regulatory Guide 1.86 (Ref. 7)). Otherwise, the applicant would potentially be violating the provisions of Subpart I of 10 CFR Part 20 by allowing removal of licensed material to unrestricted areas without ensuring appropriate controls and surveillance. The applicant needs to propose appropriate limits for the "limited release" application. The staff considers this an open issue. Depending upon the limits proposed by the applicant for the "limited release," the applicant may no longer wish to distinguish between limited and unlimited releases.

6.2 Potential Radiation Exposure

The potential doses from intake of radioactive materials and exposure to external radiation need to be assessed in evaluating the safety and environmental protection performance of the proposed facility. NRC requirements in 10 CFR 20.1302(b) require that licensees show compliance with the annual dose limit to members of the public in 10 CFR 20.1301 by either (1) demonstrating by measurements or calculations that the effective dose equivalent to the individual, likely to receive the highest dose from the licensed operation, does not exceed the annual dose limit in 20.1301, or (2) demonstrating that the annual average concentration of radioactive materials released at the unrestricted area boundary does not exceed the values specified in Table 2 of Appendix B to 10 CFR 20. In addition, licensees need to comply with the occupational dose limits in 10 CFR 20.1201 and other provisions of Subpart C of 10 CFR Part 20.

Although compliance with the occupational dose limits is routinely demonstrated through the use of personnel dosimetry, radiological dose assessment may be used prospectively to evaluate the safety aspects of proposed activities. For both occupational and public protection, assessment of potential radiation exposure is also useful in selecting appropriate operational conditions and design parameters.

The potential radiation exposures at the proposed facility may be classified into two categories based on the type of radionuclide releases and direct radiation. The first category is associated with normal operational conditions where exposures are caused from normal releases of radioactive materials or gamma exposure at the site. The second category of radiation exposure results from unusual operational conditions or accidents at the site. NRC requirements in 10 CFR Part 20 apply to normal operating conditions. Staff also reviewed potential exposures from the second category to ensure that such events are unlikely or their consequences have been appropriately mitigated.

6.2.1 Exposures from Normal Operating Conditions

6.2.1.1 Potential Doses to Members of the Public

The applicant has presented in Appendix A a comprehensive analysis of potential environmental pathways of public exposure at the site boundary and in the surrounding area within an 80 Km (50 mi) radius. This section presents Envirocare's analysis of mechanisms of radiation exposure at the site boundary (for individual members of the public) from environmental pathways, such as atmospheric releases of radioactivity, atmospheric transport, and particulate deposition or inhalation.

Envirocare depicted two mechanisms of radiation exposure to potential members of the public including: (1) internal exposure resulting from inhalation of airborne materials and ingestion of contaminated food (specifically meat consumed from livestock grazing on contaminated soils), and (2) external exposure resulting from exposure to the airborne materials and to ground-deposited contamination. The applicant has considered the direct gamma exposure mechanism, from waste on site, as insigificant because the direct gamma exposure rate is greatly attenuated by the great distance between the waste embankment and the boundary of the site. The applicant also considered that ingestion of contaminated drinking water and exposure pathways resulting from soil irrigation are also not viable because of the inferior quality of groundwater and the unlikely potential for recharge at the site.

The major sources of exposures resulting from radionuclide releases under normal operating conditions were summarized in section 17.1.3 of the license application. These sources include:

(1) Exposures from inhalation of radon (^{222}Rn) and thoron (^{220}Rn) released from:

- uncovered waste,
- covered waste, and
- waste processing (e.g., waste unloading, storage, and emplacement).

(2) Exposures from inhalation of radioactive particulates released into the atmosphere from:

- waste unloading,
- waste storage,
- high-activity waste emplacement, and
- low-activity waste emplacement.

(3) Exposures from direct external radiation emitted from:

- Ground-deposited radionuclides, and
- Airborne radioactive particulates.

(4) Internal exposure from ingestion of contaminated foods.

The applicant has employed six assessment steps in estimating the total effective dose equivalent (TEDE) to an individual member of the public at the site boundary and the population dose to the public residing within 80 Km (50 mi) from the proposed disposal facility. NRC's requirements in 10 CFR Part 20 limit individual doses, but do not specifically restrict population doses. Therefore, population dose estimates will not be considered in this assessment.

The six assessment steps include:

1. Estimation of radon (^{222}Rn) and thoron (^{220}Th) releases.
2. Estimation of rate radioactive particulate releases.

3. Estimation of concentration of airborne radioactive particulates.

4. Calculation of surface activity from ground deposition of airborne radioactive particulates.

5. Calculation of radiation doses for individual pathways of exposure.

6. Calculation of the total effective dose equivalent (TEDE) by summing doses from all pathways of exposure.

Each assessment step mentioned above will be discussed separately.

6.2.1.1.1 Estimation of Radon (^{222}Rn) and Thoron (^{220}Rn) Release Rates

Radon (^{222}Rn) and thoron (^{220}Rn) gases will be released in the atmosphere from decay of ^{226}Ra and ^{224}Ra, respectively. The applicant has analyzed radon and thoron releases for each of the following sources:

6.2.1.1.1.1 Radon and thoron releases from uncovered waste in the embankment

Radon and thoron releases from uncovered waste in the disposal cell are largely dependent on the following parameters:

<u>Physical characteristics of the waste</u>

The applicant has provided justifications for selection of parameters that affect the release of radon and thoron by comparing such parameters with measured values for tailings in different sites. Based on this justification, the applicant has selected the following values for the major physical parameters:

- Emanating power for radon and thoron: 0.25 and 0.1 respectively;
- Bulk diffusion coefficient: 2.5×10^{-3} cm^2/s;
- Porosity: 0.3;
- Bulk density: 1.6 g/cm^3;
- Moisture content in waste and cover: 10 and 18%, respectively;
- Compaction of tailings and cover: 90 and 95%, respectively; and

- Thickness of waste to be disposed in the cell: 10 m (33 ft).

<u>Concentrations of ^{226}Ra and ^{224}Ra in the waste</u>

The applicant has assumed that 90% of the waste (categorized by the applicant as low-activity waste) will have an average bulk-specific activity of 500 pCi/g for ^{226}Ra and ^{224}Ra. The remaining 10% of the waste is categorized as high-activity waste with bulk-specific activity of more than 1000 pCi/g of any radionuclide in the U or Th decay chains. This waste will not be placed within 3.3 m of the elevation of the radon barrier.

NRC staff reviewed the physical parameters selected by the applicant and compared them with the generic parameters reported in NRC Regulatory Guide 3.64 (Ref. 8), parameters that are commonly selected for the generic assessments of radon and thoron releases from uranium milling (Ref. 9), and parameters selected in the design and analysis of the proposed embankment design for the uranium tailings from the Vitro site (Ref. 10). Based on this information, the staff prepared the following table (Table 11), which compares the physical parameters selected by the applicant with the parameters reported in references 8, 9, and 10.

Table 11

Comparison of the Physical Parameters for Radon and Thoron Releases

Parameter/Unit	Regulatory Guide 3.64	Envirocare Value	Vitro Site Value	FGEIS Values
Bulk Density (g/cm^3)	1.6	1.6	1.5	1.6
Decay Coeff. (s^{-1}) Radon Thoron	2.1E-6 1.27E-2	2.1F-6 1.27E-2	2.1E-6 1.27E-2	2.1E-6 1.27E-2
Emanating power (dimensionless)	0.35	0.25	0.22	0.2
Moisture Content (Wt. %)	6	10	10	10
Porosity (dimensionless)	0.4	0.3	0.43	na
Diffusion Coeff. (cm^2/s)	na	2.5E-3	2.3E-2	1.4E-2

* na: not available.

From the above comparison, NRC staff concludes that some of the parameters selected by the applicant are not sufficiently conservative or necessarily representative of conditions at the proposed disposal facility. The applicant's proposed values are acceptable for the bulk density, the radioactive decay coefficients, and the emanating power. The remaining parameters, specifically moisture content, the porosity, and the diffusion coefficient, however, are not sufficiently conservative nor have they been justified based on site-specific analysis. The values selected by the applicant for these parameters may result in underestimating the release rates of thoron and radon at the site, thereby resulting in lower projected doses to workers and any member of the public at the boundary of the facility.

Therefore, the applicant should select conservative physical parameter values or justify alternative values based on site-specific analysis. NRC staff considers selection of conservative physical parameters or alternate values based on a site-specific analysis in the calculations of radon and thoron release rates an open issue.

Calculation of the Diffusion Coefficient and Specific Flux

As described above, the applicant selected a diffusion coefficient based on values identified in previous studies. The applicant did not follow the procedures described in NRC Regulatory Guide 3.64, which indicates that if measurements for the radon diffusion coefficients are not available, the coefficients should be estimated using the correlation function:

$$D = 0.07 \exp(-4(m - mn^2 + m^5))$$

Where D is the diffusion coefficient; m is the moisture saturation fraction; and n is the porosity.

NRC staff calculated the diffusion coefficient using a calculated saturation fraction of 0.24 (calculated by assuming a porosity of 0.4; bulk density of 1.6 g/cm^3; moisture content of 6%; and using equation 8 in the Regulatory Guide 3.64). Based on these assumptions and the equations above, the calculated value for the diffusion coefficient is 3.1×10^{-2} cm^2/s. In contrast, the applicant proposed a value of 2.5×10^{-3} cm^2/s, which would result in considerably lower estimated radon fluxes. Therefore, the applicant should either select a conservative diffusion coefficient of similar value or measure the diffusion coefficient for the waste. NRC staff considers this is an open issue.

The applicant calculated radon and thoron specific fluxes for various assumed values of emanating power, diffusion coefficient, density, and porosity. The applicant used the mathematical formulation reported by Kraner et. al. (1964) (Ref. 10); Momeni et. al. (1979) (Ref. 11); and

Rogers and Nelson (1981) (Ref. 12). These formulations were provided on pages 31 of Appendix A of the license application. The following formulation was used by the applicant for the uncovered waste:

$$\phi_t(Z) = \frac{10^4 \epsilon \rho_t [Ra] \sqrt{\lambda D_t / P_t}}{\tanh(Z) \sqrt{\lambda P_t / D_t}}$$

Where $\phi(Z)$ is the flux at the surface of the waste of thickness Z, ϵ is the emanation power, ρ_t is the bulk density of the tailings, [Ra] is the concentration of ^{226}Ra in the waste (pCi/g), λ is the decay constant of ^{222}Rn, D_t is the bulk diffusion coefficient of radon in the tailings, and P_t is the bulk porosity of the tailing.

The equation used to calculate radon flux from the bare (uncovered) tailings in the NRC Regulatory Guide 3.64 is:

$$J_t = 10^4 R_t \rho_t E_t \sqrt{\lambda D_t} \tanh(x_t \sqrt{\lambda / D_t})$$

Where J_t is the radon flux from the bare tailings source (pCi/m^2.s); R_t is the specific activity of ^{226}Ra in the tailings (pCi/g); ρ_t is the dry bulk density of the tailings; λ is the ^{222}Rn decay constant; D_t is the diffusion coefficient of radon in the total pore space of the tailings (cm^2/s); and x_t is the thickness of the tailings. The two equations are not consistent. Therefore, the applicant needs to explain the derivations of the equation presented on page 31 of Appendix A, and to compare this equation with equation 9 of NRC Regulatory Guide 3.64. NRC staff considers the mathematical formulation of radon flux calculation for the uncovered waste as an open issue.

The applicant used the above (non-conservative) physical parameters and the erroneous mathematical formulation to calculate radon and thoron fluxes. The applicant's calculations showed average specific fluxes for radon and thoron around 0.8 pCi/m^{2}s and 80 pCi/m^{2}s, respectively.

The NRC staff estimates, using equation 9, of Regulatory Guide 3.64 that radon and thoron fluxes may be more than ten times larger that estimated by the applicant. For example, using the standard relationship between ^{226}Ra concentration and radon flux of approximately 1 pCi/m^{2}s per pCi/g, the estimated radon flux for 500 pCi/g ^{226}Ra would be on the order of 500 pCi/m^{2}s (Ref. 14). Therefore, the applicant needs to re-calculate radon and thoron fluxes using conservative physical parameters and appropriate mathematical formulations, such as the reference values and equations in NRC Regulatory Guide 3.64 or measured parameters for bulk average 11e.(2) waste to be disposed of at the site.

The applicant calculated the radon and thoron specific fluxes for various waste thicknesses containing 1 pCi/g of ^{226}Ra. In these calculations, the applicant varied the diffusion coefficient in the range $2x10^{-3}$ to $6x10^{-3}$; the porosity in the range 0.25 to 0.5; and the emanating power in the range 0.2 to 0.3. The radon flux calculated under these variable conditions was in the range 0.48-0.77 pCi $^{222}Rn/m^2s$ per pCi/g ^{226}Ra. Thoron specific flux calculated for 0.4 cm thick waste containing 1 pCi/g of ^{224}Ra was determined in the range 44-76 pCi $^{222}Rn/m^2s$.

Calculation of Release Rates

Finally, the applicant calculated the radon and thoron annual release rates by assuming an average radon and thoron specific fluxes of 0.51 and 78 pCi/m^2s, respectively. The applicant also assumed only one-third of the disposal area would be uncovered at any one time and the disposal operation will be carried out nine months per year. Thus, the area (12,767 m^2) was multiplied by a factor of 9/12 to obtain a figure of $9.582x10^3$ m^2 for the uncovered area, which emanates radon and thoron. For the radon release rate calculation, this figure was multiplied by the specific flux (0.51 pCi/m^2s) and then by the average concentration of ^{226}Ra of 500 pCi/g. The release rate was then calculated at 2.45 $\mu Ci/s$. For thoron, the same methodology (assuming an average thoron specific flux of 78 pCi/m^2 and uncovered area of $9.582x10^3$ m^2) was applied, and the release rate was calculated at 192 $\mu Ci/s$. Using this approach, the applicant concluded (Appendix A, page 57) that the annual expected radon and thoron releases from uncovered waste in the embankment would be 38.4 and 3000 Ci ^{222}Rn and ^{220}Rn, respectively.

The applicant used larger assumed release rates for radon and thoron in modeling atmospheric transport using the MILDOSE-AREA and TDAD codes, respectively. For ^{222}Rn releases, the applicant assumed a value of 77.5 Ci/yr in the MILDOS-AREA code. The applicant used a value of 3408 Ci/yr for ^{220}Rn in the TDAD code[3].

