Rec'd 9-15-99

ENVIROCARE OF UTAH, INC.
THE SAFE ALTERNATIVE

CD99-461

40-8989

September 14, 1999

Mr. John J. Surmeier, Chief
Uranium Recovery and Low-Level Waste Branch
Division of Waste Management
Office of Nuclear Materials Safety and Safeguards
US Nuclear Regulatory Commission
M S T-7J9
11545 Rockville Pike
Washington, DC 20555-0001

SUBJECT: Courtesy Copy of SERP Panel Findings

Dear Mr. John J. Surmeier:

Envirocare of Utah, Inc. (Envirocare) hereby submits copies of the following SERP Findings as a courtesy:

11e.(2) 99-012
11e.(2) 99-014

If you have questions, please call me at (801) 532-1330.

Sincerely,

Mark Ledoux, CHP
Corporate Radiation Safety Officer

CC: Region IV Administrator

NL10'/1

Enclosures

Envirocare of Utah, Inc. 11e.(2) License

SAFETY AND ENVIRONMENTAL REVIEW PANEL (SERP)
PANEL FINDINGS

In accordance with the Safety and Environmental Review Panel (SERP) Administrative Procedure, ADMIN-5, the panel has determined that the provisions of the procedure have been met to change Envirocare 11e.(2) License, SMC-1559, as follows:

Docket Number:	11e.(2)99-012
Date:	July 12, 1999
Action Requested:	To authorize relocation of Environmental Monitoring Station A-5 for the proposed expansion of the LARW Embankment 200 feet to the South.

The SERP has reviewed the requested action and the requirements of License Condition 9.4 (SMC-1559) and finds as follows:

a. **The change does not conflict with any requirement specifically stated in the license (excluding material referenced in License Condition 9.3), or impair the licensee's ability to meet all applicable NRC regulations.**

The license conditions have been reviewed and it has been determined that the requirements of license condition 9.4 have not been met, indicating that granting approval for the relocation of environmental monitoring station A-5 for the proposed expansion of the LARW Embankment 200 feet to the South does require a request to amend the license. The requested change does not conflict with any required actions pursuant to 10 CFR Part 20.

b. **There is no degradation in the essential safety or environmental commitments in the license application, or provided by the approved reclamation plan.**

This revision does not affect the essential safety or environmental commitments in the license application, or provided by the approved reclamation plan. As such, a safety and environmental analysis was not necessary for the determination of the request.

c. **The changes are consistent with the conclusions of actions analyzed and selected in the site Environmental Impact Statement (NUREG-1476) dated August 1993, and the Safety Evaluation Report dated January 1994.**

The EIS and the SER have been reviewed to confirm that the requested action is not consistent with the conclusions of these documents.

d. **In consideration of the above, the SERP has determined that the change does require an amendment to Radioactive Materials License SMC-1559.**

Discussion:

A review of the 11e.(2) License has been performed to determine if any License Condition specifically addresses environmental monitoring locations. Environmental Monitoring Station A-5 will require relocation due to a proposed 200-foot expansion of the LARW Embankment. Station A-5 will be moved approximately 250 feet west. License Condition 11.4(b)2) states the following: "Shall monitor the effluent release of airborne particulates as per Section 7.4 of the license application (see License Condition 9.3) at the air sampling stations listed in Table 7.2 of the license application (see License condition 9.3). Section 7.4 of the license application states that environmental monitoring shall occur at the locations listed in Table 7.1 (this table provides northing and easting coordinates for each station). Page 5-22 of the EIS discusses the environmental monitoring locations and provides figure 5.2 showing the locations. Page 136 of the FSER states that "The on-site monitoring stations will be located at A2, A3, A5, A6, A7 and A11 through A13 (see Table 7.1 of the license application for coordinates of these stations)."

Due to conflict with the regulatory requirements stated in the License, EIS, and SER, an amendment request for the relocation of environmental station A-5 will be required. Therefore, the SERP hereby denies the request.

Michael J Zumwalt ______ 7/15/99

Mike Zumwalt, Chief Financial Officer — Date

______ 7-15-99

Mark Ledoux, CRSO — Date

Paul J Larsen ______ 7/15/99

Paul Larsen, Director of Operations — Date

(Temporary member on behalf of the Vice President of Operations)

Bob Reifsnyder PT via telecom ______ 7/15/99

Bob Reifsnyder, Corporate Quality Assurance Manager — Date

(Temporary member)

Envirocare of Utah, Inc.
The Safe Alternative

SERP Request Form
EC98005, Revision 1

Date:	7-9-99	Docket Number: (to be assigned)	11e.(2)99-012
Requested By: Mark Ledoux		Department: Health Physics	

Revision/Change Requested: Review 11e.(2) license for ramifications of the 200ft extension of LARW. This must be signed and approved prior to submittal of 200ft LARW amendment

Technical Justification\References:

Ensure compliance with EIS since site (environmentally) is considered a whole.

Affected Documents: (Identify any and all documents that will be affected by this change request)

SERP Coordinator:	Accept ✓	Reject ☐	SERP Scheduled: 7-13-99
Signature: Reese Parker		Date: 7-9-99	

Attach all supplemental information necessary to support the requested change; including a statement of determination when a license amendment is not necessary in your opinion.

Envirocare of Utah, Inc. 11e.(2) License

SAFETY AND ENVIRONMENTAL REVIEW PANEL (SERP)
PANEL FINDINGS

In accordance with the Safety and Environmental Review Panel (SERP) Administrative Procedure, ADMIN-5, the panel has determined that the provisions of the procedure have been met to change Envirocare 11e.(2) License, SMC-1559, as follows:

Docket Number:	11e.(2)99-014
Date:	September 7, 1999
Action Requested:	To authorize reorganization as proposed in Section 18 of the License Application and approve revision to Section 17 of the License Application. Changes included in the revisions affect the Radiation Safety and Operations Areas.

The SERP has reviewed the requested action and the requirements of License Condition 9.4 (SMC-1559) and finds as follows:

a. **The change does not conflict with any requirement specifically stated in the license (excluding material referenced in License Condition 9.3), or impair the licensee's ability to meet all applicable NRC regulations.**

The license conditions have been reviewed and it has been determined that the requirements of license condition 9.4 have been met, indicating that granting approval for the changes in organization for the Radiation Safety and Operations Area does not require a request to amend the license. The requested change does not conflict with any required actions pursuant to 10 CFR Part 20.

b. **There is no degradation in the essential safety or environmental commitments in the license application, or provided by the approved reclamation plan.**

This revision does not affect the essential safety or environmental commitments in the license application, or provided by the approved reclamation plan. As such, a safety and environmental analysis was not necessary for the determination of the request.

c. **The changes are consistent with the conclusions of actions analyzed and selected in the site Environmental Impact Statement (NUREG-1476) dated August 1993, and the Safety Evaluation Report dated January 1994.**

The EIS and the SER have been reviewed to confirm that the requested action is consistent with the conclusions of these documents.

d. **In consideration of the above, the SERP has determined that the change does not require an amendment to Radioactive Materials License SMC-1559.**

Discussion:

SERP Request Docket # 11e.(2) 99-013 requests that Section 18 and Section 17 of the License Application be revised to reflect the proposed organizational structure. The specific changes involve positions in the Operations and Radiation Safety Areas. The position of Deputy Corporate Radiation Safety Officer has been separated into two parts: one for Operations, and one for licensing. Also, the positions of Assistant Radiation Safety Officers have been instituted to assist the Area Facility Managers in the performance of Operation tasks. The Assistant Radiation Safety Officers will report to the Site RSO for radiation safety issues and the Facility Managers for operational issues. Additional changes included in the redline version are those that were included in SERP 11e.(2)99-006. SERP 99-006 was denied and a formal amendment request is to be submitted to the NRC. Changes included in SERP 99-006 were the change in authority over ground water and the CRSO.

Because the request does not conflict with any regulatory requirements, the SERP hereby approves the request.

Michael J. Zumwalt — 9-9-99
Mike Zumwalt, Chief Financial Officer — Date

9-9-99
Mark Ledoux, CRSO — Date

9/10/99
Paul Larsen, Director of Operations — Date
(Temporary member on behalf of the Vice President of Operations)

R. Reifsnyder (via telecom) — 9/7/99
Bob Reifsnyder, Corporate Quality Assurance Manager — Date
(Temporary member)

SECTION 18. ORGANIZATION

18.1 ORGANIZATIONAL STRUCTURE

18.1.1 Design, Construction and Pre-Operational Responsibilities

The operations and design of the Clive facility is described in detail in Sections 4 and 16. The waste material is placed in an earthen embankment, compacted in place, and covered with barriers to reduce radon emanation below Commission guidelines and to protect the embankment from the effects of weather erosion.

During the development and preparation of this application, Envirocare has utilized the services of the following consultants/contractors:

1. Donald W. Hendricks, CHP, President
   DON HENDRICKS AND ASSOCIATES, INC.
   609 No. Crestline Drive
   Las Vegas, Nevada 89107
   702/878-4420

2. Jeff Throckmorton, CIH, President
   HEALTH & SAFETY SERVICES, INC.
   10508 Aberdeen Lane
   Highland, Utah 84003
   801/756-0063

3. Gary M. Sandquist, Ph.D.
   1738 Ramona Avenue
   Salt Lake City, Utah 84108-3110
   801/486-8521

4. Craig B. Forster, Ph.D.
   3479 East Quad Road
   Salt Lake City, Utah
   801/581-3864

5. Stanley L. Plaisier, P.E.
   BINGHAM ENVIRONMENTAL, INC.
   5160 West Wiley Post Way
   Salt Lake City, Utah 84116
   801/532-2230

6. T. Leslie Youd, Ph.D.
1132 East 1010 North
Orem, Utah 84057
804/378-6327

7. Blair McDonald, P.E.
343 South 1000 East
Salt Lake City, Utah 84102

Envirocare of Utah, Inc., with the assistance of these consultants, developed the personnel monitoring systems, data/record keeping systems, disposal material analysis and handling procedures, environmental monitoring systems, employee training, and general health and safety procedures and other technical supporting information for the 11e.(2) disposal project.

18.1.2 Operational Phase

The operational phase is also the construction phase of this proposed disposal project, in that the disposal project is discussed in Section 4 and 16.

A conceptual organizational chart is included as Figure 18.1, showing by responsibility the major divisions of Envirocare:

1. The peripheral activities of Scheduling, Accounting, and Marketing are represented on the organizational chart but do not need to be further described in this application.

2. President. The President oversees and provides direction and leadership for the operation. At a minimum, the president will:

   a. Promulgate company policies that identify his commitment to safety, the importance of compliance with requirements, the employees responsibilities to identify safety concerns to management, the need for adherence to company procedures, etc.
   b. Visit the site and observe the operations at least quarterly.
   c. Receive for his review summary audit reports, follow-up reports, close-out reports, NRC inspection reports and State inspection reports to ensure operations are conducted in accordance with Envirocare's high standard for quality and safety.

3. Corporate Radiation Safety Officer (CRSO) - Responsible to the Sr. Vice President of Compliance and Development ~~of Compliance and Licensing~~ and works very closely with the Director of Operations and Site ~~(Field)~~ Radiation Safety Officer (SRSO). The CRSO is

responsible for implementation of and compliance with all protocols and procedures of the radioactive materials license, including, health and safety monitoring, environmental monitoring, training, and personnel monitoring. The CRSO ensures that adequate instrumentation and equipment is used and that adequate measurements are made to ensure that all applicable standards for personnel exposures to radiation and radioactive materials are satisfied including:

- Shipping and Receiving of Radioactive Materials
- Airborne radioactivity
- Surface contamination
- Internal and external exposures
- Effluents
- Environmental monitoring
- ~~Ground water monitoring~~

The CRSO shall also be responsible for the annual report which summarizes all of the previously mentioned information. The annual report will be provided to the President, ~~and~~ the Sr. Vice President of Compliance and Development, and the Sr. Vice President of Operations and Business Development ~~President of Operations~~ for review and appropriate actions.

The CRSO has authority to terminate any activities on the site that are deemed to be unsafe. The CRSO may also suspended activities until hazard-abatement measures have been performed. The CRSO is responsible for health physics and radiation protection, training, and safety review.

It is anticipated that the CRSO will work 20 hours per week on issues related to the 11e.(2) project. The remainder of his time will be used to work on issues related to the Low Activity Radioactive Waste (LARW) project currently operating at the Clive site.

4. Site ~~(Field)~~ Radiation Safety Officer (SRSO) - The SRSO is responsible to the CRSO and works very closely with the Site Facility Manager. The SRSO or designee is responsible for on-site radiation safety and implementation of and compliance with all protocols and procedures of the radioactive materials license, including health and safety monitoring, environmental monitoring, training, and personnel monitoring. The SRSO determines whether adequate instrumentation and equipment are being used and whether adequate measurements are made to ensure that all applicable standards for personnel exposures to radiation and radioactive materials are satisfied. The SRSO is also responsible for oversight

of gamma spectral analysis, the environmental program, and instrument program. The SRSO provides technical direction for radiological laboratory functions.

The SRSO has authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. This determination may be made unilaterally or upon receiving reports of suspect conditions from other site supervisors, contractors, visitors or employees.

It is anticipated that the SRSO will work 20 hours per week on issues related to the 11e.(2) project.

5. ~~Operations~~ Assistant Radiation Safety Officers (~~O~~ARSO). Assistant Radiation Safety Officers are designated to each area of operation (i.e., Mixed Waste Treatment, Mixed Waste Disposal,LARW/11e.(2).The ~~O~~ARSO's ~~is~~ are responsible for managing the health physics team, performing daily site inspections, and observing field operations. The ~~O~~ARSO's can serve as acting SRSO and report~~s~~ to the SRSO.

The ~~O~~ARSO's haves authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. This determination may be made unilaterally or upon receiving reports of suspect conditions from other site supervisors, contractors, visitors, or employees.

~~6. Lab Assistant Radiation Safety Officer (LARSO). The LARSO is responsible for oversight of gamma spectral analysis, environmental program, and instrument program. Provides technical direction for radiological laboratory functions. The LARSO can serve as acting SRSO and reports to the SRSO.~~

~~The LARSO has authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. This determination may be unilaterally or upon receiving reports of suspect conditions from other site supervisors, contractors, visitors, or employees.~~

6~~7~~. The Environmental Coordinator is responsible to the ~~LARSO~~SRSO. The Environmental Coordinator has authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. The Environmental Coordinator is charged with carrying out the environmental monitoring activities on site including:

a. Implement applicable radiation control regulations and all provisions of radioactive material license.
b. Data base management/record keeping to document all environmental monitoring activities at the site.
c. Analysis of disposed material to document receipt and disposition.
d. Analysis of disposal material to document receipt and disposition
e. Other duties as assigned

7~~8~~. ~~Radiation Technicians (~~Health Physics Specialists ~~II)~~ are responsible to the ~~O~~appropriate ARSO for the Area assigned (i.e., Mixed Waste Treatment, Mixed Waste Disposal, or LARW 11e.(2)) and are trained by and have their work reviewed by the SRSO. Health Physics Specialists ~~II~~ have direct access to the Facility~~Site~~ Manager and SRSO on matters dealing with radiological safety. Health Physics Specialists ~~II~~ will work on both the 11e(2), Mixed Waste~~, and~~, and LARW operations. Health Physics Specialists ~~II~~ have the authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. They are charged with carrying out the health physics activities on site including:

a. Implement applicable radiation control regulations and all provisions of radioactive material license.
b. Personnel monitoring of Envirocare and contractor employees.
c. Assist in conducting training for new employees or refresher training for incumbent employees.
d. Supervision of truck/equipment decontamination facility.
e. Data base management/record keeping to document all disposal and health physics activities on site.
f. ~~f.~~ Perform reviews of previous radiation dose records with individual site workers.
g. Maintain continuous surveillance of site operating conditions and act to prevent actions which might result in the release or spread of radioactivity.
h~~g~~. Other duties as assigned.

8~~9~~. ~~Radiation Monitors (Health Physics Specialists I)~~Access Control Technicians are responsible to the ~~L~~ARSO of the Mixed Waste Treatment area or the ARSO of the LARW 11e.(2) Area. They are charged with carrying out minimal health physics activities on site:

a. Implement applicable radiation control regulations and all provisions of radioactive material license.

b. Access Control monitoring of Envirocare and contractor employees.

c. Manning of Access Control portal.

~~d. Manning of truck/equipment decontamination facility.~~

~~e. Maintain continuous surveillance of site operating conditions and act to prevent actions which might result in the release or spread of radioactivity.~~

d~~f~~. Perform and document weekly surveys of radiation dose rates and surface contamination in assigned areas.

e~~g~~. Other duties as assigned.

9~~10~~. Sr. Vice President of Operations and Business Development ~~of Operations - The~~ The Sr. Vice President of Operations and Business Development ~~of Operations~~ reports to the President of Envirocare. The Sr. Vice President of Operations and Business Development ~~of Operations~~ is responsible for the overall management of direct operations and support functions for the disposal facility. The Sr. Vice President of Operations and Business Development ~~of Operations~~ works closely with other corporate personnel to ensure that all operations are conducted in a planned and safe manner in accordance with all regulatory requirements.

The Sr. Vice President of Operations and Business Development ~~of Operations~~ shall establish and promulgate departmental employee policy when needed. ~~This position is directly responsible for negotiating contracts with subcontractors.~~ The Sr. Vice President of Operations and Business Development shall also ~~be responsible~~be responsible for investigating innovative methods of improving operations and/or efficiency.

10~~1~~. Sr. Vice President of Compliance and Development ~~of Compliance and Licensing -~~ The Sr. Vice President of Compliance and Development ~~of Compliance and Licensing~~ reports to the President of Envirocare. The Sr. Vice President of Compliance and Development ~~of Compliance and Licensing~~ oversees and directs compliance, licensing, and permitting activities at Envirocare; including such areas as quality assurance, radiation safety, environmental monitoring, ground water monitoring, safety, training, and regulatory affairs. ~~, and personnel management.~~

The Sr. Vice President of Compliance and Development ~~of Compliance and Licensing~~ shall oversee and facilitate permit and license renewals, modifications, and amendments. This position will set compliance objectives jointly with the Operations Department personnel. Direction and support will be provided for policy development and site training to assist in ensuring compliance.

1~~2~~1. Director of Operations - The Director of Operations must be an experienced Civil Engineer, or other relevant engineering degree. The Director of Operations reports to the Sr. Vice President of Operations and Business Development~~of Operations~~ and is charged with the responsibilities of the operations of the waste disposal site in an efficient and safe manner in accordance with design specifications and all applicable regulations.

The Director of Operations is responsible for ~~the management of site structural engineering, soil mechanics, materials, and operations support.~~site operations including laboratory management, cell construction, waste management and disposal. The Director of Operations is directly responsible for negotiating contracts with subcontractors.

12. The Corporate Engineering Manager - The Corporate Engineering Manager performs certification of engineering design drawings, project plans, construction reports, and As-Built Drawings. The Corporate Engineering Manager is responsible for the management of technical and engineering support, including site structural engineering, soil mechanics, materials, and hydraulic engineering. The Corporate Engineering Manager provides or procures services from internal resources or technical contractors as necessary; provides tehcnical and engineering support for the operation including site layout and design reviews; and approves with QA oversight, those designs and specifications.

~~12~~13. The Site Facility Manager- The Site Facility Manager is responsible for the day-to-day operation of the Clive facility. The Site Facility Manager is to work closely with the SRSO to assure that all aspects of site operation are conducted according to the applicable regulations. The Site Facility Manager has limited specific responsibilities so that his efforts can be used in ensuring the effectiveness of the overall operational activities at the site. The Site Facility Manager is also responsible for the management of the site maintenance support and fire protection.

14. ~~14.~~ Production ~~(Site)~~ Engineer - The Production Engineer is responsible to the ~~Director of Operations~~Corporate Engineering Manager and is ~~in charge of the daily on-site supervision of the construction/disposal activities, including interfacing with the Construction Contractor, checking field~~

~~procedures and record keeping to assure proper quality assurance.~~responsible for overseeing the production, scheduling, and coordination aspects of facility construction with the exception of QA (which is the responsibility of the QAM). During construction, the Production Engineer will regularly inspect the construction site. The Production Engineer will coordinate the selection of the construction contractor(s) and administration of the construction contract, including any changes. The Production Engineer will review proposed design, engineering, or construction changes and submit these changes to the Corporate Engineering Manager for approval.

~~15~~15. Site Engineer – The Site Engineer is responsible for construction quality control, overseeing the production, scheduling and coordination aspects of facility construction, with the exception of QA (which is the responsibility of the QAM). During construction, the Site Engineer will regularly inspect the construction site. The Site Engineer will coordinate the selection of the construction contractor(s) and administration of the construction contract, including any changes. The Site Engineer will review proposed design, engineering, or construction changes and submit these changes to the Corporate Engineering Manager for approval.

16. Construction Contractor - responsible to Site Facility Manager to perform construction, ~~earth moving~~earth moving, and disposal activities in accordance with approved procedures and specifications. -The Construction Contractor is also charged with maintaining compliance with all provisions of UOSHA and making records available for review by the Industrial Hygiene Consultant.

~~16~~17. The ~~Permitting and Compliance~~Compliance and Permitting Manager – The Compliance and Permitting Manager is responsible for ~~the operation of the Hazardous Waste portions of the site operation.~~ Initiating, producing, and obtaining appropriate licenses and permits. The Compliance and Permitting Manager oversees the administration of the Air Quality Program and the preparation of all reports submitted in accordance with Envirocare's licenses and permits. The ~~Permitting and Compliance~~Compliance and Permitting Manager has the authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until abatement measures have been performed.

~~17~~18. The Corporate Quality Assurance Manager ("CQAM") is responsible for ensuring that the quality assurance requirements outlined in the Quality Assurance Program Document (QAPD) are implemented. The reporting relationships shown in Figure 18.1 allow the CQAM sufficient authority and autonomy to implement and direct the QAPD; to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions independent of undue influences, and responsibilities, such as costs and schedules. As such, the CQAM reports directly to the Sr. Vice President of Compliance and Development ~~of Compliance and Licensing~~ in implementing the QAPD.

~~18~~19. Outside Contractual Assistance.

As indicated in Section 18.1.1, Envirocare has access to qualified consultants to assist in the development and implementation of radiological health and safety plans, environmental monitoring programs, industrial hygiene and safety programs. These consultants will be utilized extensively to provide reviews of safety, employee training, evaluation of fire protection systems, and quality assurance reviews in addition to continuous operations support. These contractors are responsible to the President of Envirocare.

~~Envirocare will hold staff meetings, generally once a week, to coordinate all phases of Envirocare's operations. Staff meetings will include the Vice President of Compliance and Licensing, Vice President of Operations, CRSO, Director of Operations, and other personnel. Envirocare will hold operations staff meetings, generally once a week, to coordinate operations at Envirocare's facilities. Operations staff meetings will include the Director of Operations, SRSO, Site Manager, and other personnel.~~

All Envirocare management personnel and personnel with safety responsibilities will have free access to each other to resolve immediate safety, operational or other issues.

In order to more fully outline the responsibilities assigned, the following chart is provided with the applicable assignments:

RESPONSIBILITY	POSITION
Structural, soil mechanics, materials, hydraulic engineering	E
Health physics, radiation protection	R
Maintenance Support	S
Operations Support	S
Quality Assurance	Q
Training	V
Safety Review	R
Fire Protection	E
Outside Contractual Assistance	~~P~~O
R-Corporate Radiation Safety Officer	

Q-Corporate Quality Assurance Manager
E- ~~Production~~ Corporate Engineering Manager
S-Site Facility Manager
V- Sr. Vice President of Compliance and Development ~~of Compliance and Licensing~~
~~P- President~~ O-Director of Operations

18.2 QUALIFICATIONS OF APPLICANT

Envirocare is cognizant of the radiological nature of the disposal materials to be handled in this operation. Envirocare feels a major emphasis lies in the selection of the CRSO, as well as the Director of Operations and the construction contractor.

18.2.1 Corporate Radiation Safety Officer

The Corporate Radiation Safety Officer (CSRO) will have the following minimum qualifications:

1. B.S. graduate in Engineering, Chemistry, Physics, or physical science-related field; and,
2. Five years of supervisory experience in NORM, uranium mining/milling operations, UMTRA Projects or other related fields where handling and/or disposal of low level radioactive materials are involved.

18.2.2 ~~Field (~~Site~~)~~ Radiation Safety Officer (SRSO)

The Site Radiation Safety Officer (SRSO) will have the following minimum qualifications:

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.
2. Two years of supervisory experience in uranium mining/milling operations, UMTRA Projects, or NORM disposal operations where handling and/or disposal of low-activity or low-level radioactive materials are involved.

~~18.2.3 Laboratory Supervisor~~

1. ~~Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.~~
2. ~~Ability to learn and understand radiation safety principles and practices.~~

~~3. Ability to follow protocol and procedures, and maintain environmental monitoring schedules established by the CRSO.~~
~~4. Ability to work with contractor personnel and supervise radiation monitor(s) during operations.~~

18.2.3~~4~~ ~~Radiation Technicians (~~Health Physics Specialist ~~II)~~

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to follow protocol and procedures, and maintain health physics schedules established by the CRSO.

4. Ability to work with contractor personnel and supervise radiation monitor(s) during operations.

18.2.4~~5~~ ~~Radiation Monitor(s) (Health Physics Specialists I)~~Access Control Technician

1. Ability to learn and understand radiation safety principles and practices.
2. Ability to follow protocol and procedures, and maintain schedules established by the CRSO.
3. Ability to work with contractor personnel and oversee work areas, such as the unloading and wash down facilities.

18.2.5~~6~~ Director of Operations

The Director of Operations will have the following minimum qualifications:
~~As indicated earlier, the Director of Operations must be a~~

1. Civil Engineer, or other relevant engineering degree, with three years of experience in earth-moving construction projects~~. and must also be~~

2. -basically familiar with the principles of radiation safety, as applied to these types of projects.

18.2.6~~7~~ Site Facility Manager

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to manage the operations at the site. To set schedules for personnel and complete assignments in a timely manner.
4. ~~4.~~ Ability to work with contractor personnel and supervise their work during operations.

18.2.7. Corporate Engineering Manager

The Corporate Engineering Manager will have the following minimum qualifications:

1. A Bachelor's degree in an engineering field

2. At least six years experience

3. Shall be a Utah certified professional engineer

18.2.~~8~~ 8 ~~Site (~~Production~~)~~ Engineer ~~or Site Engineering Technician~~

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; and one year of experience as a engineering technician or equivalent.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to follow protocol and procedures, and maintain construction operations and records as established by the Director of Operations
4. ~~4.~~ Ability to work with contractor personnel and supervise construction operations~~-~~.

18.2.~~9~~ 9 Site Engineer

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; and one year of experience as a engineering technician or equivalent.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to follow protocol and procedures, and maintain construction operations and records as established by the Director of Operations.
4. Ability to work with contractor personnel and supervise construction operations.

18.2.10 Construction Contractor

The construction contractor will be required to operate in accordance with the construction operation safety plan that includes, a comprehensive radiation safety/health physics plan. In addition, the construction contractor must demonstrate a willingness and commitment to comply with certain provisions, as outlined in Section 7, to which contractors may not normally be subjected:

1. Radiation monitoring of all construction personnel.
2. Decontamination and frisk-monitoring of personnel at access control portal.
3. Maintenance of Personnel in/out logs at access control.
4. Wearing protective clothing.
5. Decontamination of all vehicles and equipment prior to leaving the restricted area(s).
6. Making available to the Industrial Hygiene Consultant any requested records pertaining to employee exposure to occupational hazards, and to employee accidents.

18.2.~~10~~ 11 ~~Permitting & Compliance~~Compliance and Permitting Manager

The ~~Permitting and Compliance~~Compliance and Permitting Manager will have the following minimum qualifications:

1. B.S. graduate in Engineering, Chemistry, Physics, or physical science-related field; and,
2. Supervisory experience in hazardous waste operations, where handling and/or disposal of hazardous materials are involved.

18.2.~~11~~ 12 Corporate Quality Assurance Manager

The Corporate Quality Assurance Manager will have the following minimum qualifications:

1. Undergraduate technical degree, preferably in a science or engineering field, or a closely associated discipline, or equivalent technical experience.
2. For construction QA, the CQAM should have an understanding of materials testing methods for soil classification and compaction, of surveying methods for establishing the location of point coordinates and elevations, and of general construction techniques.
3. For laboratory QA, the CQAM should have an understanding of laboratory safety, methodology, and general chemistry concepts.

4. For health physics, the CQAM should have an understanding of industrial health and safety concerns, testing techniques, and ALARA concepts.

18.3 TRAINING PROGRAM

The training program for all contractor employees, Envirocare personnel and outside contractors/consultants is addressed in Section 17.5.6.3. All persons using or working with the radioactive material receive training which is commensurate with the materials he/she will be handling.

At the date of this submittal, Envirocare is current with the training requirements outlined in Section 17.5.6.3.

18.4 EMERGENCY PLANNING

The maximum credible accident at the Envirocare site would be the accidental dumping of a load at some location other than the disposal cell. The model used to calculate the permitted radionuclides in waste accepted at the site was designed to limit total occupational doses to 5 rem per year. If a load containing waste with the maximum permitted concentration was accidentally dumped, requiring its removal to the disposal cell, and if a full day is assumed for its removal, the maximum predicted dose to an employee would be 0.025 rem. Considering that most of the land within 10 miles of the site is under Bureau of Land Management (BLM) control and that there are no nearby residents, any dose received by a person outside of the controlled area would be a small fraction of 0.025 rem. Envirocare has an emergency response plan which is incorporated as part of the training procedures in Appendix C.

18.5 REVIEW AND AUDIT

The construction review and audit requirements are addressed in Section 14.1.4. The radiation safety audits are described in Section 14.7.

18.6 FACILITY ADMINISTRATIVE AND OPERATING PROCEDURES

18.6.1 Scope of work

At this time it is impossible to exactly state the amount of waste material to be handled or buried in a year. It is stated elsewhere in this application, that Envirocare anticipates approximately 500,000 tons per year. It is also impossible to estimate the time frame or schedule(s) for arrival of the material at the site.

18.6.2 Administrative Procedures

All personnel who work at the Envirocare facility will be required to abide by all site regulations and all requirements of this application. All violations of these requirements will be recorded on site violation forms and turned in to the Director of Operations. The implementation of this program will be under the direction of the Director of Operations

18.6.3 Operating Procedures

As described in the previous sections there are several people on the site who have the authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. Examples of situations that would require that the site be closed until remediation of the problem would be:

1. Windy conditions which cause unsafe conditions.
2. Construction equipment operating in an unsafe condition.
3. Lack of trained personnel to operate the site.

18.6.4 Required Personnel

Envirocare will only perform specific operational activities when the trained personnel responsible for these activities are on site. For example, a Field Testing Inspector or equivalent must be on site whenever material is to be placed on a portion of the embankment that needs soil density verification.

Whenever the Clive facility is in full operation the SRSO or authorized designee must be present on site.

18.7 Safety and Environmental Review Panel

Envirocare will establish a "Safety and Environmental Review Panel (SERP)." The SERP shall consist of a minimum of three individuals. One member of the SERP shall have expertise in management and will be responsible for managerial and financial approval changes; one member shall have expertise in operations and/or construction and shall have expertise in implementation of any changes; and, one member shall be the Corporate Radiation Safety Officer or equivalent. Other members of the SERP may be utilized as appropriate, to address technical aspects, in areas, such as health physics, groundwater hydrology, surface water hydrology, specific earth sciences, and others. Temporary members, or permanent members, other than those specified above, may be consultants. The SERP shall convene at least monthly to review, evaluate and make determinations regarding the licensing requirements for the following actions, or address other matters pertaining to the SERP.

(1) Make changes in the facility or process, as presented in the application.
(2) Make changes in the procedures presented in the application.
(3) Conduct tests or experiments not presented in the application.

Envirocare will file an application for an amendment to the license, unless the following conditions are satisfied.