NRC staff reviewed the applicant's calculations of estimated radon and thoron release rates by using the same calculational methodology to confirm the applicant's calculations. Using the release rates given in Table 3.8 of Appendix A (2.45 and 192 $\mu Ci/s$), the annual releases of radon and thoron would be 77.26 and $6.05x10^3$ Ci, respectively. Thus, the annual release rate input values to TDAD code should be nearly twice of the input value used by the applicant. Further, NRC staff considers

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[3]The applicant assumed that the disposal area is composed of 16 segments of source areas. Each segment contributes 213 pCi/y to the total thoron annual release from the disposal area.

the applicant's selection of one-third of the annual disposal area to calculate the annual release rate for radon and thoron is not conservative because the applicant has not proposed specific procedures that would restrict the open area of the embankment. In addition, radon and thoron would continue to be released over the inactive, exposed portions of the embankment, wherever the final radon barrier has not been placed. Indeed, the applicant used an annual disposal area of 38,000 m^2 for radon and thoron atmospheric concentration calculations (using MILDOS-AREA and TDAD). However, for calculation of the input values of the annual release rate of the source, only one-third of that area was accounted for. NRC staff made rough estimate of ^{222}Rn releases based on a generic value of radon flux for bulk waste of 1 pCi/g per pCi/g of ^{226}Ra (Ref. 14). Considering an annual disposal area of 38,300 m^2, and assuming an average concentration of ^{226}Ra of 500 pCi/g, the total annual radon release would be around 600 Ci. Assuming one third of the area will be exposed at any one time, annual radon releases would be a minimum of 200 Ci (in contrast to the applicant's estimates of 77.5 and 38.4 Ci). Therefore, the applicant's reported estimates for annual radon releases are not conservative and unrealistic. Based on the NRC staff's analysis, the applicant should re-examine and verify his calculations of radon and thoron release rates. **NRC staff considers this an open issue.**

6.2.1.1.1.2 Radon and Thoron Releases from Receiving and Unloading the Waste

The applicant has employed the following assumptions in calculating radon and thoron release rates at the rollover pad:

- The time period between receiving and unloading the waste is 10 days.
- The total mass of waste to be handled annually at the rollover pad is $4.5X10^{11}$ g.
- The fraction of radon or thoron generated is 1.0 (i.e., all radon and thoron produced via decay is released).
- The emanating powers of radon and thoron are 0.25 and 0.1, respectively.
- The decay factors (λ) for each radon and thoron are: 0.181 d^{-1} and $1.09x10^3$ d^{-1}, respectively.
- The ^{226}Ra and ^{224}Ra concentrations in 90% of the waste are 500 pCi/g and in 10% of the waste are 1000 pCi/g.

NRC staff reviewed the above assumptions based on the general conventional practices in mill tailings operations (Ref. 9) and on the generic assumptions made by the applicant for the facility operation (see section 6.1.2 of the SER for specific details). Based on this review, the applicant's assumptions appear to be reasonably conservative, especially in assuming that the fraction of radon and thoron released is 1.0. The applicant's assumed emanating powers of radon and thoron, however, are not sufficiently conservative as was discussed in the previous section.

<u>Calculation of Release Rates</u>

The applicant calculated the radon and thoron release from receiving and unloading the waste using the relationship:

$$Q = \epsilon[Ra]\ M\ f\ (1-e^{-\lambda T})$$

where Q is the release, ϵ is the emanating power, M is the mass of waste, f is the fraction of radon generated, λ is the decay factor, and T is the waste processing time. The above equation was approximated by the applicant to:

$$Q = \epsilon[Ra]\ Mf\lambda T$$

to calculate radon release and to:

$$Q_{max} = \epsilon[Ra]Mf$$

to calculate thoron release. Using the above equations, the applicant obtained release values from waste unloading of 119.96 and 24.75 Ci/yr for radon and thoron, respectively.

NRC staff reviewed the calculated release rates by conducting the calculations using the above equations and the same assumptions proposed by the applicant. NRC staff compared the applicant's approach for calculating the release rates with the approach used by NRC in Ref. 9. Based on this review, the NRC staff confirmed the estimated release rate for radon from the unloading operation. However, for calculating the thoron release, the NRC staff noted that the ratio of releases for radon and thoron appeared to be inverted and may have been caused by the applicant's manipulation of the equations. The staff expects that thoron releases would exceed those of radon given the ratio between the decay constants and the specific fluxes for the two radionuclides. Therefore, the applicant needs to justify eliminating the decay constant and time parameters from the equation and reassess potential thoroon releases from unloading operations. **NRC staff considers this an open issue.**

NRC staff also examined the use of the applicant's values as input in the dose assessment codes MILDOS-AREA and TDAD for annual radon and thoron releases from the waste unloading operations. The applicant used 115 Ci/yr for radon releases in the MILDOS-AREA code and 25 Ci/yr for thoron releases in the TDAD code. NRC staff review indicates that the applicant's calculations for radon release are correct and consistent with values estimated using the methodology presented in Ref. 9. Therefore, the annual radon release rates from the unloading operation is accepted by NRC staff.

In contrast, the thoron release rates need to be justified. If the same equation is used for estimation of thoron release rate (i.e., $Q = \epsilon[Ra] Mf\lambda T$), the estimated thoron release rate will be 2.45×10^5 Ci, which is also 10,000 times larger than the value calculated and assumed by the applicant. [Boby check this]

6.2.1.1.1.3 Radon and Thoron Releases from Storage

For estimating radon and thoron releases from waste storage, the applicant assumed that 10% of the waste (equivalent to 4.4×10^{10} g annually) will be stored onsite awaiting disposal or processing. The applicant also assumed the storage period would not exceed 90 days. The applicant presented the estimated release rates on page 74 of Appendix A. The applicant presented, without analysis, estimated release rates of radon and thoron during the waste storage period as 5.62 and 2.3 Ci/yr for radon and thoron, respectively. The applicant also stated (Appendix A, page 74) that the annual release rate for radon during storage 86.9 Ci/y and 1600 Ci/y for thoron. The applicant contemplated that storage of waste increases releases of radon and thoron into the environment. The applicant did not explain the origin or the discrepancy between the two sets of values for the release rates of radon and thoron.

NRC staff reviewed the above release rates and estimated release rates using the approach used in Ref. 9. NRC staff examined the consistency of the release rate inputs in the dose assessment codes (MILDOS-AREA and TDAD) and those reported by the applicant in Appendix A (page 74). The applicant used annual release rates of 100 and 1600 Ci/yr for radon and thoron, respectively, in assessments of atmospheric transport using the MILDOS-AREA and TDAD computer codes. The applicant used the same values reported in the text and those used as inputs to the codes.

NRC staff analyzed the basis for the applicant's estimate of the release rates. The staff's analysis indicates that the applicant presumes a two-step process for waste storage. First, the waste is transported from the unloading to the storage pad within a day. Second, the waste will be stored at the storage pad for a period not to exceed 90 days. For the first step, the applicant assumed a one-day operation, and the mass of waste is 4.5×10^{10} g annually. The applicant apparently used the equation:

$Q_{max} = \epsilon[Ra]Mf$

to calculate the release rates for both radon and thoron for the transportation step. The applicant assumed a mass of waste of $4.5x10^{10}$ g annually and emanating powers of 0.25 and 0.1 for radon and thoron, respectively. For the second step, the applicant used for radon release rate calculation the equation: $Q = \epsilon[Ra]\ Mf\lambda T$. NRC staff independently calculated the release rates from using the same assumptions and formulations and obtained a comparable value for the radon release r... NRC staff calculated an annual radon release rate of about 92 Ci vr. This value is close to the value reported by the applicant 86 9 Ci/y. For thoron release, the applicant reported a release rate of 1600 Ci/y. NRC's estimate of the thoron release rate, however, is considerably different. The staff used the above two equations to calculate thoron release rate and obtained values of $2.2x10^{5}$ Ci/y (using the equation $Q = \epsilon[Ra]\ Mf\lambda T$). Therefore, it is apparent that the applicant is not consistent in his assumptions for calculation of thoron release rates. The applicant needs to reassess the estimated release rate for thoron from waste in storage and justify the basis of the calculations. Beause estimates of thoron releases from storage may significantly affect worker and public exposures, NRC staff considers this an open issue.

6.2.1.1.1.4 Radon and Thoron Releases from High-Activity Waste

The applicant estimated annual radon and thoron release rates from disposal of high-activity waste at 15.4 Ci/yr ^{222}Rn and 120 Ci/yr ^{220}Rn (Appendix A, page 57). The applicant did not provide an explanation of how these estimates were derived.

NRC staff analyzed the release rates by considering the data presented in Table 3.8 of Appendix A. The applicant provided a radon release rate of 4.9 μCi/s for a radon flux of 510 pCi/m^{2}s assuming a ^{226}Ra concentration of 1000 pCi/g. Multiplying this release rate (4.9 $\mu Ci.s^{-1}$) by $3.15x10^{7}$ (s/yr) to convert the time from seconds into one year and dividing by a factor of 10 (since the high activity waste is assumed to represent 10% of the bulk waste), the annual release of radon from the high activity waste will be about 15.5 Ci. This figure is consistent with the value reported by the applicant.

For thoron annual release rate estimation, the NRC staff used the same approach with the release rate of 384 μC/s (Appendix A, Table 3.8). The staff calculated an annual thoron release rate of 121 Ci for the high-activity waste, which is consistent with the value reported by the applicant.

NRC staff also examined the input values used in the computer models of atmospheric transport. The applicant assumed annual release rates of

34.2 and 150 Ci/yr for radon and thoron, respectively, in the MILDOS-AREA and TDAD codes.
Thus, it is apparent that the applicant used higher release rates as input values to the codes than the values estimated in the text (Appendix A, page 57). It should be noted however, that the above release rates were based on a calculated specific flux rate. The calculation of specific flux rate was based on non-conservative assumptions as was discussed above. For example, the applicant used a non-conservative diffusion coefficient value of 2.5×10^{-3} cm^2/s rather than 3.1×10^{-2} cm^2/s. Using a more conservative diffusion coefficient value would increase the annual release rate for radon and thoron from the high activity waste by a factor of 10 or more. Therefore, the applicant needs to revise calculations of radon and thoron specific flux and employ conservative assumptions or justify appropriate factors on a site-specific basis. Adequate estimation of radon and thoron release rates is considered by NRC staff an open issue.

6.2.1.1.1.5 Radon and Thoron Release from the Covered Waste

The applicant assessed radon and thoron releases after the waste has been covered by a radon barrier to demonstrate compliance with NRC requirements in 10 CFR 40, Appendix A, Criterion 6. Appendix A requires "reasonable assurance of control of radiological hazards to limit releases of radon-222 from uranium byproduct materials and radon-220 from thorium byproduct materials, to the atmosphere so as to not exceed 20 pCi/m^2s." Compliance with the specifications for the radon barrier is addressed in Section [illegible] of this SER.

Based on the radiation protection review covered in this section, however, the NRC concluded that the applicant used non-conservative physical parameters in analyzing radon and thoron flux through the barrier. The most significant parameter in this regard is the radon and thoron diffusion coefficient, which appears to be underestimated by a factor of 10. The applicant calculated radon specific flux and transport through the proposed 2.13 m-thick radon barrier. The applicant assumed a non-conservative value (18%) for the moisture content in cover material. Although this value was used in the UMTRA Remedial Action Plan for the Vitro tailings (DOE/AL-0141), this value is highly non-conservative as compared to the reference values identified in NRC Regulatory Guide 3.64. The applicant concluded in that 10 cm of soil cover would practically eliminate thoron transport to the atmosphere. The applicant also concluded that radon transport through the 2.13 m radon barrier would be 11.1 $pCi/m^2.s$. Contribution of high activity waste to radon release rate was estimated at 5% of total radon release. Using this derived radon flux value, the applicant concluded that the proposed radon barrier design complies with the 20 pCi/m^2s limit of Criterion 6, Appendix A, 10 CFR Part 40.

However, NRC staff views the analysis of conducted by the applicant as non-conservative because of the low estimates used by the applicant for physical parameters such as diffusion coefficient and the high estimate of moisture content. Less conservative values for these parameters could significantly affect the required thickness and design specifications for the radon barrier. Therefore, the applicant needs to assess the design parameters for the radon barrier. The staff considers this open issue.

The applicant did not calculate radon and thoron annual release rates from the disposal facility after placing the cover and also did not conduct any dose assessments in this regard. The applicant apparently, assumed that complying with the radon and thoron release limits in 10 CFR Part 40, Appendix A ensures adequate protection of workers and the public. However, portions of the embankment will be covered with the radon barrier while disposal operations continue at the site. Releases of radon from the covered portions will continue to contribute to airborne releases of radon, which could affect worker and public exposures.

6.2.1.1.2 Estimation Rates of Releases of Radioactive Particulates

Envirocare estimated the rates of radioactive particulate releases into the atmosphere from four sources (Appendix A, page 75-79): (a) Waste unloading; (b) Waste storage; (c) High-activity waste emplacement; and (d) waste emplacement.

6.2.1.1.2.1 Particulate Releases from Waste Unloading

In estimating radionuclide particulate releases from waste unloading, the applicant assumed an average dust load in the vicinity of the earth moving equipment of 10 mg/m^3. The applicant selected this value for airborne particulates based on an EPA assessment of fugitive dust emissions from coal mines (Ref. 15). The applicant also assumed an average bulk specific activity of 500 pCi/g for each radionuclide and 30% of the dust load is respirable (i.e., less than 10 μm in diameter). Further, the applicant considered that the respirable fraction would have 2.4 times higher specific activity (Ref. 9). Based on these assumptions, the applicant estimated the rate of release of the respirable fraction at 1.6 x 10^{-2} Ci/yr for each radionuclide. For the remainder of dust particles larger than 10 μm, but less than 100 μm, the applicant estimated a release rate of 1.6 x 10^{-2} Ci/yr. The high-activity waste contribution for particles less than 100 μm was also estimated at 1.3 x 10^{-2} Ci/yr. The applicant concluded that the estimated total release rate from the waste unloading operation is 3.2 x 10^{-2} Ci/yr. However, the applicant did not show how the release rates were derived for the high-activity wastes.

NRC staff reviewed the applicant's estimates of release rates using the information provided by the applicant, considering the annual mass of waste to be handled (4.5×10^{11} g), the average radionuclide concentration in the waste (500 pCi/g), and the assumed dust loading factor. NRC's estimates confirmed the particulate release estimates calculated by the applicant for waste unloading operations. The NRC staff also considred the estimated release of 0.1 g/Kg of waste processed based on the combined actions of wind and machinery (Ref. 9, Ref. 15, and Ref. 16). Thus, the NRC staff estimated an annual particulate release rate (Q) in Ci as follows:

Q = Waste Mass (g) X Radionuclide Concentration (Ci/g) X 0.001 (Kg/g) x 0.1 (g/Kg)

The factor 0.001 is used to convert the mass to kilogram unit, and the factor 0.1 is the dust loading factor. The annual release rate calculated using above equation is 2.25×10^{-2} Ci. Using the applicant's assumption that 30% of the airborne particulates are <10 μm in diameter, and a 2.4 times higher specific activity (based on Ref. 9), the annual particulate release rate at this size fraction would be 1.62×10^{-2} Ci. Using the same approach, the remainder of the dust loading (10-100 μm in diameter) to the annual particulate release in the unloading operation would be 1.58×10^{-2} Ci. This figure is also consistent with the applicant's release figure. Therefore, the total airborne particulare releases from unloading operations would be on the order of 3.2×10^{-2} Ci/y.