(1) The change, test or experiment does not conflict with any requirement specifically stated in this license (excluding the License Condition referencing the License Application or Reclamation Plan), or impair the licensee's ability to meet all applicable NRC regulations.
(2) There is no degradation in the essential safety or environmental commitments in the license application, or provided by the approved reclamation plan.
(3) The change, test, or experiment is consistent with the conclusions of actions analyzed and selected in the Final Environmental Impact Statement dated August 1993 (NUREG-1476).

Envirocare will maintain records of any changes made pursuant to this section. These records shall include written safety and environmental evaluation, made by the SERP, that provide the basis for the determination that the change is in compliance with the requirements referred to above. Envirocare will furnish, in the annual report to NRC, a description of such change, tests, or experiments, including a summary of the safety and environmental evaluation of each. Envirocare will annually submit changed pages to its license application to reflect changes made under this section.

SECTION 17. SAFETY ASSESSMENT

17.1 RELEASE OF RADIOACTIVITY

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

17.1.1 Characterization of Waste

17.1.1.1 Radionuclides

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Summarizing the data presented above, the following radiological waste character is anticipated for the Envirocare 11e.(2) disposal site. Considering the relative proportions of lower and higher activity waste at the site, Envirocare estimates that the overall average concentration for any radionuclide will be approximately 500 pCi/g; however, individual sites may vary widely around that average, as described above. Because of this, individual shipments of wastes may contain higher average concentrations of Ra-226 and Th-232. In the context of waste deliveries to the disposal site a shipment is taken to mean a single waste-hauling truck or rail car from a single generator. Weighted average concentrations in a shipment must not exceed 4,000 pCi/g for natural uranium or any radionuclide in the Ra-226 series; 60,000 pCi/g of thorium-230; or 6,000 pCi/g for any radionuclide within the thorium series, although they may be present at those concentrations together.

A conservatively-high estimate of the volume of material to be handled and disposed of at the site would be one-half million (500,000) tons/year. Assuming an average Ra-226 and Th-232 concentration of 500 pCi/g, the estimated annual average total activity disposed of would be 227 Curies for each of the radionuclides. Since the daughter products may be assumed to be in secular equilibrium, there would be approximately 227 Curies of each of the other important radionuclides, such as Ra-228 and Ra-224. The amount of Uranium would be expected to be less than 25 percent that of Ra-226. The average Th-230 concentration is expected to be similar to that of Ra-226 and will depend upon the disequilibrium of the radionuclides in that decay series. The actual amount of radioactivity disposed of in a given year will vary around the estimated 227 curies per radionuclide as actual concentrations and disposal amounts vary.

17.1.1.2 Chemical Constituents in the Waste

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

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

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

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

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

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

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

<u>METALS</u>
Arsenic
Barium
Beryllium
Cadmium
Chromium
Cyanide
Fluorine
Lead
Mercury
Molybdenum
Nickel
Selenium
Silver

<u>VOLATILE ORGANICS</u>
Acetone
2-Butanone
Chloroform
Carbon disulfide
1,2-Dichloroethane
Methylene chloride
Naphtha

<u>SEMI-VOLATILE ORGANICS</u>
Diethylphthalate
2-Methylnaphthalene

<u>RADIONUCLIDES</u>
Radium-226
Radium-228
Thorium-230
Thorium-232
Uranium

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

17.1.2 Infiltration

Section 4.1.1 discusses principal design features to minimize water infiltration into the embankment and disposed materials. As indicated in that section, calculations in Appendix M demonstrate that the amount of precipitation that infiltrates into the

embankment and percolates to the shallow groundwater under proposed conditions is negligible.

17.1.3 Radionuclide Release - Normal Conditions

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

1. Release of interstitially trapped radon and thoron gas when handling bulk wastes.
2. Exhalation of radon gas from embankment area(s) that have not been covered with the compacted clay radon barrier.
3. Exhalation of radon gas from embankment area(s) that have been covered with the compacted clay radon barrier.
4. Exhalation of thoron gas from the top layer of embankment areas which have not been covered with a layer of non-thorium-containing waste or clean clay.
5. Localized resuspension of dust from waste handling operations.
6. Windblown materials from the embankment and unloading area.

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

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

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

17.1.3.1 Off-site Impacts from Normal Operations

M&A (Appendices A and A-1) provided estimates of projected radionuclide release rates and radiological impacts during site operations, assuming waste which exhibits the radiological characteristics estimated for the overall 11e.(2) profile (500,000 tons per year of waste containing 500 pCi/g of each of the radionuclides in the uranium and thorium series). While these Appendices demonstrate compliance with 10 CFR 20.1301 and 10 CFR 20.1302 under the assumed conditions, they do not completely serve the purpose of evaluating the variable characteristics of waste quantities and

radionuclide concentrations which are expected to occur annually, or over shorter periods of time. M&A performed a sensitivity analysis of Envirocare's waste management procedures and waste characteristics (Appendix A-2). This analysis permits each waste handling procedure, from receipt to final closure, to be evaluated for its environmental impact while handling any quantity of wastes at any specified radioactivity concentration. Output from the analysis of Appendix A-2 will be used as input to the calculational spreadsheet described in Appendix A-3 to provide guidance to Envirocare planners in scheduling waste shipments and planning waste handling operations to meet the effluent concentration limits of Table 2, Appendix B to 10 CFR 20.1001 - 20.2401. The application of Appendices A-2 and A-3 to waste management will allow Envirocare to manage wastes within an envelope of quantities and radioactivity characteristics during the year while meeting the overall environmental results of Appendices A and A-1.

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

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

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

The dose calculations above, from Appendices A and A-1, were based on a single assumed average concentration in waste with an annual total of 500,000 tons of waste disposed, or an annual disposal of 227 Ci of each of the radionuclides in the uranium and thorium series. Occupational and environmental doses are shown to be almost completely dependent upon the total amount of radioactivity managed. While the use of Appendices A-2 and A-3 provide considerable flexibility in waste management, the reliance upon the modelling of Appendices A and A-1 will assure that occupational and environmental 1impacts are as described in those appendices. With this option, Envirocare can safely dispose of any combination of radioactivity concentrations up the shipment limits of 4,000 pCi/g for natural uranium and any radionuclide in the ^{226}Ra series; 60,000 pCi/g of thorium-230; and 6,000 pCi/g for any radionuclide in the thorium series. Application of this approach would

automatically restrict the amount of waste which could be received at higher concentrations.

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

17.1.3.2 Occupational Radiation Exposures

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

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

$$ {}_{b}D = 0.23\, E_b\, c $$

where: ${}_{b}D$ = Dose rate from an infinite cloud (rad/s)
E_b = Average beta energy per disintegration (MeV/dis)
c = Concentration of the beta emitting isotope in the cloud (Ci/m^3)

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

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

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

periods, Envirocare will closely monitor internal and external exposures to maintain TEDE as low as reasonably achievable and, in all cases, below the standards of 10 CFR 20.1201.

17.1.4 Radionuclide Release - Accidents or Unusual Operation Conditions

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

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

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

1. on-site truck turnover or collision
2. train derailment
3. flooding
4. tornado.

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

occurred into the Great Salt Lake. Impacts of these events were found to be negligible compared to the impacts from normal operations.

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

The following accidental on-site releases have been evaluated:

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

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

1.3 x 10^{-6}/km x 53 loads/day x 260 days/year x 1 km/load
= 1.8 x 10^{-2} or about 1.8%.

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

For a theoretical truck accident involving a yellowcake shipment, a 24-hour release period, all particles in the respirable range, and a population density of 7.5 persons per square mile, NRC estimated 50 year dose commitments to the lungs of the general public in the range of 0.7 to 9 person-rem. The yellowcake specific activity is about 6.77 x 10^5 pCi/g while the average uranium or thorium concentrations expected at Envirocare would be 500 pCi/g, or a factor of 1300 lower. Individual shipments to Envirocare might have ^{226}Ra concentrations as high as 4,000 pCi/g, or similar to those found in uranium mill tailings. Concentrations of ^{232}Th in a small

fraction of shipments could be as high as 6,000 pCi/g. The dose per unit intake via inhalation is higher for Th-232 wastes than for yellowcake by up to a factor of 1000, depending upon the chemical form and radionuclide mix. Therefore, the postulated off-site public doses could be approximately an order of magnitude higher than for a yellowcake spill under the same circumstances. However, the population distribution around the Clive site is insignificant compared to that in the NUREG calculation and, therefore, the off-site population dose would be inconsequential.

For on-site workers, there would be a very short exposure time since there would be no reason to stand downwind for 24 hours (or even one hour). Assuming an accident involving the spill of a load of waste with a concentration of 15,000 pCi/g; a period of three hours for cleanup with no use of respiratory protection; an airborne concentration of 1 mg/m^3; and a respiratory rate of 1.2 m^3/h a total of 54 pCi of each nuclide would be inhaled. Comparing these to the ALI's from Appendix B of 10 CFR 20.1.001 - 4201, the sum of fractions is 0.022. The external gamma dose, using the relationship of 3.1 mrem/h/pCi/g for Ra-226 from Appendix A Section 3.7.3 and doubling for the contribution from Ra-228, would be less than 140 mrem. Such a dose added to the projected maximum TEDE of 1,032 mrem/y would still be well within the permissible annual exposures for radiation workers. In actual fact, no workers would be present under such conditions without respiratory protection and would not be standing on the spilled waste for more than a few minutes.

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

<u>Train derailment:</u>

The probability of a train derailment occurring on the Clive site is not readily calculable. However, because of the short length of track involved, the small amount of train movement, the low train speeds compared to truck speeds, and the relatively small number of cars compared to truck shipments, the probability of a derailment should be much less than the probability of a truck accident.

The dose to the workers and to the population should be much less than that for an off-site derailment and spillage event since trained workers and equipment would be available to immediately use dust control measures to control releases and cleanup the spill. The DOE, as discussed above, concluded that the dose to cleanup workers and nearby residents from such an off-site spill was insignificant. As a worst case,

the same assumptions could be applied as for the truck accident scenario above, with the same low total dose to emergency response teams.

Flooding:

Flood control features for both the Vitro and Clive sites have been designed and constructed to prevent erosion or off-site transport of wastes from the sites by overland flooding. Details of the flood control features are provided in Appendix F. No off-site transport of radioactive waste by flooding is anticipated. Cleanup of contamination caused by dispersion of stored or already disposed waste within the controlled area by flooding would replace placement of waste as an activity and radiation doses to workers would be the same as, or lower than, those received during normal operations.

Tornado:

From NUREG-0706 the probability of tornado occurrence in Utah is probably in the range of 1 to 5 x 10^{-4}. NUREG-0706 also estimates the consequences of a tornado striking a model uranium mill. In this case about 12.6 tons of yellowcake is entrained in the vortex, the vortex dissipates at the site boundary, all of the yellowcake is respirable in size, and the cloud is dispersed as a volume source by the prevailing winds. Settling velocity is negligible. The model predicts a maximum exposure at 2.5 miles from the mill, where the 50 year dose commitment is estimated to be 0.83 micro-rem. At the fence line (1600 feet) the dose is estimated to be 0.22 micro-rem. Our wastes would have average activities considerably less than this but as discussed above, the TEDE per unit intake is higher, resulting in comparable doses at receptor locations. Since there are no nearby population groups, the significance of this very small potential dose is even more insignificant.

Severe Winds

In the preceding discussion of airborne exposures resulting from tornadoes it was concluded that the maximum 50-year dose commitment at 2.5 miles would be less than 1 micro-rem. That conclusion is derived from a NUREG-0706 analysis of tornado-dispersed yellowcake from a uranium mill and is considered of a comparable magnitude to the transport of Th-232 waste from the Clive Site under similar conditions.

While severe winds on the order of 35 m/s have been recorded in the vicinity, the occurrence is infrequent and the duration is short. Assuming an order of magnitude increase in airborne concentrations during severe wind conditions which occur approximately one percent of the time, the time-weighted average off-site exposure would increase by only 10 percent. This would result in a maximum additional

annual collective TEDE of less than 1 mrem to current nearby population groups (See Table 3.20, revised, Appendix A-1)

17.1.4.1 Transfer Mechanism - Groundwater

The possibility of contamination releases to known water resources is highly unlikely. Without extensive treatment, use of the water in the South Clive area would appear to be confined to very limited industrial uses. There is minimal potential for degradation of water quality in the vicinity of the south Clive site inasmuch as the water at the site has been characterized as a brine, with levels of many constituents often exceeding EPA primary or secondary drinking water standards by a large amounts.

Envirocare has commissioned a hydrogeologic study to more accurately describe the possibility of groundwater contamination. This report by Bingham Environmental (Appendix GG) concluded that it would take 400 to 600 years for leachate to travel through the unsaturated zone and then another 800 years to reach the nearest off-site well. No non-radiological constituent would reach the Ground water in less than 700 years.on site would be 191 years. Using this estimate, it would take well over 1,000 years for any groundwater from the 11e.(2) cell to reach the boundary of the Envirocare facility.

17.1.4.2 Transfer Mechanism - Air

Because of the location of the South Clive facility, the meteorological characteristics of the area, and the lack of population within 20 miles of the facility, the impact of air as a transfer mechanism for radioactivity is limited. The modelling study conducted by Momeni & Associates (Appendix A) concluded that the annual population TEDE (exclusive of doses to workers at the nearby hazardous waste operations) after 16 years of operation would be 0.016 person-rem/year. Calculated TEDE to the nearby hazardous waste workers would add approximately 0.5 person-rem/year.

17.1.4.3 Transfer Mechanism - Surface Water

The probability of contamination through surface water is highly unlikely inasmuch as there are no surface waters at the site. As is stated previously, "No surface-water bodies are present on the South Clive site. The nearest stream channel ends about 2 miles east of the site and is typical of all the drainage along the transportation corridors within about 20 miles of the South Clive site. Stream flows from higher elevations usually evaporate and infiltrate into the ground before reaching lower, flatter land. The stream channels are well defined in their upper reaches, but as they

approach the flatland the size of the channel reduces until there is no evidence of a stream."

17.1.4.4 Other Transfer Mechanisms

Because of the location of the South Clive facility, the sparse biota in the area, and the lack of population within 20 miles of the facility, the impacts of other transfer mechanisms such as gamma radiation through air and transfer of radioactivity through biotic pathways are very small.

17.1.5 Radionuclide Transport

The most significant radioactivity transport mechanisms are air, groundwater, surface water, direct radiation and biotic pathways. The five periods of principal concern to NRC (NUREG-1199) include the operational, closure, observational and surveillance, active institutional control, and passive institutional control periods. In reality, the periods of real concern should be operational and post-closure.

During the closure period one would not ordinarily expect continuing shipments of waste so exposures from air, surface water, direct radiation, and biotic pathways should be less than exposures received during the operational period. No new wastes are being received, old wastes are being covered, and the surface is being decontaminated.

During the observational and surveillance, active institutional control, and passive institutional control periods the site has already been decontaminated, wastes are covered and there should again be no changes in exposures.

The evaluations of Appendix A address exposure pathways for operational periods and were compared to regulatory standards. Results were used to determine potential exposures to on-and off-site personnel. As discussed in Sections 17.1.3.2 and 17.1.4.2, projected doses to on-site radiation workers are 1 rem/year or less and the annual regional population TEDE to off-site residents and nearby industrial workers is approximately 0.5 rem.

17.1.6 Assessment of Impacts and Regulatory Compliance

The M&A report addresses the specific impacts of releases under normal operating conditions. Release mechanisms were evaluated, exposures to workers and the public assessed, and the results compared to applicable standards and regulations. It was concluded that with the proposed waste characteristics and operating procedures, exposures to the workers and the public will be within acceptable limits

and the design will limit the radon flux to 20 pCi/m^2s as proposed in 10 CFR Part 40, Appendix A.

While the exposures to site custodial personnel during the active institutional control period were not specifically evaluated, all waste will have been covered, gamma exposure rates will be near background, and radon emission rates will be limited to the design criterion of 20 pCi/m^2s. There is no reason to believe that exposures during this period will be more than a small fraction of those to the workers during operations.

For a discussion of impacts of releases due to accidents or unusual operating conditions see Section 17.1.4. In general, because of the relatively low radionuclide concentrations of the Clive wastes, it is difficult to postulate an on-site accident that could cause significant exposures to on- or off-site personnel.

17.2 LONG-TERM STABILITY

The embankment design will provide long-term stability and be relatively maintenance-free after site closure. Long-term stability is discussed in detail in Sections 4 and 6.

17.3 CONSTRUCTION SAFETY

Envirocare has implemented a construction safety plan which covers both Envirocare and contractor employees. While the prime contractor is responsible for developing his own safety and health plan, Envirocare performs safety inspections of the contractor's on-site operations to assure compliance with UOSHA and Envirocare regulations. The content of the plan includes:

1. Purpose/Goals - Envirocare and Contractor commit to the following goals:
   a. Safe and health working conditions for all on-site personnel.
   b. Protection of the general public.
   c. Compliance with all governmental safety and health regulations.
   d. Reduce liability to Envirocare and contractor to a minimum.
2. Establish an organizational chart to define responsibilities for safety and health program direction and enforcement.
3. Emergency Medical Care.
4 Pre-planning for unusual occurrences.
5. Safety & Health training program.
6. Control and monitoring of Safety and Health Plan.
7. Industrial Hygiene plan.

8. Corporate Safety Program.

Any contractor that performs work for Envirocare on this project must formally take responsibility to obey all site rules. Any contractor who is to work for an extended period of time at the site must submit their own Health and Safety Program. All OSHA regulations will be under the jurisdiction of UOSHA. The Corporate RSO is responsible for overall development, direction and coordination of the Safety and Health Plan. The Site Manager is responsible for on-site implementation and enforcement of all safety and health provisions. It is recognized that industrial accidents pose a greater risk to employees than radiation risks and a significant effort is made to ensure a safe workplace. Employees are instructed to bring all health and safety concerns to their supervisor or the Site Manager. Unresolved concerns may be brought to the attention of UOSHA for immediate reconciliation.

The Safety and Health Plan relies on identification of risks, development of procedures to control those risks and to comply with UOSHA regulations, pre-employment safety training, continuing on-the-job safety training and on-going safety inspections of all operations. Radiation Technicians (Health Physics Specialist II), who are already trained in radiation safety, are also given responsibility to enforce all safety regulations.

17.4 Radiation Safety and Health Physics

17.4.1 Radiation Protection Policy

It is the policy of Envirocare, to maintain personnel/occupational radiation exposures as low as reasonably achievable (ALARA). Because of the nature of the 11e.(2) wastes, experience has shown that radiation exposures are normally low and Envirocare is committed to continuing to minimize exposures to the workers and the environment.

The average annual dose for 294 workers involved in the Vitro Remedial Action Project during 1986 was 50 mrem, with maximum exposures of 250 mrem. This maximum value is only 5% of the radiation dose standard of 10 CFR 20.101. Envirocare's experience with handling similar materials at it's LLRW facility was even better in that the highest total dose received during any year of Envirocare's five years (1988-1992) of operation was 200 mrem and the average annual dose equivalent was less than 50 mrem. The data are presented in Table 17.7.

In keeping with the ALARA principle, any reported personnel exposures in excess of 50 mrem/month will be investigated and documented by the Corporate Radiation Safety Officer(CRSO).

Procedures and methods to keep internal exposures ALARA include:

a. Dust suppression on all operational roads by application of magnesium chloride or watering at 2-hour intervals.
b. Speed limit of 35 mph on roads treated by dust suppressants; 10 mph on infrequently used roads.
c. Stopping operations in high wind conditions (all operations cease at winds of 40 mph; radiation safety personnel have authority to stop operations at lower wind speeds if dusting or other safety considerations warrant).
d. Placement of radon barrier on portions of the cell as they are completed.
e. Weekly area radiation surveys with investigation of increasing levels to determine the cause.
f. Requiring workers to wear respirations in areas of potential high dust concentrations, for example, the rollover and selected heavy equipment operations.
g. Pre-planning tasks which have the potential for higher than normal exposures to limit exposures through efficient use of time and handling procedures.

The Site Radiation Safety Officer (SRSO) will have the day-to-day responsibility for maintaining occupational and environmental radiation exposures ALARA, consulting such guidance documents as NRC Regulatory Guide 8.31, "Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Mills Will Be As Low As Reasonably Achievable" and draft Guide DG-8013, "ALARA Levels for Effluents from Materials Facilities." The SRSO will document ALARA activities including:

a. Reviews of new proposed disposal contracts to assure that Envirocare's procedures, facilities, and equipment are appropriate and sufficient to limit exposures to workers and the environment;
b. Monthly reviews of work area, perimeter, and environmental air monitoring results noting trends and adjusting work procedures when practical to further reduce potential exposures; and
c. Monthly reviews of work area gamma-ray exposure rates and advising the Site Manager(SM) on operational changes that will reduce exposures to ALARA levels.

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

17.4.2 Restricted and Controlled Areas

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

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

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

17.4.3 Radiation Dose Limits

17.4.3.1 Occupational Dose Limits for Adults

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

a. Annual limit will be the more limiting of:

1. The total effective dose equivalent (TEDE) equal to 5 rems; or
2. The sum of the deep-dose equivalent (DDE) and the committed dose equivalent (CDE) to any individual organ or tissue other than the lens of the eye being equal to 50 rems.

b. The annual limits to the lens of the eye, to the skin, and to the extremities, are:

1. An eye dose equivalent of 15 rems; and
2. A shallow dose equivalent of 50 rems to the skin or to any extremity.

c. Doses received in excess of the annual limits must be subtracted from the limits for planned special exposures that an individual may receive during the current year and during an individual's lifetime.

d. For soluble uranium, the intake by any individual is limited to 10 milligrams in any week in consideration of chemical toxicity.

17.4.3.2 Occupational Dose Limits to Minors

The annual occupational dose limits for minors are 10 percent of the annual dose limits specified for adults. The Site Radiation Safety Officer (SRSO) will review any work assignment given to minors to assure that exposures are maintained ALARA and within this guidance.

17.4.3.3 Dose Limit to an Embryo/Fetus

The dose equivalent to the embryo/fetus will be limited to 0.5 rem during the entire pregnancy in accordance with 10 CFR 20.1208. Envirocare's policy is to inform female workers of the regulations regarding protection of the embryo/fetus and to ask them to inform Envirocare in writing, upon discovery or suspicion of a pregnancy. The Corporate Radiation Safety Officer (CRSO) will review the work assignments and normally offer the woman the opportunity to take available positions in non-radiation areas for the duration of the pregnancy. If no positions are available, the CRSO will counsel the individual to assure an understanding by the individual of the additional risks of continued employment. If the woman continues to work in the radiation area, the SRSO will monitor the work assignments and activities to assure that the TEDE to the embryo/fetus is ALARA and limited to 0.5 rem.

17.4.3.4 Planned Special Exposures

Envirocare does not anticipate authorizing planned special exposures since the radiation levels and radioactive constituent concentrations in 11e.(2) byproduct material are low. In the event that circumstances warrant a planned special exposure, Envirocare will do so in full compliance with the guidance in 10 CFR 20.1206.

17.4.3.5 Summation of Occupational Internal and External Doses

Guidance for the summation of the internal and external dose equivalents are specified in 10 CFR 20.1202. Summation is not required if either the external or internal radiation exposures are not likely to exceed 10 percent of the limit. This includes occupational exposures to adults as well as minors and to the embryo/fetus.

It is unlikely that exposures to workers at the Envirocare facility will exceed 10 percent of the allowable limits for direct radiation as well as internal radiation. Data for the UMTRA Project disposal at Clive show that the average annual dose equivalent from direct radiation was 50 mrem, with a maximum individual dose equivalent of 250 mrem. Envirocare has been operating the LARW facility beginning in 1988. The maximum individual dose equivalent from 1988-1992 was 200 mrem. Similarly the lapel sample and work area monitoring results indicate that the airborne particulate concentrations are near background levels.

Should Envirocare find that summation of occupational internal and external doses is necessary, the following method will be employed:

a. Should the internal dose as determined by air monitoring results, bioassay, or other means - as well as the dose from external sources as determined by radiation dosimeters - likely exceed 10 percent of the allowable limits, the Committed Effective Dose Equivalent (CEDE) will be added to the Deep Dose Equivalent (DDE) and compared with the Total Effective Dose Equivalent (TEDE) limit of 5 rem for adults and 0.5 rem for minors and the fetus/embryo.

b. If the only intake of radionuclides is by inhalation, the procedure specified in 10 CFR 20.1202(b) may be applied. The TEDE limit will not be exceeded, according to this procedure, if the sum of the DDE divided by the TEDE limit and one of the following, does not exceed unity:

   1. The sum of the fractions of the inhalation ALI for each radionuclide, or
   2. The total number of derived air concentration-hours (DAC-hours) for all radionuclides divided by 2,000, or
   3. 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 of 50 rem.

c. If the intake by oral ingestion exceeds 10 per cent of the oral ALI, Envirocare will account for this intake and include it in demonstrating compliance with the limits.

d. If intake occurs via wounds or skin absorption, Envirocare will evaluate these intakes and include these in the calculation of the TEDE.

17.4.3.6 Determination of Prior Occupational Dose

If any employee is anticipated to receive an occupational dose in excess of 10 percent of the limits presented in this Section, Envirocare will determine the previous radiation exposure for use in limiting the annual dose equivalent to the allowable limits and for planning special exposures.

Determination of prior occupational exposures will be done by

1. Obtaining a written signed statement from the employee or his most immediate employer, that discloses the nature and the amount of any occupational dose that the individual may have received during the current year; and
2. Obtaining or attempting to obtain from the employee's most recent employer, a written signed statement in the form of an NRC Form 4 or an equivalent form, showing the life-time occupational exposure history. In case this cannot be done, the guidance in 10 CFR 20.2104 will be followed.

17.4.3.7 Radiation Dose Limits for Individual Members of the Public

Operations will be conducted such that the additional dose equivalent to individual members of the public will be limited in accordance with the limits of 10 CFR 20.1301, 10 CFR 61, and 10 CFR 40, Appendix A. The limits are:

a. The total effective dose equivalent to individual members of the public from the licensed operation will not exceed 25 mrem per year above natural background levels, radon and radon daughters excepted.
b. Radon and radon daughters will be limited to levels specified in Table 2 of 10 CFR [20.1001-20.2401], Appendix B.
c. The total effective dose equivalent limit to occupants in the controlled area (other than restricted areas) will not exceed 100 mrem per year above background levels.
d. The dose equivalent in any unrestricted area from external sources will not exceed 0.002 rem in any one hour.

Table 3.12, revised, Appendix A-1, shows the calculated concentrations of particulate radioactivity at the site boundaries. The projected concentrations are in the range of ambient background concentrations and are well below the concentration limits of Appendix B to 10 CFR 20.1001-20.2041. Airborne particulate monitoring will be performed to confirm those predictions.

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

17.4.4 Internal Radiation Dose Assessment

17.4.4.1 Calculation of Internal Radiation Exposure from Inhalation

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

DAC-hours = (C/DAC) x t

where:
C = airborne concentration of radionuclides in mCi/ml
DAC = Derived Air Concentration in mCi/ml
t = time of exposure in hours

The total exposure is equal to the sum of such calculations for all radionuclides present.

17.4.4.2 Calculation of Internal Dose from Inhalation

In order to assure compliance with the occupational dose limit, the committed dose equivalent (CDE) to any organ and the Committed Effective Dose Equivalent (CEDE) will be calculated for each radionuclide as follows:

Committed Dose Equivalent (mrem) = C x t x R x f_{CDE} / PF

where
C = concentration in mCi/ml
t = exposure time in hours
R = inhalation rate, 1.2 E+06 ml/h
f_{CDE} = exposure to dose conversion factor, in mrem/mCi, for the maximally exposed organ
PF = respirator protection factor as given in Appendix A to 10 CFR 20.1001-20.2401

The total committed dose equivalent for any organ is obtained by summing the contribution from each radionuclide of significance. Since the physical and chemical form of the radionuclides will normally not be characterized,

the exposure to dose conversion factor for the most restrictive lung clearance class (Day, Week, Year) for the maximally exposed organ will be used. Using Table 2.1 of ICRP Publication 30 or Table 13.24.2 in The Health Physics and Radiological Health Handbook (Revised Edition, Scinta, Inc), it is apparent that the dose to the endosteal lining of the bone from the thorium in 11e.(2) material is dominant for most lung clearance classes. For 11e.(2) material having high concentrations of insoluble uranium, it may, however, be possible that a combination of radionuclides could result in a larger dose to the lung. Therefore, data on which to calculate the organ doses is included below and unless the specific chemical form and lung class is known, the calculations will be made for both organs to assure that the 50 rem CDE limit has not been exceeded.

Choosing the listed lung clearance classes for maximizing the dose to the endosteum, the following CDE to the endosteum per unit intake will be used.

Radionuclide	Class	f_{CDE} for ENDOSTEUM(mrem/mCi)
U-nat	D	3.82E+4
U-234	D	4.03E+4
U-235	D	3.74E+4
U-238	D	3.62E+4
Th-230	W	7.99E+6
Th-232	W	4.11E+7
Ra-226	W	2.81E+4
Ra-228	W	2.41E+4
Pb-210	D	2.02E+5
Po-210	D	1.49E+3

Choosing the lung classes for maximizing the dose to the lungs for each radionuclide of interest, the following f_{CDE} will be applied.

Radionuclide	Class	f_{CDE} for lung (mrem/mCi)
U-nat	Y	1.00E+6
U-234	Y	1.11E+6
U-235	Y	1.04E+6
U-238	Y	1.00E+6
Th-230	Y	1.11E+6
Th-232	Y	3.48E+6
Ra-226	W	5.96E+4
Ra-228	W	2.67E+4
Pb-210	D	1.18E+3
Po-210	W	4.81E+4

The formula for the CEDE is similar to the above with the exception that "f" is the exposure to committed effective dose equivalent conversion factor, f_{CEDE} in mrem/mCi. It is chosen for the lung clearance class that maximizes the CEDE for each radionuclide. It is also based on NRC recommended Organ Dose Weighting Factors rather than the factors in ICRP Publication 26.
Again taking the values from the Health Physics and Radiological Health Handbook, the following data from Table 13.24.2 will result in maximizing the calculated CEDE:

Radionuclide	Class	f_{CEDE} IN (mrem/mCi)
U-nat	Y	1.25E+5
U-234	Y	1.32E+5
U-235	Y	1.23E+5
U-238	Y	1.18E+5
Th-230	Y	2.62E+5
Th-232	Y	1.15E+6
Ra-226	W	8.58E+3
Ra-228	W	4.77E+3
Pb-210	D	1.36E+4
Po-210	D	8.58E+3
Rn-220*		
Rn-222*		

*The internal dose from Radon-220 and Rn-222 for occupational workers will be calculated for occupational exposures using the relationship that either the ALI for radon or the WLM limits for radon daughters is equivalent to a TCEDE of 5 rem (see 10 CFR [20.1001-20.240-1], App B).

In order to determine which of the two limits (TEDE of 5 rem/year or sum of the deep-dose equivalent and the CDE to any organ of 50 rem) are the most restrictive for the particular mix of radionuclides, the TEDE and the CDE to the maximally exposed organs are calculated as described above. The DDE is added to the CDE and compared to the 50 rem organ dose limit; the TEDE is compared to the 5 rem annual limit. The calculations will be made for all employees according to the requirements in 10 CFR 20.1202.

The radionuclide mix will either be determined by estimating the volume-weighted radionuclide mix using waste characterization data or by a laboratory analysis of composites of work area or personnel monitor air filters.

The tables above normally use the maximum dose equivalent per unit intake. When uranium tailings are being handled, dose equivalent values for the

lung clearance class "D" will be used for uranium and "Y" for thorium since this is standard practice based on industry data. Other modifications to the parameter values will be made when the information is available.

17.4.4.3 Calculation of Internal Dose from Oral Ingestion

The ingestion of radionuclides at the Envirocare site is controlled primarily by restricting eating and drinking to monitored clean areas. In addition, the use of respiratory protection in the most highly contaminated areas minimizes the potential for contaminating the face and transfer of material from other parts of the body to the mouth.

While it is unlikely, the internal dose will be calculated and included in the employees total dose assessment should Envirocare be made aware of such occurrence. An assessment of the radionuclide intake will be made and the respective Committed Dose Equivalent per Unit Intake via Ingestion factor will be use to calculate the CDE and CEDE (see ICRP Tables or Table 13.24.21 in the Health Physics and Radiological Health Handbook).

17.4.4.4 Calculation of Internal Dose from Intake through Wounds or Skin Absorption

Employees at the Envirocare site are normally protected from intake through wounds and skin absorption by wearing protective clothing. Should an accident result in an open wound, the CRSO or Site RSO will inform the attending physician of the fact for his guidance in effecting removal or reduction of the amount of radioactive material remaining in the wound. The CRSO will perform an investigation and estimate the intake using data from wound monitoring or other available information.

The CDE to any organ will be estimated using methods similar to those used in NCRP Report 111, Developing Radiation Emergency Plans for Academic, Medical or Industrial Facilities. August, 1991. Table 4.2 provides values of maximum committed dose equivalent to any organ for adults per unit intake. These were derived by taking the ICRP Publication 30 values for ingestion and dividing by the gut transfer factor f_1. Envirocare will use a similar approach by estimating the radionuclide mixture and intake for each radionuclide, and calculating the CDE to each organ using appropriate f_1 values and CDE per unit intake for each radionuclide of significance via the ingestion pathway.

Calculated CDE's will be compared to the standards of 10 CFR 20.1201. Additional efforts at reducing dose will be based on total CDE and the potential for reducing the CDE through available means.

17.4.4.5 Bioassay

All permanent employees working at the site will be required to participate in a urine bioassay program to assist in evaluating internal deposition of radionuclides. A baseline urine sample will be collected upon employment and annually thereafter. Samples will be routinely analyzed for gross beta (minus K-40), Ra-226, isotopic thorium, and total uranium.

An increase above baseline levels equal to three standard deviations of the baseline values for the entrance bioassay will trigger an investigation of the work activities, an increased frequency of sampling, and a more detailed analysis to estimate the intake and resultant dose equivalent.

For those personnel working directly with the waste, a quarterly sampling program will be instituted. At this time it is anticipated that most waste will be similar in physical and chemical composition to uranium mill tailings. A urine bioassay action level of 1.5 mg/l has been derived for natural uranium (U-nat) at which time better controls on intake must be instituted. This action level was derived (see below) assuming chronic exposure to airborne tailings where the quarterly intake is equal to ten percent of the TEDE. A similar derivation for other radiological mixes may be required and a different action level used when large quantities of other 11e.(2) materials are being handled.

Based on experience at Envirocare's NORM disposal facility, it is unlikely that any employee's bioassay results will be above the action level. If any result does exceed the action level, the causes for such a level will be investigated and steps will be taken to reduce the employee's future exposure to inhaled or ingested radioactive materials.
A special bioassay sampling will be done for all personnel involved in an incident determined by the CRSO as having a potential for a significant intake of radionuclides. Twenty-four hour fecal and urine samples will be collected on a periodic basis until activities are below the minimum detectable levels or a determination is made that continued monitoring is not necessary. If the waste contained high Th-232 concentrations, lung or whole-body counting techniques may be employed to measure deposition in the body.

Excretion models will be used along with waste characterization data, bioassay data, and operational data to estimate the radionuclide intake and the resultant dose to the organs. Methods recommended in NCRP Report No. 87, "Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition" will be used. The guidance of 10 CFR 20.1201 will be followed in cases where significant organ doses or TEDE's are found.

Derivation of Action Level for Uranium Tailings

The worker exposure pathway for radionuclides under normal operations is via the inhalation pathway. Routine chronic exposure to radionuclides is limited by dust control measures and use of respiratory protection. However, to check the adequacy of these measures, *in vivo* or *in vitro* methods may be employed periodically, as determined by the CRSO, to assure that intakes are a small fraction of the regulatory limits.

No single method exists that will adequately detect intakes of potential 11e.(2) radionuclides at levels near the allowable limit of intake (ALI). Bioassay methods work well for the normally soluble uranium isotopes but fail to detect the insoluble thorium isotopes. Similarly, whole-body counting or lung counting methods may detect levels of Th-232 and Ra-226 (Radon daughters) at or near the ALI, depending upon the distribution in the body but fail to detect Th-230, Ra-228 or other alpha or beta emitting radionuclides. For acute intakes, analysis of the feces is normally more sensitive than other methods, while for chronic intakes it is not a viable method.

Section 17.1.1 presents a review of potential wastes for disposal at Clive. Most of those wastes are expected to contain significant weight percentages of uranium which may be used as an indicator to estimate other radionuclide intakes within the mixture.

This method is presently being used at the UMTRA sites and is described in Reif, 1992. Calculations similar to the approach in that reference will be used to develop an action level for Clive for the case where wastes similar to uranium mill tailings are being handled. Changes will be made to reflect the recent NRC regulatory requirement to limit the TEDE to 5 rem per year.

Ref Reif, R. H., Turner, J. B. and D. S. Carlson. "Uranium in Vitro Bioassay Action Level Used to Screen Workers for Chronic Inhalation Intakes of Uranium Mill Tailings", Health Physics Vol. 63, No. 4 (1992) p398.

Reif(1992) develops a radionuclide mix for mill tailings based upon actual data from the UMTRA sites. This radionuclide mix will also be assumed in this analysis and is presented in Table 17.4.1.

Presented in Table 17.4.1 are the TEDE factors per unit uptake for each of the radionuclides that contribute more than 1 percent of the TEDE.

Table 17.4.1 TEDE Per Unit Intake for Uranium Mill Tailings

Radionuclide	Lung Class	Relative Activity	TEDE (mrem/mCi)
U-nat	D	2	2.6 E+3
Th-230	Y	13	2.6 E+5
Ra-226	W	13	8.6 E+3
Pb-210	D	13	1.4 E+4
Po-210	D	13	9.4 E+3

Using the radionuclide mix in the above table, the U-nat intake equal to a TEDE of 500 mrem for the mixture was calculated to be 260 pCi. The next step is to estimate the quantity that will be transferred to the blood and eliminated via the urine.

ICRP Publication 30 uses a fractional transfer factor of inhaled activity to blood for long-lived Class D radionuclides as:

Fraction = $0.48 + 0.15\ f_1$, where f_1 is the fraction entering the blood via the GI tract. For Class D uranium, f_1 is equal to 0.05.

The concentration of U-nat in the urine at the end of a 90-day chronic exposure period, is approximately equal to the product of the daily intake rate and the intake retention fraction divided by the daily urine volume. Within the accuracy of the model, it will be assumed that all of the uranium in the blood is eliminated via the urine and thus the retention fraction is equal to 0.49.

If we assume a three-month chronic exposure at which the employee received an intake of tailings equal to 10 % of the allowable annual TEDE, the uranium concentration in a 24-hour voiding urine sample can be calculated as follows:

> (260 pCi/90 days)(0.49)/(1.4 liters/day) = 1.0 pCi/liter, where the 1.4 l/day is the daily urine produced by standard man, 90 days is the exposure time, and 0.49 is the intake retention fraction.

This concentration of U-nat in urine is equivalent to 1.5 mg/l, a level easily detectible using fluorimetry analysis.

17.4.5 Assessing Dose Equivalent from External Radiation Sources

All personnel entering the restricted area are required to wear radiation dosimeters at all times.

17.4.5.1 Permanent Employees

Permanent employees are issued a thermoluminescent dosimeter (TLD) badge provided by Envirocare. These badges are exchanged on a quarterly basis or read as soon as practical upon termination of employment. Badges are selected that measure the skin dose equivalent (shallow dose) as well as the deep dose equivalent for compliance with 10 CFR 20.1203 and 10 CFR 20.1502 and are worn in the proper place as instructed by the RSO. All badges, along with control badges, are maintained at the manned access control point when the employee is not at work.

Processing is done by a dosimetry processor holding accreditation from the National Voluntary Laboratory Accreditation Program of the National Institute of Standards and Technology appropriate for the radiation fields at the Envirocare site.

It is not anticipated that the measurement of the shallow dose equivalent will be difficult since the very soft beta radiations will be absorbed by the protective clothing of the employees as well as the relative large thickness of the air between the personnel and the waste. A periodic review of the appropriateness of the TLD program will be made by the CRSO with necessary measurements to document the findings. The use of thin window ion chambers or other methods will be used to measure the ratio of total dose rate to penetrating dose rate for each waste type at the worker's point of maximum exposure. This will be compared to the shallow and deep dose equivalent measured by the worker's personal dosimeter.

Should the CRSO determine that it is necessary to measure the shallow dose rather than use TLD devices, Envirocare will implement a procedure to calculate the shallow dose by applying a correction factor to the TLD reading(s). All exposures will be recorded when received from the dosimetry vendor to demonstrate compliance with the standards. In the event that an individual loses the personal TLD, the SRSO or his designee will investigate the potential exposure conditions and provide an estimate of the exposure.

All employees will notify their supervisor immediately upon discovery that a TLD has been lost. A new dosimeter will be issued immediately.

At this time, it is not anticipated that extremity monitoring will be necessary. However, the SRSO will monitor the work activity and if extremity monitoring is warranted, appropriate dosimeters will be obtained from the dosimetry vendor.

NRC Regulatory Guide 8.30 discusses the concern for measuring the shallow dose from yellow cake where the contact dose rate is approximately 150 mrad/hour and the dose at 30 cm is approximately 1 mrad/hour. While Envirocare understands this concern, we do not believe that the beta dose will be significant in the 11e.(2) wastes received at the site. Disposal of 11e.(2) material will normally be depleted in uranium isotopes and the disposal of separated uranium will be limited by the concentration limits in the waste acceptance criteria which is small compared to the approximately 600,000 pCi/g in yellowcake or other uranium compounds. During the waste handling operations at Envirocare, direct contact with the waste is normally not made and the combination of low activities, large distances, and protective clothing will limit the shallow dose equivalent to acceptable levels for the wastes containing uranium compounds.

Because of the low radionuclide activities in the waste, there is little potential for a significant penetrating or non-penetrating external radiation dose from airborne radioactive material. The deep dose equivalent component of this small dose, will be included in the employee's personal dosimeter reading.

17.4.5.2 Visitors and Temporary Employees

Individuals who are visiting the site on a limited basis will be issued a pocket dosimeter to record exposure. The dosimeter is read upon exiting the controlled area and recorded on the Access Log. In the case of individuals visiting as a group, one pocket dosimeter may be used providing they stay together.

17.4.6 Radiation Monitoring

17.4.6.1 Equipment, Instrumentation, and Facilities

Health Physics instrumentation selected for this program includes the portable and laboratory equipment described below.

a. Berthold Model 1043AS hand and foot monitor - 1 each. Selected as a sensitive personnel portal monitor capable of measuring alpha and

beta contamination levels simultaneously and independently and providing both a direct printer record of each survey and a computer record for each individual using coded identification badges.

b. Ludlum Model 19 Micro-R Survey Meters - 3 each. Selected as the basic survey meter for gamma exposure rates for area surveys and incoming shipments. Due to the low exposure rates encountered, a scintillation survey meter capable of performing accurate measurements in the range of background is required. The selected meters are rugged, dependable, easy to use, and feature a range of 0 to 5,000 mR/h over 5 ranges.

c. Berthold Model 122 contamination survey meter - 3 each. This meter measures alpha and beta surface contamination independently and provides a direct readout of area contamination levels. It operates over a wider range of temperature conditions than other survey meters and is well suited for field use in meeting the release standards presented in Section 17.4.7.1.

d. Ludlum Model 177 Ratemeter with Model 44-9 Pancake G-M Detector - 3 each. Selected as a portal frisker for personnel surveys due to the high sensitivity of the pancake detector and alarm-ratemeter capability of the ratemeter.

   The thin-window GM detector is sensitive to alpha, beta, and gamma radiation. The radiation types can be determined by selective use of shielding.

e. Ludlum Model 9 Ion Chamber Survey Meter - 2 each. Selected to provide a wide range of exposure rate measurements with little dependence on gamma energy. This instrument is rugged and reliable, and has a range of 0 - 5 R/h over 4 ranges.

f. Self-Reading Dosimeters (Victoreen 541R or equivalent or Bicron Model PDM-207 or equivalent). Selected to provide detection capability of approximately 1 mR over a scale of 0 - 200 mR. Used to record exposures to visitors and temporary employees while in the controlled area.

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

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

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

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

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

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

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

All instruments will be efficiency checked or source checked prior to use on a daily basis. Alpha and beta laboratory counters will be efficiency checked each day that they are in use. Portal monitors will be source checked at the beginning of each day using a source that is adequate to indicate an alarm. The response of hand-held radiation detection instruments will be compared to known sources prior to each use.

The respiratory protection equipment and protective clothing are located in the change room in the Administration building. Portable radiation instruments and laboratory instruments are located in the radiological laboratory in the Administration Building.

17.4.6.2 Area Radiation Surveys

Routine external gamma surveys using a gamma scintillation survey meter will be conducted and documented in areas involving disposal material in accordance with the type, frequency, and location(s) listed in Table 17.8. Addition area gamma surveys will be performed during daily operations as considered necessary by health physics personnel.

Routine wipe surveys for surface contamination will be conducted as listed in Table 17.8. The wipes will be analyzed for gross alpha contamination using a Ludlum Model 1000 Scaler or equal with a Model 43-10 alpha scintillation probe or equal. They will also be analyzed for gross beta contamination using a Ludlum Model 2200 scaler or equivalent and a Model 120 gas flow proportional counter or equivalent.

17.4.6.3 Airborne Particulate Radioactivity Monitoring

Work areas and boundary areas will be monitored for airborne radioactive particulates. The continuous airborne particulate samplers operated on site as part of the environmental monitoring program (see Section 7) will provide an overall average of the concentrations of airborne radioactivity. In addition to the fixed-location environmental stations, work-place samples will be collected to better assess potential exposure to employees.

On-site air particulate samples will be collected by means of F & J Specialty Products, CO. Model FJ-28B Low Volume Air Sampler, or equivalent, operating at 60 liters per minute (lpm) with a 2-inch diameter glass fiber filter. This sampler was selected on the basis of its demonstrated reliability, continuous flow control, and ability to collect sufficient sample during the weekly sample period to meet the sensitivity requirements set forth in Section 7.3.1. The sampling locations, shown in Figure 7.1, were selected to monitor airborne particulate radioactivity at site boundary locations as well as near on-site operational areas such as the rollover, disposal cell and haulways.

Work area samples will be collected with FJ-HV-1 high volume air samplers, or equivalent. The FJ-HV-1 sampler collects samples at 120 lpm and is used as a moveable sampler to collect airborne particulates at locations where a.c. power is available, or by means of a portable generator. For locations where a.c. power is not available, battery-powered portable samplers capable of collecting at least 20 lpm will be used.

Both samplers were selected to collect sufficient sample on a 2-inch glass fiber filter to permit detection levels comparable to Table 1 of 10 CFR {20.1001-2401], Appendix B, making estimation of potential exposures sufficiently sensitive for occupational exposures.

The a.c.-powered samplers will be used at locations such as the rollover, along haul ways, or near excavation and disposal activities to collect 8-hour, work-day samples. Samples will be collected daily at two locations during periods of high work activity and a minimum of twice each week during periods of low work activity. During the winter months when disposal work has been terminated, no measurements will be made. Sample collection data will include a short statement of weather conditions during collection so that results may be compared to prevailing conditions.
At the end of the sampling period, air particulate samples will be stored in envelopes and marked with the pertinent information. After a delay of seven days, air filters will be counted for gross alpha and beta levels. Gross alpha activity levels will be compared to the DAC for Th-232 of 5

E-13 mCi/ml; gross beta activity levels will be compared to the DAC for Pb-210 of 1E-10 mCi/ml.

After counting, filters will be stored in closed containers for future analysis. If warranted by calculations of probable exposure, the composite filters will be analyzed for Th-230, Th-232, Pb-210, Ra-228, and Ra-226, to provide precise data on radionuclide concentrations in the work environment and potential levels of internal exposure. Results of the isotopic analyses will be compared to limits provided in 10 CFR [20.1001-20.2401], Appendix B.

Gross alpha concentrations of 5 E-13 mCi/ml or gross beta concentrations of 1 E-10 mCi/ml on individual air filters are considered "action levels", and will trigger the following response by the Site Radiation Safety Officer:

1. The SRSO will evaluate site conditions to determine whether additional dust suppression methods are needed, whether posting for airborne radioactivity (20 CFR 10.1902) is required, and whether respiratory protection requirements are adequate.
2. The sample will be analyzed by gamma spectrometry and, if necessary, by radiochemical separation and laboratory analysis to determine the activities of the radionuclides present.
3. If it is confirmed that any employees exceeded the concentration limits of 10 CFR [20.1001-20.2401], Appendix B, Table 1, considering any respiratory protection devices, special urine/or fecal samples may be collected from the most significantly exposed employee to determine the extent of radionuclide uptake due to inhalation of dust. The situation will be investigated to determine the cause for such concentrations and the means of reducing such exposures in the future.

Air sampling results for airborne particulates and radon will be used to calculate internal doses to employees. Those employees in assignments most likely to receive exposure to higher concentrations of airborne particulates will be required to routinely wear respirators.

17.4.6.4 Personnel Contamination Monitoring

The use of protective clothing should minimize the potential for skin contamination. However, all personnel working in the restricted areas will be required to be monitored before leaving the access control area and must meet the release standards of Table 17.6. A hand and foot monitor sensitive to both alpha and beta contamination will be used for routine monitoring for contamination of personnel.

Workers involved in handling material will be required to wash exposed skin (hands and face) before they leave the site. In addition, showers are provided in the change area for use by all workers, as may be required by individual conditions, when exiting the site.

Workers are advised to consider any measurable contamination on their person as excessive and the goal is to keep such contamination below detectable levels.

Personnel will be expected to accomplish this by washing exposed areas of the skin with soap and water. If this does not reduce the levels below the standards of Table 17.6, the SRSO will be notified and other attempts will be made. Special radiation decontamination cleansers will be used to reduce skin contamination levels. Personnel with skin contamination will not be allowed to leave the site without approval of the CRSO.

All personal contaminated clothing or personal articles that cannot be decontaminated below the limits of Table 17.6 will be retained at the site and managed as radioactive waste.

All personnel contamination events will be documented.

The accident evaluation of Section 17.1.4 and the routine worker evaluation of Appendix A show that it is extremely unlikely that any employee could receive a lung burden of radioactivity which would require any action. If such an event did happen, the individual involved would be transported to a facility to receive a whole-body count to evaluate the potential dose. Subsequent actions, such as reassignment to a function not involving radiation exposure would be considered.

A worker might be injured in an accident that would result in the impaction of radioactive waste into a wound. Envirocare policy is to attempt to monitor injured employees before they are transported to medical care. In any case, the treating physician is informed that the injury involves possible radioactive contamination. Because the radionuclides involved are relatively insoluble, normal cleansing of the wound should remove most, if not all, of the radioactivity. A radiation survey will be used to estimate the remaining radioactivity and potential doses calculated as described in 17.4.4.4. The need for additional treatment would be based on the results of the monitoring.

Bioassay samples will be used, as necessary to help determine the body burden of any radioactivity which might have resulted from an unusual inhalation situation or wound.

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

17.4.4 Internal Radiation Dose Assessment

17.4.4.1 Calculation of Internal Radiation Exposure from Inhalation

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

DAC-hours = (C/DAC) x t

where: C = airborne concentration of radionuclides in mCi/ml
DAC = Derived Air Concentration in mCi/ml
t = time of exposure in hours

The total exposure is equal to the sum of such calculations for all radionuclides present.

17.4.4.2 Calculation of Internal Dose from Inhalation

In order to assure compliance with the occupational dose limit, the committed dose equivalent (CDE) to any organ and the Committed Effective Dose Equivalent (CEDE) will be calculated for each radionuclide as follows:

Committed Dose Equivalent (mrem) = C x t x R x f_{CDE} / PF

where C = concentration in mCi/ml
t = exposure time in hours
R = inhalation rate, 1.2 E+06 ml/h
f_{CDE} = exposure to dose conversion factor, in mrem/mCi, for the maximally exposed organ
PF = respirator protection factor as given in Appendix A to 10 CFR 20.1001-20.2401

The total committed dose equivalent for any organ is obtained by summing the contribution from each radionuclide of significance. Since the physical and chemical form of the radionuclides will normally not be characterized,

the exposure to dose conversion factor for the most restrictive lung clearance class (Day, Week, Year) for the maximally exposed organ will be used. Using Table 2.1 of ICRP Publication 30 or Table 13.24.2 in The Health Physics and Radiological Health Handbook (Revised Edition, Scinta, Inc), it is apparent that the dose to the endosteal lining of the bone from the thorium in 11e.(2) material is dominant for most lung clearance classes. For 11e.(2) material having high concentrations of insoluble uranium, it may, however, be possible that a combination of radionuclides could result in a larger dose to the lung. Therefore, data on which to calculate the organ doses is included below and unless the specific chemical form and lung class is known, the calculations will be made for both organs to assure that the 50 rem CDE limit has not been exceeded.

Choosing the listed lung clearance classes for maximizing the dose to the endosteum, the following CDE to the endosteum per unit intake will be used.

Radionuclide	Class	f_{CDE} for ENDOSTEUM (mrem/mCi)
U-nat	D	3.82E+4
U-234	D	4.03E+4
U-235	D	3.74E+4
U-238	D	3.62E+4
Th-230	W	7.99E+6
Th-232	W	4.11E+7
Ra-226	W	2.81E+4
Ra-228	W	2.41E+4
Pb-210	D	2.02E+5
Po-210	D	1.49E+3

Choosing the lung classes for maximizing the dose to the lungs for each radionuclide of interest, the following f_{CDE} will be applied.

Radionuclide	Class	f_{CDE} for lung (mrem/mCi)
U-nat	Y	1.00E+6
U-234	Y	1.11E+6
U-235	Y	1.04E+6
U-238	Y	1.00E+6
Th-230	Y	1.11E+6
Th-232	Y	3.48E+6
Ra-226	W	5.96E+4
Ra-228	W	2.67E+4
Pb-210	D	1.18E+3
Po-210	W	4.81E+4

The formula for the CEDE is similar to the above with the exception that "f" is the exposure to committed effective dose equivalent conversion factor, f_{CEDE} in mrem/mCi. It is chosen for the lung clearance class that maximizes the CEDE for each radionuclide. It is also based on NRC recommended Organ Dose Weighting Factors rather than the factors in ICRP Publication 26.

Again taking the values from the Health Physics and Radiological Health Handbook, the following data from Table 13.24.2 will result in maximizing the calculated CEDE:

Radionuclide	Class	f_{CEDE} IN (mrem/mCi)
U-nat	Y	1.25E+5
U-234	Y	1.32E+5
U-235	Y	1.23E+5
U-238	Y	1.18E+5
Th-230	Y	2.62E+5
Th-232	Y	1.15E+6
Ra-226	W	8.58E+3
Ra-228	W	4.77E+3
Pb-210	D	1.36E+4
Po-210	D	8.58E+3
Rn-220*		
Rn-222*		

*The internal dose from Radon-220 and Rn-222 for occupational workers will be calculated for occupational exposures using the relationship that either the ALI for radon or the WLM limits for radon daughters is equivalent to a TCEDE of 5 rem (see 10 CFR [20.1001-20.240-1], App B).

In order to determine which of the two limits (TEDE of 5 rem/year or sum of the deep-dose equivalent and the CDE to any organ of 50 rem) are the most restrictive for the particular mix of radionuclides, the TEDE and the CDE to the maximally exposed organs are calculated as described above. The DDE is added to the CDE and compared to the 50 rem organ dose limit; the TEDE is compared to the 5 rem annual limit. The calculations will be made for all employees according to the requirements in 10 CFR 20.1202.

The radionuclide mix will either be determined by estimating the volume-weighted radionuclide mix using waste characterization data or by a laboratory analysis of composites of work area or personnel monitor air filters.

The tables above normally use the maximum dose equivalent per unit intake. When uranium tailings are being handled, dose equivalent values for the

lung clearance class "D" will be used for uranium and "Y" for thorium since this is standard practice based on industry data. Other modifications to the parameter values will be made when the information is available.

17.4.4.3 Calculation of Internal Dose from Oral Ingestion

The ingestion of radionuclides at the Envirocare site is controlled primarily by restricting eating and drinking to monitored clean areas. In addition, the use of respiratory protection in the most highly contaminated areas minimizes the potential for contaminating the face and transfer of material from other parts of the body to the mouth.

While it is unlikely, the internal dose will be calculated and included in the employees total dose assessment should Envirocare be made aware of such occurrence. An assessment of the radionuclide intake will be made and the respective Committed Dose Equivalent per Unit Intake via Ingestion factor will be use to calculate the CDE and CEDE (see ICRP Tables or Table 13.24.21 in the Health Physics and Radiological Health Handbook).

17.4.4.4 Calculation of Internal Dose from Intake through Wounds or Skin Absorption

Employees at the Envirocare site are normally protected from intake through wounds and skin absorption by wearing protective clothing. Should an accident result in an open wound, the CRSO or Site RSO will inform the attending physician of the fact for his guidance in effecting removal or reduction of the amount of radioactive material remaining in the wound. The CRSO will perform an investigation and estimate the intake using data from wound monitoring or other available information.

The CDE to any organ will be estimated using methods similar to those used in NCRP Report 111, Developing Radiation Emergency Plans for Academic, Medical or Industrial Facilities. August, 1991. Table 4.2 provides values of maximum committed dose equivalent to any organ for adults per unit intake. These were derived by taking the ICRP Publication 30 values for ingestion and dividing by the gut transfer factor f_1. Envirocare will use a similar approach by estimating the radionuclide mixture and intake for each radionuclide, and calculating the CDE to each organ using appropriate f_1 values and CDE per unit intake for each radionuclide of significance via the ingestion pathway.

Calculated CDE's will be compared to the standards of 10 CFR 20.1201. Additional efforts at reducing dose will be based on total CDE and the potential for reducing the CDE through available means.

17.4.4.5 Bioassay

All permanent employees working at the site will be required to participate in a urine bioassay program to assist in evaluating internal deposition of radionuclides. A baseline urine sample will be collected upon employment and annually thereafter. Samples will be routinely analyzed for gross beta (minus K-40), Ra-226, isotopic thorium, and total uranium.

An increase above baseline levels equal to three standard deviations of the baseline values for the entrance bioassay will trigger an investigation of the work activities, an increased frequency of sampling, and a more detailed analysis to estimate the intake and resultant dose equivalent.

For those personnel working directly with the waste, a quarterly sampling program will be instituted. At this time it is anticipated that most waste will be similar in physical and chemical composition to uranium mill tailings. A urine bioassay action level of 1.5 mg/l has been derived for natural uranium (U-nat) at which time better controls on intake must be instituted. This action level was derived (see below) assuming chronic exposure to airborne tailings where the quarterly intake is equal to ten percent of the TEDE. A similar derivation for other radiological mixes may be required and a different action level used when large quantities of other 11e.(2) materials are being handled.

Based on experience at Envirocare's NORM disposal facility, it is unlikely that any employee's bioassay results will be above the action level. If any result does exceed the action level, the causes for such a level will be investigated and steps will be taken to reduce the employee's future exposure to inhaled or ingested radioactive materials.
A special bioassay sampling will be done for all personnel involved in an incident determined by the CRSO as having a potential for a significant intake of radionuclides. Twenty-four hour fecal and urine samples will be collected on a periodic basis until activities are below the minimum detectable levels or a determination is made that continued monitoring is not necessary. If the waste contained high Th-232 concentrations, lung or whole-body counting techniques may be employed to measure deposition in the body.

Excretion models will be used along with waste characterization data, bioassay data, and operational data to estimate the radionuclide intake and the resultant dose to the organs. Methods recommended in NCRP Report No. 87, "Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition" will be used. The guidance of 10 CFR 20.1201 will be followed in cases where significant organ doses or TEDE's are found.
Derivation of Action Level for Uranium Tailings

The worker exposure pathway for radionuclides under normal operations is via the inhalation pathway. Routine chronic exposure to radionuclides is limited by dust control measures and use of respiratory protection. However, to check the adequacy of these measures, *in vivo* or *in vitro* methods may be employed periodically, as determined by the CRSO, to assure that intakes are a small fraction of the regulatory limits.

No single method exists that will adequately detect intakes of potential 11e.(2) radionuclides at levels near the allowable limit of intake (ALI). Bioassay methods work well for the normally soluble uranium isotopes but fail to detect the insoluble thorium isotopes. Similarly, whole-body counting or lung counting methods may detect levels of Th-232 and Ra-226 (Radon daughters) at or near the ALI, depending upon the distribution in the body but fail to detect Th-230, Ra-228 or other alpha or beta emitting radionuclides. For acute intakes, analysis of the feces is normally more sensitive than other methods, while for chronic intakes it is not a viable method.

Section 17.1.1 presents a review of potential wastes for disposal at Clive. Most of those wastes are expected to contain significant weight percentages of uranium which may be used as an indicator to estimate other radionuclide intakes within the mixture.

This method is presently being used at the UMTRA sites and is described in Reif, 1992. Calculations similar to the approach in that reference will be used to develop an action level for Clive for the case where wastes similar to uranium mill tailings are being handled. Changes will be made to reflect the recent NRC regulatory requirement to limit the TEDE to 5 rem per year.

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Reif(1992) develops a radionuclide mix for mill tailings based upon actual data from the UMTRA sites. This radionuclide mix will also be assumed in this analysis and is presented in Table 17.4.1.

Presented in Table 17.4.1 are the TEDE factors per unit uptake for each of the radionuclides that contribute more than 1 percent of the TEDE.

Table 17.4.1 TEDE Per Unit Intake for Uranium Mill Tailings

Radionuclide	Lung Class	Relative Activity	TEDE (mrem/mCi)
U-nat	D	2	2.6 E+3
Th-230	Y	13	2.6 E+5
Ra-226	W	13	8.6 E+3
Pb-210	D	13	1.4 E+4
Po-210	D	13	9.4 E+3

Using the radionuclide mix in the above table, the U-nat intake equal to a TEDE of 500 mrem for the mixture was calculated to be 260 pCi. The next step is to estimate the quantity that will be transferred to the blood and eliminated via the urine.

ICRP Publication 30 uses a fractional transfer factor of inhaled activity to blood for long-lived Class D radionuclides as:

Fraction = $0.48 + 0.15\ f_1$, where f_1 is the fraction entering the blood via the GI tract. For Class D uranium, f_1 is equal to 0.05.

The concentration of U-nat in the urine at the end of a 90-day chronic exposure period, is approximately equal to the product of the daily intake rate and the intake retention fraction divided by the daily urine volume. Within the accuracy of the model, it will be assumed that all of the uranium in the blood is eliminated via the urine and thus the retention fraction is equal to 0.49.

If we assume a three-month chronic exposure at which the employee received an intake of tailings equal to 10 % of the allowable annual TEDE, the uranium concentration in a 24-hour voiding urine sample can be calculated as follows:

> (260 pCi/90 days)(0.49)/(1.4 liters/day) = 1.0 pCi/liter, where the 1.4 l/day is the daily urine produced by standard man, 90 days is the exposure time, and 0.49 is the intake retention fraction.

This concentration of U-nat in urine is equivalent to 1.5 mg/l, a level easily detectible using fluorimetry analysis.

17.4.5 Assessing Dose Equivalent from External Radiation Sources

All personnel entering the restricted area are required to wear radiation dosimeters at all times.

17.4.5.1 Permanent Employees

Permanent employees are issued a thermoluminescent dosimeter (TLD) badge provided by Envirocare. These badges are exchanged on a quarterly basis or read as soon as practical upon termination of employment. Badges are selected that measure the skin dose equivalent (shallow dose) as well as the deep dose equivalent for compliance with 10 CFR 20.1203 and 10 CFR 20.1502 and are worn in the proper place as instructed by the RSO. All badges, along with control badges, are maintained at the manned access control point when the employee is not at work.