6.2.1.1.2.2 Particulate Releases from Waste Storage

The applicant estimated the release rate of radioactive particulates from waste storage would be 2×10^{-2} Ci/yr. However, the applicant did not provide information on how this release rate was estimated. NRC staff developed an independent estimate of particulate releases using the approach that was discussed previously for estimating particulate releases from unloading. Based on these calculations, the staff estimates that annual release rates of radioactive particulates will be less than 0.01 Ci for ^{226}Ra, ^{230}Th, ^{232}Th, and other important radionuclides in the ^{226}Ra and ^{232}Th decay chains. Therefore, the staff concluded that the applicant's estimate of annual release rate of radioactive particulates from storage appears reasonable, even though Envirocare did not justify this estimate.

6.2.1.1.2.3 Particulate Releases from Emplacement of High-Activity Waste

The applicant stated that particulate release rates from emplacement of high-activity wastes are not expected to exceed 2×10^{-3} Ci/yr. The applicant assumed that the area designated for disposal of high-activity waste will be a small portion of the active waste emplacement area. The applicant also assumed that high-activity waste will be promptly covered by thin layer of clean soil or of low-activity waste. However, the applicant did not attempt to justify the release rate estimate for particulate releases based on calculations or operational experience at analogous facilities. Although high-activity waste will be disposed along with the low-activity waste in the disposal embankment, the applicant did not use the MILDOS-AREA and the TDAD codes to calculate the annual releases of radioactivity in a similar manner as was done for the low activity waste.

NRC staff reviewed the applicant's proposed estimate of particulate release rates from emplacement of high-activity waste. Based on the review, the staff concluded [Boby, fill in the blank here; did the applicant's estimate appear reasonable? based on what? did we independently assess?]

6.2.1.1.2.4 Particulate Releases from Waste Emplacement

The applicant did not specifically identify or estimate an annual release of radioactive particulates from the emplacement of waste in the active disposal area. The applicant, however, calculated the airborne concentrations of radionuclides in the ^{226}Ra and ^{232}Th decay chains (along with ^{230}Th and lesser amounts of ^{238}U and ^{234}U) using the computer codes MILDOS-AREA and TDAD. The two codes are modifications of the UDAD (Uranium Dispersion and Dosimetry) code (Ref. XX). The applicant justified the use of these codes on the basis that the same methodology and procedure was used by NRC in the preparation of the FGEIS on uranium milling (Ref. 9). The applicant calculated the projected airborne concentrations (pCi/m^3) of radionuclides in the uranium decay chain using an input concentration value of 500 pCi/g for each of the radionuclides: ^{238}U, ^{230}Th, ^{226}Ra, and ^{210}Pb. The applicant used the TDAD (Thorium Dispersion and Decay) code to estimate particulate releases from emplacing waste at the same concentration value (500 pCi/g) for each of the major radionuclides in the thorium decay chain: ^{232}Th, ^{228}Ra, ^{228}Th, ^{224}Ra. The projected airborne concentrations of the radionuclides and associated doses from inhalation and ground deposition are discussed below.

NRC staff independently estimated the annual particulate release rates from the emplacement of waste in the active disposal area. The staff calculated the estimates using the methodology described above for estimating particulate releases from unloading operations at the proposed facility. The calculations assumed a resuspension rate of 0.1 g per kilogram of waste emplaced in the embankment based on Reference 9. The staff estimates that the total particulate release rates will be less than 2.2×10^{-2} Ci/y for each radionuclide in the U and Th decay chains.

6.2.1.1.3 Estimation of Concentration of Airborne Radionuclides

The applicant estimated concentrations of airborne radionuclides in the uranium and thorium series by modeling atmospheric transport using MILDOS-AREA and TDAD codes. The **MILDOS-AREA** code is Version III of UDAD (Uranium Dispersion and Dosimetry) code (Ref. XX). The applicant used this code to calculate particulate and radon releases and transport in the atmosphere for radionuclides in the uranium decay chain.

MILDOS-AREA code was developed by the Argonne national Laboratory (Ref. XX). It is a revision of the MILDOS code, as described in NUREG/CR-2011 (Ref. XX), which was developed for the NRC by the Pacific Northwestern Laboratory (PNL). MILDOS-AREA provides an improved capability for handling large-area sources to compute environmental radiation doses from uranium recovery operations. It estimates the radiological impacts of airborne emissions from uranium mining and milling facilities or any other large-area sources involving emissions of radionuclides in the ^{238}U decay series.

The code treats the atmospheric dispersion from area source using a finite-element integration approach for the virtual-point method when specified by the user. The code has the option of using "Martin-Tickvart Dispersion Coefficients" instead of the "Briggs Coefficients" for a given source. The first option is recommended specifically for ground-level sources, whereas the second option is typically used for elevated sources.

MILDOS-AREA provides the user with option of employing a Finite-Element Integration Method for area sources. Thus, in the new model a "Sector-Averaged Gaussian Plume Dispersion" is used to calculate concentrations of radioactive materials from the fixed point sources. The conventional Gaussian Plume model is an analytical solution for the atmospheric dispersion of a point source. The Finite-Element Integration Scheme approach used in MILDOS-AREA partitions the large-area sources into triangles, rectangles, or elements of other selected shapes. In MILDOS-AREA, the plume is assumed to be reflected by boundaries. The code also considers the mean morning and afternoon mixing heights.

The code accounts for transport processes, such as dry deposition of particulates, resuspension, radioactive decay, decay product in-growth, and plume reflection. Variation of the emission rates of the sources is taken as a step function of time. The major exposure (or impact) pathways considered in the code include: inhalation, external exposure (from ground concentration and cloud immersion), and ingestion. The code calculates individual and population dose commitments using the inhalation dose conversion factors using the dosimetric model from ICRP Report No. 30 (Ref. XX). Values of 50-year committed dose equivalent (H_{50}) per unit intake given in ICRP Report No. 30 are for radioactive particulates with an "Activity Median Aerodynamic Diameter" (AMAD) of 1 μm.

The applicant calculated the values of H_{50} for an aerosol with a different AMAD using the following relationship:

$$\frac{H_{50}(AMAD)}{H_{50}(1\mu m)} = f_{N-P}\left(\frac{D_{n-p}(AMAD)}{D_{N-P}(1\mu m)}\right) + f_{T-B}\left(\frac{D_{T-B}(AMAD)}{T_{T-B}(1\mu m)}\right) + f_{p}\left(\frac{D_{p}(AMAD)}{D_{p}(1\mu m)}\right)$$

Where the terms: N-P is the Nasal Pharyngeal region in the lung; the T-B is the Trachea-bronchus region; and the P is the Pulmonary region. The term f_i is the fraction of the committed dose equivalent in the reference tissue resulting from deposition in the respiratory region *i*. The term D_i is the deposition probability in respiratory region *i* for a given AMAD ((Ref. XX) ICRP Task Group on Lung Dynamics, 1966). The term *i* is the respiratory region index N-P, T-B, and P. The calculations were based on linear approximation from the 12-point AMAD values to provide a reasonably accurate description of the effective dose equivalent factor as a function of particle size. The age-specific dose factors were calculated using the methodology from ICRP Publications 26 and 30 ((Ref. XX) ICRP 1977, 1979) and a report prepared by Oak Ridge National Laboratory for the NRC ((Ref. XX) Cristy et. al. 1986).

A validation study was conducted for the MILDOS-AREA program ((Ref. XX) Yuan, Y.C. et. al. 1989 Appendix B). In this study, airborne ^{222}Rn concentrations and working levels were calculated with MILDOS-AREA program and compared both with measured concentrations and with results from the computer code AIRDOS-EPA ((Ref. XX) Moore, 1979 as updated in 1988). The site selected for the validation study was the uranium mill tailings impoundments area in Monticello, Utah. The code AIRDOS-EPA was chosen for comparison because it has been widely used to calculate atmospheric dispersion of radionuclides from variety of sources. The validation effort indicated that MILDOS-AREA results generally agreed within ±50% with measured ^{222}Rn concentrations at the Monticello site. Comparison with AIRDOS-EPA code show that ^{222}Rn concentrations calculated by the two codes are in very good agreement for distances greater than 1 Km from the origin. At distances very close to the origin, however, AIRDOS-EPA results are much higher for some locations (by as much as 50% or more) than the results obtained from MILDOS-AREA code. This is due to source analysis limitations inherent in the AIRDOS-EPA code.

The applicant used the TDAD (Thorium Dispersion and Dosimetry) code to estimate radiation doses to individuals and to the general population from the airborne release and transport of radioactive particulates and thoron from processing and disposal of wastes for radionuclides in the thorium decay series. The code was developed for NRC from the Version IX of the UDAD code, as described in NUREG/CR-0553 ((Ref. XX)), by Argonne National Laboratory. The UDAD code was used by NMSS for calculation of potential radiological impacts of uranium milling for individual mills and for the FGEIS on uranium milling (Ref. 9).

The UDAD code calculates exposures via inhalation, ingestion, and external exposures. The code calculates atmospheric concentrations of radioactive particulates and radon from specific sources by means of dispersion-deposition-resuspension model. The code accounts for source depletion due to deposition, fallout, and decay in a sector-averaged Gaussian plume dispersion model. The deposition velocity is estimated on the basis of particle size, density, and physical and environmental conditions. Dose conversion factors for the calculation of inhalation dose rates were based on the ICRP Task Group Lung Model (Ref. XX).

The major input data for the UDAD include: source data, receptor data, meteorological data, and population data. The code calculates particulate aerodynamic sizes for five size classes. A density value is assigned to each of the five classes. The rate of release from an area source may be either specified by the user or internally computed by the code. The code uses the three basic modes of particulate movements including: surface creep, saltation, and airborne suspension. The particulate flux model ((Ref. XX) Travis, 1974 and Gillet 1973) used in the code depends on the average tailings grain density and diameter, the surface roughness height, the percent of tailing mass that has a diameter smaller than 20 μm, and the activity of the <20 μm size fraction. The rate of horizontal particle movement in the saltation process is directly related to the cube of the surface shear velocity (U) as: $q = C_h . U^3$; where C_h is a constant dependent on the soil and surface conditions. The surface flux from a contaminated surface to the air is calculated based on the deposition velocity (m/sec), resuspension rate (sec^{-1}), and air concentration (activity/m^3). For the special case of no weathering and a steady-state condition, the resuspension factor is expressed in terms of the resuspension rate and the deposition velocity. The code calculates the concentration of radionuclides on the ground resulting from continuous deposition, ingrowth of decay products, radioactive decay, and the weathering process. The detailed mathematical formulation is given Sections

4.4 and 4.5 of NUREG/CR-0553 (Ref. XX).

The TDAD code is a modification of UDAD code and was developed recently by consultants to the applicant to calculate particulate release (Boby, and thoron transport?) into the atmosphere and transport for radionuclides in the thorium decay chain. TDAD code treats dispersion, resuspension, ground-deposition, and dosimetry in a similar fashion as the UDAD code.

In applying the two codes, the applicant made the following assumptions:

(1) Radioactive particulates have a range of particle sizes with 30% of the material have an aerodynamic diameter of less than 7.7 μm and 70% of the material with an aerodynamic diameter of 54 μm.

(2) The specific gravity of the fine particle fraction is 2.4 g/cm^3; hence, the specific activity is 2.4 times the bulk-specific activity of the waste (which is assumed to have a density around 1 g/cm^3).

(3) The annual disposal rate of waste will be 4.5×10^5 tonnes (5×10^5 tons).

(4) The annual disposal area was assumed to be 38,472 m^2 (Appendix A, page 8). The active disposal area for the 11 e.(2) waste will not exceed one-third of the annual disposal area at any one time (Appendix A, page 8).

(5) Waste disposal will continue for 16 years at the above annual rate. For the MILDOS-AREA modeling, the applicant has divided the disposal operation into eight phases; each phase (a two-year period) terminates with the completion of the radon barrier over that portion of the embankment and the placement of the surface erosion barrier. For the modeling using the TDAD code, the operation was assumed to be continuous for 16 years.

(6) There are no other sources of airborne particulate releases above background at the site. The applicant did not consider any particulate releases from the adjacent waste disposal facilities (Envirocare NORM and MIXED waste disposal facilities, and the disposal facility from the VITRO waste) in evaluating projected doses from the disposal facility for 11e.(2) waste.

(7) The radionuclides included in the analysis for the uranium decay series included: ^{238}U, ^{234}U, ^{230}Th, ^{226}Ra, ^{222}Rn, ^{214}Bi,

^{214}Pb, ^{210}Po, and ^{210}Pb. Radionuclides in the thorium decay series included: ^{232}Th, ^{228}Ra, ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{212}Bi, ^{212}Po, and ^{208}Tl.

(8) All radionuclides in the uranium and thorium decay series were assumed to be present in secular equilibrium with the bulk-average concentration for each of radionuclide 500 pCi/g. [Boby, However, the concentrations of ^{238}U and ^{234}U in the waste was assumed to be 125 pCi/g (25% of the ^{226}Ra concentration) due to partial removal of the uranium at the generator facilities. Ingrowth and decay of all radionuclides were accounted for during transport in the atmosphere.

The applicant estimated airborne particulate concentrations using MILDOS-AREA code in five directions from the site (N, E, W, S, and Maximum transport direction) and at 12 distances up to 80 Km. Table 12 lists the distances and coordinates (x, y) of these locations with respect to the SW corner of the facility as the point of origin. The applicant used the TDAD code to calculate particulate concentrations at 15 specific distances. The applicant also calculated radon and thoron decay product concentrations in WL units using both codes.

Table 13. Receptor Locations (X, Y, and direct distances are relative to SW corner of the facility) used in MILDOS-AREA and TDAD Codes

Receptor Location	Grid Location		Distance from site (km)
	X (km)	Y (km)	
East Boundary	1.6	0.9	1.84
West Boundary	0.0	0.9	0.9
North Boundary	0.9	2.7	2.85
South Boundary	0.9	0.0	0.9
Administration Building	0.9	1.6	1.84
Guard Trailer	0.8	1.2	1.44
USPCI Facility	-1.6	0.6	1.71
Industrial Location	10.4	6.8	12.43

The results, presented by the applicant (Table 3.12, Appendix A, page 85, and Attachments C & D), indicated that the maximum ^{226}Ra particulate concentration in air of 2.4×10^{-3} pCi/m^3 would be at the "Guard Trailer" location. Concentrations of ^{226}Ra at four locations at the site boundary were in the range of 3.0×10^{-4} to 8.5×10^{-4} pCi/m^3. The concentration of ^{226}Ra at the administrative building was 1.5×10^{-3} pCi/m^3. Particulate concentrations of the other radionuclide in the uranium decay series (e.g., ^{238}U, ^{230}Th, and ^{210}Pb) were similar to ^{226}Ra concentration. The code also calculated airborne particulate concentrations of ^{222}Rn and decay products (e.g., ^{218}Po, ^{214}Pb, ^{214}Bi, ^{210}Pb, ^{210}Bi, and ^{210}Po.) at the above mentioned eight receptor locations. ^{222}Rn concentrations also reached a maximum value at the Guard Trailer at 21.5 pCi/m^3 and at the Administrative building at 12.0 pCi/m^3. The concentration of ^{222}Rn at the site boundary was in the range of 3.0 - 10.4 pCi/m^3.