Processing is done by a dosimetry processor holding accreditation from the National Voluntary Laboratory Accreditation Program of the National Institute of Standards and Technology appropriate for the radiation fields at the Envirocare site.

It is not anticipated that the measurement of the shallow dose equivalent will be difficult since the very soft beta radiations will be absorbed by the protective clothing of the employees as well as the relative large thickness of the air between the personnel and the waste. A periodic review of the appropriateness of the TLD program will be made by the CRSO with necessary measurements to document the findings. The use of thin window ion chambers or other methods will be used to measure the ratio of total dose rate to penetrating dose rate for each waste type at the worker's point of maximum exposure. This will be compared to the shallow and deep dose equivalent measured by the worker's personal dosimeter.

Should the CRSO determine that it is necessary to measure the shallow dose rather than use TLD devices, Envirocare will implement a procedure to calculate the shallow dose by applying a correction factor to the TLD reading(s). All exposures will be recorded when received from the dosimetry vendor to demonstrate compliance with the standards. In the event that an individual loses the personal TLD, the SRSO or his designee will investigate the potential exposure conditions and provide an estimate of the exposure.

All employees will notify their supervisor immediately upon discovery that a TLD has been lost. A new dosimeter will be issued immediately.

At this time, it is not anticipated that extremity monitoring will be necessary. However, the SRSO will monitor the work activity and if extremity monitoring is warranted, appropriate dosimeters will be obtained from the dosimetry vendor.

NRC Regulatory Guide 8.30 discusses the concern for measuring the shallow dose from yellow cake where the contact dose rate is approximately 150 mrad/hour and the dose at 30 cm is approximately 1 mrad/hour. While Envirocare understands this concern, we do not believe that the beta dose will be significant in the 11e.(2) wastes received at the site. Disposal of 11e.(2) material will normally be depleted in uranium isotopes and the disposal of separated uranium will be limited by the concentration limits in the waste acceptance criteria which is small compared to the approximately 600,000 pCi/g in yellowcake or other uranium compounds. During the waste handling operations at Envirocare, direct contact with the waste is normally not made and the combination of low activities, large distances, and protective clothing will limit the shallow dose equivalent to acceptable levels for the wastes containing uranium compounds.

Because of the low radionuclide activities in the waste, there is little potential for a significant penetrating or non-penetrating external radiation dose from airborne radioactive material. The deep dose equivalent component of this small dose, will be included in the employee's personal dosimeter reading.

17.4.5.2 Visitors and Temporary Employees

Individuals who are visiting the site on a limited basis will be issued a pocket dosimeter to record exposure. The dosimeter is read upon exiting the controlled area and recorded on the Access Log. In the case of individuals visiting as a group, one pocket dosimeter may be used providing they stay together.

17.4.6 Radiation Monitoring

17.4.6.1 Equipment, Instrumentation, and Facilities

Health Physics instrumentation selected for this program includes the portable and laboratory equipment described below.

a. Berthold Model 1043AS hand and foot monitor - 1 each. Selected as a sensitive personnel portal monitor capable of measuring alpha and

beta contamination levels simultaneously and independently and providing both a direct printer record of each survey and a computer record for each individual using coded identification badges.

b. Ludlum Model 19 Micro-R Survey Meters - 3 each. Selected as the basic survey meter for gamma exposure rates for area surveys and incoming shipments. Due to the low exposure rates encountered, a scintillation survey meter capable of performing accurate measurements in the range of background is required. The selected meters are rugged, dependable, easy to use, and feature a range of 0 to 5,000 mR/h over 5 ranges.

c. Berthold Model 122 contamination survey meter - 3 each. This meter measures alpha and beta surface contamination independently and provides a direct readout of area contamination levels. It operates over a wider range of temperature conditions than other survey meters and is well suited for field use in meeting the release standards presented in Section 17.4.7.1.

d. Ludlum Model 177 Ratemeter with Model 44-9 Pancake G-M Detector - 3 each. Selected as a portal frisker for personnel surveys due to the high sensitivity of the pancake detector and alarm-ratemeter capability of the ratemeter.

   The thin-window GM detector is sensitive to alpha, beta, and gamma radiation. The radiation types can be determined by selective use of shielding.

e. Ludlum Model 9 Ion Chamber Survey Meter - 2 each. Selected to provide a wide range of exposure rate measurements with little dependence on gamma energy. This instrument is rugged and reliable, and has a range of 0 - 5 R/h over 4 ranges.

f. Self-Reading Dosimeters (Victoreen 541R or equivalent or Bicron Model PDM-207 or equivalent). Selected to provide detection capability of approximately 1 mR over a scale of 0 - 200 mR. Used to record exposures to visitors and temporary employees while in the controlled area.

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

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

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

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

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

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

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

All instruments will be efficiency checked or source checked prior to use on a daily basis. Alpha and beta laboratory counters will be efficiency checked each day that they are in use. Portal monitors will be source checked at the beginning of each day using a source that is adequate to indicate an alarm. The response of hand-held radiation detection instruments will be compared to known sources prior to each use.

The respiratory protection equipment and protective clothing are located in the change room in the Administration building. Portable radiation instruments and laboratory instruments are located in the radiological laboratory in the Administration Building.

17.4.6.2 Area Radiation Surveys

Routine external gamma surveys using a gamma scintillation survey meter will be conducted and documented in areas involving disposal material in accordance with the type, frequency, and location(s) listed in Table 17.8. Addition area gamma surveys will be performed during daily operations as considered necessary by health physics personnel.

Routine wipe surveys for surface contamination will be conducted as listed in Table 17.8. The wipes will be analyzed for gross alpha contamination using a Ludlum Model 1000 Scaler or equal with a Model 43-10 alpha scintillation probe or equal. They will also be analyzed for gross beta contamination using a Ludlum Model 2200 scaler or equivalent and a Model 120 gas flow proportional counter or equivalent.

17.4.6.3 Airborne Particulate Radioactivity Monitoring

Work areas and boundary areas will be monitored for airborne radioactive particulates. The continuous airborne particulate samplers operated on site as part of the environmental monitoring program (see Section 7) will provide an overall average of the concentrations of airborne radioactivity. In addition to the fixed-location environmental stations, work-place samples will be collected to better assess potential exposure to employees.

On-site air particulate samples will be collected by means of F & J Specialty Products, CO. Model FJ-28B Low Volume Air Sampler, or equivalent, operating at 60 liters per minute (lpm) with a 2-inch diameter glass fiber filter. This sampler was selected on the basis of its demonstrated reliability, continuous flow control, and ability to collect sufficient sample during the weekly sample period to meet the sensitivity requirements set forth in Section 7.3.1. The sampling locations, shown in Figure 7.1, were selected to monitor airborne particulate radioactivity at site boundary locations as well as near on-site operational areas such as the rollover, disposal cell and haulways.

Work area samples will be collected with FJ-HV-1 high volume air samplers, or equivalent. The FJ-HV-1 sampler collects samples at 120 lpm and is used as a moveable sampler to collect airborne particulates at locations where a.c. power is available, or by means of a portable generator. For locations where a.c. power is not available, battery-powered portable samplers capable of collecting at least 20 lpm will be used.

Both samplers were selected to collect sufficient sample on a 2-inch glass fiber filter to permit detection levels comparable to Table 1 of 10 CFR {20.1001-2401], Appendix B, making estimation of potential exposures sufficiently sensitive for occupational exposures.

The a.c.-powered samplers will be used at locations such as the rollover, along haul ways, or near excavation and disposal activities to collect 8-hour, work-day samples. Samples will be collected daily at two locations during periods of high work activity and a minimum of twice each week during periods of low work activity. During the winter months when disposal work has been terminated, no measurements will be made. Sample collection data will include a short statement of weather conditions during collection so that results may be compared to prevailing conditions.
At the end of the sampling period, air particulate samples will be stored in envelopes and marked with the pertinent information. After a delay of seven days, air filters will be counted for gross alpha and beta levels. Gross alpha activity levels will be compared to the DAC for Th-232 of 5

E-13 mCi/ml; gross beta activity levels will be compared to the DAC for Pb-210 of 1E-10 mCi/ml.

After counting, filters will be stored in closed containers for future analysis. If warranted by calculations of probable exposure, the composite filters will be analyzed for Th-230, Th-232, Pb-210, Ra-228, and Ra-226, to provide precise data on radionuclide concentrations in the work environment and potential levels of internal exposure. Results of the isotopic analyses will be compared to limits provided in 10 CFR [20.1001-20.2401], Appendix B.

Gross alpha concentrations of 5 E-13 mCi/ml or gross beta concentrations of 1 E-10 mCi/ml on individual air filters are considered "action levels", and will trigger the following response by the Site Radiation Safety Officer:

1. The SRSO will evaluate site conditions to determine whether additional dust suppression methods are needed, whether posting for airborne radioactivity (20 CFR 10.1902) is required, and whether respiratory protection requirements are adequate.
2. The sample will be analyzed by gamma spectrometry and, if necessary, by radiochemical separation and laboratory analysis to determine the activities of the radionuclides present.
3. If it is confirmed that any employees exceeded the concentration limits of 10 CFR [20.1001-20.2401], Appendix B, Table 1, considering any respiratory protection devices, special urine/or fecal samples may be collected from the most significantly exposed employee to determine the extent of radionuclide uptake due to inhalation of dust. The situation will be investigated to determine the cause for such concentrations and the means of reducing such exposures in the future.

Air sampling results for airborne particulates and radon will be used to calculate internal doses to employees. Those employees in assignments most likely to receive exposure to higher concentrations of airborne particulates will be required to routinely wear respirators.

17.4.6.4 Personnel Contamination Monitoring

The use of protective clothing should minimize the potential for skin contamination. However, all personnel working in the restricted areas will be required to be monitored before leaving the access control area and must meet the release standards of Table 17.6. A hand and foot monitor sensitive to both alpha and beta contamination will be used for routine monitoring for contamination of personnel.

Workers involved in handling material will be required to wash exposed skin (hands and face) before they leave the site. In addition, showers are provided in the change area for use by all workers, as may be required by individual conditions, when exiting the site.

Workers are advised to consider any measurable contamination on their person as excessive and the goal is to keep such contamination below detectable levels.

Personnel will be expected to accomplish this by washing exposed areas of the skin with soap and water. If this does not reduce the levels below the standards of Table 17.6, the SRSO will be notified and other attempts will be made. Special radiation decontamination cleansers will be used to reduce skin contamination levels. Personnel with skin contamination will not be allowed to leave the site without approval of the CRSO.

All personal contaminated clothing or personal articles that cannot be decontaminated below the limits of Table 17.6 will be retained at the site and managed as radioactive waste.

All personnel contamination events will be documented.

The accident evaluation of Section 17.1.4 and the routine worker evaluation of Appendix A show that it is extremely unlikely that any employee could receive a lung burden of radioactivity which would require any action. If such an event did happen, the individual involved would be transported to a facility to receive a whole-body count to evaluate the potential dose. Subsequent actions, such as reassignment to a function not involving radiation exposure would be considered.

A worker might be injured in an accident that would result in the impaction of radioactive waste into a wound. Envirocare policy is to attempt to monitor injured employees before they are transported to medical care. In any case, the treating physician is informed that the injury involves possible radioactive contamination. Because the radionuclides involved are relatively insoluble, normal cleansing of the wound should remove most, if not all, of the radioactivity. A radiation survey will be used to estimate the remaining radioactivity and potential doses calculated as described in 17.4.4.4. The need for additional treatment would be based on the results of the monitoring.

Bioassay samples will be used, as necessary to help determine the body burden of any radioactivity which might have resulted from an unusual inhalation situation or wound.

Any employees who are believed to have received a TEDE of greater than 200 mrem from any source in one quarter will be notified and will assist in determining the source of the exposure and in finding a way to reduce future exposures.

17.4.6.5 Occupational Radon and Radon Daughter Monitoring

The handling of large quantities of Ra-226 and Th-232 bearing materials is expected to release Rn-222 (radon) and Rn-220 (thoron). The concentrations will vary depending upon the type of waste handled.

The occupational limit for radon daughter exposure is four (4) WLM while the limit for thoron daughter exposure is 12 WLM.
The occupational exposure limit for radon without daughters present is 4,000 pCi/l while for radon with all daughters present (100 % equilibrium) is 30 pCi/l. The exposure limit for thoron without daughters is 7,000 pCi/l and 9 pCi/l with daughters in equilibrium.

All work areas, including the administration building, will be monitored for radon and thoron using pairs of E-Perm ion chambers. One chamber responds to radon and thoron, the other responds primarily to radon. The readings along with the difference in the readings are used to calculate the radon and thoron concentrations. The minimum detectable concentration varies with the mixture of radon and thoron. If only radon is present, the MDC is approximately 500 pCi/liter-hours, or 0.75 pCi/l-month, where a month is considered continuous exposure for 4 weeks. If only thoron is present, the MDC is approximately 3.6 pCi/l-month. Detectors will be placed in the work areas and read weekly. While the measured average concentrations will be for 24 hours/day rather than the average for the work day, the results should be conservative in that the meteorology of the site is expected to enhance the levels at night.

Due to the long exposure times for the E-Perms, other measurements of the work area environment will be made to assess the workers exposure to radon and thoron and their daughter products. The E-Perm results of the radon and thoron measurements will be supplemented by grab samples for radon and thoron concentration and grab samples for radon and thoron WL determinations. If exposures are likely to exceed 10 percent of the allowable limits over a 40 hour exposure period, the grab sample results will be used to estimate the radon daughter equilibrium and the E-Perm radon concentration

results will be used to calculate a monthly average WL for radon and thoron. The radon and thoron WL results will then be used in determining the internal dose equivalents for the workers.

The occupational limit for radon daughter exposure is four (4) working months (WLM) per year, which is equivalent to a DAC of 30 pCi/l of Rn-222 in equilibrium with its daughters.

Instant WL Monitors or grab sample techniques will be used to monitor the work area on a weekly basis during periods of calm winds. For work areas routinely falling below 10 percent of the WL limits for radon and thoron daughters(0.03 WL and 0.1 WL for radon and thoron, respectively), the exposure will not be considered in the dosimetry program, provided there are no minors or declared pregnant women in the area (see 10 CFR 20.1205 (g)).

If grab samples are taken, the Ogden method, [Ogden, T.L. (1974). "*A method for measuring the working level values of mixed radon and thoron daughters in coal mine air.*" Ann Occ. Hyg. **17**, 23.] [Ogden, T.L. (1977). *"Radon and thoron daughter working levels from ordinary industrial hyqiene samples"* Ann. Occ. Hyg. **20**, 49.] will be used to measure radon and thoron daughter - WL concentrations with sample collection volumes and counting times sufficient to provide a lower limit of detection (sensitivity) of better than 0.03 WL (See NRC Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills" and the references cited therein). Instant WL meters or continuous WL monitors will be used only if the equivalent sensitivity can be achieved.

17.4.6.6 Environmental Monitoring Program

The environmental monitoring program is presented in Section 7.

17.4.7 Personnel Protection and Contamination Control

17.4.7.1 Access Control

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

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

1. Permanent employees will be issued a thermoluminescent dosimeter (TLD) badge provided by Envirocare. These dosimeters will be exchanged and

returned to the vendor on a quarterly basis. Permanent employees will pick up and turn in their dosimeters at the beginning and end of their working day at the manned access control point.

2. Individuals who are visiting the site on a limited basis will be issued a pocket dosimeter to record exposure. Visitors will pick up and turn in their pocket dosimeters at the manned access control point when they enter and exit the site. The dosimeters will be read as the individual leaves the site and recorded in the Access Log.
3. A group of visitors may all use the exposure from either one TLD or one pocket dosimeter in a situation where the entire group is to stay in the same vicinity while in the restricted area.

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

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

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

The quarterly dosimeter results from the quarterly exchange of dosimeters will be promptly recorded by the Site Radiation Safety Officer (SRSO), or his designee. The data will then be reviewed by the SRSO. Higher than expected personnel exposures will be further investigated by the Corporate Radiation Safety Officer (CRSO) and/or a contractor consultant.

All exiting employees must be surveyed for contamination using an alpha sensitive instrument. Records are maintained of the number of employees found to be contaminated and the level of contamination.

Personnel or materials leaving the restricted area will be required to meet the conditions of the following table (see Section 16.3 for equipment/vehicle decontamination procedures):

Table 17.5 SURFACE CONTAMINATION LEVELS ON EQUIPMENT, CLOTHING AND PERSONNEL TO BE RELEASED WITHOUT RESTRICTIONS FROM RESTRICTED AREA

Column I	Column II	Column III	
Nuclide[a]	**Average[b,d,f]**	**Maximum[b,d,f]**	**Removable[b,e,f]**
U-nat,U-235,U-238, and associated decay	5,000 dpm alpha/100cm^2	15,000 dpm alpha/100cm^2	1,000 dpm alpha/100cm^2 products
Transuranics, Ra-226, Ra-228,Th-230,Th-228, Pa-231,Ac-227,I-125, I-129	100 dpm/ 100 cm^2	300 dpm/ 100 cm^2	20 dpm/ 100 cm^2
Th-nat,Th-232,Sr-90 Ra-223,Ra-224,U-232 I-126,I-131, I-133	1,000 dpm/ 100 cm^2	3,000 dpm/ 100 cm^2	200 dpm/ 100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except SR-90 and others noted above	5,000 dpm beta-gamma/100 cm^2	15,000 dpm beta-gamma/100 cm^2	1,000 dpm beta-gamma/100 cm^2

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

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

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

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

e. The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping the area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of know efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

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

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

There will be no high or very high radiation areas on site due to the concentration limitations in the waste acceptance criteria. As shown in Section 17.1.4, even with

wastes as high as 15,000 pCi/g of each radionuclide the external gamma exposure rate would not exceed 50 mR/h. Therefore, no special access control procedures as required in 10 CFR [20.1601-20.1602] will be developed.

17.4.7.2 Protective Clothing and Change Facilities

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

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

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

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

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

17.4.7.3 Respiratory Protection Program

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

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

recordkeeping. The CRSO is responsible for administering the respiratory protection program.

17.4.7.4 Dust Control Measures

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

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

   a) Date of application
   b) Number of treatments
   c) Rainfall received
   d) Time of day treatments were made

17.4.7.5 Envirocare Site Regulations

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

17.4.8 Health and Safety Training

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

persons to enter the restricted area is related to the activities for which the person will enter the restricted area.

There are three categories of restricted-area functions:

(1) Permanent Employee
(2) Temporary Worker
(3) Visitor

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

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

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

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

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

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

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

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

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

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

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

The training is meant to educate the employees in the fundamentals of handling radioactive materials, to provide information on the ways and means of minimizing exposure, and to inform employees of practices and programs aimed at preventing possible spread of contamination.

The semi-annual refresher sessions for permanent employees will be provided to keep the employees aware of the nature of the material with which they have daily contact. The semi-annual refresher course will be a one-hour review of the topics discussed in the 3-hour training.

The Restricted Area Entrance Training will be given on site by the CRSO or SRSO, or any Envirocare Health Physics Specialist II. During this training, procedures and precautions will be explained and the trainees will be required to read and sign either the release form or a training roster form. The training records will be maintained by the SRSO.

In addition to the above training all Envirocare site employees will be required to attend at least 20 hours of training annually taught by qualified personnel. This training will be tailored to the specific employees needs and duties and will cover such topics as general

occupational safety, radiological safety, and training on any specific items such as new procedures or safety deficiencies.

17.4.9 Staffing and Personnel

17.4.9.1 Responsibilities

The Corporate Radiation Safety Officer (CRSO) is responsible for assuring that the environmental health and safety requirements at the site are being met and, in particular, the operations at the site are in compliance with Nuclear Regulatory Commission License Requirements. All health and safety related procedural changes are approved by the CRSO.

The Site Radiation Safety Officer (SRSO) has the day-to-day radiation safety responsibilities and reports to the CRSO while working very closely with the Site Manager. Assisting the SRSO are ~~Radiation Monitors (Health Physics Specialist I),~~ Access Control Technicians, Health Physics Specialists ~~II~~, and an Environmental Coordinator . The Environmental Coordinator is responsible for conducting the routine environmental monitoring program and performing certain laboratory analyses.

17.4.9.2 Certification for ~~Health Physics Specialist I~~ Access Control Technician and Health Physic Specialist ~~II~~

All personnel must be certified before they can be classified as either an ~~Health Physics Specialist I~~ Access Control Technician or a Health Physics Specialist II. This certification will include training and testing beyond that given in the restricted-area training program. Specific training and experience requirements for the positions, entrance training, on-the-job training, and examinations are listed in the Procedures Manual, Appendix B. The following is a summary of requirements for certification in those areas:

~~Health Physics Specialist I~~ Access Control Technician

1. 20 classroom hours of training in areas of chemistry, physics, radiation safety, construction safety, operation of equipment and site operations.
2. Pass a written exam designed specifically for ~~Health Physics Specialists I~~ access Control Technician.
3. Pass, to the satisfaction of the Site Radiation Safety Officer, a practical test designed to assure that candidate possesses knowledge for all equipment is being handled properly and all duties can be performed effectively.

Health Physics Specialist ~~II~~

1. 40 classroom hours of training in areas of chemistry, physics, radiation safety, construction safety, operation of equipment and site operations.
2. Pass a written exam designed specifically for Health Physics Specialist ~~II~~ .

3. Pass a laboratory test designed to assure that all equipment is being handled properly and all duties can be performed effectively.

In addition to the certification, each ~~Health Physics Specialist I~~Access Control Technician and Health Physics Specialist ~~II~~ must maintain certification by completing the annual training described in Section 17.4.6.3.

SECTION 18. ORGANIZATION

18.1 ORGANIZATIONAL STRUCTURE

18.1.1 Design, Construction and Pre-Operational Responsibilities

The operations and design of the Clive facility is described in detail in Sections 4 and 16. The waste material is placed in an earthen embankment, compacted in place, and covered with barriers to reduce radon emanation below Commission guidelines and to protect the embankment from the effects of weather erosion.

During the development and preparation of this application, Envirocare has utilized the services of the following consultants/contractors:

1. Donald W. Hendricks, CHP, President
   DON HENDRICKS AND ASSOCIATES, INC.
   609 No. Crestline Drive
   Las Vegas, Nevada 89107
   702/878-4420

2. Jeff Throckmorton, CIH, President
   HEALTH & SAFETY SERVICES, INC.
   10508 Aberdeen Lane
   Highland, Utah 84003
   801/756-0063

3. Gary M. Sandquist, Ph.D.
   1738 Ramona Avenue
   Salt Lake City, Utah 84108-3110
   801/486-8521

4. Craig B. Forster, Ph.D.
   3479 East Quad Road
   Salt Lake City, Utah
   801/581-3864

5. Stanley L. Plaisier, P.E.
   BINGHAM ENVIRONMENTAL, INC.
   5160 West Wiley Post Way
   Salt Lake City, Utah 84116
   801/532-2230

6. T. Leslie Youd, Ph.D.
   1132 East 1010 North
   Orem, Utah 84057
   804/378-6327

7. Blair McDonald, P.E.
   343 South 1000 East
   Salt Lake City, Utah 84102

Envirocare of Utah, Inc., with the assistance of these consultants, developed the personnel monitoring systems, data/record keeping systems, disposal material analysis and handling procedures, environmental monitoring systems, employee training, and general health and safety procedures and other technical supporting information for the 11e.(2) disposal project.

18.1.2 Operational Phase

The operational phase is also the construction phase of this proposed disposal project, in that the disposal project is discussed in Section 4 and 16.

A conceptual organizational chart is included as Figure 18.1, showing by responsibility the major divisions of Envirocare:

1. The peripheral activities of Scheduling, Accounting, and Marketing are represented on the organizational chart but do not need to be further described in this application.

2. President. The President oversees and provides direction and leadership for the operation. At a minimum, the president will:

   a. Promulgate company policies that identify his commitment to safety, the importance of compliance with requirements, the employees responsibilities to identify safety concerns to management, the need for adherence to company procedures, etc.
   b. Visit the site and observe the operations at least quarterly.
   c. Receive for his review summary audit reports, follow-up reports, close-out reports, NRC inspection reports and State inspection reports to ensure operations are conducted in accordance with Envirocare's high standard for quality and safety.

3. Corporate Radiation Safety Officer (CRSO) - Responsible to the Sr. Vice President of Compliance and Development and works very closely with the Director of Operations and Site Radiation Safety Officer (SRSO). The CRSO is responsible for implementation of

and compliance with all protocols and procedures of the radioactive materials license, including, health and safety monitoring, environmental monitoring, training, and personnel monitoring. The CRSO ensures that adequate instrumentation and equipment is used and that adequate measurements are made to ensure that all applicable standards for personnel exposures to radiation and radioactive materials are satisfied including:

- Shipping and Receiving of Radioactive Materials
- Airborne radioactivity
- Surface contamination
- Internal and external exposures
- Effluents
- Environmental monitoring
-

The CRSO shall also be responsible for the annual report which summarizes all of the previously mentioned information. The annual report will be provided to the President, the Sr. Vice President of Compliance and Development, and the Sr. Vice President of Operations and Business Development for review and appropriate actions.

The CRSO has authority to terminate any activities on the site that are deemed to be unsafe. The CRSO may also suspended activities until hazard-abatement measures have been performed. The CRSO is responsible for health physics and radiation protection, training, and safety review.

It is anticipated that the CRSO will work 20 hours per week on issues related to the 11e.(2) project. The remainder of his time will be used to work on issues related to the Low Activity Radioactive Waste (LARW) project currently operating at the Clive site.

4. Site Radiation Safety Officer (SRSO) - The SRSO is responsible to the CRSO and works very closely with the Site Facility Manager. The SRSO or designee is responsible for on-site radiation safety and implementation of and compliance with all protocols and procedures of the radioactive materials license, including health and safety monitoring, environmental monitoring, training, and personnel monitoring. The SRSO determines whether adequate instrumentation and equipment are being used and whether adequate measurements are made to ensure that all applicable standards for personnel exposures to radiation and radioactive materials are satisfied. The SRSO is also responsible for oversight of gamma spectral analysis,

the environmental program, and instrument program. The SRSO provides technical direction for radiological laboratory functions.

The SRSO has authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. This determination may be made unilaterally or upon receiving reports of suspect conditions from other site supervisors, contractors, visitors or employees.

It is anticipated that the SRSO will work 20 hours per week on issues related to the 11e.(2) project.

5. Assistant Radiation Safety Officers (ARSO). Assistant Radiation Safety Officers are designated to each area of operation (i.e., Mixed Waste Treatment, Mixed Waste Disposal,LARW/11e.(2).The ARSO's are responsible for managing the health physics team, performing daily site inspections, and observing field operations. The ARSO's can serve as acting SRSO and report to the SRSO.

   The ARSO's have authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. This determination may be made unilaterally or upon receiving reports of suspect conditions from other site supervisors, contractors, visitors, or employees.

6. The Environmental Coordinator is responsible to the SRSO. The Environmental Coordinator has authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. The Environmental Coordinator is charged with carrying out the environmental monitoring activities on site including:

   a. Implement applicable radiation control regulations and all provisions of radioactive material license.
   b. Data base management/record keeping to document all environmental monitoring activities at the site.
   c. Analysis of disposed material to document receipt and disposition.
   d. Analysis of disposal material to document receipt and disposition
   e. Other duties as assigned

7. Health Physics Specialists are responsible to the appropriate ARSO for the Area assigned (i.e., Mixed Waste Treatment, Mixed Waste Disposal, or LARW/11e.(2) and are trained by and have their work reviewed by the SRSO. Health Physics Specialists have direct access

access to the Facility Manager and SRSO on matters dealing with radiological safety. Health Physics Specialists will work on both the 11e(2), Mixed Waste, and LARW operations. Health Physics Specialists have the authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. They are charged with carrying out the health physics activities on site including:

a. Implement applicable radiation control regulations and all provisions of radioactive material license.
b. Personnel monitoring of Envirocare and contractor employees.
c. Assist in conducting training for new employees or refresher training for incumbent employees.
d. Supervision of truck/equipment decontamination facility.
e. Data base management/record keeping to document all disposal and health physics activities on site.
f. Perform reviews of previous radiation dose records with individual site workers.
g. Maintain continuous survaillance of site operating conditions and act to prevent actions which might result in the release or spread of radioactivity.
h. Other duties as assigned.

8. Access Control Technicians are responsible to the ARSO of the Mixed Waste Treatment area or the ARSO of the LARW/11e.(2) Area. They are charged with carrying out minimal health physics activities on site:

a. Implement applicable radiation control regulations and all provisions of radioactive material license.
b. Access Control monitoring of Envirocare and contractor employees.
c. Manning of Access Control portal.
d. Perform and document weekly surveys of radiation dose rates and surface contamination in assigned areas.
e. Other duties as assigned.

9. Sr. Vice President of Operations and Business Development.- The Sr. Vice President of Operations and Business Development reports to the President of Envirocare. The Sr. Vice President of Operations and Business Development is responsible for the overall management of direct operations and support functions for the disposal facility. The Sr. Vice President of Operations and Business Development works closely with other corporate personnel to ensure

that all operations are conducted in a planned and safe manner in accordance with all regulatory requirements.

The Sr. Vice President of Operations and Business Development shall establish and promulgate departmental employee policy when needed. The Sr. Vice President of Operations and Business Development shall also be responsible for investigating innovative methods of improving operations and/or efficiency.

10. Sr. Vice President of Compliance and Development.- The Sr. Vice President of Compliance and Development reports to the President of Envirocare. The Sr. Vice President of Compliance and Development oversees and directs compliance, licensing, and permitting activities at Envirocare; including such areas as quality assurance, radiation safety, environmental monitoring, ground water monitoring, safety, training, and regulatory affairs.

    The Sr. Vice President of Compliance and Development shall oversee and facilitate permit and license renewals, modifications, and amendments. This position will set compliance objectives jointly with the Operations Department personnel. Direction and support will be provided for policy development and site training to assist in ensuring compliance.

11. Director of Operations - The Director of Operations must be an experienced Civil Engineer, or other relevant engineering degree. The Director of Operations reports to the Sr. Vice President of Operations and Business Developmentand is charged with the responsibilities of the operations of the waste disposal site in an efficient and safe manner in accordance with design specifications and all applicable regulations.

    The Director of Operations is responsible for site operations including laboratory management, cell construction, waste management and disposal. The Director of Operations is directly responsible for negotiating contracts with subcontractors.

12. The Corporate Engineering Manager – The Corporate Engineering Manager perofrms certification of engineering design drawings, project plans, construction reports, and As-Built Drawings. The Corporate Engineering Manager is responsible for the management of technical and engineering support, including site structural engineering, soil mechanics, materials, and hydraulic engineering. The Corporate Engineering Manager provides or procures services from internal resources or technical contractors as necessary; provides tehcnical and engineering support for the operation

including site layout and design reviews; and approves with QA oversight, those designs and specifications.

13. The Site Facility Manager– The Site Facility Manager is responsible for the day-to-day operation of the Clive facility. The Site Facility Manager is to work closely with the SRSO to assure that all aspects of site operation are conducted according to the applicable regulations. The Site Facility Manager has limited specific responsibilities so that his efforts can be used in ensuring the effectiveness of the overall operational activities at the site. The Site Facility Manager is also responsible for the management of the site maintenance support and fire protection.

14. Production Engineer - The Production Engineer is responsible to the Corporate Engineering Manager and is responsible for overseeing the production, scheduling, and coordination aspects of facility construction with the exception of QA (which is the responsibility of the QAM). During construction, the Production Engineer will regularly inspect the construction site. The Production Engineer will coordinate the selection of the construction contractor(s) and administration of the construction contract, including any changes. The Production Engineer will review proposed design, engineering, or construction changes and submit these changes to the Corporate Engineering Manager for approval.

15. Site Engineer – The Site Engineer is responsible for construction quality control, overseeing the production, scheduling and coordination aspects of facility construction, with the exception of QA (which is the responsibility of the QAM). During construction, the Site Engineer will regularly inspect the construction site. The Site Engineer will coordinate the selection of the construction contractor(s) and administration of the construction contract, including any changes. The Site Engineer will review proposed design, engineering, or construction changes and submit these changes to the Corporate Engineering Manager for approval.

16. Construction Contractor - responsible to Site Facility Manager to perform construction, earth moving, and disposal activities in accordance with approved procedures and specifications. The Construction Contractor is also charged with maintaining compliance with all provisions of UOSHA and making records available for review by the Industrial Hygiene Consultant.

17. The Compliance and Permitting Manager – The Compliance and Permitting Manager is responsible for Initiating, producing, and

obtaining appropriate licenses and permits. The Compliance and Permitting Manager oversees the adminstration of the Air Quality Program and the preparation of all reports submitted in accordance with Envirocare's licenses and permits. The Compliance and Permitting Manager has the authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until abatement measures have been performed.

18. The Corporate Quality Assurance Manager ("CQAM") is responsible for ensuring that the quality assurance requirements outlined in the Quality Assurance Program Document (QAPD) are implemented. The reporting relationships shown in Figure 18.1 allow the CQAM sufficient authority and autonomy to implement and direct the QAPD; to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions independent of undue influences, and responsibilities, such as costs and schedules. As such, the CQAM reports directly to the Sr. Vice President of Compliance and Development in implementing the QAPD.

19. Outside Contractual Assistance.

    As indicated in Section 18.1.1, Envirocare has access to qualified consultants to assist in the development and implementation of radiological health and safety plans, environmental monitoring programs, industrial hygiene and safety programs. These consultants will be utilized extensively to provide reviews of safety, employee training, evaluation of fire protection systems, and quality assurance reviews in addition to continuous operations support. These contractors are responsible to the President of Envirocare.

All Envirocare management personnel and personnel with safety responsibilities will have free access to each other to resolve immediate safety, operational or other issues.

In order to more fully outline the responsibilities assigned, the following chart is provided with the applicable assignments

RESPONSIBILITY	POSITION
Structural, soil mechanics, materials, hydraulic engineering	E
Health physics, radiation protection	R
Maintenance Support	S
Operations Support	S
Quality Assurance	Q
Training	V
Safety Review	R
Fire Protection	E
Outside Contractual Assistance	O
R-Corporate Radiation Safety Officer Q-Corporate Quality Assurance Manager E- Corporate Engineering Manager S-Site Facility Manager V- Sr. Vice President of Compliance and Development O-Director of Operations	

18.2 QUALIFICATIONS OF APPLICANT

Envirocare is cognizant of the radiological nature of the disposal materials to be handled in this operation. Envirocare feels a major emphasis lies in the selection of the CRSO, as well as the Director of Operations and the construction contractor.

18.2.1 Corporate Radiation Safety Officer

The Corporate Radiation Safety Officer (CSRO) will have the following minimum qualifications:

1. B.S. graduate in Engineering, Chemistry, Physics, or physical science-related field; and,
2. Five years of supervisory experience in NORM, uranium mining/milling operations, UMTRA Projects or other related fields where handling and/or disposal of low level radioactive materials are involved.

18.2.2 Site Radiation Safety Officer (SRSO)

The Site Radiation Safety Officer (SRSO) will have the following minimum qualifications:

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.

2. Two years of supervisory experience in uranium mining/milling operations, UMTRA Projects, or NORM disposal operations where handling and/or disposal of low-activity or low-level radioactive materials are involved.

18.2.3 Health Physics Specialist

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to follow protocol and procedures, and maintain health physics schedules established by the CRSO.
4. Ability to work with contractor personnel and supervise radiation monitor(s) during operations.

18.2.4 Access Control Technician

1. Ability to learn and understand radiation safety principles and practices.
2. Ability to follow protocol and procedures, and maintain schedules established by the CRSO.
3. Ability to work with contractor personnel and oversee work areas, such as the unloading and wash down facilities.

18.2.5 Director of Operations

The Director of Operations will have the following minimum qualifications:

1. Civil Engineer, or other relevant engineering degree, with three years of experience in earth-moving construction projects
2. basically familiar with the principles of radiation safety, as applied to these types of projects.

18.2.6 Site Facility Manager

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.
2. Ability to learn and understand radiation safety principles and practices.

3. Ability to manage the operations at the site. To set schedules for personnel and complete assignments in a timely manner.
4. Ability to work with contractor personnel and supervise their work during operations.

18.2.7. Corporate Engineering Manager

The Corporate Engineering Manager will have the following minimum qualifications:

1. A Bachelor's degree in an engineering field

2. At least six years experience

3. Shall be a Utah certified professional engineer

18.2.8 Production Engineer

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; and one year of experience as a engineering technician or equivalent.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to follow protocol and procedures, and maintain construction operations and records as established by the Director of Operations
4. Ability to work with contractor personnel and supervise construction operations.

18.2.9 Site Engineer

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; and one year of experience as a engineering technician or equivalent.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to follow protocol and procedures, and maintain construction operations and records as established by the Director of Operations.
4. Ability to work with contractor personnel and supervise construction operations.

18.2.10 Construction Contractor

The construction contractor will be required to operate in accordance with the construction operation safety plan that includes, a comprehensive radiation safety/health physics plan. In addition, the construction contractor must demonstrate a willingness and commitment to comply with certain provisions, as outlined in Section 7, to which contractors may not normally be subjected:

1. Radiation monitoring of all construction personnel.
2. Decontamination and frisk-monitoring of personnel at access control portal.
3. Maintenance of Personnel in/out logs at access control.
4. Wearing protective clothing.
5. Decontamination of all vehicles and equipment prior to leaving the restricted area(s).
6. Making available to the Industrial Hygiene Consultant any requested records pertaining to employee exposure to occupational hazards, and to employee accidents.

18.2.11 Compliance and Permitting Manager

The Compliance and Permitting Manager will have the following minimum qualifications:

1. B.S. graduate in Engineering, Chemistry, Physics, or physical science-related field; and,
2. Supervisory experience in hazardous waste operations, where handling and/or disposal of hazardous materials are involved.

18.2.12 Corporate Quality Assurance Manager

The Corporate Quality Assurance Manager will have the following minimum qualifications:

1. Undergraduate technical degree, preferably in a science or engineering field, or a closely associated discipline, or equivalent technical experience.
2. For construction QA, the CQAM should have an understanding of materials testing methods for soil classification and compaction, of surveying methods for establishing the location of point coordinates and elevations, and of general construction techniques.
3. For laboratory QA, the CQAM should have an understanding of laboratory safety, methodology, and general chemistry concepts.
4. For health physics, the CQAM should have an understanding of industrial health and safety concerns, testing techniques, and ALARA concepts.

18.3 TRAINING PROGRAM

The training program for all contractor employees, Envirocare personnel and outside contractors/consultants is addressed in Section 17.5.6.3. All persons using or working with the radioactive material receive training which is commensurate with the materials he/she will be handling.

At the date of this submittal, Envirocare is current with the training requirements outlined in Section 17.5.6.3.

18.4 EMERGENCY PLANNING

The maximum credible accident at the Envirocare site would be the accidental dumping of a load at some location other than the disposal cell. The model used to calculate the permitted radionuclides in waste accepted at the site was designed to limit total occupational doses to 5 rem per year. If a load containing waste with the maximum permitted concentration was accidentally dumped, requiring its removal to the disposal cell, and if a full day is assumed for its removal, the maximum predicted dose to an employee would be 0.025 rem. Considering that most of the land within 10 miles of the site is under Bureau of Land Management (BLM) control and that there are no nearby residents, any dose received by a person outside of the controlled area would be a small fraction of 0.025 rem. Envirocare has an emergency response plan which is incorporated as part of the training procedures in Appendix C.

18.5 REVIEW AND AUDIT

The construction review and audit requirements are addressed in Section 14.1.4. The radiation safety audits are described in Section 14.7.

18.6 FACILITY ADMINISTRATIVE AND OPERATING PROCEDURES

18.6.1 Scope of work

At this time it is impossible to exactly state the amount of waste material to be handled or buried in a year. It is stated elsewhere in this application, that Envirocare anticipates approximately 500,000 tons per year. It is also impossible to estimate the time frame or schedule(s) for arrival of the material at the site.

18.6.2 Administrative Procedures

All personnel who work at the Envirocare facility will be required to abide by all site regulations and all requirements of this application. All violations of these requirements will be recorded on site violation forms and turned in to the Director of Operations. The implementation of this program will be under the direction of the Director of Operations

18.6.3 Operating Procedures

As described in the previous sections there are several people on the site who have the authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. Examples of situations that would require that the site be closed until remediation of the problem would be:

1. Windy conditions which cause unsafe conditions.
2. Construction equipment operating in an unsafe condition.
3. Lack of trained personnel to operate the site.

18.6.4 Required Personnel

Envirocare will only perform specific operational activities when the trained personnel responsible for these activities are on site. For example, a Field Testing Inspector or equivalent must be on site whenever material is to be placed on a portion of the embankment that needs soil density verification.

Whenever the Clive facility is in full operation the SRSO or authorized designee must be present on site.

18.7 Safety and Environmental Review Panel

Envirocare will establish a "Safety and Environmental Review Panel (SERP)." The SERP shall consist of a minimum of three individuals. One member of the SERP shall have expertise in management and will be responsible for managerial and financial approval changes; one member shall have expertise in operations and/or construction and shall have expertise in implementation of any changes; and, one member shall be the Corporate Radiation Safety Officer or equivalent. Other members of the SERP may be utilized as appropriate, to address technical aspects, in areas, such as health physics, groundwater hydrology, surface water hydrology, specific earth sciences, and others. Temporary members, or permanent members, other than those specified above, may be consultants. The SERP shall convene at least monthly to review, evaluate and make determinations regarding the licensing requirements for the following actions, or address other matters pertaining to the SERP.

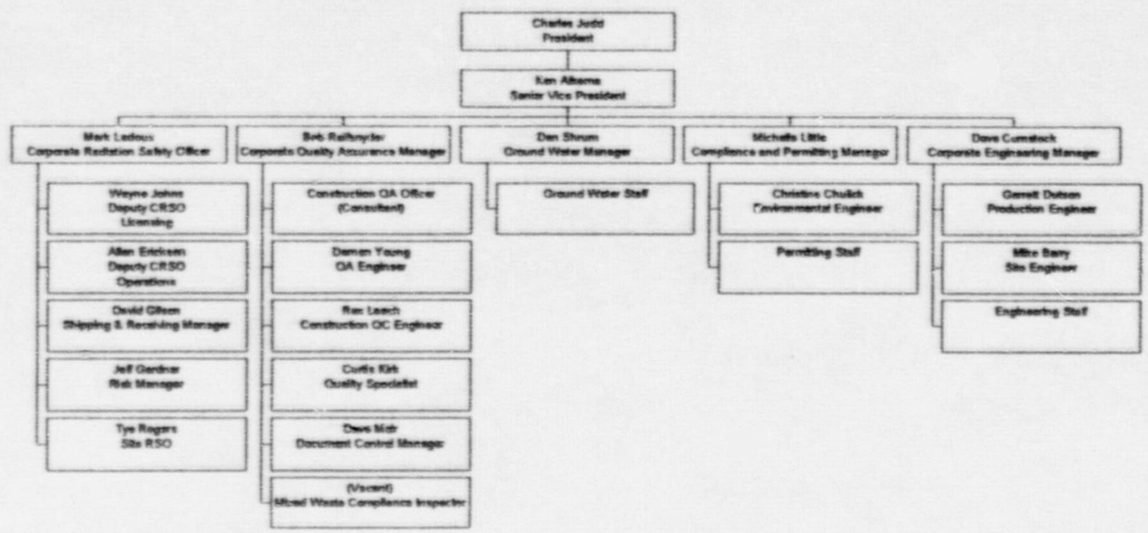
(1) Make changes in the facility or process, as presented in the application.
(2) Make changes in the procedures presented in the application.
(3) Conduct tests or experiments not presented in the application.

Envirocare will file an application for an amendment to the license, unless the following conditions are satisfied.

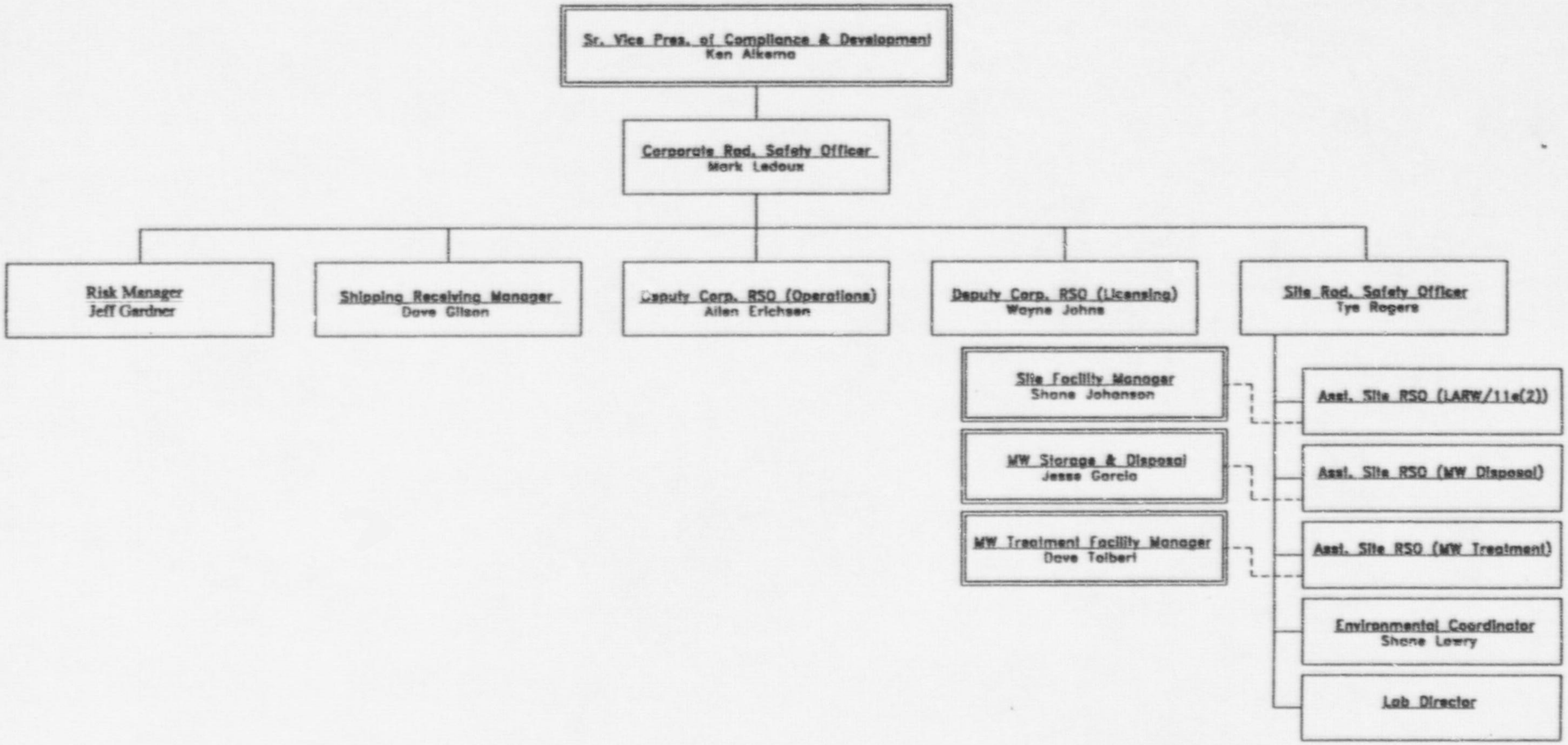
(1) The change, test or experiment does not conflict with any requirement specifically stated in this license (excluding the License Condition referencing the License Application or Reclamation Plan), or impair the licensee's ability to meet all applicable NRC regulations.
(2) There is no degradation in the essential safety or environmental commitments in the license application, or provided by the approved reclamation plan.
(3) The change, test, or experiment is consistent with the conclusions of actions analyzed and selected in the Final Environmental Impact Statement dated August 1993 (NUREG-1476).

Envirocare will maintain records of any changes made pursuant to this section. These records shall include written safety and environmental evaluation, made by the SERP, that provide the basis for the determination that the change is in compliance with the requirements referred to above. Envirocare will furnish, in the annual report to NRC, a description of such change, tests, or experiments, including a summary of the safety and environmental evaluation of each. Envirocare will annually submit changed pages to its license application to reflect changes made under this section.

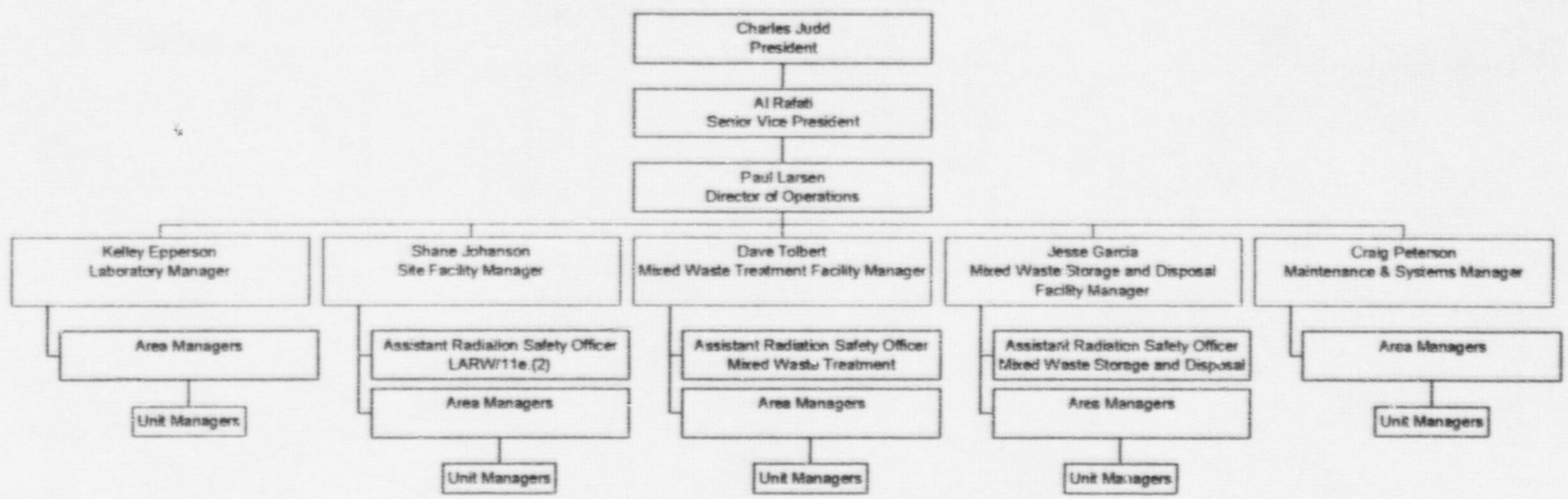
Corporate Development



Radiation Safety

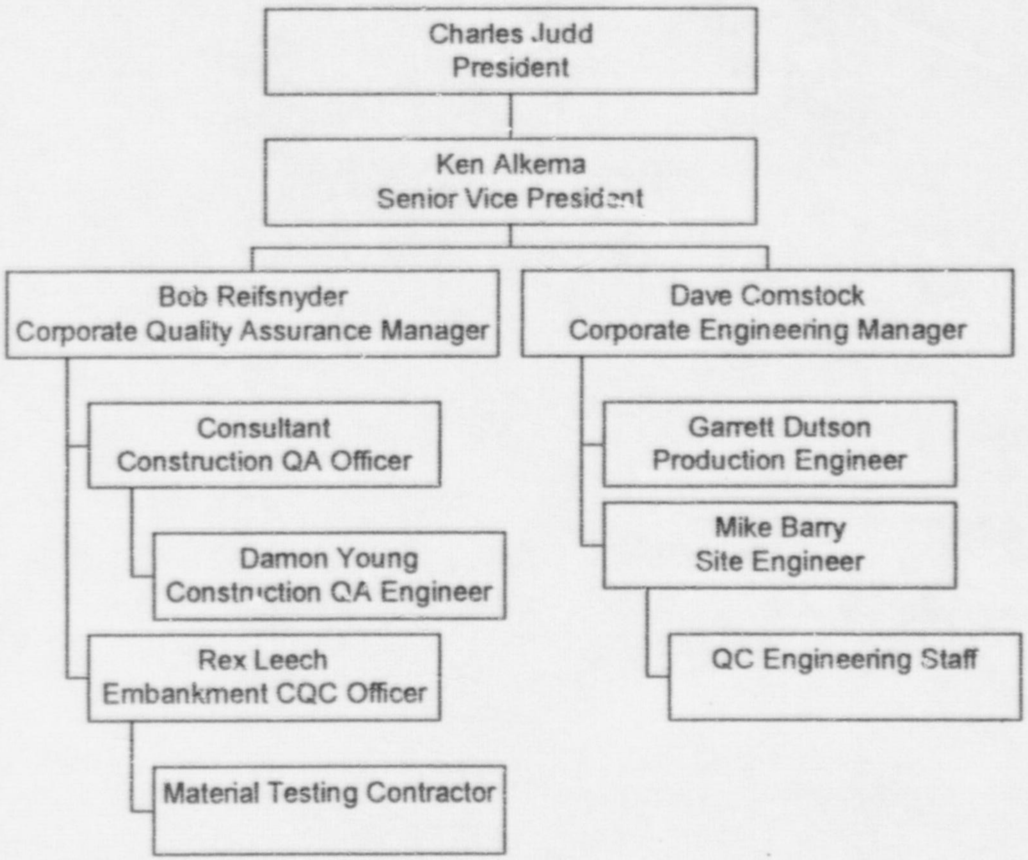


Operations

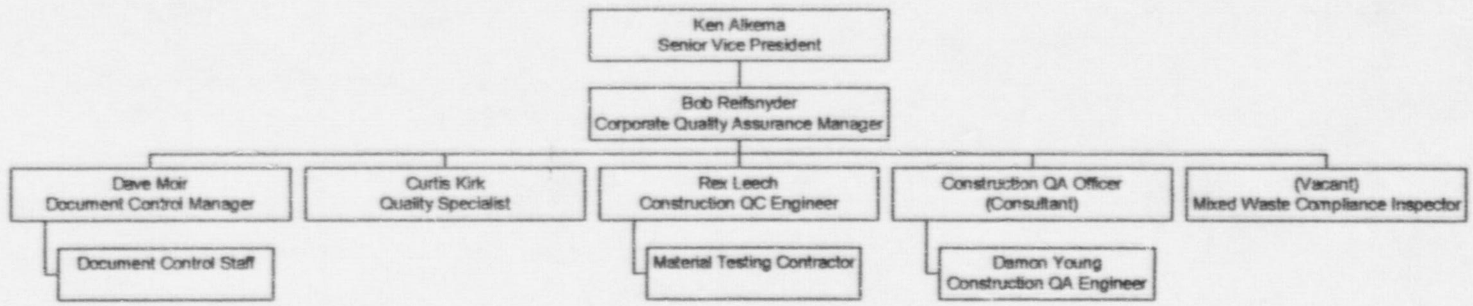


Charles Judd
President
Al Rafati
Senior Vice President
Paul Larsen
Director of Operations
Kelley Epperson
Laboratory Manager
Area Managers
Unit Managers
Shane Johanson
Site Facility Manager
Assistant Radiation Safety Officer
LARW/11e.(2)
Area Managers
Unit Managers
Dave Tolbert
Mixed Waste Treatment Facility Manager
Assistant Radiation Safety Officer
Mixed Waste Treatment
Area Managers
Unit Managers
Jesse Garcia
Mixed Waste Storage and Disposal
Facility Manager
Assistant Radiation Safety Officer
Mixed Waste Storage and Disposal
Area Managers
Unit Managers
Craig Peterson
Maintenance & Systems Manager
Area Managers
Unit Managers

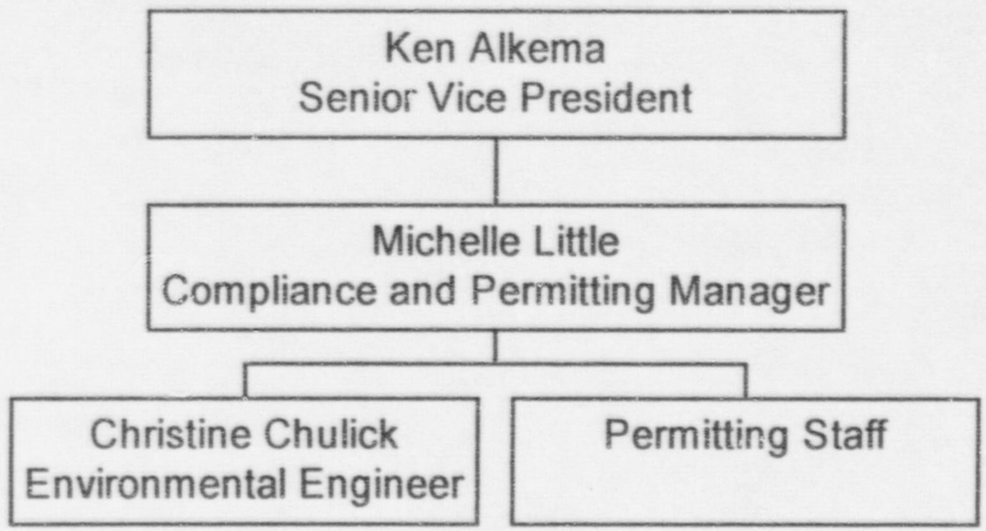
Construction Quality Assurance



Quality Assurance



Compliance and Permitting



SECTION 17. SAFETY ASSESSMENT

17.1 RELEASE OF RADIOACTIVITY

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

17.1.1 Characterization of Waste

17.1.1.1 Radionuclides

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Summarizing the data presented above, the following radiological waste character is anticipated for the Envirocare 11e.(2) disposal site. Considering the relative proportions of lower and higher activity waste at the site, Envirocare estimates that the overall average concentration for any radionuclide will be approximately 500 pCi/g; however, individual sites may vary widely around that average, as described above. Because of this, individual shipments of wastes may contain higher average concentrations of Ra-226 and Th-232. In the context of waste deliveries to the disposal site a shipment is taken to mean a single waste-hauling truck or rail car from a single generator. Weighted average concentrations in a shipment must not exceed 4,000 pCi/g for natural uranium or any radionuclide in the Ra-226 series; 60,000 pCi/g of thorium-230; or 6,000 pCi/g for any radionuclide within the thorium series, although they may be present at those concentrations together.

A conservatively-high estimate of the volume of material to be handled and disposed of at the site would be one-half million (500,000) tons/year. Assuming an average Ra-226 and Th-232 concentration of 500 pCi/g, the estimated annual average total activity disposed of would be 227 Curies for each of the radionuclides. Since the daughter products may be assumed to be in secular equilibrium, there would be approximately 227 Curies of each of the other important radionuclides, such as Ra-228 and Ra-224. The amount of Uranium would be expected to be less than 25 percent that of Ra-226. The average Th-230 concentration is expected to be similar to that of Ra-226 and will depend upon the disequilibrium of the radionuclides in that decay series. The actual amount of radioactivity disposed of in a given year will vary around the estimated 227 curies per radionuclide as actual concentrations and disposal amounts vary.

17.1.1.2 Chemical Constituents in the Waste

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

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

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

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

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

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

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

METALS
Arsenic
Barium
Beryllium
Cadmium
Chromium
Cyanide
Fluorine
Lead
Mercury
Molybdenum
Nickel
Selenium
Silver

VOLATILE ORGANICS
Acetone
2-Butanone
Chloroform
Carbon disulfide
1,2-Dichloroethane
Methylene chloride
Naphtha

SEMI-VOLATILE ORGANICS
Diethylphthalate
2-Methylnaphthalene

RADIONUCLIDES
Radium-226
Radium-228
Thorium-230
Thorium-232
Uranium

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

17.1.2 Infiltration

Section 4.1.1 discusses principal design features to minimize water infiltration into the embankment and disposed materials. As indicated in that section, calculations in Appendix M demonstrate that the amount of precipitation that infiltrates into the

embankment and percolates to the shallow groundwater under proposed conditions is negligible.

17.1.3 Radionuclide Release - Normal Conditions

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

1. Release of interstitially trapped radon and thoron gas when handling bulk wastes.
2. Exhalation of radon gas from embankment area(s) that have not been covered with the compacted clay radon barrier.
3. Exhalation of radon gas from embankment area(s) that have been covered with the compacted clay radon barrier.
4. Exhalation of thoron gas from the top layer of embankment areas which have not been covered with a layer of non-thorium-containing waste or clean clay.
5. Localized resuspension of dust from waste handling operations.
6. Windblown materials from the embankment and unloading area.

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

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

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

17.1.3.1 Off-site Impacts from Normal Operations

M&A (Appendices A and A-1) provided estimates of projected radionuclide release rates and radiological impacts during site operations, assuming waste which exhibits the radiological characteristics estimated for the overall 11e.(2) profile (500,000 tons per year of waste containing 500 pCi/g of each of the radionuclides in the uranium and thorium series). While these Appendices demonstrate compliance with 10 CFR 20.1301 and 10 CFR 20.1302 under the assumed conditions, they do not completely serve the purpose of evaluating the variable characteristics of waste quantities and

radionuclide concentrations which are expected to occur annually, or over shorter periods of time. M&A performed a sensitivity analysis of Envirocare's waste management procedures and waste characteristics (Appendix A-2). This analysis permits each waste handling procedure, from receipt to final closure, to be evaluated for its environmental impact while handling any quantity of wastes at any specified radioactivity concentration. Output from the analysis of Appendix A-2 will be used as input to the calculational spreadsheet described in Appendix A-3 to provide guidance to Envirocare planners in scheduling waste shipments and planning waste handling operations to meet the effluent concentration limits of Table 2, Appendix B to 10 CFR 20.1001 - 20.2401. The application of Appendices A-2 and A-3 to waste management will allow Envirocare to manage wastes within an envelope of quantities and radioactivity characteristics during the year while meeting the overall environmental results of Appendices A and A-1.

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

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

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

The dose calculations above, from Appendices A and A-1, were based on a single assumed average concentration in waste with an annual total of 500,000 tons of waste disposed, or an annual disposal of 227 Ci of each of the radionuclides in the uranium and thorium series. Occupational and environmental doses are shown to be almost completely dependent upon the total amount of radioactivity managed. While the use of Appendices A-2 and A-3 provide considerable flexibility in waste management, the reliance upon the modelling of Appendices A and A-1 will assure that occupational and environmental limpacts are as described in those appendices. With this option, Envirocare can safely dispose of any combination of radioactivity concentrations up the shipment limits of 4,000 pCi/g for natural uranium and any radionuclide in the ^{226}Ra series; 60,000 pCi/g of thorium-230; and 6,000 pCi/g for any radionuclide in the thorium series. Application of this approach would

automatically restrict the amount of waste which could be received at higher concentrations.

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

17.1.3.2 Occupational Radiation Exposures

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

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

$$_{b}D = 0.23\, E_b\, c$$

where: $_{b}D$ = Dose rate from an infinite cloud (rad/s)
E_b = Average beta energy per disintegration (MeV/dis)
c = Concentration of the beta emitting isotope in the cloud (Ci/m^3)

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

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

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

periods, Envirocare will closely monitor internal and external exposures to maintain TEDE as low as reasonably achievable and, in all cases, below the standards of 10 CFR 20.1201.

17.1.4 Radionuclide Release - Accidents or Unusual Operation Conditions

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

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

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

1. on-site truck turnover or collision
2. train derailment
3. flooding
4. tornado.

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

occurred into the Great Salt Lake. Impacts of these events were found to be negligible compared to the impacts from normal operations.