Airborne particulate concentrations of radionuclides in the thorium chain (^{232}Th, ^{228}Ra, and ^{224}Ra) were all similar. The ^{224}Ra concentration was in the range of 1.26×10^{-3} to 9.7×10^{-6} pCi/m^3. The maximum estimated particulate concentration of 1.26×10^{-3} pCi/m^3 for ^{224}Ra was estimated at the administration building. The guard trailer receptor location was projected to have particulate concentration of 8.0×10^{-4} pCi/m^3. Thoron (^{220}Rn) and thoron daughters (^{216}Po, ^{212}Pb, ^{212}Bi, ^{210}Pb, and ^{210}Po) concentrations were much higher; thus, ^{220}Rn concentration at the Guard trailer site was the highest at 36.2 pCi/m^3, and concentration at the site boundary was in the range of 0.07 to 20.1 pCi/m^3. It should be pointed out that concentrations of other radionuclides in the uranium decay chain (e.g. ^{238}U, ^{234}Th, and ^{230}Th and ^{210}Pb) and in the thorium decay chain (e.g. 232Th, ^{228}Ra and ^{224}Ra) were assumed to be equivalent to ^{226}Ra and ^{232}Th concentrations, respectively. The applicant concluded that particulate releases of radionuclides (including radon and thoron) in the U and Th decay chains to the air associated with 11e.(2) waste disposal operations would not exceed background levels at all distances beyond 1 Km from the site. Further, the applicant concluded that the projected airborne concentrations of ^{226}Ra at the site boundary would be of the same order of magnitude as those detected elsewhere in the western United States.

NRC staff reviewed the applicant's airborne particulate modeling and results. The review included assessment of the proposed assumptions and input values to the codes, assessment of limited validation of the code, and comparison of modeling results with rough estimates of releases based on the calculational methodologies presented in Reference 9 and estimates of radioactive particulate releases for the Vitro site ((Ref. XX) FEIS, DOE/EIS-0099-F, 1984). Based on this review, NRC staff concluded the following:

1. The applicant has assumed a non-conservative particle size distribution for airborne radioactive particulates, namely 30% of the particulates are less than 7 μm and 70% are greater than or equal to 54 μm. NRC staff believes that selection of this particle size is non-conservative and may not adequately represent the particle size fractions that will be observed at the proposed disposal facility. Particle size fractions may significantly affect doses via inhalation by controlling the depositional location of the particulates within the pulmonary system.

   NRC's limits on airborne effluent releases in Appendix B of 10 CFR Part 20 and ICRP recommendations (Boby, in Report 30?) (Ref. XX) are based on a standard particulate size of 1 μm "Activity Median Aerodynamic Diameter" (AMAD) for use in calculating doses from inhalation (see the introduction to Appendix B of Part 20). In the absence of specific justifications of alternative particulate size distributions, the applicant should either assume 1 μm AMAD or justify alternative particle size distribution based on site specific measurements. NRC staff considers this an open issue.

2. The applicant did not include any particulate releases, in the calculations of airborne particulate concentrations, from Envirocare's adjacent waste disposal facilities for NORM, low-level waste, and MIXED wastes, or the disposal

embankment for the Vitro uranium mill tailings. Although particulate releases from the Vitro embankment are likely to be negligible because the residual radioactive materials have already been covered by a radon barrier, particulate releases from the other active waste facilities could be significant. If they are significant, operations at the proposed disposal facility for 11e.(2) waste may have to be curtailed or modified to account for occupational and public doses resulting from these operations to ensure that cumulative exposures from licensed and urlicensed operations do not exceed the dose limits in Part 20. In addition, these particulate releases could also mask monitoring data collected for the proposed 11e.(2) waste disposal facility because the same radionuclides are expected to be contained in the wastes. Therefore, the applicant needs to estimate airborne particulate releases and associated doses from adjacent waste disposal operations to demonstrate compliance with the dose limits in 10 CFR Part 20 and to avoid any masking effects on environmental monitoring data. NRC staff considers this an open issue.

3. The MILDOS-AREA and TDAD codes used in the calculation of airborne particulate concentrations may produce estimated concentrations with large uncertainties for short distances (e.g. < 1 Km). As was discussed above, the codes are anticipated to produce results with large uncertainties at such locations. NRC staff anticipates the uncertainties associated with calculated concentrations of airborne particulates may be more than 50% based on the results of the previous validation studies against available monitoring data. Several of the most important receptor locations (e.g., guard trailer, administration building, and the site boundary) are within a distance of 0.9 to 2.0 Km from the embankment area.

4. The applicant did not compare the estimated airborne particulate concentrations for the different radionuclides with the air effluent concentrations listed in Table 2, Column 1 of Appendix B to 10 CFR Part 20. NRC staff compared the applicant's estimated particulate concentrations with the concentration limits in Appendix B. Table 14 compares the projected concentrations for ^{226}Ra, ^{222}Rn, and ^{220}Rn at four locations: East Boundary, West Boundary, Administration Building, and the Guard Trailer.

Table 14. Comparison of Calculated Particulate Concentrations with NRC's 10 CFR Part 20 Effluent Concentration Limits

Radionuclide	Location	Calculated Concentration (pCi/m^3)	Calculated Concentration (μCi/ml)	Part 20 Appendix B Limit (μCi/ml)
^{226}Ra	East B	8.5E-4	8.5E-16	9E-13
	West B	5.5E-4	5.5E-16	9E-13
	Adm. Bldg.	1.5E-3	1.5E-15	9E-13
	Guard Trailer	2.4E-3	2.4E-15	9E-13
^{222}Rn	East B	10.4	1.0E-11	1E-10
	West B	4.8	4.8E-12	1E-10
	Adm. Bldg	12.0	1.2E-11	1E-10
	Guard Trailer	21.5	2.2E-11	1E-10
2[illegible]	East B	6.7	6.7E-12	3E-11
	West B	20.1	2.0E-11	3E-11
	Adm. Bldg	4.6	4.6E-12	3E-11
	Guard Trailer	36.2	3.6E-11*	3E-11

* Value exceeds the effluent concentration limit of 10 CFR Part 20, Appendix B, Table 2, Column 1

From the data presented in Table 14, it is evident that ^{226}Ra calculated concentrations are well below the effluent concentration limits of Appendix B. In addition, calculated ^{222}Rn concentrations are also below the limits by a small margin. However, calculated concentrations of ^{220}Rn are either very close to or exceed the limit. Considering the non-conservative assumptions in the applicant's calculations (e.g., particle size, selection of location for the source origin, estimation of annual release rates), additional particulate releases from adjacent waste disposal facilities, and large uncertainties associated with the model results close to the source, there is reasonable doubt whether the applicant will be able to comply with the air effluent concentration limits in Appendix B of 10 CFR Part 20, at least for the proposed facility as is currently designed and operated. The applicant needs to demonstrate that radioactive particulate releases will not exceed the concentration limits in Appendix B or develop an alternative design and operational procedures to ensure the proposed facility will operate in compliance with 10 CFR Part 20. The NRC staff considers this an open issue. In addition, particulate monitoring will likely be necessary to ensure that airborne particulates remain acceptably low and to demonstrate, in accordance with §20.1302(a), that doses to workers and individual members of the public do not exceed the limits in 10 CFR Part 20.

As an alternative compliance demonstration approach to §20.1302(b)(2) and the concentration limits in Appendix B of 10 CFR Part 20, the total effective dose equivalent to the individual likely to receive the highest dose from licensed operation does not exceed the annual dose limit in §20.1301(a). Potential doses to members of the public will be considered below.

6.2.1.1.4 Estimation of Surface Activity from Ground Deposition of Airborne Radioactive Particulates

The applicant used MILDOS-AREA and TDAD codes for calculating radioactive particulate ground deposition of uranium and thorium decay chains, respectively. Using the MILDOS-AREA code, the applicant calculated total deposition rate (pCi/m^2-yr) as a function of distance for ^{238}U, ^{230}Th, ^{226}Ra, and ^{210}Pb. The applicant also calculated the ground-deposited activities for ^{232}Th, ^{228}Th, and ^{228}Ra using the TDAD code. The applicant calculated ground-deposited activities for each particulate radionuclide in five directions and at 12 distances. The ground-deposited activity ranged from more than 10,000 pCi/m^2-yr on soil at the site to approximately 2 pCi/m^2-yr on soil at distance 20 Km downwind from the site. The maximum ^{226}Ra ground-deposited activity was calculated at the Guard Trailer at 2.6 x 10^4 pCi/m^2-yr. At the boundary of the site, ground-deposited activity ranged from 2.7 x 10^3 to 8.6 x 10^3 pCi/m^2-yr. The applicant concluded that the ground-deposited thorium activities would be similar to ground-deposited activities for radionuclides in the ^{238}U decay chain.

The NRC staff reviewed the applicant's modeling approach and estimated ground deposition rates for airborne particulates. Except for the non-conservative parameters and other omissions and uncertainties noted above for atmospheric releases of particulates, the staff concluded that the applicant's estimates for ground deposition of particulates were acceptable.

6.2.1.1.5 Estimation of Doses From Exposure to Environmental Pathways for Each Individual Pathway of Exposure

The applicant used the estimated radon and thoron concentrations, airborne particulate concentrations, and ground deposition rates described above to estimate potential doses to members of the public from operation of the proposed disposal facility. The applicant used the computer codes MILDOS-AREA and TDAD to model the transport of the radioactive materials and estimate dose conversion factors for each exposed individual organ. The applicant calculated total effective dose equivalents by multiplying the organ doses by the appropriate organ weighting factors (§20.1003) and then summing the products to yield a total effective dose equivalent. The applicant considered exposures from the following pathways:

- Inhalation of airborne radionuclides (radon, thoron, and airborne particulates);
- Ingestion of contaminated food (from consumption of ingested meat of livestock grazed on contaminated forage); and
- External exposure.

The applicant calculated the annual dose commitments and annual population dose commitments for each of these exposure pathways. In a similar approach, as discussed above for particulate releases, the applicant assessed potential exposure for eight different phases of waste disposal operation, for each pathway of exposure, and for each source of release of radioactivity. The applicant did not include exposure via ingestion of contaminated groundwater or use of the water to irrigate crops or water livestock for the reasons discussed in Chapter 3 of the SER.

The MILDOS-AREA and TDAD codes calculate dose conversion factors for radionuclides in the uranium and thorium decay chains, respectively, for individual organs (Attachment C, pages 8-11) and estimate the annual committed dose-equivalent (ACDE) for each organ. The integration period for the calculation of the committed dose is 50 years. The applicant assessed ACDEs for different ages: infant, child, teenager, and adult. For the uranium series, dose calculations were made (using MILDOS-AREA) for: bone, average-lung, bronchi, liver, kidney, and effective whole body (see Attachment C of the license application). For the thorium series radionuclides, the applicant calculated ACDEs (using TDAD) for nine organs: skin, ovaries, testes, small intestines, lung, red marrow, skeleton, spleen, and whole body (see Attachment D of the license application). Table 15 summarizes the results of ACDE calculations for selected organs in mrem per year from inhalation of airborne radionuclides in the uranium and thorium series in adult individuals:

Table 15. Summa., of the Annual Committed Dose Equivalent Rates (mrem/y) to Selected Organs from Inhalation of Airborne Radionuclides in the Uranium and Thorium Series*

Series	Location	Bone	Lung**	Liver	Kidney	Bronchi	ACDE Rates (mrem /y)
		mrem/y					
U	East B	7.2E0	2.9E0	4.6E-1	3.3E-1	na	1.5E1
	West B	4.6E0	1.8E0	3.0E-1	2.1E-1	na	7.2E0
	Adm.Bldg	1.3E1	5.0E0	8.0E-1	5.8E-1	na	1.8E1
	Grd.TRLR	2.0E1	8.0E0	1.3E0	9.4E-1	na	3.2E1
Th	East B	8.9E1	2.5E-2	3.45E0	1.7E1	3.1E-3	4.2E0
	West B	5.5E1	1.5E-2	2.1E0	1.0E1	1.9E-3	1.3E1
	Adm.Bldg	1.5E2	4.0E-2	5.9E0	2.8E1	5.1E-3	2.9E0
	GRD.TRLR	2.4E2	6.6E-2	9.2E0	4.4E1	8.1E-2	2.3E1

*Dose to epithelium tissues from radon and thoron decay products are not listed in this table. These doses are listed on pages 285-289 (License Application Attachment C) and page 156 of Attachment D, respectively.
** Pulmonary region alone

The applicant calculated an ACDE of 22.6 mrem/yr from thoron decay products at the guard trailer, and of 31.6 mrem/yr from the radon decay products. Considering the 12% organ weighting factor for the lung (§20.1003), the annual committed effective dose equivalent would be 2.7 and 3.8 mrem/yr for thoron and radon, respectively. Thus, a person located continuously at the security guard trailer (according to the calculations) would receive a CEDE dose from inhalation of radon and thoron of about 6.5 mrem/yr.

The NRC staff believes this calculation is non-conservative because a large particle sizes assumed for particulates and the other factors discussed above (e.g., lower diffusion coefficient for radon and thoron). Therefore, the staff expects that projected doses at the location of the guard trailer from inhalation of radon and thoron, assuming for example a much smaller particle size of 1 μm AMAD, may be considerably greater than projected by the applicant.

The applicant assessed potential exposures via the ingestion and agricultural pathways. NRC staff reviewed the applicant's ingestion pathway analysis and concluded it is generally acceptable. This potential exposure pathway is considered of negligible significance at the proposed facility because of the absence of good quality surface water, poor quality groundwater, limited existing and project future agricultural land use, and the remoteness of this arid site. Thus, the NRC staff considers it unlikely that members of the public may be exposed to radionuclides in the near vicinity of the site via the ingestion pathway.

The applicant assessed potential external exposure to members of the public due to radiation emitted from the airborne and ground-deposited radionuclides. The applicant has also calculated (using MILDOS-AREA and TDAD codes) the dose-conversion factors for exposure to airborne (cloud) and ground-deposited (ground radionuclides). This analysis was conducted for each phase of the waste disposal operation and for only the adult age group. The applicant's results indicated that at the location of the guard trailer the external whole body dose rate for adult individual exposed to airborne and ground-deposited radionuclides in the uranium series is about 4.7 mrem/yr. These calculated rates did not include contribution from direct gamma radiation from the waste and sky-shine (e.g., reflection and scattering of gamma radiation in the atmosphere above the waste embankment).

The direct exposure rate from the exposed waste (containing 500 pCi/g ^{226}Ra) at the edge of the waste embankment was estimated by the applicant to be about 13.6 R/yr. At 90 m, the distance to the closest boundary from the embankment, the direct gamma exposure rate would decrease to less than 2 mR/yr. The applicant estimated the total external dose rate at the nearest site boundaries (the southern and western boundaries) would be 2.0 mrem/yr from waste containing radionuclides in the uranium series only. The direct exposure from radionuclides in the thorium series were estimated by the applicant to be of the same order magnitude as that for the uranium series radionuclides.

NRC staff reviewed the applicant's assessment of projected doses from exposure to direct gamma radiation from the waste handling and emplacement operations, from atmospheric transpot of radioactive particlates, and from ground-deposited activity. The staff confirmed the applicant's estimates of direct gamma exposure rates by calculating projected doses are various distances using the inverse square formula (i.e., exposure rate decreases proportional to the square of the separaton distance). Although the applicant used several non-conservative assumptions in the transport calculations, as described above, the staff generally concluded that the applicant's estimates of direct gamma exposures were reasonable and representative for projected conditions at the site. The staff particularly notes that the applicant did not include estimates of direct gamma exposures associated with potential exposures from waste handling and disposal activities at the adjacent waste disposal facilities for NORM, low-level waste, and MIXED wastes, or the disposal embankment for the Vitro uranium mill tailings.

6.2.1.1.6 Calculation of the Total Effective Dose Equivalent (TEDE) to an Individual Member of the Public from All Pathways

The applicant has presented in Appendix A (Section 3.8, page 107) estimates of the total effect e dose equivalent (TEDE) at various locations surrounding the proposed facility. The applicant based these estimates on the results of the modeling using the MILDOS-AREA and TDAD computer codes as described above. The applicant's estimated TEDE from the uranium series radionuclides at the guard trailer location was 12 mrem/yr (this dose includes doses from radon and daughters). The total effective dose equivalent at the same location from exposure to airborne, ground-deposited, and inhaled radionuclides in the thorium decay series was estimated to be 11.6 mrem/yr (direct gamma exposure was not considered by the applicant). The applicant summarized the total effective dose equivalent estimates from the uranium and thorium series as follows:

Location*	Projected Doses to Potential Members of the Public TEDE (mrem/yr)
East B	22.2
West B	12.7
South B	17.1
Admin. Bldg.	40.8
Guard Trailer	75.4
USPCI	18.9
Industry	0.7

* See Table 12 for distances.

Based on the above results from modeling using the MILDOS-AREA and TDAD codes, the applicant concluded that the proposed facility should be able to comply with the public dose limit in 10 CFR 20.1301 (as provided in §20.1302(b)(1)) because the projected dose to members of the public should be less than 100 mrem/yr.

The NRC staff has reviewed the applicant's dose assessment and projected doses to members of the public at the proposed facility. NRC's public dose limits in 10 CFR Part 20 are intended to protect the most exposed member of the public. As such, the reqirements are written in terms of protecting an actual individual. Members of the public (i.e., persons other than individuals employed by Envirocare and who are exposed to radiation incident to their occupation) do not live in the immediate vicinity (i.e., within 1 km) of the proposed facility. The security guard that resides at the guard's trailer is considered a worker and is protected by the occupational dose limits in 10 CFR Part 20 because the guard is exposed to radiation incident to performing assigned duties. However, any individual who may live or visit with the guard at the guard trailer would be considered a member of the general public, except to the extent that that individal may also be employed by Envirocare as a radiation worker. Consequently, the NRC staff determined that the location of the maximum exposed individual member of the public, if one existed near the site, would be at the guard's trailer.

NRC staff reviewed each step of the applicant's dose assessment and identified numerous deficiencies, limitations, and uncertainties associated with the assessment methodology and results, as described in the preceding sections. Based on this review, the staff concluded the following:

1. In spite of several non-conservative assumptions, the applicant demonstrated that radionuclide releases to the atmosphere near the site boundary either barely meet or exceed the effluent concentration values as listed in Appendix B of 10 CFR Part 20. Consequently, attempts to demonstrate compliance with the public dose limit in §20.1301 need to rely on assessments of projected doses to the most exposed member of the public.

2. The applicant's assessment did not assess the total projected doses from releases of all radionuclides by summing the projected contribution of each radionuclide. In addition, the applications of the MILDOS-AREA and TDAD codes to this facility at distances close to the source may produce results with large uncertainties. Ccupled with the non-conservative assumptions discussed previously and the omission of any doses from waste emplacment and handling opeations at adjacent facilties, the applicant has not demonstrated compliance with reasonable assurance with the public dose limits in 10 CFR 20.1301 for the facility as presently proposed in terms of design and operations, as well as assessed.

Based on the above discussion, NRC staff believes that the applicant needs to demonstrate compliance with the public dose limits in accordance with §20.1301 and 20.1302. NRC staff considers this an open issue.

6.2.1.2 Occupational Exposure

NRC's regulations in 10 CFR 20.1201 require that licensee's control occupational dose to individual adults to the more limiting of the total effective dose equivalent of 5 rems (0.05 Sv) or other dose limits described therein. The regulations in §20.1201(d) allow licensees to demonstrate compliance with the occupational dose limits using the Derived Air Concentrations (DACs) in Table 1 of Appendix B of 10 CFR 20. In addition to the annual dose limits, Part 20 requires licensees to limit the soluble uranium intake by an individual to 10 mg per week to avoid chemically toxic effects on workers (§20.1201(e)). This section describes and assesses the applicant's program for protecting workers against radiological hazards and toxic effects associated with uranium intake.

The principal pathways of occupational exposure at the applicant's proposed disposal facility for 11e.(2) waste include:

- Direct external exposure to waste and ground-deposited dust;
- Internal exposure from inhalation of airborne radioactive particulates and radon and thoron and their decay products released from the waste during receipt, storage, processing, and disposal of the waste.

6.2.1.2.1 Direct Gamma Exposure

The applicant did not employ any model or code to assess direct gamma exposure to workers. The applicant determined that it would not be practical to project potential external exposure for each planned activity. The applicant would prefer to rely on personnel monitoring to demonstrate compliance with the occupational dose limit of 10 CFR Part 20.1201 after facility operations and waste handling commences. The applicant intends to issue dosimeters to measure external radiation exposures (thermoluminescent dosimeters (TLDs)) to all staff working within the boundary of the site of the proposed disposal facility for 11e.(2) waste. Personnel will not be monitored for doses from beta radiation directly. Instead, Envirocare intends to measure beta radiation intensity at the working environment using appropriate radiation survey instruments, such as thin-window Geiger-Mueller probes, ion chambers, and scintillation detectors. Envirocare proposes to measure radiation intensity with open and closed windows and estimate the beta radiation intensity from the difference between the measurements.

The applicant has proposed that personnel monitoring not be segregated into each type of activity. In other words, Envirocare plans to use the same personnel dosimetry for each radiation worker regardless of the type of job activities and will not distinguish sources of worker external exposure. The applicant clearly stated that some of the radiation worker activities will be associated with disposal work at the NORM and Mixed waste disposal facilities located adjacent to the applicant's proposed disposal facility for 11e.(2) waste. However, the applicant also commits to ensure that the dose to any worker will not exceed the occupational dose limits in 10 CFR 20.1201, regardless of the alleged source of the exposure.

The applicant analyzed gamma exposure rates to workers based on operational activities at the Envirocare facility. The predicted average occupational exposure rates (mrem/yr) for each of the three activities (unloading, storing, and placing waste) was the same for the projected deep dose equivalent at about 400 mrem/yr. Details of occupational exposure rates based on activity type were provided by the applicant in Table 8.22, Appendix A, page 124.

The applicant estimated gamma exposure rates (at a distance of 1 m) from receiving the waste containing 500 pCi/g of ^{226}Ra and ^{228}Ra at 4.2×10^{-4} and 2.6×10^{-4} $\mu R/hr$, respectively [Boby, check these values; they appear to be extremely low]. The applicant projected gamma exposure rates up to 5 mR/hr. However, the applicant expects that the average exposure rate would be on the order of 0.2 mR/hr. Based on 2000 hours annual working time, the estimated deep dose equivalent from receiving the waste to be about 0.4 rem/yr.

In estimating the dose to the front end loader operator in unloading and transporting the waste, the applicant assumed a distance of 4 m between the operator and the waste. The applicant estimated a gamma exposure rate to the front end loader operator at about 400 mrem/yr.

The applicant estimated an external radiation dose rate to the equipment operator, involved in waste storage, at 100 mrem/yr. The waste disposal process requires spreading and compacting of the waste in the embankment using specialized heavy equipment. The applicant estimated an exposure rate of 400 mrem/yr for an individual operating heavy machinery over the waste surface. The applicant assumed a shielding factor of 10 times due to the steel construction of the equipment.

NRC staff reviewed the applicant's estimates of worker doses from direct gamma exposures during unloading, transporting, storage, and emplacement of the waste. At the average concentrations assumed for the facility (i.e., 500 pCi/g ^{226}Ra and other associated radionuclides), the staff estimates that the direct gamma doses may be greater than projected by the applicant. For example, using the conversion factor provided in (Ref. XX) [Boby, see NCRP report that Wrenn relied on for Molycorp's conversion factor], the exposure rate 1 m above the surface of the embankment where waste was being emplaced would be expected to be on the order of 500 $\mu R/hr$. If an individual were exposed continuously while working at the site at that location for 2000 hours per year, the approximate estimated dose would be 1 rem (0.01 Sv). In addition, NRC staff believes that applicant's assumed shielding factor is non-conservative and leads to underestimated doses from direct gamma exposure. Nevertheless, in accordance with §20.1502(a), the applicant will monitor doses from external gamma exposure to workers and is required to comply with the dose limits in Part 20. Therefore, the applicant will need to implement controls and additional protective measures (e.g., greater shielding, restricted access, time constraints) to limit external doses based on personnel dosimetry and other monitoring.

6.2.1.2.2 Internal Dose from Inhalation

The applicant has estimated annual intakes of radionuclides via inhalation by workers using the following relationship:

Intake (pCi/yr) = Activity Concentration (pCi/m^3) x Inhalation Rate (m^3/hr) x Work Period (hr/yr)

The activity concentration was calculated by the applicant by assuming a dust loading of 1 mg/m^3 (based on (Ref. XX)). The activity of each radionuclide in the waste is assumed to be 500 pCi/g for ^{226}Ra, ^{232}Th, and decay products in secular equilibrium. Thus, the airborne activity in the dust loading would be 0.5 pCi/m^3 for each radionuclide, which corresponds to an airborne concentration of 5.0×10^{-13} µCi/ml. The applicant, however, incorrectly calculated a concentration of 0.05 pCi/m^3 and converted it to an airborne concentration of 5.0×10^{-14} µCi/ml. The inhalation rate of the industrial worker was assumed to be 1.2 m^3/hr; assuming that occupational worker spends 2000 hr/yr on the job, the total amount of air inhaled would be 2.4×10^3 m^3/yr, which corresponds to an intake rate of 1.2×10^{-4} µCi/yr. Using this relationship, the applicant estimated the average worker intake at 120 pCi/yr. However, using the above relationship and proper conversions this intake should be 1200 pCi/yr (1.2×10^{-3} µCi/yr).

The DAC limits in Table 1 of 10CFR Part 20, Appendix A are listed in the following table for comparison with the calculated airborne concentrations:

Table 15. Comparison of the Applicants Estimated DACs with 10 CFR Part 20, Appendix B

Radionuclide	10 CFR Part 20 DACs (μCi/ml)	Applicant's Estimated Airborne Concentrations (μCi/ml)*	Estimated DAC-hours (Assuming 2000 hours per year)
^{220}Rn	1.0 WL	9E-5 WL	0.18
^{222}Rn	0.33 WL	?	?
^{226}Ra	3E-10	5E-13	3.4
^{228}Ra	5E-10	5E-13	2
^{230}Th	3E-12	5E-13	340
^{228}Th	7E-12	5E-13	140
^{232}Th	5E-13	5E-13	2000

* Based on corrected conversion of the dust loading from 0.05 to 0.5 pCi/m^3

Even though the applicant projected air concentrations less than the DAC values, Envirocare proposed standard operational procedures that would require using respiratory protection with a protection factor of at least 10 in the dust-forming areas. In the dose assessment of inhalation exposure to radionuclides in the ^{232}Th and ^{226}Ra decay chains, the applicant assumed use of respiratory protection with a protection factor of at least 10. The applicant concluded that the annual committed effective dose equivalent from inhalation of airborne particulates would be 325.2 mrem/yr. Without respiratory protection, the projected CEDE from inhalation alone would be estimated at 3252 mrem/yr. However, the applicant did not explain how these doses were derived.

The applicant's dose assessment only considered particulate releases from the low-activity waste (500 pCi/g). For handling the high-activity waste, the applicant estimated an additional dose of 130 mrem/yr. Thus the applicant's total estimated CEDE from inhalation of airborne particulates from handling and emplacement of the combined waste would be on the order of 4,553 mrem/yr without respiratory protection and 455 mrem/yr with respiratory protection.

The applicant also estimated doses from radon and thoron releases from the waste by calculating concentration levels as a function of distance and direction from the waste. The applicant employed the same procedure for calculation of exposure in terms of WL and WLM as that detailed in UDAD code manual (pages 15, and 63-64). The applicant calculated the average thoron exposure to workers (assuming continuous occupation) at 0.1 and 0.5 Km from the embankment to be about 6.1×10^{-3} and 3.2×10^{-3} WLM, with an average of 4.6×10^{-3} WLM for the facility. Considering a working period of 2000 hours per year, the applicant estimated the average occupational exposure to each worker would be 1.1×10^{-3} WLM. The applicant assumed each WLM corresponded to a 5 rem dose to the bronchial epithelium to estimate the radiation dose rate to workers [Boby, check this conversion; NCRP 78 gives a conversion factor in the range of 12.6 to 14.2 rem per WLM for radon decay products and 2.3 rem per WLM for thoron decay products]. Thus, the committed effective dose equivalent to the bronchial epithelium of individual worker was estimated to be 13.4 ?mrem/yr. Similarly, the applicant estimated the average dose rate to workers from radon decay products at 45.0 ?mrem/yr. Thus, the applicant's predicted dose rate (CED) to the bronchial epithelium for the average radiation worker from inhalation of radon and thoron at the Envirocare facility was would be about 58.4? mrem/yr. [Boby, ensure that these are all CEDEs]

The applicant committed on Appendix A, page 122 to ensure compliance with the occupational dose limits in 10 CFR Part 20 by estimating the sum of the committed effective dose equivalents from each inhaled radionuclide and show the compliance with:

$$\Sigma_i C_i/(DAC)_i \le \text{Criteria or } \Sigma_i \text{ intake}/ALI_i \le 1 \text{ Criteria}$$

Where C_i is the concentration (μCi/ml) of radionuclide i and $(DAC)_i$ is the derived annual concentration (μCi/ml) inhalation limit of radionuclide i listed in 10 CFR Part 20, Appendix B, Table 1, Column 3. The $(ALI)_i$ is the annual inhalation limit of intake (μCi) for radionuclide i as listed in 10 CFR Part 20, Appendix B, Table 1, Column 2. The applicant will also limit the duration of worker exposure during operations that result in heavy airborne dust releases and will use respiratory protection for all workers involved in the operation of dust generating machines. The applicant also reiterated that during receiving, relocating, and disposing of higher specific activity waste, workers involved in such operations will use full-face respirators with a protection factor of 50.