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

The following accidental on-site releases have been evaluated:

On-site truck turnover or collision

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

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

1.3×10^{-6}/km x 53 loads/day x 260 days/year x 1 km/load
$= 1.8 \times 10^{-2}$ or about 1.8%.

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

For a theoretical truck accident involving a yellowcake shipment, a 24-hour release period, all particles in the respirable range, and a population density of 7.5 persons per square mile, NRC estimated 50 year dose commitments to the lungs of the general public in the range of 0.7 to 9 person-rem. The yellowcake specific activity is about 6.77×10^{5} pCi/g while the average uranium or thorium concentrations expected at Envirocare would be 500 pCi/g, or a factor of 1300 lower. Individual shipments to Envirocare might have ^{226}Ra concentrations as high as 4,000 pCi/g, or similar to those found in uranium mill tailings. Concentrations of ^{232}Th in a small

fraction of shipments could be as high as 6,000 pCi/g. The dose per unit intake via inhalation is higher for Th-232 wastes than for yellowcake by up to a factor of 1000, depending upon the chemical form and radionuclide mix. Therefore, the postulated off-site public doses could be approximately an order of magnitude higher than for a yellowcake spill under the same circumstances. However, the population distribution around the Clive site is insignificant compared to that in the NUREG calculation and, therefore, the off-site population dose would be inconsequential.

For on-site workers, there would be a very short exposure time since there would be no reason to stand downwind for 24 hours (or even one hour). Assuming an accident involving the spill of a load of waste with a concentration of 15,000 pCi/g; a period of three hours for cleanup with no use of respiratory protection; an airborne concentration of 1 mg/m^3; and a respiratory rate of 1.2 m^3/h a total of 54 pCi of each nuclide would be inhaled. Comparing these to the ALI's from Appendix B of 10 CFR 20.1.001 - 4201, the sum of fractions is 0.022. The external gamma dose, using the relationship of 3.1 mrem/h/pCi/g for Ra-226 from Appendix A Section 3.7.3 and doubling for the contribution from Ra-228, would be less than 140 mrem. Such a dose added to the projected maximum TEDE of 1,032 mrem/y would still be well within the permissible annual exposures for radiation workers. In actual fact, no workers would be present under such conditions without respiratory protection and would not be standing on the spilled waste for more than a few minutes.

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

Train derailment:

The probability of a train derailment occurring on the Clive site is not readily calculable. However, because of the short length of track involved, the small amount of train movement, the low train speeds compared to truck speeds, and the relatively small number of cars compared to truck shipments, the probability of a derailment should be much less than the probability of a truck accident.

The dose to the workers and to the population should be much less than that for an off-site derailment and spillage event since trained workers and equipment would be available to immediately use dust control measures to control releases and cleanup the spill. The DOE, as discussed above, concluded that the dose to cleanup workers and nearby residents from such an off-site spill was insignificant. As a worst case,

the same assumptions could be applied as for the truck accident scenario above, with the same low total dose to emergency response teams.

Flooding:

Flood control features for both the Vitro and Clive sites have been designed and constructed to prevent erosion or off-site transport of wastes from the sites by overland flooding. Details of the flood control features are provided in Appendix F. No off-site transport of radioactive waste by flooding is anticipated. Cleanup of contamination caused by dispersion of stored or already disposed waste within the controlled area by flooding would replace placement of waste as an activity and radiation doses to workers would be the same as, or lower than, those received during normal operations.

Tornado:

From NUREG-0706 the probability of tornado occurrence in Utah is probably in the range of 1 to 5×10^{-4}. NUREG-0706 also estimates the consequences of a tornado striking a model uranium mill. In this case about 12.6 tons of yellowcake is entrained in the vortex, the vortex dissipates at the site boundary, all of the yellowcake is respirable in size, and the cloud is dispersed as a volume source by the prevailing winds. Settling velocity is negligible. The model predicts a maximum exposure at 2.5 miles from the mill, where the 50 year dose commitment is estimated to be 0.83 micro-rem. At the fence line (1600 feet) the dose is estimated to be 0.22 micro-rem. Our wastes would have average activities considerably less than this but as discussed above, the TEDE per unit intake is higher, resulting in comparable doses at receptor locations. Since there are no nearby population groups, the significance of this very small potential dose is even more insignificant.

Severe Winds

In the preceding discussion of airborne exposures resulting from tornadoes it was concluded that the maximum 50-year dose commitment at 2.5 miles would be less than 1 micro-rem. That conclusion is derived from a NUREG-0706 analysis of tornado-dispersed yellowcake from a uranium mill and is considered of a comparable magnitude to the transport of Th-232 waste from the Clive Site under similar conditions.

While severe winds on the order of 35 m/s have been recorded in the vicinity, the occurrence is infrequent and the duration is short. Assuming an order of magnitude increase in airborne concentrations during severe wind conditions which occur approximately one percent of the time, the time-weighted average off-site exposure would increase by only 10 percent. This would result in a maximum additional

annual collective TEDE of less than 1 mrem to current nearby population groups (See Table 3.20, revised, Appendix A-1)

17.1.4.1 Transfer Mechanism - Groundwater

The possibility of contamination releases to known water resources is highly unlikely. Without extensive treatment, use of the water in the South Clive area would appear to be confined to very limited industrial uses. There is minimal potential for degradation of water quality in the vicinity of the south Clive site inasmuch as the water at the site has been characterized as a brine, with levels of many constituents often exceeding EPA primary or secondary drinking water standards by a large amounts.

Envirocare has commissioned a hydrogeologic study to more accurately describe the possibility of groundwater contamination. This report by Bingham Environmental (Appendix GG) concluded that it would take 400 to 600 years for leachate to travel through the unsaturated zone and then another 800 years to reach the nearest off-site well. No non-radiological constituent would reach the Ground water in less than 700 years.on site would be 191 years. Using this estimate, it would take well over 1,000 years for any groundwater from the 11e.(2) cell to reach the boundary of the Envirocare facility.

17.1.4.2 Transfer Mechanism - Air

Because of the location of the South Clive facility, the meteorological characteristics of the area, and the lack of population within 20 miles of the facility, the impact of air as a transfer mechanism for radioactivity is limited. The modelling study conducted by Momeni & Associates (Appendix A) concluded that the annual population TEDE (exclusive of doses to workers at the nearby hazardous waste operations) after 16 years of operation would be 0.016 person-rem/year. Calculated TEDE to the nearby hazardous waste workers would add approximately 0.5 person-rem/year.

17.1.4.3 Transfer Mechanism - Surface Water

The probability of contamination through surface water is highly unlikely inasmuch as there are no surface waters at the site. As is stated previously, "No surface-water bodies are present on the South Clive site. The nearest stream channel ends about 2 miles east of the site and is typical of all the drainage along the transportation corridors within about 20 miles of the South Clive site. Stream flows from higher elevations usually evaporate and infiltrate into the ground before reaching lower, flatter land. The stream channels are well defined in their upper reaches, but as they

approach the flatland the size of the channel reduces until there is no evidence of a stream."

17.1.4.4 Other Transfer Mechanisms

Because of the location of the South Clive facility, the sparse biota in the area, and the lack of population within 20 miles of the facility, the impacts of other transfer mechanisms such as gamma radiation through air and transfer of radioactivity through biotic pathways are very small.

17.1.5 Radionuclide Transport

The most significant radioactivity transport mechanisms are air, groundwater, surface water, direct radiation and biotic pathways. The five periods of principal concern to NRC (NUREG-1199) include the operational, closure, observational and surveillance, active institutional control, and passive institutional control periods. In reality, the periods of real concern should be operational and post-closure.

During the closure period one would not ordinarily expect continuing shipments of waste so exposures from air, surface water, direct radiation, and biotic pathways should be less than exposures received during the operational period. No new wastes are being received, old wastes are being covered, and the surface is being decontaminated.

During the observational and surveillance, active institutional control, and passive institutional control periods the site has already been decontaminated, wastes are covered and there should again be no changes in exposures.

The evaluations of Appendix A address exposure pathways for operational periods and were compared to regulatory standards. Results were used to determine potential exposures to on-and off-site personnel. As discussed in Sections 17.1.3.2 and 17.1.4.2, projected doses to on-site radiation workers are 1 rem/year or less and the annual regional population TEDE to off-site residents and nearby industrial workers is approximately 0.5 rem.

17.1.6 Assessment of Impacts and Regulatory Compliance

The M&A report addresses the specific impacts of releases under normal operating conditions. Release mechanisms were evaluated, exposures to workers and the public assessed, and the results compared to applicable standards and regulations. It was concluded that with the proposed waste characteristics and operating procedures, exposures to the workers and the public will be within acceptable limits

and the design will limit the radon flux to 20 pCi/m^{2}s as proposed in 10 CFR Part 40, Appendix A.

While the exposures to site custodial personnel during the active institutional control period were not specifically evaluated, all waste will have been covered, gamma exposure rates will be near background, and radon emission rates will be limited to the design criterion of 20 pCi/m^{2}s. There is no reason to believe that exposures during this period will be more than a small fraction of those to the workers during operations.

For a discussion of impacts of releases due to accidents or unusual operating conditions see Section 17.1.4. In general, because of the relatively low radionuclide concentrations of the Clive wastes, it is difficult to postulate an on-site accident that could cause significant exposures to on- or off-site personnel.

17.2 LONG-TERM STABILITY

The embankment design will provide long-term stability and be relatively maintenance-free after site closure. Long-term stability is discussed in detail in Sections 4 and 6.

17.3 CONSTRUCTION SAFETY

Envirocare has implemented a construction safety plan which covers both Envirocare and contractor employees. While the prime contractor is responsible for developing his own safety and health plan, Envirocare performs safety inspections of the contractor's on-site operations to assure compliance with UOSHA and Envirocare regulations. The content of the plan includes:

1. Purpose/Goals - Envirocare and Contractor commit to the following goals:
    a. Safe and health working conditions for all on-site personnel.
    b. Protection of the general public.
    c. Compliance with all governmental safety and health regulations.
    d. Reduce liability to Envirocare and contractor to a minimum.
2. Establish an organizational chart to define responsibilities for safety and health program direction and enforcement.
3. Emergency Medical Care.
4 Pre-planning for unusual occurrences.
5. Safety & Health training program.
6. Control and monitoring of Safety and Health Plan.
7. Industrial Hygiene plan.

8. Corporate Safety Program.

Any contractor that performs work for Envirocare on this project must formally take responsibility to obey all site rules. Any contractor who is to work for an extended period of time at the site must submit their own Health and Safety Program.
All OSHA regulations will be under the jurisdiction of UOSHA. The Corporate RSO is responsible for overall development, direction and coordination of the Safety and Health Plan. The Site Manager is responsible for on-site implementation and enforcement of all safety and health provisions. It is recognized that industrial accidents pose a greater risk to employees than radiation risks and a significant effort is made to ensure a safe workplace. Employees are instructed to bring all health and safety concerns to their supervisor or the Site Manager. Unresolved concerns may be brought to the attention of UOSHA for immediate reconciliation.

The Safety and Health Plan relies on identification of risks, development of procedures to control those risks and to comply with UOSHA regulations, pre-employment safety training, continuing on-the-job safety training and on-going safety inspections of all operations. Radiation Technicians (Health Physics Specialist II), who are already trained in radiation safety, are also given responsibility to enforce all safety regulations.

17.4 Radiation Safety and Health Physics

17.4.1 Radiation Protection Policy

It is the policy of Envirocare, to maintain personnel/occupational radiation exposures as low as reasonably achievable (ALARA). Because of the nature of the 11e.(2) wastes, experience has shown that radiation exposures are normally low and Envirocare is committed to continuing to minimize exposures to the workers and the environment.

The average annual dose for 294 workers involved in the Vitro Remedial Action Project during 1986 was 50 mrem, with maximum exposures of 250 mrem. This maximum value is only 5% of the radiation dose standard of 10 CFR 20.101. Envirocare's experience with handling similar materials at it's LLRW facility was even better in that the highest total dose received during any year of Envirocare's five years (1988-1992) of operation was 200 mrem and the average annual dose equivalent was less than 50 mrem. The data are presented in Table 17.7.

In keeping with the ALARA principle, any reported personnel exposures in excess of 50 mrem/month will be investigated and documented by the Corporate Radiation Safety Officer(CRSO).

Procedures and methods to keep internal exposures ALARA include:

a. Dust suppression on all operational roads by application of magnesium chloride or watering at 2-hour intervals.
b. Speed limit of 35 mph on roads treated by dust suppressants; 10 mph on infrequently used roads.
c. Stopping operations in high wind conditions (all operations cease at winds of 40 mph; radiation safety personnel have authority to stop operations at lower wind speeds if dusting or other safety considerations warrant).
d. Placement of radon barrier on portions of the cell as they are completed.
e. Weekly area radiation surveys with investigation of increasing levels to determine the cause.
f. Requiring workers to wear respirations in areas of potential high dust concentrations, for example, the rollover and selected heavy equipment operations.
g. Pre-planning tasks which have the potential for higher than normal exposures to limit exposures through efficient use of time and handling procedures.

The Site Radiation Safety Officer (SRSO) will have the day-to-day responsibility for maintaining occupational and environmental radiation exposures ALARA, consulting such guidance documents as NRC Regulatory Guide 8.31, "Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Mills Will Be As Low As Reasonably Achievable" and draft Guide DG-8013, "ALARA Levels for Effluents from Materials Facilities." The SRSO will document ALARA activities including:

a. Reviews of new proposed disposal contracts to assure that Envirocare's procedures, facilities, and equipment are appropriate and sufficient to limit exposures to workers and the environment;
b. Monthly reviews of work area, perimeter, and environmental air monitoring results noting trends and adjusting work procedures when practical to further reduce potential exposures; and
c. Monthly reviews of work area gamma-ray exposure rates and advising the Site Manager(SM) on operational changes that will reduce exposures to ALARA levels.

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

17.4.2 Restricted and Controlled Areas

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

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

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

17.4.3 Radiation Dose Limits

17.4.3.1 Occupational Dose Limits for Adults

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

a. Annual limit will be the more limiting of:

1. The total effective dose equivalent (TEDE) equal to 5 rems; or
2. The sum of the deep-dose equivalent (DDE) and the committed dose equivalent (CDE) to any individual organ or tissue other than the lens of the eye being equal to 50 rems.

b. The annual limits to the lens of the eye, to the skin, and to the extremities, are:

1. An eye dose equivalent of 15 rems; and
2. A shallow dose equivalent of 50 rems to the skin or to any extremity.

c. Doses received in excess of the annual limits must be subtracted from the limits for planned special exposures that an individual may receive during the current year and during an individual's lifetime.

d. For soluble uranium, the intake by any individual is limited to 10 milligrams in any week in consideration of chemical toxicity.

17.4.3.2 Occupational Dose Limits to Minors

The annual occupational dose limits for minors are 10 percent of the annual dose limits specified for adults. The Site Radiation Safety Officer (SRSO) will review any work assignment given to minors to assure that exposures are maintained ALARA and within this guidance.

17.4.3.3 Dose Limit to an Embryo/Fetus

The dose equivalent to the embryo/fetus will be limited to 0.5 rem during the entire pregnancy in accordance with 10 CFR 20.1208. Envirocare's policy is to inform female workers of the regulations regarding protection of the embryo/fetus and to ask them to inform Envirocare in writing, upon discovery or suspicion of a pregnancy. The Corporate Radiation Safety Officer (CRSO) will review the work assignments and normally offer the woman the opportunity to take available positions in non-radiation areas for the duration of the pregnancy. If no positions are available, the CRSO will counsel the individual to assure an understanding by the individual of the additional risks of continued employment. If the woman continues to work in the radiation area, the SRSO will monitor the work assignments and activities to assure that the TEDE to the embryo/fetus is ALARA and limited to 0.5 rem.

17.4.3.4 Planned Special Exposures

Envirocare does not anticipate authorizing planned special exposures since the radiation levels and radioactive constituent concentrations in 11e.(2) byproduct material are low. In the event that circumstances warrant a planned special exposure, Envirocare will do so in full compliance with the guidance in 10 CFR 20.1206.

17.4.3.5 Summation of Occupational Internal and External Doses

Guidance for the summation of the internal and external dose equivalents are specified in 10 CFR 20.1202. Summation is not required if either the external or internal radiation exposures are not likely to exceed 10 percent of the limit. This includes occupational exposures to adults as well as minors and to the embryo/fetus.

It is unlikely that exposures to workers at the Envirocare facility will exceed 10 percent of the allowable limits for direct radiation as well as internal radiation. Data for the UMTRA Project disposal at Clive show that the average annual dose equivalent from direct radiation was 50 mrem, with a maximum individual dose equivalent of 250 mrem. Envirocare has been operating the LARW facility beginning in 1988. The maximum individual dose equivalent from 1988-1992 was 200 mrem. Similarly the lapel sample and work area monitoring results indicate that the airborne particulate concentrations are near background levels.

Should Envirocare find that summation of occupational internal and external doses is necessary, the following method will be employed:

a. Should the internal dose as determined by air monitoring results, bioassay, or other means - as well as the dose from external sources as determined by radiation dosimeters - likely exceed 10 percent of the allowable limits, the Committed Effective Dose Equivalent (CEDE) will be added to the Deep Dose Equivalent (DDE) and compared with the Total Effective Dose Equivalent (TEDE) limit of 5 rem for adults and 0.5 rem for minors and the fetus/embryo.

b. If the only intake of radionuclides is by inhalation, the procedure specified in 10 CFR 20.1202(b) may be applied. The TEDE limit will not be exceeded, according to this procedure, if the sum of the DDE divided by the TEDE limit and one of the following, does not exceed unity:

   1. The sum of the fractions of the inhalation ALI for each radionuclide, or
   2. The total number of derived air concentration-hours (DAC-hours) for all radionuclides divided by 2,000, or
   3. 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 of 50 rem.

c. If the intake by oral ingestion exceeds 10 per cent of the oral ALI, Envirocare will account for this intake and include it in demonstrating compliance with the limits.

d. If intake occurs via wounds or skin absorption, Envirocare will evaluate these intakes and include these in the calculation of the TEDE.

17.4.3.6 Determination of Prior Occupational Dose

If any employee is anticipated to receive an occupational dose in excess of 10 percent of the limits presented in this Section, Envirocare will determine the previous radiation exposure for use in limiting the annual dose equivalent to the allowable limits and for planning special exposures.

Determination of prior occupational exposures will be done by

1. Obtaining a written signed statement from the employee or his most immediate employer, that discloses the nature and the amount of any occupational dose that the individual may have received during the current year; and
2. Obtaining or attempting to obtain from the employee's most recent employer, a written signed statement in the form of an NRC Form 4 or an equivalent form, showing the life-time occupational exposure history. In case this cannot be done, the guidance in 10 CFR 20.2104 will be followed.

17.4.3.7 Radiation Dose Limits for Individual Members of the Public

Operations will be conducted such that the additional dose equivalent to individual members of the public will be limited in accordance with the limits of 10 CFR 20.1301, 10 CFR 61, and 10 CFR 40, Appendix A. The limits are:

a. The total effective dose equivalent to individual members of the public from the licensed operation will not exceed 25 mrem per year above natural background levels, radon and radon daughters excepted.
b. Radon and radon daughters will be limited to levels specified in Table 2 of 10 CFR [20.1001-20.2401], Appendix B.
c. The total effective dose equivalent limit to occupants in the controlled area (other than restricted areas) will not exceed 100 mrem per year above background levels.
d. The dose equivalent in any unrestricted area from external sources will not exceed 0.002 rem in any one hour.

Table 3.12, revised, Appendix A-1, shows the calculated concentrations of particulate radioactivity at the site boundaries. The projected concentrations are in the range of ambient background concentrations and are well below the concentration limits of Appendix B to 10 CFR 20.1001-20.2041. Airborne particulate monitoring will be performed to confirm those predictions.

Any employees who are believed to have received a TEDE of greater than 200 mrem from any source in one quarter will be notified and will assist in determining the source of the exposure and in finding a way to reduce future exposures.

17.4.6.5 Occupational Radon and Radon Daughter Monitoring

The handling of large quantities of Ra-226 and Th-232 bearing materials is expected to release Rn-222 (radon) and Rn-220 (thoron). The concentrations will vary depending upon the type of waste handled.

The occupational limit for radon daughter exposure is four (4) WLM while the limit for thoron daughter exposure is 12 WLM.
The occupational exposure limit for radon without daughters present is 4,000 pCi/l while for radon with all daughters present (100 % equilibrium) is 30 pCi/l. The exposure limit for thoron without daughters is 7,000 pCi/l and 9 pCi/l with daughters in equilibrium.

All work areas, including the administration building, will be monitored for radon and thoron using pairs of E-Perm ion chambers. One chamber responds to radon and thoron, the other responds primarily to radon. The readings along with the difference in the readings are used to calculate the radon and thoron concentrations. The minimum detectable concentration varies with the mixture of radon and thoron. If only radon is present, the MDC is approximately 500 pCi/liter-hours, or 0.75 pCi/l-month, where a month is considered continuous exposure for 4 weeks. If only thoron is present, the MDC is approximately 3.6 pCi/l-month. Detectors will be placed in the work areas and read weekly. While the measured average concentrations will be for 24 hours/day rather than the average for the work day, the results should be conservative in that the meteorology of the site is expected to enhance the levels at night.

Due to the long exposure times for the E-Perms, other measurements of the work area environment will be made to assess the workers exposure to radon and thoron and their daughter products. The E-Perm results of the radon and thoron measurements will be supplemented by grab samples for radon and thoron concentration and grab samples for radon and thoron WL determinations. If exposures are likely to exceed 10 percent of the allowable limits over a 40 hour exposure period, the grab sample results will be used to estimate the radon daughter equilibrium and the E-Perm radon concentration

results will be used to calculate a monthly average WL for radon and thoron. The radon and thoron WL results will then be used in determining the internal dose equivalents for the workers.

The occupational limit for radon daughter exposure is four (4) working months (WLM) per year, which is equivalent to a DAC of 30 pCi/l of Rn-222 in equilibrium with its daughters.

Instant WL Monitors or grab sample techniques will be used to monitor the work area on a weekly basis during periods of calm winds. For work areas routinely falling below 10 percent of the WL limits for radon and thoron daughters(0.03 WL and 0.1 WL for radon and thoron, respectively), the exposure will not be considered in the dosimetry program, provided there are no minors or declared pregnant women in the area (see 10 CFR 20.1205 (g)).

If grab samples are taken, the Ogden method, [Ogden, T.L. (1974). "*A method for measuring the working-level values of mixed radon and thoron daughters in coal mine air.*" Ann Occ. Hyg. **17**, 23.] [Ogden, T.L. (1977). "*Radon and thoron daughter working levels from ordinary industrial hygiene samples*" Ann. Occ. Hyg. **20**, 49.] will be used to measure radon and thoron daughter - WL concentrations with sample collection volumes and counting times sufficient to provide a lower limit of detection (sensitivity) of better than 0.03 WL (See NRC Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills" and the references cited therein). Instant WL meters or continuous WL monitors will be used only if the equivalent sensitivity can be achieved.

17.4.6.6 Environmental Monitoring Program

The environmental monitoring program is presented in Section 7.

17.4.7 Personnel Protection and Contamination Control

17.4.7.1 Access Control

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

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

1. Permanent employees will be issued a thermoluminescent dosimeter (TLD) badge provided by Envirocare. These dosimeters will be exchanged and

returned to the vendor on a quarterly basis. Permanent employees will pick up and turn in their dosimeters at the beginning and end of their working day at the manned access control point.

2. Individuals who are visiting the site on a limited basis will be issued a pocket dosimeter to record exposure. Visitors will pick up and turn in their pocket dosimeters at the manned access control point when they enter and exit the site. The dosimeters will be read as the individual leaves the site and recorded in the Access Log.
3. A group of visitors may all use the exposure from either one TLD or one pocket dosimeter in a situation where the entire group is to stay in the same vicinity while in the restricted area.

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

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

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

The quarterly dosimeter results from the quarterly exchange of dosimeters will be promptly recorded by the Site Radiation Safety Officer (SRSO), or his designee. The data will then be reviewed by the SRSO. Higher than expected personnel exposures will be further investigated by the Corporate Radiation Safety Officer (CRSO) and/or a contractor consultant.

All exiting employees must be surveyed for contamination using an alpha sensitive instrument. Records are maintained of the number of employees found to be contaminated and the level of contamination.

Personnel or materials leaving the restricted area will be required to meet the conditions of the following table (see Section 16.3 for equipment/vehicle decontamination procedures):

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

Column I	Column II	Column III	
Nuclide[a]	Average[b,d,f]	Maximum[b,d,f]	Removable[b,e,f]
U-nat,U-235,U-238, and associated decay	5,000 dpm alpha/100cm^2	15,000 dpm alpha/100cm^2	1,000 dpm alpha/100cm^2 products
Transuranics, Ra-226, Ra-228,Th-230,Th-228, Pa-231,Ac-227,I-125, I-129	100 dpm/ 100 cm^2	300 dpm/ 100 cm^2	20 dpm/ 100 cm^2
Th-nat,Th-232,Sr-90 Ra-223,Ra-224,U-232 I-126,I-131, I-133	1,000 dpm/ 100 cm^2	3,000 dpm/ 100 cm^2	200 dpm/ 100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except SR-90 and others noted above	5,000 dpm beta-gamma/100 cm^2	15,000 dpm beta-gamma/100 cm^2	1,000 dpm beta-gamma/100 cm^2

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

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

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

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

e. The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping the area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of know efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

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

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

There will be no high or very high radiation areas on site due to the concentration limitations in the waste acceptance criteria. As shown in Section 17.1.4, even with

wastes as high as 15,000 pCi/g of each radionuclide the external gamma exposure rate would not exceed 50 mR/h. Therefore, no special access control procedures as required in 10 CFR [20.1601-20.1602] will be developed.

17.4.7.2 Protective Clothing and Change Facilities

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

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

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

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

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

17.4.7.3 Respiratory Protection Program

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

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

recordkeeping. The CRSO is responsible for administering the respiratory protection program.

17.4.7.4 Dust Control Measures

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

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

   a) Date of application
   b) Number of treatments
   c) Rainfall received
   d) Time of day treatments were made

17.4.7.5 Envirocare Site Regulations

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

17.4.8 Health and Safety Training

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

persons to enter the restricted area is related to the activities for which the person will enter the restricted area.

There are three categories of restricted-area functions:

(1) Permanent Employee
(2) Temporary Worker
(3) Visitor

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

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

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

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

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

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

CATEGORY	Restricted Area Safety Training 1-hr	Read/Sign Site Regs	3-hour Rad Safety Training	Refresher or Repeat After
Permanent Employee	Yes	Yes	Yes	6 months* Refresher
Temporary Worker	Yes	Yes	No	1 week Repeat**
Visitor	No	Yes	No	3 months Repeat
* Refresher course for permanent employees is one-hour review course ** After a temporary worker has received training for three weeks of restricted-area work within any one-year period, the temporary worker must receive the permanent employee training prior to performing additional work within the one-year period.				

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

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

The training is meant to educate the employees in the fundamentals of handling radioactive materials, to provide information on the ways and means of minimizing exposure, and to inform employees of practices and programs aimed at preventing possible spread of contamination.

The semi-annual refresher sessions for permanent employees will be provided to keep the employees aware of the nature of the material with which they have daily contact. The semi-annual refresher course will be a one-hour review of the topics discussed in the 3-hour training.

The Restricted Area Entrance Training will be given on site by the CRSO or SRSO, or any Envirocare Health Physics Specialist II. During this training, procedures and precautions will be explained and the trainees will be required to read and sign either the release form or a training roster form. The training records will be maintained by the SRSO.

In addition to the above training all Envirocare site employees will be required to attend at least 20 hours of training annually taught by qualified personnel. This training will be tailored to the specific employees needs and duties and will cover such topics as general

occupational safety, radiological safety, and training on any specific items such as new procedures or safety deficiencies.

17.4.9 Staffing and Personnel

17.4.9.1 Responsibilities

The Corporate Radiation Safety Officer (CRSO) is responsible for assuring that the environmental health and safety requirements at the site are being met and, in particular, the operations at the site are in compliance with Nuclear Regulatory Commission License Requirements. All health and safety related procedural changes are approved by the CRSO.

The Site Radiation Safety Officer (SRSO) has the day-to-day radiation safety responsibilities and reports to the CRSO while working very closely with the Site Manager. Assisting the SRSO are ~~Radiation Monitors (Health Physics Specialist I),~~ Access Control Technicians, Health Physics Specialists ~~II~~, and an Environmental Coordinator . The Environmental Coordinator is responsible for conducting the routine environmental monitoring program and performing certain laboratory analyses.

17.4.9.2 Certification for ~~Health Physics Specialist I~~ Access Control Technician and Health Physic Specialist ~~II~~

All personnel must be certified before they can be classified as either an ~~Health Physics Specialist I~~ Access Control Technician or a Health Physics Specialist II. This certification will include training and testing beyond that given in the restricted-area training program. Specific training and experience requirements for the positions, entrance training, on-the-job training, and examinations are listed in the Procedures Manual, Appendix B. The following is a summary of requirements for certification in those areas:

~~Health Physics Specialist I~~Access Control Technician

1. 20 classroom hours of training in areas of chemistry, physics, radiation safety, construction safety, operation of equipment and site operations.
2. Pass a written exam designed specifically for ~~Health Physics Specialists I~~access Control Technician.
3. Pass, to the satisfaction of the Site Radiation Safety Officer, a practical test designed to assure that candidate possesses knowledge for all equipment is being handled properly and all duties can be performed effectively.

Health Physics Specialist ~~II~~

1. 40 classroom hours of training in areas of chemistry, physics, radiation safety, construction safety, operation of equipment and site operations.
2. Pass a written exam designed specifically for Health Physics Specialist ~~II~~ .

3. Pass a laboratory test designed to assure that all equipment is being handled properly and all duties can be performed effectively.

In addition to the certification, each ~~Health Physics Specialist I~~Access Control Technician and Health Physics Specialist ~~II~~ must maintain certification by completing the annual training described in Section 17.4.6.3.

SECTION 18. ORGANIZATION

18.1 ORGANIZATIONAL STRUCTURE

18.1.1 Design, Construction and Pre-Operational Responsibilities

The operations and design of the Clive facility is described in detail in Sections 4 and 16. The waste material is placed in an earthen embankment, compacted in place, and covered with barriers to reduce radon emanation below Commission guidelines and to protect the embankment from the effects of weather erosion.

During the development and preparation of this application, Envirocare has utilized the services of the following consultants/contractors:

1. Donald W. Hendricks, CHP, President
   DON HENDRICKS AND ASSOCIATES, INC.
   609 No. Crestline Drive
   Las Vegas, Nevada 89107
   702/878-4420

2. Jeff Throckmorton, CIH, President
   HEALTH & SAFETY SERVICES, INC.
   10508 Aberdeen Lane
   Highland, Utah 84003
   801/756-0063

3. Gary M. Sandquist, Ph.D.
   1738 Ramona Avenue
   Salt Lake City, Utah 84108-3110
   801/486-8521

4. Craig B. Forster, Ph.D.
   3479 East Quad Road
   Salt Lake City, Utah
   801/581-3864

5. Stanley L. Plaisier, P.E.
   BINGHAM ENVIRONMENTAL, INC.
   5160 West Wiley Post Way
   Salt Lake City, Utah 84116
   801/532-2230

6. T. Leslie Youd, Ph.D.
   1132 East 1010 North
   Orem, Utah 84057
   804/378-6327

7. Blair McDonald, P.E.
   343 South 1000 East
   Salt Lake City, Utah 84102

Envirocare of Utah, Inc., with the assistance of these consultants, developed the personnel monitoring systems, data/record keeping systems, disposal material analysis and handling procedures, environmental monitoring systems, employee training, and general health and safety procedures and other technical supporting information for the 11e.(2) disposal project.

18.1.2 Operational Phase

The operational phase is also the construction phase of this proposed disposal project, in that the disposal project is discussed in Section 4 and 16.

A conceptual organizational chart is included as Figure 18.1, showing by responsibility the major divisions of Envirocare:

1. The peripheral activities of Scheduling, Accounting, and Marketing are represented on the organizational chart but do not need to be further described in this application.