With respect to intake of radioactive material by inhalation, NRC states in §20.1202(b) that the total effective dose equivalent limit will not be exceeded if the sum of the deep-dose equivalent divided by the total effective dose equivalent limit and one of the following does not exceed unity:

- The sum of the fractions of the inhalation ALI for each radionuclide, or
- The total number of DAC-hours for all radionuclides divided by 2000,

or

- The sum of the calculated committed effective dose equivalents to all significantly irradiated organs or tissues calculated from bioassay data using appropriate biological models and expressed as a fraction of the annual limit.

NRC staff compared the applicant's estimated internal exposures against the DACs for the different radionuclides. The applicant claimed that the estimated radioactive particulate concentrations are at the most 10 times lower than the DACs in Table 1 of Appendix B of 10 CFR Part 20. Using the corrected airborne activity concentration value of 0.5 pCi/m^3 and using the above relationships, however, the applicant's estimated airborne concentrations would be either at or slightly above the DACs for ^{230}Th and ^{232}Th. By summing the DAC-hours listed in the table above, the total exposure would be 2485 DAC-hours, which exceeds the limits in §20.1202(b). This calculation assumed continuous exposure at the calculated airborne concentrations for a 2000-hour work year, which is unlikely. Nevertheless, the calculations are based on a modeling effort that used non-conservative parameter values and has several significant uncertainties, especially close to the waste handling areas as described above.

Therefore, the use of respiratory protection and other controls (such as restricting the amount of exposure time) will likely be necessary in accordance with the provisions §20.1702 to limit airborne intakes of radioactive material. The applicant committed to measure the dust loading and airborne radioactive particulate concentrations, and radon and thoron and decay products, at the disposal site using portable generators and particulate and gas sampling. The applicant should follow NRC Regulatory Guide 8.25 (1992) for appropriate guidance on the design and implementation of an effective air sampling program in the work place. The applicant also committed to certain measures to mitigate dust suspension. These controls and measures may have to be adjusted based on operational experience, variations in waste concentrations, and personnel monitoring (including bioassay) to ensure that worker doses do not exceed the radiation protection and uranium toxicity limits in 10 CFR Part 20.

In addition, the applicant did not address the issue of the soluble uranium intake by worker, which must be less than 10 mg soluble uranium in a week in accordance with 10 CFR 20.1201(e). The applicant did not assess the solubility of the uranium that may be released from the handling the waste nor show that cumulative uptake of uranium via inhalation will not exceed the 10 mg weekly limit. The applicant needs to address this issue and propose specific actions that will be taken if uranium intakes are projected to exceed the 10 mg limit. NRC staff considers this an open issue.

6.2.1.2.3 Total Dose

The applicant will have to limit the sum of the deep dose equivalent and committed effective dose equivalent to no more than 5 rem per year from all sources under the applicant's control at the site. The analysis of doses from inhalation of radioactive materials above indicates that doses via this pathway may alone be close to the 5 rem per year limit. In addition, direct gamma doses also may be significant depending on the effectiveness of the applicant's control program. Further, operation of adjacent disposal facilities for low-level radioactive waste, mixed waste, and naturally occurring radioactive material may also contribute doses to individuals that work at the proposed site. Consequently, the applicant will need to implement an effective combination of procedural controls, mitigative measures, and personnel and environmental monitoring to ensure that total doses to workers do not exceed the occupational dose and chemical toxicity limits in 10 CFR Part 20.

6.2.2 Exposures Resulting from Accidental or Unusual Operating Conditions

Accidental or unusual releases of radioactive material could occur at the proposed facility. Although the radiation protection standards in 10 CFR Part 20 do not specifically apply to such conditions, the applicant assessed the radiological consequences for several accident scenarios. These scenarios included the following: 1) on-site truck turnover or collision, 2) train derailment, 3) flooding, and 4) tornado or severe winds.

6.2.2.1 On-site Truck Turnover or Collision

The applicant used the methodology described in Reference XX (NUREG-0706) in assessing the accidental exposure due to a truck turnover or collision on site. The probability selected for the truck accident was 1.3×10^{-6}/km based on (Ref. XX). The applicant assumed the number of truckloads per day was 53 and the travel distance was 1 km. Thus, the probability of an accident in any one year was calculated at 1.8×10^{-2} or 1.8%. Envirocare assumed 40 pounds or less of 11e.(2) waste may become airborne immediately and, if the spill was not contained or otherwise controlled, the release fraction over a 24-hour period might increase to 360 pounds. A comparison was made in the license application (page 17-18) with a truck accident involving a yellow-cake shipment in an area with population density of 7.5 persons per square mile. For such an accident, the 50-year dose commitment to the lungs was estimated at 0.7 to 9 person-rem. Since the specific activity for Envirocare waste is far less than the yellow-cake (at least 20 times lower), the applicant estimated the dose to the public from a truck accident involving a spill of 11e.(2) waste would be between 0.03 to 0.4 person-rems. The applicant further stressed that the actual off-site population dose would be zero because there are no residents in the nearby area.

The applicant has also estimated projected doses to workers involving an accident with a load of waste shipment with waste of concentration 15,000 pCi/g. The applicant assumed a three-hour period for the cleanup with no use of respiratory protection. The projected maximum TEDE to the radiation worker involved in such accident was 1,032 mrem (over 3 hours). The applicant concluded that such doses are within the annual dose limit for radiation worker (10 CFR Part 20.1201).

NRC staff reviewed Envirocare's assessment of the potential adverse consequences of a truck accident on site. Although the applicant did not specifically justify each of the parameter values assumed in the assessment, the staff considers the analogy to a yellowcake spill and qualitative assessment of radiological releases from on-site truck turnover appropriate because the concentration of the 11e.(2) waste and population density are expected to be far less than values assumed in the analysis. Use of representative values for both of these factors would reduce the projected population doses associated with a truck accident. In addition, the applicant has committed to using respiratory protection in any activities involving release of radioactive particulates at the proposed facility. Consequently, projected doses to workers should be well within the dose limits of 10 CFR Part 20 for this accident scenario because of the protective measures that the licensee has committed to in the license application.

Train derailment

Envirocare used the argument that the probability of a train derailment is far less than that for a truck accident at the site. As in the case of the truck accident, Envirocare assumed there is no population in the nearby area and hence there should be negligible dose to the public. The dose to workers was not estimated or calculated although the applicant stated that: "no significant dose to workers for the same reasons discussed under truck accidents". The applicant, however, referred to truck accident scenario to estimate doses from such an accident.

NRC staff has reviewed the applicant's brief assessment of potential doses to workers associated with a train derailment. Although the probability of a train derailment may be less than was calculated for the truck accident (i.e., less than 2×10^{-2} per year), the derailment would probably involve a much larger quantity of contaminated material. The typical number of rail cars in a train at the envirocare facility, other than proposed shipments of contaminated materials from the West Chicago, is 3 to 4 cars. Consequently, the potential releases to the environment, airborne concentration values, and doses to the worke could also be considerably greater than were estimated for the truck accident. However, the probable extent of a spill of waste from train derailment at the proposed site would likely be limited for the following reasons: (1) the distance of the rail spur on site is less than the road length; (2) the rail cars are typically covered, so a derailment may not result in a spill of contaminated material; (3) the speed of the trains on the rail spur is limited, so it is unlikely that the train would be moving faster than a nominal speed (e.g., less than 5 miles per hour); and (4) if a derailment occurred, the cars would likely fall off the tracks and remain upright, rather than toppling over and spilling their contents. If a derailment did result in a large spill of contaminated material, the licensee would be required to remove the waste in accordance with release limits for unrestricted areas and dispose of the waste within the embankment. The doses to workers from activities involved in cleaning up the spill would be similar to projected doses associated with unloading and emplacement of the 11e.(2) waste in the disposal embankment. The applicant has committed in the license application to limit such doses to ensure compliance with 10 CFR Part 20. Therefore, the staff concluded that the applicant's assessment of the truck accident scenario provides a reasonable estimate for the probable impacts associated with a train derailment.

Flooding

Due to flood control design features (Appendix F of the license application), Envirocare concluded that no off-site transport of radioactive waste by flooding will be anticipated. NRC staff reviewed and confirmed the applicant's analysis of flooding in section XX of this SER.

Tornado and Severe Winds

The applicant assessed potential consequences associated with severe weather at the proposed facility. The applicant employed the probability figure of 1 to 5 x 10^{-4} from (Ref. XX) (NUREG-0706) for tornados in Utah. Envirocare also used NRC's estimates from the GEIS for uranium milling (Ref. XX), which predicts a maximum exposure at 2.5 miles (50-year dose commitment) of 0.83 micro-rem from the release of yellow cake from a model uranium mill. Considering population of zero density in the nearby areas and low specific activity of the waste, Envirocare concluded that the dose for off-site residents resulting from tornado would be zero. The applicant also concluded that doses to workers caught in the tornado would be trivial compared to physical hazards associated with the severe weather, without providing any estimates for the doses.

The applicant also assessed potential consequences of severe wind using the predicted airborne exposure resulting from tornadoes (50-year dose commitment at 2.5 miles was estimated at less than 1 microrem). Appendix I of the license application estimated a continuous exposure to a plume from a 3 m/s wind at a 500-m distance to the nearest resident. Envirocare concluded that the anticipated dose to an off-site resident as a result of infrequent severe winds would be on the order of microrem/yr.

The NRC staff reviewed the applicant's assessment of potential consequences of severe winds and tornados and concluded that Envirocare's assessment is reasonably conservative and appropriate. The applicant assumed yellowcake in the assessment, which would tend to overestimate potential doses because of the high concentration of uranium in the yellowcake compared with the concentrations of radionuclides in the 11e.(2) waste at the proposed facility. Although the yellowcake would not contain as high concentrations of decay products from uranium and the radionuclides in the thorium decay chain, the differences in projected doses should not be significant because the much higher concentration of uranium in the yellowcake should compensate. In addition, the staff concurs with the applicant's determination that physical hazards associated with tornados would likely overwhelm concerns about projected radiological doses to workers during the severe weather.

6.3 Instrumentation, Methods, and Equipment

NRC regulations in 10 CFR Part 20 require that licensees conduct radiation surveys and apply process or other engineering controls to ensure that workers and members of the public are protected in accordance with the dose limits of Part 20. For example, §20.1501(a) requires licensees to conduct surveys to evaluate the extent of radiation levels, concentrations and quantities of radioactive materials, and potential radiological hazards that could be present. In addition, §20.1701 requires licensees to use engineering or other process controls to control concentrations of radioactive materials in the air, including control of access, limitation of exposure times, and the use of respiratory protection. The applicant has proposed to use radiation protection instrumentation, methods, and equipment in the application in accordance with the provisions of Part 20.

The applicant proposed portable and laboratory equipment for surveying and detecting radiation and radioactive materials in Section 17.5.5.4.3 of the license application (pp. 17-34 to 17-36). The survey and detection equipment proposed by the applicant includes a full range of instruments and detectors that are consistently used throughout the industry. The applicant has also committed to use respiratory protection equipment and protective clothing to ensure protection of workers and limit the internal and external exposures to radioactive materials.

Envirocare committed that all survey and monitoring equipment will be periodically calibrated by staff licensed by the State of Utah. The calibration will be performed within the tolerance sensitivity specified by the manufacturer of the equipment. The frequency of calibration will be daily for the multi channel-analyzers and associated equipment, and semi-annually for the G-M survey meters, the Micro-R meters, and the Alpha survey meters.

The analytical balances and the dosimeters will be calibrated on monthly and yearly bases, respectively. Calibration records for all equipment will be kept at the Site Administration Office and at the Company Main Offices in Salt Lake City. All equipment will be identified by serial number, person performing calibration, date of previous calibration, and the date for next calibration. A record of equipment damage will also be kept with the equipment file.

NRC staff has reviewed the proposed health physics instrumentation and calibration procedures and concluded, in gene al, that the applicant's proposed approaches are appropriate and adequate to demonstrate compliance with 10 CFR Part 20. The NRC staff based its conclusion on a variety of observations, including: (1) the proposed methods generally represent state-of-the-art field instrumentation for radiological survey applications and monitoring of operating facilities, (2) the methods conform reasonably with standard industry practice for similar types of operations, and (3) the instrumentation will be adequate with respect to required sensitivity levels for their applications over the range of radionuclides and waste types to be received at the proposed facility.

NRC staff considers the instrument calibration program proposed by Envirocare is adequate and in compliance with 10 CFR 20.1501(b). Envirocare indicated that calibration and management of monitoring equipment will be based on applicable guidance in NRC Regulatory Guides (e.g. Regulatory Guides 4.14, 8.25, and DG-80030). Based on its review, of the type of equipment to be used for radiation surveys and measurements, NRC staff suggests that the response of survey instruments be checked against a known source prior to each usage (see NRC Regulatory Guide 8.30). This check should be supplemented at 6-month intervals by calibrating each instrument at two points separated by at least 50% of each linear scale or at one point near the midpoint of each decade on logarithmic scales that are routinely used. Air flow rates through filters should be determined by calibrating pumps with the filter paper in place at least once every six months to a ±20% accuracy. The fluorimetric analysis for uranium should be calibrated by running a known standard uranium, traceable to NIST, and a blank with each batch. Alpha counting systems used for radon decay product measurements should be calibrated at least monthly using a known, traceable standard alpha source.

6.4 Radiation Protection Program

Envirocare presented a radiation safety and environmental program (section 17.4 of the license application) that included a variety of procedures and methods for ensuring protection of workers and members of the public. Envi-ocare addressed all of the applicable radiation protection standards in 10 CFR Part 20.

In general, the applicant's radiation protection program appears to be sufficiently developed to address the specific dose limits and radiation protection provisions in 10 CFR Part 20, including occupational dose limits for adults, summation of external and internal doses, determination of airborne doses and other internal exposures, planned special exposures, occupational dose limits for minors, protection of an embryo/fetus for a declared pregnant woman, and dose limits for members of the public (as described in Section 6.2 of this SER). Detailed analysis of the compliance status with respect to each requirement is described in Appendix A of this SER.