2. President. The President oversees and provides direction and leadership for the operation. At a minimum, the president will:

   a. Promulgate company policies that identify his commitment to safety, the importance of compliance with requirements, the employees responsibilities to identify safety concerns to management, the need for adherence to company procedures, etc.
   b. Visit the site and observe the operations at least quarterly.
   c. Receive for his review summary audit reports, follow-up reports, close-out reports, NRC inspection reports and State inspection reports to ensure operations are conducted in accordance with Envirocare's high standard for quality and safety.

3. Corporate Radiation Safety Officer (CRSO) - Responsible to the Sr. Vice President of Compliance and Development and works very closely with the Director of Operations and Site Radiation Safety Officer (SRSO). The CRSO is responsible for implementation of

and compliance with all protocols and procedures of the radioactive materials license, including, health and safety monitoring, environmental monitoring, training, and personnel monitoring. The CRSO ensures that adequate instrumentation and equipment is used and that adequate measurements are made to ensure that all applicable standards for personnel exposures to radiation and radioactive materials are satisfied including:

- Shipping and Receiving of Radioactive Materials
- Airborne radioactivity
- Surface contamination
- Internal and external exposures
- Effluents
- Environmental monitoring
-

The CRSO shall also be responsible for the annual report which summarizes all of the previously mentioned information. The annual report will be provided to the President, the Sr. Vice President of Compliance and Development, and the Sr. Vice President of Operations and Business Development for review and appropriate actions.

The CRSO has authority to terminate any activities on the site that are deemed to be unsafe. The CRSO may also suspended activities until hazard-abatement measures have been performed. The CRSO is responsible for health physics and radiation protection, training, and safety review.

It is anticipated that the CRSO will work 20 hours per week on issues related to the 11e.(2) project. The remainder of his time will be used to work on issues related to the Low Activity Radioactive Waste (LARW) project currently operating at the Clive site.

4. Site Radiation Safety Officer (SRSO) - The SRSO is responsible to the CRSO and works very closely with the Site Facility Manager. The SRSO or designee is responsible for on-site radiation safety and implementation of and compliance with all protocols and procedures of the radioactive materials license, including health and safety monitoring, environmental monitoring, training, and personnel monitoring. The SRSO determines whether adequate instrumentation and equipment are being used and whether adequate measurements are made to ensure that all applicable standards for personnel exposures to radiation and radioactive materials are satisfied. The SRSO is also responsible for oversight of gamma spectral analysis,

the environmental program, and instrument program. The SRSO provides technical direction for radiological laboratory functions.

The SRSO has authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. This determination may be made unilaterally or upon receiving reports of suspect conditions from other site supervisors, contractors, visitors or employees.

It is anticipated that the SRSO will work 20 hours per week on issues related to the 11e.(2) project.

5. Assistant Radiation Safety Officers (ARSO). Assistant Radiation Safety Officers are designated to each area of operation (i.e., Mixed Waste Treatment, Mixed Waste Disposal,LARW/11e.(2).The ARSO's are responsible for managing the health physics team, performing daily site inspections, and observing field operations. The ARSO's can serve as acting SRSO and report to the SRSO.

   The ARSO's have authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. This determination may be made unilaterally or upon receiving reports of suspect conditions from other site supervisors, contractors, visitors, or employees.

6. The Environmental Coordinator is responsible to the SRSO. The Environmental Coordinator has authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. The Environmental Coordinator is charged with carrying out the environmental monitoring activities on site including:

   a. Implement applicable radiation control regulations and all provisions of radioactive material license.
   b. Data base management/record keeping to document all environmental monitoring activities at the site.
   c. Analysis of disposed material to document receipt and disposition.
   d. Analysis of disposal material to document receipt and disposition
   e. Other duties as assigned

7. Health Physics Specialists are responsible to the appropriate ARSO for the Area assigned (i.e., Mixed Waste Treatment, Mixed Waste Disposal, or LARW/11e.(2) and are trained by and have their work reviewed by the SRSO. Health Physics Specialists have direct access

access to the Facility Manager and SRSO on matters dealing with radiological safety. Health Physics Specialists will work on both the 11e(2), Mixed Waste, and LARW operations. Health Physics Specialists have the authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. They are charged with carrying out the health physics activities on site including:

a. Implement applicable radiation control regulations and all provisions of radioactive material license.
b. Personnel monitoring of Envirocare and contractor employees.
c. Assist in conducting training for new employees or refresher training for incumbent employees.
d. Supervision of truck/equipment decontamination facility.
e. Data base management/record keeping to document all disposal and health physics activities on site.
f. Perform reviews of previous radiation dose records with individual site workers.
g. Maintain continuous survaillance of site operating conditions and act to prevent actions which might result in the release or spread of radioactivity.
h. Other duties as assigned.

8. Access Control Technicians are responsible to the ARSO of the Mixed Waste Treatment area or the ARSO of the LARW/11e.(2) Area. They are charged with carrying out minimal health physics activities on site:

a. Implement applicable radiation control regulations and all provisions of radioactive material license.
b. Access Control monitoring of Envirocare and contractor employees.
c. Manning of Access Control portal.
d. Perform and document weekly surveys of radiation dose rates and surface contamination in assigned areas.
e. Other duties as assigned.

9. Sr. Vice President of Operations and Business Development.- The

Sr. Vice President of Operations and Business Development reports to the President of Envirocare. The Sr. Vice President of Operations and Business Development is responsible for the overall management of direct operations and support functions for the disposal facility. The Sr. Vice President of Operations and Business Development works closely with other corporate personnel to ensure that all operations are conducted in a planned and safe manner in accordance with all regulatory requirements.

The Sr. Vice President of Operations and Business Development shall establish and promulgate departmental employee policy when needed. The Sr. Vice President of Operations and Business Development shall also be responsible for investigating innovative methods of improving operations and/or efficiency.

10. Sr. Vice President of Compliance and Development.- The Sr. Vice President of Compliance and Development reports to the President of Envirocare. The Sr. Vice President of Compliance and Development oversees and directs compliance, licensing, and permitting activities at Envirocare; including such areas as quality assurance, radiation safety, environmental monitoring, ground water monitoring, safety, training, and regulatory affairs.

    The Sr. Vice President of Compliance and Development shall oversee and facilitate permit and license renewals, modifications, and amendments. This position will set compliance objectives jointly with the Operations Department personnel. Direction and support will be provided for policy development and site training to assist in ensuring compliance.

11. Director of Operations - The Director of Operations must be an experienced Civil Engineer, or other relevant engineering degree. The Director of Operations reports to the Sr. Vice President of Operations and Business Developmentand is charged with the responsibilities of the operations of the waste disposal site in an efficient and safe manner in accordance with design specifications and all applicable regulations.

    The Director of Operations is responsible for site operations including laboratory management, cell construction, waste management and disposal. The Director of Operations is directly responsible for negotiating contracts with subcontractors.

12. The Corporate Engineering Manager – The Corporate Engineering Manager perofrms certification of engineering design drawings, project plans, construction reports, and As-Built Drawings. The Corporate Engineering Manager is responsible for the management of technical and engineering support, including site structural engineering, soil mechanics, materials, and hydraulic engineering. The Corporate Engineering Manager provides or procures services from internal resources or technical contractors as necessary; provides tehcnical and engineering support for the operation

including site layout and design reviews; and approves with QA oversight, those designs and specifications.

13. The Site Facility Manager– The Site Facility Manager is responsible for the day-to-day operation of the Clive facility. The Site Facility Manager is to work closely with the SRSO to assure that all aspects of site operation are conducted according to the applicable regulations. The Site Facility Manager has limited specific responsibilities so that his efforts can be used in ensuring the effectiveness of the overall operational activities at the site. The Site Facility Manager is also responsible for the management of the site maintenance support and fire protection.

14. Production Engineer - The Production Engineer is responsible to the Corporate Engineering Manager and is responsible for overseeing the production, scheduling, and coordination aspects of facility construction with the exception of QA (which is the responsibility of the QAM). During construction, the Production Engineer will regularly inspect the construction site. The Production Engineer will coordinate the selection of the construction contractor(s) and administration of the construction contract, including any changes. The Production Engineer will review proposed design, engineering, or construction changes and submit these changes to the Corporate Engineering Manager for approval.

15. Site Engineer – The Site Engineer is responsible for construction quality control, overseeing the production, scheduling and coordination aspects of facility construction, with the exception of QA (which is the responsibility of the QAM). During construction, the Site Engineer will regularly inspect the construction site. The Site Engineer will coordinate the selection of the construction contractor(s) and administration of the construction contract, including any changes. The Site Engineer will review proposed design, engineering, or construction changes and submit these changes to the Corporate Engineering Manager for approval.

16. Construction Contractor - responsible to Site Facility Manager to perform construction, earth moving, and disposal activities in accordance with approved procedures and specifications. The Construction Contractor is also charged with maintaining compliance with all provisions of UOSHA and making records available for review by the Industrial Hygiene Consultant.

17. The Compliance and Permitting Manager – The Compliance and Permitting Manager is responsible for Initiating, producing, and

obtaining appropriate licenses and permits. The Compliance and Permitting Manager oversees the adminstration of the Air Quality Program and the preparation of all reports submitted in accordance with Envirocare's licenses and permits. The Compliance and Permitting Manager has the authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until abatement measures have been performed.

18. The Corporate Quality Assurance Manager ("CQAM") is responsible for ensuring that the quality assurance requirements outlined in the Quality Assurance Program Document (QAPD) are implemented. The reporting relationships shown in Figure 18.1 allow the CQAM sufficient authority and autonomy to implement and direct the QAPD; to identify quality problems; to initiate, recommend, or provide solutions; and to verify implementation of solutions independent of undue influences, and responsibilities, such as costs and schedules. As such, the CQAM reports directly to the Sr. Vice President of Compliance and Development in implementing the QAPD.

19. Outside Contractual Assistance.

    As indicated in Section 18.1.1, Envirocare has access to qualified consultants to assist in the development and implementation of radiological health and safety plans, environmental monitoring programs, industrial hygiene and safety programs. These consultants will be utilized extensively to provide reviews of safety, employee training, evaluation of fire protection systems, and quality assurance reviews in addition to continuous operations support. These contractors are responsible to the President of Envirocare.

All Envirocare management personnel and personnel with safety responsibilities will have free access to each other to resolve immediate safety, operational or other issues.

In order to more fully outline the responsibilities assigned, the following chart is provided with the applicable assignments:

RESPONSIBILITY	POSITION
Structural, soil mechanics, materials, hydraulic engineering	E
Health physics, radiation protection	R
Maintenance Support	S
Operations Support	S
Quality Assurance	Q
Training	V
Safety Review	R
Fire Protection	E
Outside Contractual Assistance	O
R-Corporate Radiation Safety Officer Q-Corporate Quality Assurance Manager E- Corporate Engineering Manager S-Site Facility Manager V- Sr. Vice President of Compliance and Development O-Director of Operations	

18.2 QUALIFICATIONS OF APPLICANT

Envirocare is cognizant of the radiological nature of the disposal materials to be handled in this operation. Envirocare feels a major emphasis lies in the selection of the CRSO, as well as the Director of Operations and the construction contractor.

18.2.1 Corporate Radiation Safety Officer

The Corporate Radiation Safety Officer (CSRO) will have the following minimum qualifications:

1. B.S. graduate in Engineering, Chemistry, Physics, or physical science-related field; and,
2. Five years of supervisory experience in NORM, uranium mining/milling operations, UMTRA Projects or other related fields where handling and/or disposal of low level radioactive materials are involved.

18.2.2 Site Radiation Safety Officer (SRSO)

The Site Radiation Safety Officer (SRSO) will have the following minimum qualifications:

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.

2. Two years of supervisory experience in uranium mining/milling operations, UMTRA Projects, or NORM disposal operations where handling and/or disposal of low-activity or low-level radioactive materials are involved.

18.2.3 Health Physics Specialist

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to follow protocol and procedures, and maintain health physics schedules established by the CRSO.
4. Ability to work with contractor personnel and supervise radiation monitor(s) during operations.

18.2.4 Access Control Technician

1. Ability to learn and understand radiation safety principles and practices.
2. Ability to follow protocol and procedures, and maintain schedules established by the CRSO.
3. Ability to work with contractor personnel and oversee work areas, such as the unloading and wash down facilities.

18.2.5 Director of Operations

The Director of Operations will have the following minimum qualifications:

1. Civil Engineer, or other relevant engineering degree, with three years of experience in earth-moving construction projects
2. basically familiar with the principles of radiation safety, as applied to these types of projects.

18.2.6 Site Facility Manager

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; or two years of experience in the nuclear field.
2. Ability to learn and understand radiation safety principles and practices.

3. Ability to manage the operations at the site. To set schedules for personnel and complete assignments in a timely manner.
4. Ability to work with contractor personnel and supervise their work during operations.

18.2.7. Corporate Engineering Manager

The Corporate Engineering Manager will have the following minimum qualifications:

1. A Bachelor's degree in an engineering field

2. At least six years experience

3. Shall be a Utah certified professional engineer

18.2.8 Production Engineer

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; and one year of experience as a engineering technician or equivalent.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to follow protocol and procedures, and maintain construction operations and records as established by the Director of Operations
4. Ability to work with contractor personnel and supervise construction operations.

18.2.9 Site Engineer

1. Two years post-high school education with emphasis in sciences, engineering, and/or mathematics; and one year of experience as a engineering technician or equivalent.
2. Ability to learn and understand radiation safety principles and practices.
3. Ability to follow protocol and procedures, and maintain construction operations and records as established by the Director of Operations.
4. Ability to work with contractor personnel and supervise construction operations.

18.2.10 Construction Contractor

The construction contractor will be required to operate in accordance with the construction operation safety plan that includes, a comprehensive radiation safety/health physics plan. In addition, the construction contractor must demonstrate a willingness and commitment to comply with certain provisions, as outlined in Section 7, to which contractors may not normally be subjected:

1. Radiation monitoring of all construction personnel.
2. Decontamination and frisk-monitoring of personnel at access control portal.
3. Maintenance of Personnel in/out logs at access control.
4. Wearing protective clothing.
5. Decontamination of all vehicles and equipment prior to leaving the restricted area(s).
6. Making available to the Industrial Hygiene Consultant any requested records pertaining to employee exposure to occupational hazards, and to employee accidents.

18.2.11 Compliance and Permitting Manager

The Compliance and Permitting Manager will have the following minimum qualifications:

1. B.S. graduate in Engineering, Chemistry, Physics, or physical science-related field; and,
2. Supervisory experience in hazardous waste operations, where handling and/or disposal of hazardous materials are involved.

18.2.12 Corporate Quality Assurance Manager

The Corporate Quality Assurance Manager will have the following minimum qualifications:

1. Undergraduate technical degree, preferably in a science or engineering field, or a closely associated discipline, or equivalent technical experience.
2. For construction QA, the CQAM should have an understanding of materials testing methods for soil classification and compaction, of surveying methods for establishing the location of point coordinates and elevations, and of general construction techniques.
3. For laboratory QA, the CQAM should have an understanding of laboratory safety, methodology, and general chemistry concepts.
4. For health physics, the CQAM should have an understanding of industrial health and safety concerns, testing techniques, and ALARA concepts.

18.3 TRAINING PROGRAM

The training program for all contractor employees, Envirocare personnel and outside contractors/consultants is addressed in Section 17.5.6.3. All persons using or working with the radioactive material receive training which is commensurate with the materials he/she will be handling.

At the date of this submittal, Envirocare is current with the training requirements outlined in Section 17.5.6.3.

18.4 EMERGENCY PLANNING

The maximum credible accident at the Envirocare site would be the accidental dumping of a load at some location other than the disposal cell. The model used to calculate the permitted radionuclides in waste accepted at the site was designed to limit total occupational doses to 5 rem per year. If a load containing waste with the maximum permitted concentration was accidentally dumped, requiring its removal to the disposal cell, and if a full day is assumed for its removal, the maximum predicted dose to an employee would be 0.025 rem. Considering that most of the land within 10 miles of the site is under Bureau of Land Management (BLM) control and that there are no nearby residents, any dose received by a person outside of the controlled area would be a small fraction of 0.025 rem. Envirocare has an emergency response plan which is incorporated as part of the training procedures in Appendix C.

18.5 REVIEW AND AUDIT

The construction review and audit requirements are addressed in Section 14.1.4. The radiation safety audits are described in Section 14.7.

18.6 FACILITY ADMINISTRATIVE AND OPERATING PROCEDURES

18.6.1 Scope of work

At this time it is impossible to exactly state the amount of waste material to be handled or buried in a year. It is stated elsewhere in this application, that Envirocare anticipates approximately 500,000 tons per year. It is also impossible to estimate the time frame or schedule(s) for arrival of the material at the site.

18.6.2 Administrative Procedures

All personnel who work at the Envirocare facility will be required to abide by all site regulations and all requirements of this application. All violations of these requirements will be recorded on site violation forms and turned in to the Director of Operations. The implementation of this program will be under the direction of the Director of Operations

18.6.3 Operating Procedures

As described in the previous sections there are several people on the site who have the authority to terminate any activities on the site that are deemed to be unsafe, or need to be suspended until hazard-abatement measures have been performed. Examples of situations that would require that the site be closed until remediation of the problem would be:

1. Windy conditions which cause unsafe conditions.
2. Construction equipment operating in an unsafe condition.
3. Lack of trained personnel to operate the site.

18.6.4 Required Personnel

Envirocare will only perform specific operational activities when the trained personnel responsible for these activities are on site. For example, a Field Testing Inspector or equivalent must be on site whenever material is to be placed on a portion of the embankment that needs soil density verification.

Whenever the Clive facility is in full operation the SRSO or authorized designee must be present on site.

18.7 Safety and Environmental Review Panel

Envirocare will establish a "Safety and Environmental Review Panel (SERP)." The SERP shall consist of a minimum of three individuals. One member of the SERP shall have expertise in management and will be responsible for managerial and financial approval changes; one member shall have expertise in operations and/or construction and shall have expertise in implementation of any changes; and, one member shall be the Corporate Radiation Safety Officer or equivalent. Other members of the SERP may be utilized as appropriate, to address technical aspects, in areas, such as health physics, groundwater hydrology, surface water hydrology, specific earth sciences, and others. Temporary members, or permanent members, other than those specified above, may be consultants. The SERP shall convene at least monthly to review, evaluate and make determinations regarding the licensing requirements for the following actions, or address other matters pertaining to the SERP.

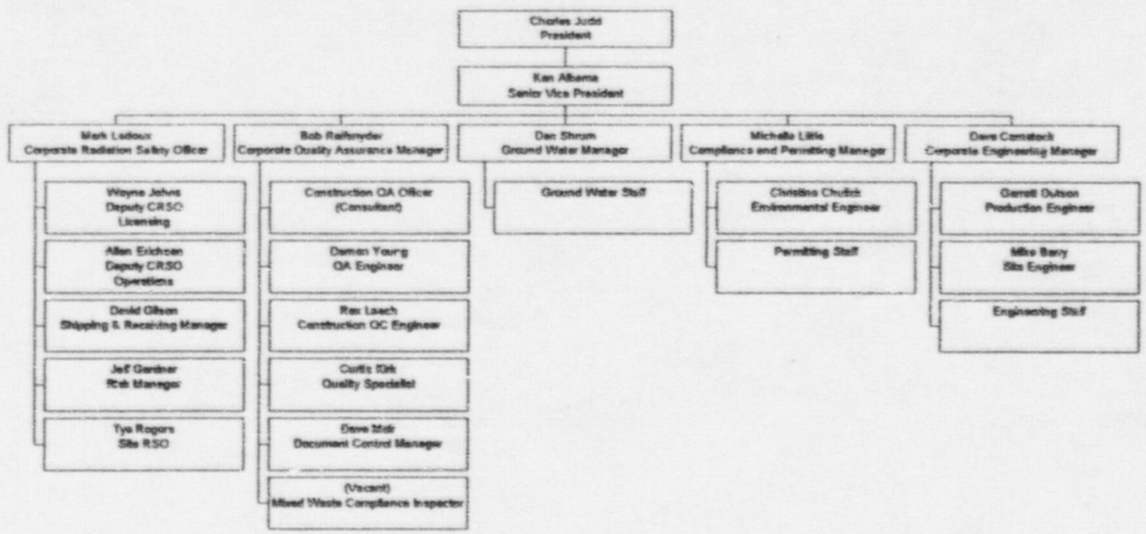
(1) Make changes in the facility or process, as presented in the application.
(2) Make changes in the procedures presented in the application.
(3) Conduct tests or experiments not presented in the application.

Envirocare will file an application for an amendment to the license, unless the following conditions are satisfied.

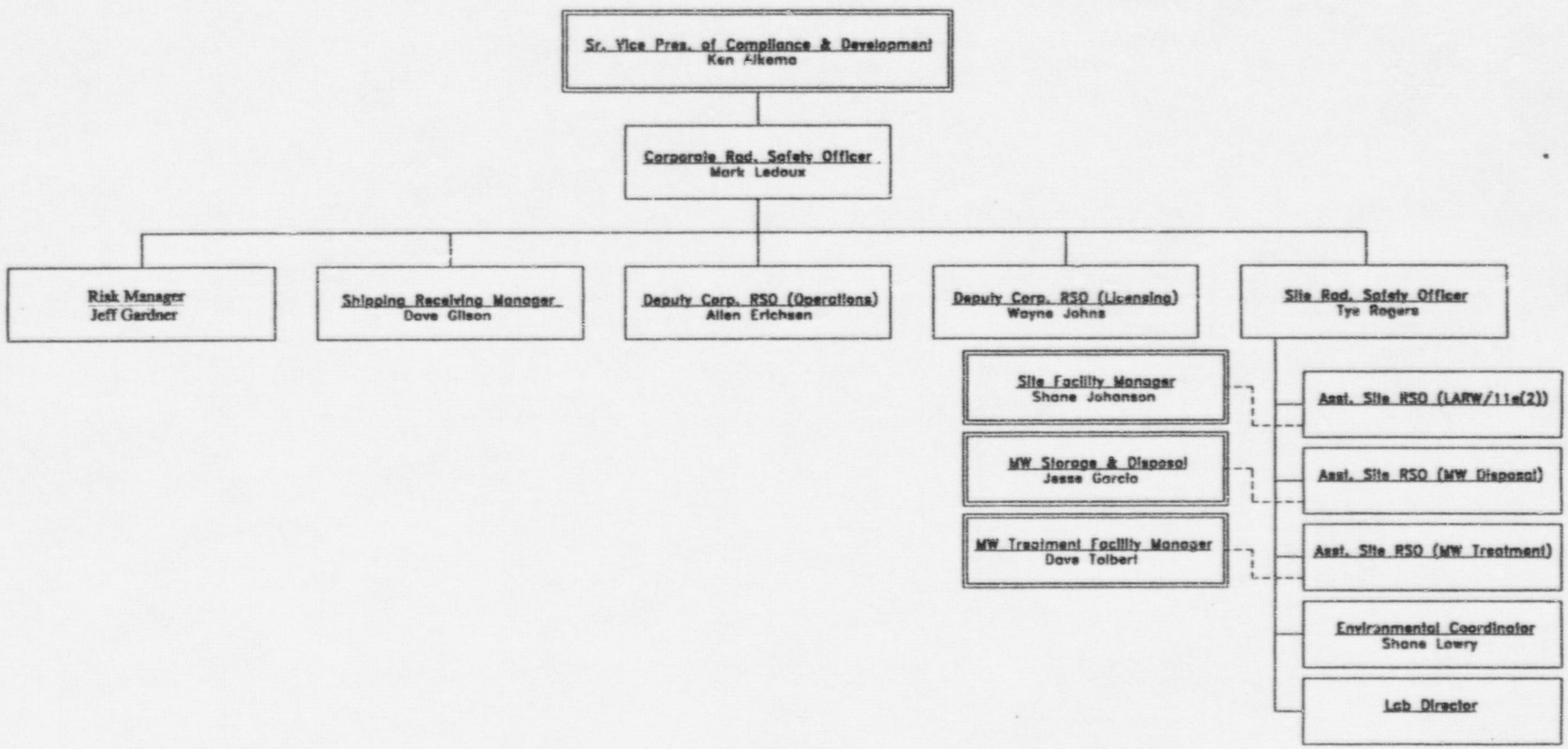
(1) The change, test or experiment does not conflict with any requirement specifically stated in this license (excluding the License Condition referencing the License Application or Reclamation Plan), or impair the licensee's ability to meet all applicable NRC regulations.
(2) There is no degradation in the essential safety or environmental commitments in the license application, or provided by the approved reclamation plan.
(3) The change, test, or experiment is consistent with the conclusions of actions analyzed and selected in the Final Environmental Impact Statement dated August 1993 (NUREG-1476).

Envirocare will maintain records of any changes made pursuant to this section. These records shall include written safety and environmental evaluation, made by the SERP, that provide the basis for the determination that the change is in compliance with the requirements referred to above. Envirocare will furnish, in the annual report to NRC, a description of such change, tests, or experiments, including a summary of the safety and environmental evaluation of each. Envirocare will annually submit changed pages to its license application to reflect changes made under this section.

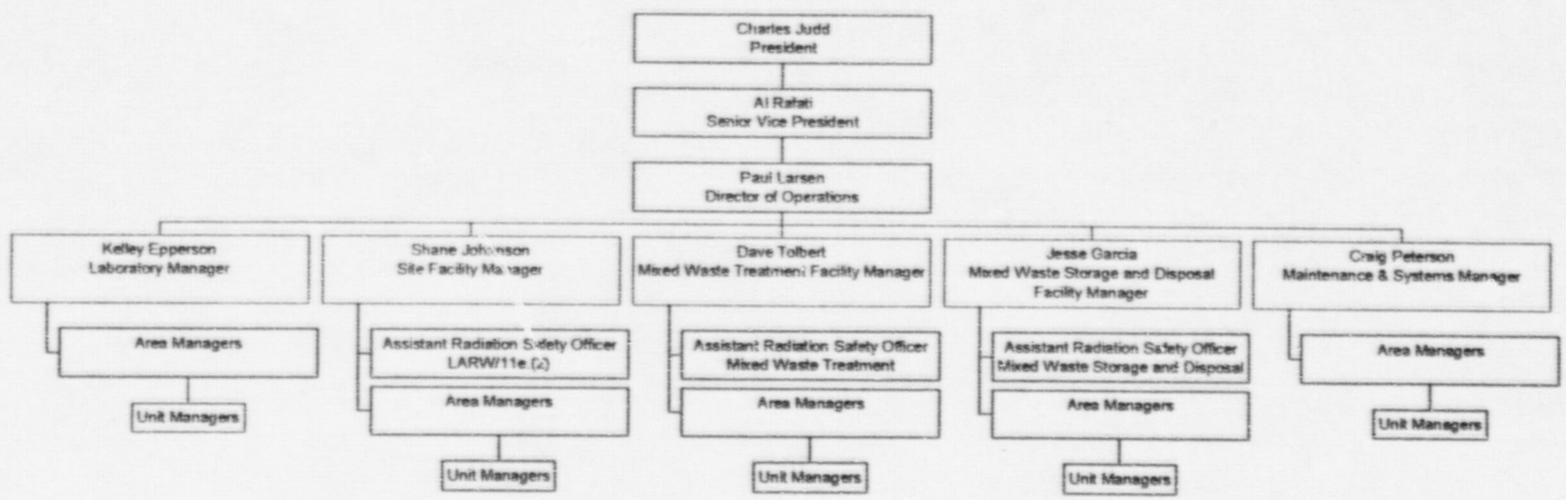
Corporate Development



Radiation Safety

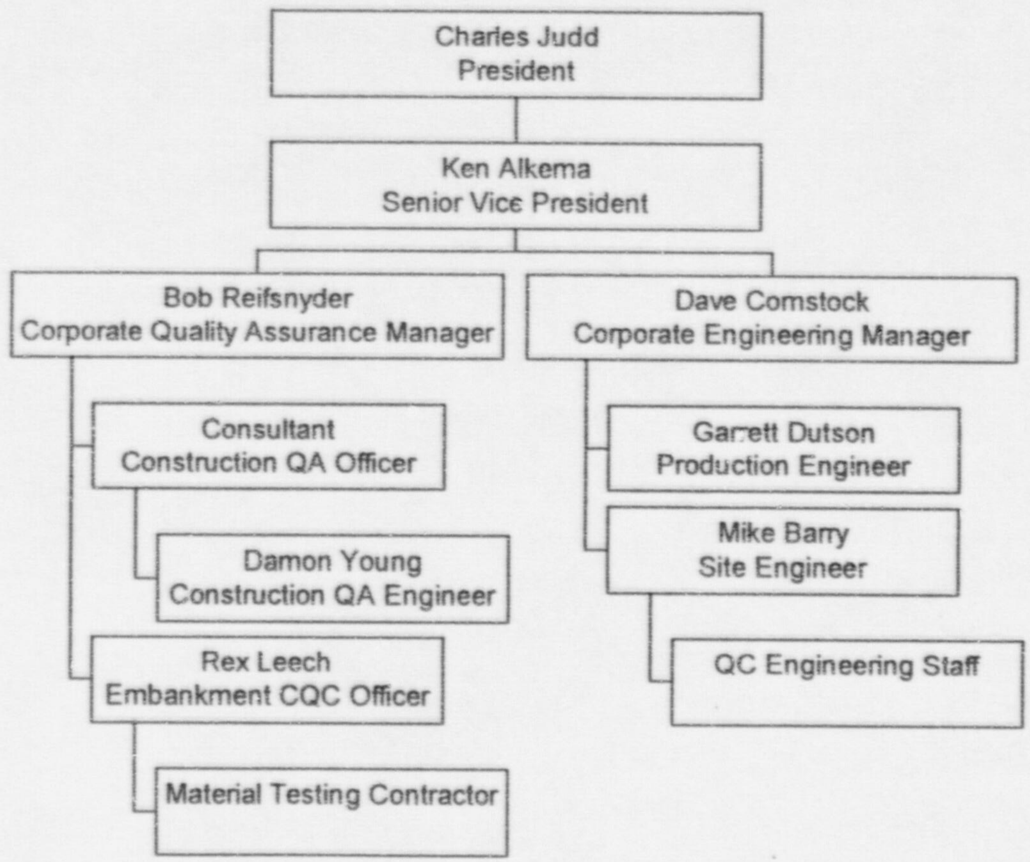


Operations

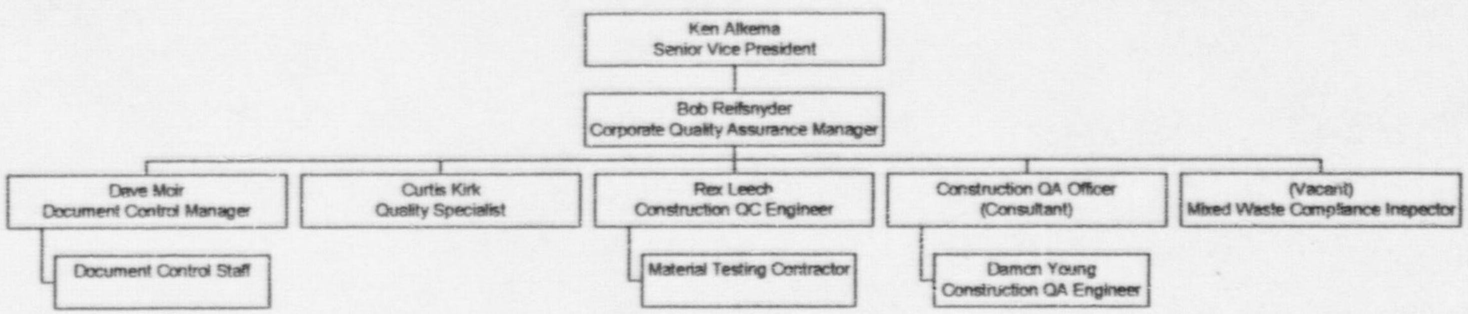


Charles Judd
President
Al Rafati
Senior Vice President
Paul Larsen
Director of Operations
Kelley Epperson
Laboratory Manager
Area Managers
Unit Managers
Shane Johnson
Site Facility Manager
Assistant Radiation Safety Officer
LARW/11e.(2)
Area Managers
Unit Managers
Dave Tolbert
Mixed Waste Treatment Facility Manager
Assistant Radiation Safety Officer
Mixed Waste Treatment
Area Managers
Unit Managers
Jesse Garcia
Mixed Waste Storage and Disposal
Facility Manager
Assistant Radiation Safety Officer
Mixed Waste Storage and Disposal
Area Managers
Unit Managers
Craig Peterson
Maintenance & Systems Manager
Area Managers
Unit Managers

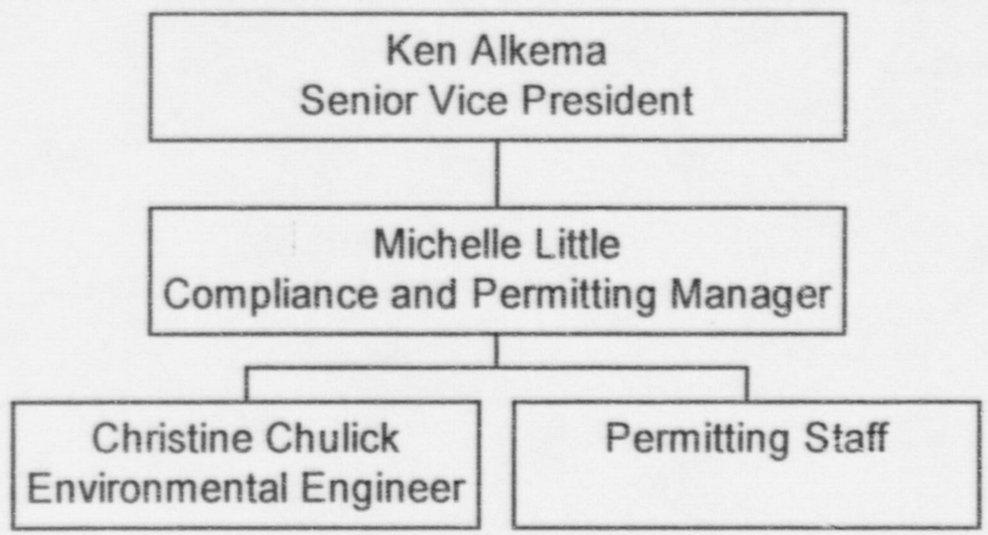
Construction Quality Assurance



Quality Assurance



Compliance and Permitting



SECTION 17. SAFETY ASSESSMENT

17.1 RELEASE OF RADIOACTIVITY

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

17.1.1 Characterization of Waste

17.1.1.1 Radionuclides

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

Summarizing the data presented above, the following radiological waste character is anticipated for the Envirocare 11e.(2) disposal site. Considering the relative proportions of lower and higher activity waste at the site, Envirocare estimates that the overall average concentration for any radionuclide will be approximately 500 pCi/g; however, individual sites may vary widely around that average, as described above. Because of this, individual shipments of wastes may contain higher average concentrations of Ra-226 and Th-232. In the context of waste deliveries to the disposal site a shipment is taken to mean a single waste-hauling truck or rail car from a single generator. Weighted average concentrations in a shipment must not exceed 4,000 pCi/g for natural uranium or any radionuclide in the Ra-226 series; 60,000 pCi/g of thorium-230; or 6,000 pCi/g for any radionuclide within the thorium series, although they may be present at those concentrations together.