6.4.1 Procedures and Methods Used in Determining the Exposures and Intake of Radioactive Materials

The applicant described procedures and methods for (1) personnel and occupational exposure monitoring, (2) area radiation monitoring, (3) environmental monitoring including general exposure rate and radionuclide concentrations in soil, water, vegetables, and wildlife, and (4) dose assessment.

6.4.1.1 Personnel and Occupational Exposure Monitoring

Envirocare committed to monitor radiation exposure of all personnel using the following methods described in section 17.4.5. (pages 17-50 to 17-53) of the license application:

a. Permanent employees will be issued TLD badges which will be examined and exchanged on a quarterly basis. The Radiation Safety Officer will keep quarterly dosimeter records for all staff. The dosimeters will be used primarily to assess direct gamma exposure.

b. Individuals visiting the site on a short-term basis will be issued a self-reading pocket dosimeter to record exposure. The dosimeters will be read as the individual leaves the site and recorded in the Site Access Log. A group of visitors may all use one TLD or one pocket dosimeter, if they will stay in one vicinity in the controlled area and are near the individual with the dosimeter.

c. All exiting employees will be surveyed for skin, hand and foot, and clothing contamination, prior to exiting the controlled area, using an instrument sensitive to alpha, beta, and gamma contamination. Records of names and number of contaminated employees and levels of contamination will be kept in the Administration Building.

d. All permanent employees will participate in a bioassay program for assessment of possible internal deposition of radionuclides. Baseline urine samples will be collected upon employment and annually thereafter. The samples will be analyzed for gross alpha, ^{226}Ra and total uranium. Sampling and analysis for ^{230}Th and ^{232}Th will be conducted upon finding an increase above baseline levels of radioactivity for gross alpha particle activity, ^{226}Ra, and total uranium.

Envirocare has also stated that they will comply with 10 CFR 20.1501(c) to ensure that all personal dosimeters be processed and evaluated by a dosimetry processor who is accredited by the "National Voluntary Laboratory Accreditation Program (NVLAP)"[4].

Further, the applicant noted that commercially available film badges and TLD personnel dosimetry often do not assess beta doses in the mixed beta-gamma field associated with uranium mill tailings and 11e.(2) byproduct material. Thus, the applicant has committed to follow NRC guidance in Regulatory Guide 8.30 by measuring worker beta dose indirectly from environmental measurements as was explained in section 6.2.2 of this SER.

The applicant has presented its bioassay program that employs the methods recommended by NCRP Report No. 87, *Use of Bioassay procedures for assessment of Internal radionuclide Deposition* (Ref. XX). The applicant indicated that the bioassay program will also be based on the methodology described in Reif et. al., 1992, (Ref. XX) for interpretation of the bioassay results.

NRC staff has reviewed the applicant's proposed personnel monitoring program. NRC staff compared the proposed program against standard industry practice, available regulatory guidance, and NRC requirements in 10 CFR Part 20. Based on this review NRC staff determined that the applicant's program should be adequate and sufficient to demonstrate compliance with the requirements of 10 CFR Part 20, particularly §§20.1501 and 1502. The bioassay interpretation procedure proposed by the applicant is appropriate for screening purposes. The applicant should consult also NRC Regulatory Guide No. DG-8009 (1991) entitled "*Interpretation of Bioassay measurements*" to ensure appropriate interpretations of the bioassay measurements.

6.4.1.2 Area Radiation Monitoring

Envirocare described the area radiation monitoring program in section 17.4.6.2 of the license application. The program includes the following aspects:

a. Gamma Exposure Rate Monitoring: perimeters of all controlled areas, the office area, and the lunch/change area will be monitored on weekly basis. Transport vehicles will also be monitored, for gamma

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[4]This is the requirement of 10 CFR Part 20, Subpart F, 20.1501 Paragraph (c).

exposure rate, upon arrival at the site and before departure. The applicant proposed to monitor gamma exposure rate quarterly in the administration building and in the security trailer. In addition, random external gamma surveys will be performed during daily operations as considered necessary by health physics personnel.

Because the security guard will reside at the trailer onsite and extended exposure to elevated direct gamma rates could result in significant doses to the guard at the trailer within the calendar quarter, the applicant needs to determine the exposure rate to the guard at the trailer location with high confidence. Therefore, Envirocare needs to continuously monitor the exposure rate at the security guard trailer rather than monitor the exposure rate quarterly as proposed in the application. This is a license condition.

b. Airborne Radioactivity Monitoring: work areas and boundary areas will be monitored for airborne radioactive particulate using high volume, fixed-head air samplers. Continuous airborne particulate sampling and monitoring will be conducted to provide overall average concentrations of radioactivity, at fixed locations of environmental monitoring. Samples will be analyzed for gross alpha particle activity, ^{226}Ra, ^{232}Th, ^{230}Th, and total uranium. Gross alpha levels will be compared with the concentration limits in note 3 of Appendix B of 10 CFR Part 20 (e.g., 6×10^{-11} $\mu Ci/ml$ for ^{238}U, ^{234}U, ^{230}Th, and ^{226}Ra; 3×10^{-11} $\mu Ci/ml$ of natural uranium, or 45 $\mu g/m^3$ of natural uranium). Breathing zone samples will be collected for equipment operators involved in handling or disposing of waste. Envirocare committed to collecting breathing zone samples. Passive environmental radon monitoring will be used at all environmental monitoring stations and indoors in the administration building and in the mobile home used by the security guard. For control of worker exposures via inhalation, the applicant proposed an administrative limit of 6 pCi/m^3 for airborne gross alpha activity.

Because the security guard will reside in the mobile home and elevated airborne particulates and concentrations of radon and thoron (with decay products) may cause significant doses to the guard, more frequent monitoring of airborne concentrations is justified at the location of the trailer. Therefore, the applicant is required to conduct frequent or continuous monitoring for airborne radioactive particulates, radon (with decay products), and thoron (with decay products) at the security guard residence to ensure that the total dose to the guard does not exceed the occupational dose limits in Part 20 and that total doses to other inhabitants or visitors (other than radiation workers) to the residents do not exceed the public dose limits in Part 20. This is a license condition.

c. Surface Contamination Sampling: contamination wipes will be taken on weekly basis in eating areas, change areas, office areas, railcar rollover platform, and all equipment and vehicles. Routine wipe surveys for surface contamination will be conducted weekly for the eating areas, change area, office areas, railcar rollover and control shack and for all equipment and vehicles. The wipes will be analyzed for gross alpha particle activity using an appropriate instrument. The wipes will also be analyzed for gross beta particle activity contamination using an appropriate instrument (e.g., gas flow proportional counter).

Surface contamination levels on equipment and clothing will be routinely examined. The levels of natural uranium, ^{235}U, ^{238}U, and associated decay products should not exceed (on the average) 5,000 dpm alpha/100 cm^2. Survey activities of ^{226}Ra, ^{228}Ra, ^{230}Th, ^{228}Th, ^{231}Pa, and ^{227}Ac, should not exceed (on the average) 100 dpm/100 cm^2. The beta-gamma emitters (with decay modes other than alpha emission or spontaneous fission) should not exceed on the average 5,000 dpm beta-gamma/100 cm^2. Surface activity of natural thorium, ^{232}Th, ^{223}Ra, ^{224}Ra, and ^{232}U should not exceed 1,000 dpm/100 cm^2 [Note: ^{232}U is not expected to be detected at the proposed facility; ^{223}Ra will probably not be detected at significant levels].

NRC staff reviewed the applicant's proposed program for area radiation monitoring and concluded that the applicant has proposed an acceptable program for conducting general area surveys and monitoring, as required by provisions in 10 CFR 20.1501 and 10 CFR 20.1502.

6.4.1.3 Environmental Monitoring and Surveillance

Envirocare presented in section 7.3 of the license application a radiological environmental monitoring program, which covers the following areas:

6.4.1.3.1 Gamma Radiation Control and Monitoring

The boundary of the site will be monitored for external gamma radiation exposure using a pressurized survey meter during periods of dry weather and using E-Perm Electric Ion Chambers or environmental thermoluminescent dosimeter (TLDs). The on-site monitoring stations will be located at A2, A3, A5, A6, A7 and A11 through A13 (see Table 7.1 of the application for coordinates of these stations). Two off-site stations, B1 and B2 (Table 7.1), will be used to establish and monitor background exposure rates during site operations. Envirocare committed to measure gamma exposure rates at 150-meter intervals along each of the eight compass directions out to a distance of 1500 m from the center of the disposal facility in accordance with NRC Regulatory Guide 4.14 (Ref. XX). Direct radiation measurements should be made in dry weather, not during periods after rainfall when soil is abnormally wet to avoid any interference by soil moisture on gamma exposure rate measurements. The environmental TLDs at all monitoring stations will be exchanged and processed at quarterly intervals.

The applicant also committed to conducting routine external gamma surveys using gamma scintillation survey meters in areas involving disposal of 11e.(2) materials as a part of the general area surveys. These external surveys were discussed in Section 6.4.1.2 of this SER.

NRC staff has reviewed the applicant's proposed environmental monitoring program for gamma exposure rate and concluded that the program is generally acceptable. Envirocare stated in the application that it will assume all potential exposures detected above background rates at the monitoring station are associated with waste handling and emplacement activities at the proposed 11e.(2) disposal facility, even though the source of the gamma radiation may be from disposal operations adjacent to the proposed disposal facility. Detection of elevated exposure rates at the boundary environmental stations in unrestricted areas may necessitate additional controls, including limitations on waste emplacement activities or additional restrictions on the acceptable concentrations for disposal, even though the source of the elevated exposure rates may be the adjacent waste disposal facilities under the control of the applicant.

6.4.1.3.2 Airborne Radioactive Monitoring

The applicant presented a sampling and monitoring program for radioactive air particulates at nine stations located at A2, A3, A5 through A7 and A10 through A13 (Table 7.1). Background samples will be collected at stations B1 and B2. The air samples will be analyzed initially for gross-alpha and gross-beta particle activity. In addition, quarterly composite samples (composed of all weekly samples collected at each specific station) will be analyzed by gamma spectrometry for identification of any gamma-emitting radionuclides. Radiochemical analysis for total U, ^{226}Ra, ^{230}Th, ^{232}Th, and ^{210}Pb will also be conducted on all composite samples.

The applicant committed that analytical techniques used will provide a minimum detectable concentration of five percent or less of the applicable Derived Airborne Concentrations (DACs) for radionuclides given in Table 2 of 10 CFR 20, Appendix B. Envirocare will establish restrictive limits of particular importance. For example, the applicant proposed to limit on airborne concentrations of alpha emitters, such as ^{230}Th, to 0.08 pCi/m^3 and beta emitters like ^{210}Pb to 4 pCi/m^3. Thus, in the proposed Envirocare monitoring program, concentrations of gross alpha greater than 0.08 pCi/m^3 or gross beta of greater than 4 pCi/m^3 will be used as monitoring set points. Samples having gross activity measurements above these levels will be analyzed using gamma spectrometry for further identification of the radionuclides present.

Airborne particulate radioactivity concentrations will also be compared with the data accumulated from background measurements. For naturally occurring radionuclides, which are normally found in air samples, a concentration in excess of the mean plus two standard deviations will be considered significant. The applicant proposed an administrative limit of 0.24 pCi/m^3 for airborne gross-alpha particle activity concentration for site boundaries.

NRC staff has reviewed the proposed environmental monitoring program for airborne radioactive particulates and determined that the program is generally acceptable under normal operating conditions. However, since the 11e.(2) facility is contiguous to other radiological waste disposal facilities that handle wastes that contain the same radionuclides (e.g. Envirocare's LLW and mixed waste and NORM disposal facilities), more frequent sampling of airborne radioactivity concentrations may be required to attempt to distinguish the radioactive emissions of the various facilities. The applicant has accepted the assumption that any detected environmental radioactivity, regardless of its source, is attributable to the 11 e.(2) waste disposal operation. Consequently, the applicant may need to take corrective measures if airborne concentrations exceed appropriate levels in accordance with 10 CFR Part 20, even though the source of the elevated airborne concentrations may be the adjacent waste disposal facilities under the control of the applicant.

6.4.1.3.3 Radon In Outdoor Air Monitoring

The applicant committed to monitor outdoor radon and decay product concentrations on a continuous basis using E-Perm Electret Ion Chambers. Radon monitoring detectors will be located at eight on-site stations [A2 through A7 and A11 through A13 (see Table 7.1 for locations)] and at two off-site stations (B1 and B2). The off-site stations will be employed to establish and monitor background levels of radon and decay products during site operations. Detectors located at these stations will be collected quarterly and analyzed to determine average radon (including decay products) concentrations (pCi/l). Any concentration levels in excess of the mean plus two standard deviation values of the background level will be considered significant [Boby - we should state the mean and standard deviation here based on the Clive monitoring data as a point of reference for how large the variability may be].

NRC staff reviewed the proposed program for monitoring ambient radon levels in outdoor air and determined that the program is generally sufficient to demonstrate compliance with 10 CFR Part 20. The applicant's approach is consistent with standard industry practice.

However, since the proposed 11e.(2) waste disposal facility is contiguous to other radiological waste disposal facilities that handle wastes that may also release radon, thoron, and their decay products (e.g. Envirocare's LLW and mixed waste and NORM disposal facilities), the applicant may need to take corrective measures if airborne concentrations of these radionuclides exceed appropriate levels in accordance with 10 CFR Part 20, even though the source of the elevated airborne concentrations may be the adjacent waste disposal facilities under the control of the applicant.

In addition, the applicant should also consider performing radon-222 flux measurements in three separate months during normal weather conditions in accordance with NRC Regulatory Guide 4.14. The purpose of the sampling is to assess the radon flux from the operating facility on a periodic basis for comparison against the radon measurements collected as part of the environmental monitoring program. The measurements are normally conducted at the center of the facility and at locations 750 and 1500 meters from the center in each of the four compass directions.

6.4.1.3.4 Soil Contamination Control and Monitoring

Soil samples will be collected quarterly and will be analyzed by gamma spectrometry to determine concentrations of gamma-emitting radionuclides and infer concentrations of other radionuclides in the decay chains that emit no or weak gamma emissions based on equilibrium relationships. The applicant will analyze all samples for ^{226}Ra and 10% of the samples for natural uranium, ^{230}Th, and ^{210}Pb. The purpose of the periodic sampling would be to detect any significant windblown transport of radioactive particulates from the disposal facility. Some selected samples will also be analyzed by alpha spectrometry for ^{230}Th, ^{232}Th, and total uranium. The applicant committed to collect and analyze samples from stations A2, A3, A5 through A7, A10 through A12 and B1 & B2 (Table 7.1 of the license application) using gamma spectrometry. ^{230}Th, ^{232}Th, and total uranium will be analyzed in soil samples collected quarterly from the vehicle decontamination area, truck shipment staging area, road from rollover to the embankment, and stations No. 5, 31, 32, and A11 through A13. The applicant has committed to take surface soil samples at 300-meter intervals to a distance of 1500 meters in each of the 8 directions from center of the disposal facility on a periodic basis. In addition, five samples will be taken at random at other locations around the site. A one-meter deep sample will be taken at distance 750 m from the site boundary in each of the 4 compass directions and at the center of the disposal area.

The NRC staff has reviewed the applicant's proposed procedures for soil sampling and analysis. The applicant's program for soil sampling is in accordance with NRC Regulatory Guide 4.14 (Ref. XX). Based on this review, the staff concluded that the procedures are generally appropriate and sufficient to comply with NRC requirements in 10 CFR Part 20. The applicant's proposed program should be adequate to detect any significant windblown transport of radioactive particulates from the disposal facility into the general environment.

6.4.1.3.5 Monitoring of Radiological Contamination in Vegetation

The applicant has committed to collect and analyze vegetation samples from local native plants twice a year. Samples will be collected from nine locations. One sample will be collected from stations A12, 30, 39, 55 and GW3 (see Table 7.1 of the application for coordinates). Samples will also be collected at four remaining locations; these locations are one mile east, west, north, or south of the site boundary (Section 7.3.5 of the license application). The latter four samples will serve as background monitoring samples. Gamma spectrometry analysis will be conducted for determination of gamma emitting radionuclides and for total uranium, ^{210}Pb, ^{210}Po, ^{226}Ra, ^{230}Th, and ^{232}Th.

NRC staff considers the above vegetation monitoring program is adequate and in accordance with NRC regulations and guidance in Regulatory Guide 4.14.

6.4.1.3.6 Wildlife Monitoring

The applicant has committed to sample wild field mice (one dozen/yr) from on site stations A11, A12, 30, and 31 and from off-site station B3 (as an upwind control). The samples will be analyzed for total U, ^{210}Pb, ^{210}Po, ^{226}Ra, ^{230}Th, and ^{232}Th (Section 7.3.6 of the license application).

NRC staff concludes that the wildlife monitoring program is adequate based on comparison with the acceptable programs described in Regulatory Guide 4.14. However, the staff also recommends that the applicant consider analyzing three samples from other predominant types of wildlife that may serve as food in the vicinity of the disposal facility, e.g. jack-rabbits and any crops or livestock, within 3 km of the site. These wildlife samples should be analyzed for total uranium, ^{210}Pb, ^{210}Po, ^{226}Ra, ^{230}Th, and ^{232}Th (NRC Regulatory Guide 4.14). Non-sacrificial sampling methods should be used to the maximum extent practical to avoid any significant impact on the diversity and number of wildlife species near the site.

6.5 Radiation Safety Program

NRC regulations in 10 CFR 20.1101(a) require each licensee to develop, document, and implement a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the requirements in Part 20.

6.5.1 Organization and Responsibilities

The applicant has described the radiation protection responsibilities of various individuals within the Envirocare organization in Chapter 18 of the license application. In the applicant's program, overall responsibility for radiation protection resides with the Corporate Quality and Radiation Safety officer (CRSO), who reports directly to the president of Envirocare.

The NRC staff reviewed the proposed organizational structure and responsibilities for the radiation safety program. Based on this review, the staff a potential problem with limited direct communication between the field radiation safety officer and the site manager. Nevertheless, the applicant has committed to close cooperation between the CRSO, site manager, and the Field Radiation Safety Officer (FRSO). Further, Envirocare will conduct weekly staff meetings, which will include the Executive Vice President, the CRSO, the operations supervisors, and other personnel. The applicant also stated that all Envirocare management and staff will have free access to each other to resolve immediate safety or operational issues. Therefore, the staff concluded that the applicant's proposed organizational structure and responsibilities should generally be adequate to ensure timely identification and resolution of issues affecting radiation safety and environmental protection.

However, the staff observed a potential problem area that the applicant should address on a continuing basis to ensure continued compliance with NRC requirements in 10 CFR Part 20. Many of the key radiation protection staff will only be committed half-time to the proposed 11.e(2) waste disposal facility. These staff positions included the CRSO and the FRSO. When not present at the 11e.(2) facility, NRC staff understands that the CRSO and FRSO would be located and accessible at the adjacent disposal facilities for LLW, mixed waste, and NORM waste. Although this level of effort should be adequate under normal operating conditions, periodic events may arise that will require simultaneous attention of the RSOs at the both facilities. For example, injuries to workers could warrant the simultaneous attention of the FRSO at both the 11e.(2) disposal facility and one or more of the other waste disposal facilities. In addition, if the FRSO is detained at the one facility, the FRSO may not be available for immediate determinations of safety hazards at the other facilities. The applicant should ensure adequate radiation protection support to the 11e.(2) waste disposal facility and adjacent facilities.

6.5.2 Staff Qualifications

Envirocare emphasized the importance of having qualified staff to manage and conduct operations of the 11e.(2) waste disposal facility due to the radiological nature of the waste. The applicant presented the qualifications of the radiation safety staff in the application.

NRC staff reviewed the proposed qualifications of key staff responsible for performing radiation safety functions. The staff compared the proposed qualifications with guidance provided in NRC Regulatory Guide 8.31 (Ref. XX). Based on this review, the staff concluded that the proposed radiation safety staff qualifications are adequate.

6.5.3 Staff Training Program

The training program, as described by the applicant in Section 17.4.8 of the license application, will be implemented under the supervision of the CRSO. Radiation safety training will be provided for all persons before they enter the controlled area. The amount of radiation safety training for any person will depend on the function and purpose of the person and type of activities to be carried out. Persons entering the controlled area will be classified into three categories: permanent employee, temporary worker, and visitor. The permanent employee is an employee hired by Envirocare for a period of 20 days or longer. A temporary worker is a service contractor, who works inside the controlled area under a contract or a service order, but is not an employee on the payroll of Envirocare. A visitor is a person whose main interest is to communicate with personnel in the controlled area and/or to observe operation of the facility.

The applicant's proposed training program includes an appropriate combination of radiation worker training, entrance training, and radiation safety training. In addition to the above training courses, the applicant has committed that all Envirocare employees will be required to attend at least 20 hours of annual training in radiation protection and safety, which will be provided by qualified personnel. This training will be tailored to the specific needs and duties of the employees. It will cover areas of occupational safety, radiological safety, and health physics procedures and techniques. Details of specific training for radiation workers, radiation monitors, health physics technicians, and security guard/radiation monitor are given in the license application Appendix B Pages 1_3-121.

The NRC staff has reviewed the applicant's proposed training program and determined that it is adequate for radiation workers and permanent employees. The staff compared the program with relevant guidance in NRC Regulatory Guide 8.31, Regulatory Guide 8.13, and Regulatory Guide 8.29 (Refs. XX). Therefore, the proposed training program should be adequate and appears to comply with NRC requirements for such training in 10 CFR

19.12.

6.5.4 Radiation Safety Posting, Access Control, Record Keeping and Reports

The applicant has stated that the entire site area will be fenced prior to receipt of 11e.(2) waste to ensure that intruders do not inadvertently gain access to the site. The fence will be posted with appropriate "Caution-Radioactive Materials" signs bearing the standard radiation symbol as required by 10 CFR 20.1901. Radiation warning signs will also be posted at all security gates and at 200-foot increments on the permanent fencing. The signs will be visible and legible from a minimum distance of 25 feet. Any embankment which has been completed will be fenced and posted with appropriate signs (e.g., "Caution-Radioactive Materials", and other warning signs as applicable). The applicant will apply a 2-foot thick erosion barrier at the boundary fence to severely limit, if not eliminate, intrusion and burrowing by small animals.

The applicant has committed that entrances into the work area will be opened only for the entrance and exit of equipment and waste from the site. All persons working in the controlled area will be required to pass through an access control gate and enter their names in the access control log. They will also be required to adhere to the access regulations. All employees and visitors will be monitored by TLDs and pocket dosimeters, respectively (see Section 6.4.1.1 of this SER). Persons who do not conform with safety and security regulations will be denied access to the controlled area of the site. Access to the site without prior training and any deviation from the dosimeter policy must have prior approval from the CRSO or the FRSO. The security guard will provide surveillance to prevent intrusion by any unauthorized persons.

The NRC staff has reviewed the proposed access control program and determined that the program is appropriate and in accordance with the provisions of 10 CFR 20.1901, 20.1902, 20.1903, and 20.1904.

6.5.5 ALARA Controls

NRC regulations in 10 CFR 20.1101(b) require that licensees use, to the extent practicable, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

The applicant proposed an ALARA program (Section 14.4.1 of the License Application) that focuses on practical aspects of reducing potential exposures and releases via dust control. The applicant will investigate any reported personnel exposure in excess of an administrative limit of 50 mrem/month in an effort to maintain occupational exposures ALARA.

Additional procedures and methods proposed by Envirocare to keep internal exposures ALARA included:

a. Dust suppression on all operational roads using magnesium chloride or watering at 2-hour intervals (License Condition).

b. Respiratory protection (at least half-face masks with a protection factor of 10) will be used by workers in areas of potential high dust concentrations, e.g. in the rollover area, emplacement areas within the embankment.

c. Prompt placement of the radon barrier on portions of the embankment as portions of the embankment are completed.

d. Suspension of waste emplacement operations under high wind conditions (i.e., winds in excess of 40 mph). The RSO may also stop the operation at lower wind speeds if dust conditions or other safety conditions warrant.

e. Speed limits of 35 mph on roads treated by dust suppressants and 10 mph on infrequently used roads.

f. Radiation surveys of the area will be conducted weekly and increasing trends in exposure rates or concentrations will be investigated.

g. Pre-planning of tasks that may have potential for higher than normal exposure to limit exposures through sufficient use of time and selection of appropriate procedures.

The applicant indicated also that the FRSO will document ALARA activities that include review of disposal contracts, monthly review of environmental air monitoring, adjustment of work procedures to reduce exposures, and review of gamma exposure rates in the working areas to reduce exposures to ALARA. Further, the applicant will conduct an audit of ALARA activities and document such activities on annual basis.

NRC staff has reviewed the applicant's proposed ALARA program as described in the application. The NRC staff compared the applicant's proposed program with relevant NRC guidance in NRC Regulatory Guides 8.31 and 8.37 (or Draft Regulatory Guide DG-8013) (Refs. XX). Although the applicant has presented some practical procedures that attempt to implement the ALARA provisions of §20.1101(b), the details of the applicant's ALARA program and audit and inspection provisions are lacking in the application [see also Section 6.4 of this SER]. In addition, the ALARA program is not complete. The applicant did not specifically address or propose adequate procedures for significant elements of an effective ALARA program, including ALARA philosophy and goals, responsibilities for overseeing and revising the ALARA program, ALARA program audit functions, respiratory protection, effluent controls, facility equipment design, ALARA training, and fire control. Therefore, as a license condition, the applicant needs to submit for NRC review and approval a more complete ALARA program prior to receiving waste at the proposed disposal facility. Once approved, the ALARA program will be established as a license condition.

6.5.6 Personnel Contamination and Decontamination

Envirocare has committed in the application to require all workers in the controllod areas to wear protective clothing or disposable coveralls to minimize any potential skin contamination and to control the spread of contamination. All permanent employees will be issued dedicated work boots that will be worn in the controlled areas. Supervisors and visitors to the site will not be required to wear protective clothing or to wash exposed skin. However, they must wear dedicated boots or boot covers and must use the hand and foot monitors prior to leaving the controlled area. Workers involved in handling material will be required to wash and survey skin (hands and face) before they leave the site. Showers will be provided in the change area for workers to be used before exiting the site. Personnel working in the embankment area will be required to be surveyed before leaving the access control area. A hand and foot monitor sensitive to both alpha and beta radiation, as well as gamma radiation, will be used for routine monitoring for contamination of personnel. All alpha contamination on skin and clothing will be considered by the applicant as removable. Envirocare will apply the limits of contamination for personnel and clothing as was given in section 6.4.1.2 of this SER.

The NRC staff has reviewed the applicant's proposed procedures for controlling personnel contamination and for decontamination. Based on this review, the staff determined that the applicant has generally presented adequate and appropriate procedures. However, the applicant's procedures are not adequate with respect to controlling internal contamination from dust inhalation and methods for decontamination [Boby, contamination of what? lungs?]. Specifically, the applicant needs to describe in more detail the potential for and procedures for controlling internal contamination of workers from dust inhalation in accordance with 10 CFR 20.1702. Staff considers these issues as open issues.

[Boby, what about Dick's concern about the location of the shower and exit point from the controlled area? Has this been resolved by the applicant?]

6.5.8 Emergency Procedures

The applicant provides in appendix C of the license application a comprehensive "Radioactive Material Accident or Emergency Contingency

Plan." Envirocare proposed that the maximum credible accident at the site would be the accidental dumping of load of radioactive waste at some location other than the disposal cell. The plan includes a description of the response procedures and responsibilities of Envirocare personnel.

The NRC staff has reviewed the proposed emergency response procedures for the proposed 11e.(2) waste disposal facility. The staff compared the licensee's response plan with plans for other facilities and standard industry practice [Boby, is this a true statement? In not, please describe how we conducted our review -- briefly]. Based on this review, the staff concluded that the applicant's proposed procedures are adequate with respect to emergency scenario presented because it includes appropriate emergency notification and response procedures and is sufficiently flexible to accommodate the potential variability of site emergencies. Given the waste characteristics and proposed operational procedures at the proposed disposal facility, the staff considers it unlikely that other types of accidents may occur, such as a fire in the embankment, accidental criticality, or chemical explosions. Therefore, it does not appear necessary to require the applicant to develop more detailed plans to address such potential emergencies.

6.5.9 Quality Control

The applicant has described a quality control program in the Application. The applicant's organizational chart (Figure 18 of the license application) indicates that the quality assurance/quality control activities are conducted under the supervision of the Field Radiation safety officer (FRSO) and the Internal Quality Assurance Auditor (IQAA). The quality control program for the health physics, environmental, and safety programs depends mainly on daily inspection of operations by the FRSO, the SM, or the RT to assure that radiation protection activities are conducted in a safe manner and in accordance with applicable requirements. These inspections cover all aspects of Section 7 of the license application and applicable regulations, including those of the NRC and the Utah Division of Radiation Control. The applicant has also committed to routinely audit all radiological records, tests, and measurements. A contracted QA Auditor will report to the president of Envirocare about any unsatisfactory work and initiate, recommend, or provide solutions to address deficiencies. The radiation and safety audit will be performed on at least quarterly intervals. The site will also be inspected at least quarterly by an industrial hygiene consultant to assure compliance of the site operation with the applicable standards of the Occupational Safety and Health Administration.

The NRC staff has reviewed the proposed Quality Control program element of the Radiation Safety Program [Boby, How did we conduct this review?

what did we compare against?]. Based on this review, the staff determined that specific elements of the QC program that are applicable to the radiation protection program area appear reasonable and sufficient to ensure safety and protection of workers and off site individuals.

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[Boby, Need to sort out references. Delete those not cited. Renumber to match citations in text and tables]

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