A conservatively-high estimate of the volume of material to be handled and disposed of at the site would be one-half million (500,000) tons/year. Assuming an average Ra-226 and Th-232 concentration of 500 pCi/g, the estimated annual average total activity disposed of would be 227 Curies for each of the radionuclides. Since the daughter products may be assumed to be in secular equilibrium, there would be approximately 227 Curies of each of the other important radionuclides, such as Ra-228 and Ra-224. The amount of Uranium would be expected to be less than 25 percent that of Ra-226. The average Th-230 concentration is expected to be similar to that of Ra-226 and will depend upon the disequilibrium of the radionuclides in that decay series. The actual amount of radioactivity disposed of in a given year will vary around the estimated 227 curies per radionuclide as actual concentrations and disposal amounts vary.

17.1.1.2 Chemical Constituents in the Waste

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

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

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

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

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

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

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

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

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

17.1.2 Infiltration

Section 4.1.1 discusses principal design features to minimize water infiltration into the embankment and disposed materials. As indicated in that section, calculations in Appendix M demonstrate that the amount of precipitation that infiltrates into the

embankment and percolates to the shallow groundwater under proposed conditions is negligible.

17.1.3 Radionuclide Release - Normal Conditions

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

1. Release of interstitially trapped radon and thoron gas when handling bulk wastes.
2. Exhalation of radon gas from embankment area(s) that have not been covered with the compacted clay radon barrier.
3. Exhalation of radon gas from embankment area(s) that have been covered with the compacted clay radon barrier.
4. Exhalation of thoron gas from the top layer of embankment areas which have not been covered with a layer of non-thorium-containing waste or clean clay.
5. Localized resuspension of dust from waste handling operations.
6. Windblown materials from the embankment and unloading area.

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

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

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

17.1.3.1 Off-site Impacts from Normal Operations

M&A (Appendices A and A-1) provided estimates of projected radionuclide release rates and radiological impacts during site operations, assuming waste which exhibits the radiological characteristics estimated for the overall 11e.(2) profile (500,000 tons per year of waste containing 500 pCi/g of each of the radionuclides in the uranium and thorium series). While these Appendices demonstrate compliance with 10 CFR 20.1301 and 10 CFR 20.1302 under the assumed conditions, they do not completely serve the purpose of evaluating the variable characteristics of waste quantities and

radionuclide concentrations which are expected to occur annually, or over shorter periods of time. M&A performed a sensitivity analysis of Envirocare's waste management procedures and waste characteristics (Appendix A-2). This analysis permits each waste handling procedure, from receipt to final closure, to be evaluated for its environmental impact while handling any quantity of wastes at any specified radioactivity concentration. Output from the analysis of Appendix A-2 will be used as input to the calculational spreadsheet described in Appendix A-3 to provide guidance to Envirocare planners in scheduling waste shipments and planning waste handling operations to meet the effluent concentration limits of Table 2, Appendix B to 10 CFR 20.1001 - 20.2401. The application of Appendices A-2 and A-3 to waste management will allow Envirocare to manage wastes within an envelope of quantities and radioactivity characteristics during the year while meeting the overall environmental results of Appendices A and A-1.

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

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

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

The dose calculations above, from Appendices A and A-1, were based on a single assumed average concentration in waste with an annual total of 500,000 tons of waste disposed, or an annual disposal of 227 Ci of each of the radionuclides in the uranium and thorium series. Occupational and environmental doses are shown to be almost completely dependent upon the total amount of radioactivity managed. While the use of Appendices A-2 and A-3 provide considerable flexibility in waste management, the reliance upon the modelling of Appendices A and A-1 will assure that occupational and environmental 1impacts are as described in those appendices. With this option, Envirocare can safely dispose of any combination of radioactivity concentrations up the shipment limits of 4,000 pCi/g for natural uranium and any radionuclide in the ^{226}Ra series; 60,000 pCi/g of thorium-230; and 6,000 pCi/g for any radionuclide in the thorium series. Application of this approach would

automatically restrict the amount of waste which could be received at higher concentrations.

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

17.1.3.2 Occupational Radiation Exposures

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

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

$${}_bD = 0.23\, E_b\, c$$

where: ${}_bD$ = Dose rate from an infinite cloud (rad/s)
E_b = Average beta energy per disintegration (MeV/dis)
c = Concentration of the beta emitting isotope in the cloud (Ci/m^3)

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

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

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

periods, Envirocare will closely monitor internal and external exposures to maintain TEDE as low as reasonably achievable and, in all cases, below the standards of 10 CFR 20.1201.

17.1.4 Radionuclide Release - Accidents or Unusual Operation Conditions

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

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

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

1. on-site truck turnover or collision
2. train derailment
3. flooding
4. tornado.

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

occurred into the Great Salt Lake. Impacts of these events were found to be negligible compared to the impacts from normal operations.

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

The following accidental on-site releases have been evaluated:

On-site truck turnover or collision

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

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

> 1.3 x 10^{-6}/km x 53 loads/day x 260 days/year x 1 km/load
> = 1.8 x 10^{-2} or about 1.8%.

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

For a theoretical truck accident involving a yellowcake shipment, a 24-hour release period, all particles in the respirable range, and a population density of 7.5 persons per square mile, NRC estimated 50 year dose commitments to the lungs of the general public in the range of 0.7 to 9 person-rem. The yellowcake specific activity is about 6.77 x 10^5 pCi/g while the average uranium or thorium concentrations expected at Envirocare would be 500 pCi/g, or a factor of 1300 lower. Individual shipments to Envirocare might have ^{226}Ra concentrations as high as 4,000 pCi/g, or similar to those found in uranium mill tailings. Concentrations of ^{232}Th in a small

fraction of shipments could be as high as 6,000 pCi/g. The dose per unit intake via inhalation is higher for Th-232 wastes than for yellowcake by up to a factor of 1000, depending upon the chemical form and radionuclide mix. Therefore, the postulated off-site public doses could be approximately an order of magnitude higher than for a yellowcake spill under the same circumstances. However, the population distribution around the Clive site is insignificant compared to that in the NUREG calculation and, therefore, the off-site population dose would be inconsequential.

For on-site workers, there would be a very short exposure time since there would be no reason to stand downwind for 24 hours (or even one hour). Assuming an accident involving the spill of a load of waste with a concentration of 15,000 pCi/g; a period of three hours for cleanup with no use of respiratory protection; an airborne concentration of 1 mg/m^3; and a respiratory rate of 1.2 m^3/h a total of 54 pCi of each nuclide would be inhaled. Comparing these to the ALI's from Appendix B of 10 CFR 20.1.001 - 4201, the sum of fractions is 0.022. The external gamma dose, using the relationship of 3.1 mrem/h/pCi/g for Ra-226 from Appendix A Section 3.7.3 and doubling for the contribution from Ra-228, would be less than 140 mrem. Such a dose added to the projected maximum TEDE of 1,032 mrem/y would still be well within the permissible annual exposures for radiation workers. In actual fact, no workers would be present under such conditions without respiratory protection and would not be standing on the spilled waste for more than a few minutes.

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

Train derailment:

The probability of a train derailment occurring on the Clive site is not readily calculable. However, because of the short length of track involved, the small amount of train movement, the low train speeds compared to truck speeds, and the relatively small number of cars compared to truck shipments, the probability of a derailment should be much less than the probability of a truck accident.

The dose to the workers and to the population should be much less than that for an off-site derailment and spillage event since trained workers and equipment would be available to immediately use dust control measures to control releases and cleanup the spill. The DOE, as discussed above, concluded that the dose to cleanup workers and nearby residents from such an off-site spill was insignificant. As a worst case,

the same assumptions could be applied as for the truck accident scenario above, with the same low total dose to emergency response teams.

Flooding:

Flood control features for both the Vitro and Clive sites have been designed and constructed to prevent erosion or off-site transport of wastes from the sites by overland flooding. Details of the flood control features are provided in Appendix F. No off-site transport of radioactive waste by flooding is anticipated. Cleanup of contamination caused by dispersion of stored or already disposed waste within the controlled area by flooding would replace placement of waste as an activity and radiation doses to workers would be the same as, or lower than, those received during normal operations.

Tornado:

From NUREG-0706 the probability of tornado occurrence in Utah is probably in the range of 1 to 5 x 10^{-4}. NUREG-0706 also estimates the consequences of a tornado striking a model uranium mill. In this case about 12.6 tons of yellowcake is entrained in the vortex, the vortex dissipates at the site boundary, all of the yellowcake is respirable in size, and the cloud is dispersed as a volume source by the prevailing winds. Settling velocity is negligible. The model predicts a maximum exposure at 2.5 miles from the mill, where the 50 year dose commitment is estimated to be 0.83 micro-rem. At the fence line (1600 feet) the dose is estimated to be 0.22 micro-rem. Our wastes would have average activities considerably less than this but as discussed above, the TEDE per unit intake is higher, resulting in comparable doses at receptor locations. Since there are no nearby population groups, the significance of this very small potential dose is even more insignificant.

Severe Winds

In the preceding discussion of airborne exposures resulting from tornadoes it was concluded that the maximum 50-year dose commitment at 2.5 miles would be less than 1 micro-rem. That conclusion is derived from a NUREG-0706 analysis of tornado-dispersed yellowcake from a uranium mill and is considered of a comparable magnitude to the transport of Th-232 waste from the Clive Site under similar conditions.

While severe winds on the order of 35 m/s have been recorded in the vicinity, the occurrence is infrequent and the duration is short. Assuming an order of magnitude increase in airborne concentrations during severe wind conditions which occur approximately one percent of the time, the time-weighted average off-site exposure would increase by only 10 percent. This would result in a maximum additional

annual collective TEDE of less than 1 mrem to current nearby population groups (See Table 3.20, revised, Appendix A-1)

17.1.4.1 Transfer Mechanism - Groundwater

The possibility of contamination releases to known water resources is highly unlikely. Without extensive treatment, use of the water in the South Clive area would appear to be confined to very limited industrial uses. There is minimal potential for degradation of water quality in the vicinity of the south Clive site inasmuch as the water at the site has been characterized as a brine, with levels of many constituents often exceeding EPA primary or secondary drinking water standards by a large amounts.

Envirocare has commissioned a hydrogeologic study to more accurately describe the possibility of groundwater contamination. This report by Bingham Environmental (Appendix GG) concluded that it would take 400 to 600 years for leachate to travel through the unsaturated zone and then another 800 years to reach the nearest off-site well. No non-radiological constituent would reach the Ground water in less than 700 years.on site would be 191 years. Using this estimate, it would take well over 1,000 years for any groundwater from the 11e.(2) cell to reach the boundary of the Envirocare facility.

17.1.4.2 Transfer Mechanism - Air

Because of the location of the South Clive facility, the meteorological characteristics of the area, and the lack of population within 20 miles of the facility, the impact of air as a transfer mechanism for radioactivity is limited. The modelling study conducted by Momeni & Associates (Appendix A) concluded that the annual population TEDE (exclusive of doses to workers at the nearby hazardous waste operations) after 16 years of operation would be 0.016 person-rem/year. Calculated TEDE to the nearby hazardous waste workers would add approximately 0.5 person-rem/year.

17.1.4.3 Transfer Mechanism - Surface Water

The probability of contamination through surface water is highly unlikely inasmuch as there are no surface waters at the site. As is stated previously, "No surface-water bodies are present on the South Clive site. The nearest stream channel ends about 2 miles east of the site and is typical of all the drainage along the transportation corridors within about 20 miles of the South Clive site. Stream flows from higher elevations usually evaporate and infiltrate into the ground before reaching lower, flatter land. The stream channels are well defined in their upper reaches, but as they

approach the flatland the size of the channel reduces until there is no evidence of a stream."

17.1.4.4 Other Transfer Mechanisms

Because of the location of the South Clive facility, the sparse biota in the area, and the lack of population within 20 miles of the facility, the impacts of other transfer mechanisms such as gamma radiation through air and transfer of radioactivity through biotic pathways are very small.

17.1.5 Radionuclide Transport

The most significant radioactivity transport mechanisms are air, groundwater, surface water, direct radiation and biotic pathways. The five periods of principal concern to NRC (NUREG-1199) include the operational, closure, observational and surveillance, active institutional control, and passive institutional control periods. In reality, the periods of real concern should be operational and post-closure.

During the closure period one would not ordinarily expect continuing shipments of waste so exposures from air, surface water, direct radiation, and biotic pathways should be less than exposures received during the operational period. No new wastes are being received, old wastes are being covered, and the surface is being decontaminated.

During the observational and surveillance, active institutional control, and passive institutional control periods the site has already been decontaminated, wastes are covered and there should again be no changes in exposures.

The evaluations of Appendix A address exposure pathways for operational periods and were compared to regulatory standards. Results were used to determine potential exposures to on-and off-site personnel. As discussed in Sections 17.1.3.2 and 17.1.4.2, projected doses to on-site radiation workers are 1 rem/year or less and the annual regional population TEDE to off-site residents and nearby industrial workers is approximately 0.5 rem.

17.1.6 Assessment of Impacts and Regulatory Compliance

The M&A report addresses the specific impacts of releases under normal operating conditions. Release mechanisms were evaluated, exposures to workers and the public assessed, and the results compared to applicable standards and regulations. It was concluded that with the proposed waste characteristics and operating procedures, exposures to the workers and the public will be within acceptable limits

and the design will limit the radon flux to 20 pCi/m^2s as proposed in 10 CFR Part 40, Appendix A.

While the exposures to site custodial personnel during the active institutional control period were not specifically evaluated, all waste will have been covered, gamma exposure rates will be near background, and radon emission rates will be limited to the design criterion of 20 pCi/m^2s. There is no reason to believe that exposures during this period will be more than a small fraction of those to the workers during operations.

For a discussion of impacts of releases due to accidents or unusual operating conditions see Section 17.1.4. In general, because of the relatively low radionuclide concentrations of the Clive wastes, it is difficult to postulate an on-site accident that could cause significant exposures to on- or off-site personnel.

17.2 LONG-TERM STABILITY

The embankment design will provide long-term stability and be relatively maintenance-free after site closure. Long-term stability is discussed in detail in Sections 4 and 6.

17.3 CONSTRUCTION SAFETY

Envirocare has implemented a construction safety plan which covers both Envirocare and contractor employees. While the prime contractor is responsible for developing his own safety and health plan, Envirocare performs safety inspections of the contractor's on-site operations to assure compliance with UOSHA and Envirocare regulations. The content of the plan includes:

1. Purpose/Goals - Envirocare and Contractor commit to the following goals:
    a. Safe and health working conditions for all on-site personnel.
    b. Protection of the general public.
    c. Compliance with all governmental safety and health regulations.
    d. Reduce liability to Envirocare and contractor to a minimum.
2. Establish an organizational chart to define responsibilities for safety and health program direction and enforcement.
3. Emergency Medical Care.
4 Pre-planning for unusual occurrences.
5. Safety & Health training program.
6. Control and monitoring of Safety and Health Plan.
7. Industrial Hygiene plan.

8. Corporate Safety Program.

Any contractor that performs work for Envirocare on this project must formally take responsibility to obey all site rules. Any contractor who is to work for an extended period of time at the site must submit their own Health and Safety Program.
All OSHA regulations will be under the jurisdiction of UOSHA. The Corporate RSO is responsible for overall development, direction and coordination of the Safety and Health Plan. The Site Manager is responsible for on-site implementation and enforcement of all safety and health provisions. It is recognized that industrial accidents pose a greater risk to employees than radiation risks and a significant effort is made to ensure a safe workplace. Employees are instructed to bring all health and safety concerns to their supervisor or the Site Manager. Unresolved concerns may be brought to the attention of UOSHA for immediate reconciliation.

The Safety and Health Plan relies on identification of risks, development of procedures to control those risks and to comply with UOSHA regulations, pre-employment safety training, continuing on-the-job safety training and on-going safety inspections of all operations. Radiation Technicians (Health Physics Specialist II), who are already trained in radiation safety, are also given responsibility to enforce all safety regulations.

17.4 Radiation Safety and Health Physics

17.4.1 Radiation Protection Policy

It is the policy of Envirocare, to maintain personnel/occupational radiation exposures as low as reasonably achievable (ALARA). Because of the nature of the 11e.(2) wastes, experience has shown that radiation exposures are normally low and Envirocare is committed to continuing to minimize exposures to the workers and the environment.

The average annual dose for 294 workers involved in the Vitro Remedial Action Project during 1986 was 50 mrem, with maximum exposures of 250 mrem. This maximum value is only 5% of the radiation dose standard of 10 CFR 20.101. Envirocare's experience with handling similar materials at it's LLRW facility was even better in that the highest total dose received during any year of Envirocare's five years (1988-1992) of operation was 200 mrem and the average annual dose equivalent was less than 50 mrem. The data are presented in Table 17.7.

In keeping with the ALARA principle, any reported personnel exposures in excess of 50 mrem/month will be investigated and documented by the Corporate Radiation Safety Officer(CRSO).

Procedures and methods to keep internal exposures ALARA include:

a. Dust suppression on all operational roads by application of magnesium chloride or watering at 2-hour intervals.
b. Speed limit of 35 mph on roads treated by dust suppressants; 10 mph on infrequently used roads.
c. Stopping operations in high wind conditions (all operations cease at winds of 40 mph; radiation safety personnel have authority to stop operations at lower wind speeds if dusting or other safety considerations warrant).
d. Placement of radon barrier on portions of the cell as they are completed.
e. Weekly area radiation surveys with investigation of increasing levels to determine the cause.
f. Requiring workers to wear respirations in areas of potential high dust concentrations, for example, the rollover and selected heavy equipment operations.
g. Pre-planning tasks which have the potential for higher than normal exposures to limit exposures through efficient use of time and handling procedures.

The Site Radiation Safety Officer (SRSO) will have the day-to-day responsibility for maintaining occupational and environmental radiation exposures ALARA, consulting such guidance documents as NRC Regulatory Guide 8.31, "Information Relevant to Ensuring that Occupational Radiation Exposures at Uranium Mills Will Be As Low As Reasonably Achievable" and draft Guide DG-8013, "ALARA Levels for Effluents from Materials Facilities." The SRSO will document ALARA activities including:

a. Reviews of new proposed disposal contracts to assure that Envirocare's procedures, facilities, and equipment are appropriate and sufficient to limit exposures to workers and the environment;
b. Monthly reviews of work area, perimeter, and environmental air monitoring results noting trends and adjusting work procedures when practical to further reduce potential exposures; and
c. Monthly reviews of work area gamma-ray exposure rates and advising the Site Manager(SM) on operational changes that will reduce exposures to ALARA levels.

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

17.4.2 Restricted and Controlled Areas

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

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

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

17.4.3 Radiation Dose Limits

17.4.3.1 Occupational Dose Limits for Adults

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

a. Annual limit will be the more limiting of:

1. The total effective dose equivalent (TEDE) equal to 5 rems; or
2. The sum of the deep-dose equivalent (DDE) and the committed dose equivalent (CDE) to any individual organ or tissue other than the lens of the eye being equal to 50 rems.

b. The annual limits to the lens of the eye, to the skin, and to the extremities, are:

1. An eye dose equivalent of 15 rems; and
2. A shallow dose equivalent of 50 rems to the skin or to any extremity.

c. Doses received in excess of the annual limits must be subtracted from the limits for planned special exposures that an individual may receive during the current year and during an individual's lifetime.

d. For soluble uranium, the intake by any individual is limited to 10 milligrams in any week in consideration of chemical toxicity.

17.4.3.2 Occupational Dose Limits to Minors

The annual occupational dose limits for minors are 10 percent of the annual dose limits specified for adults. The Site Radiation Safety Officer (SRSO) will review any work assignment given to minors to assure that exposures are maintained ALARA and within this guidance.

17.4.3.3 Dose Limit to an Embryo/Fetus

The dose equivalent to the embryo/fetus will be limited to 0.5 rem during the entire pregnancy in accordance with 10 CFR 20.1208. Envirocare's policy is to inform female workers of the regulations regarding protection of the embryo/fetus and to ask them to inform Envirocare in writing, upon discovery or suspicion of a pregnancy. The Corporate Radiation Safety Officer (CRSO) will review the work assignments and normally offer the woman the opportunity to take available positions in non-radiation areas for the duration of the pregnancy. If no positions are available, the CRSO will counsel the individual to assure an understanding by the individual of the additional risks of continued employment. If the woman continues to work in the radiation area, the SRSO will monitor the work assignments and activities to assure that the TEDE to the embryo/fetus is ALARA and limited to 0.5 rem.

17.4.3.4 Planned Special Exposures

Envirocare does not anticipate authorizing planned special exposures since the radiation levels and radioactive constituent concentrations in 11e.(2) byproduct material are low. In the event that circumstances warrant a planned special exposure, Envirocare will do so in full compliance with the guidance in 10 CFR 20.1206.

17.4.3.5 Summation of Occupational Internal and External Doses

Guidance for the summation of the internal and external dose equivalents are specified in 10 CFR 20.1202. Summation is not required if either the external or internal radiation exposures are not likely to exceed 10 percent of the limit. This includes occupational exposures to adults as well as minors and to the embryo/fetus.

It is unlikely that exposures to workers at the Envirocare facility will exceed 10 percent of the allowable limits for direct radiation as well as internal radiation. Data for the UMTRA Project disposal at Clive show that the average annual dose equivalent from direct radiation was 50 mrem, with a maximum individual dose equivalent of 250 mrem. Envirocare has been operating the LARW facility beginning in 1988. The maximum individual dose equivalent from 1988-1992 was 200 mrem. Similarly the lapel sample and work area monitoring results indicate that the airborne particulate concentrations are near background levels.

Should Envirocare find that summation of occupational internal and external doses is necessary, the following method will be employed:

a. Should the internal dose as determined by air monitoring results, bioassay, or other means - as well as the dose from external sources as determined by radiation dosimeters - likely exceed 10 percent of the allowable limits, the Committed Effective Dose Equivalent (CEDE) will be added to the Deep Dose Equivalent (DDE) and compared with the Total Effective Dose Equivalent (TEDE) limit of 5 rem for adults and 0.5 rem for minors and the fetus/embryo.

b. If the only intake of radionuclides is by inhalation, the procedure specified in 10 CFR 20.1202(b) may be applied. The TEDE limit will not be exceeded, according to this procedure, if the sum of the DDE divided by the TEDE limit and one of the following, does not exceed unity:

   1. The sum of the fractions of the inhalation ALI for each radionuclide, or
   2. The total number of derived air concentration-hours (DAC-hours) for all radionuclides divided by 2,000, or
   3. 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 of 50 rem.

c. If the intake by oral ingestion exceeds 10 per cent of the oral ALI, Envirocare will account for this intake and include it in demonstrating compliance with the limits.

d. If intake occurs via wounds or skin absorption, Envirocare will evaluate these intakes and include these in the calculation of the TEDE.

17.4.3.6 Determination of Prior Occupational Dose

If any employee is anticipated to receive an occupational dose in excess of 10 percent of the limits presented in this Section, Envirocare will determine the previous radiation exposure for use in limiting the annual dose equivalent to the allowable limits and for planning special exposures.

Determination of prior occupational exposures will be done by

1. Obtaining a written signed statement from the employee or his most immediate employer, that discloses the nature and the amount of any occupational dose that the individual may have received during the current year; and
2. Obtaining or attempting to obtain from the employee's most recent employer, a written signed statement in the form of an NRC Form 4 or an equivalent form, showing the life-time occupational exposure history. In case this cannot be done, the guidance in 10 CFR 20.2104 will be followed.

17.4.3.7 Radiation Dose Limits for Individual Members of the Public

Operations will be conducted such that the additional dose equivalent to individual members of the public will be limited in accordance with the limits of 10 CFR 20.1301, 10 CFR 61, and 10 CFR 40, Appendix A. The limits are:

a. The total effective dose equivalent to individual members of the public from the licensed operation will not exceed 25 mrem per year above natural background levels, radon and radon daughters excepted.
b. Radon and radon daughters will be limited to levels specified in Table 2 of 10 CFR [20.1001-20.2401], Appendix B.
c. The total effective dose equivalent limit to occupants in the controlled area (other than restricted areas) will not exceed 100 mrem per year above background levels.
d. The dose equivalent in any unrestricted area from external sources will not exceed 0.002 rem in any one hour.

Table 3.12, revised, Appendix A-1, shows the calculated concentrations of particulate radioactivity at the site boundaries. The projected concentrations are in the range of ambient background concentrations and are well below the concentration limits of Appendix B to 10 CFR 20.1001-20.2041. Airborne particulate monitoring will be performed to confirm those predictions.

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

17.4.4 Internal Radiation Dose Assessment

17.4.4.1 Calculation of Internal Radiation Exposure from Inhalation

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

DAC-hours = (C/DAC) x t

where:
C = airborne concentration of radionuclides in mCi/ml
DAC = Derived Air Concentration in mCi/ml
t = time of exposure in hours

The total exposure is equal to the sum of such calculations for all radionuclides present.

17.4.4.2 Calculation of Internal Dose from Inhalation

In order to assure compliance with the occupational dose limit, the committed dose equivalent (CDE) to any organ and the Committed Effective Dose Equivalent (CEDE) will be calculated for each radionuclide as follows:

Committed Dose Equivalent (mrem) = C x t x R x f_{CDE} / PF

where
C = concentration in mCi/ml
t = exposure time in hours
R = inhalation rate, 1.2 E+06 ml/h
f_{CDE} = exposure to dose conversion factor, in mrem/mCi, for the maximally exposed organ
PF = respirator protection factor as given in Appendix A to 10 CFR 20.1001-20.2401

The total committed dose equivalent for any organ is obtained by summing the contribution from each radionuclide of significance. Since the physical and chemical form of the radionuclides will normally not be characterized,

the exposure to dose conversion factor for the most restrictive lung clearance class (Day, Week, Year) for the maximally exposed organ will be used. Using Table 2.1 of ICRP Publication 30 or Table 13.24.2 in The Health Physics and Radiological Health Handbook (Revised Edition, Scinta, Inc), it is apparent that the dose to the endosteal lining of the bone from the thorium in 11e.(2) material is dominant for most lung clearance classes. For 11e.(2) material having high concentrations of insoluble uranium, it may, however, be possible that a combination of radionuclides could result in a larger dose to the lung. Therefore, data on which to calculate the organ doses is included below and unless the specific chemical form and lung class is known, the calculations will be made for both organs to assure that the 50 rem CDE limit has not been exceeded.

Choosing the listed lung clearance classes for maximizing the dose to the endosteum, the following CDE to the endosteum per unit intake will be used.

Radionuclide	Class	f_{CDE} for ENDOSTEUM (mrem/mCi)
U-nat	D	3.82E+4
U-234	D	4.03E+4
U-235	D	3.74E+4
U-238	D	3.62E+4
Th-230	W	7.99E+6
Th-232	W	4.11E+7
Ra-226	W	2.81E+4
Ra-228	W	2.41E+4
Pb-210	D	2.02E+5
Po-210	D	1.49E+3

Choosing the lung classes for maximizing the dose to the lungs for each radionuclide of interest, the following f_{CDE} will be applied.

Radionuclide	Class	f_{CDE} for lung (mrem/mCi)
U-nat	Y	1.00E+6
U-234	Y	1.11E+6
U-235	Y	1.04E+6
U-238	Y	1.00E+6
Th-230	Y	1.11E+6
Th-232	Y	3.48E+6
Ra-226	W	5.96E+4
Ra-228	W	2.67E+4
Pb-210	D	1.18E+3
Po-210	W	4.81E+4

The formula for the CEDE is similar to the above with the exception that "f" is the exposure to committed effective dose equivalent conversion factor, f_{CEDE} in mrem/mCi. It is chosen for the lung clearance class that maximizes the CEDE for each radionuclide. It is also based on NRC recommended Organ Dose Weighting Factors rather than the factors in ICRP Publication 26.

Again taking the values from the Health Physics and Radiological Health Handbook, the following data from Table 13.24.2 will result in maximizing the calculated CEDE:

Radionuclide	Class	f_{CEDE} IN (mrem/mCi)
U-nat	Y	1.25E+5
U-234	Y	1.32E+5
U-235	Y	1.23E+5
U-238	Y	1.18E+5
Th-230	Y	2.62E+5
Th-232	Y	1.15E+6
Ra-226	W	8.58E+3
Ra-228	W	4.77E+3
Pb-210	D	1.36E+4
Po-210	D	8.58E+3
Rn-220*		
Rn-222*		

*The internal dose from Radon-220 and Rn-222 for occupational workers will be calculated for occupational exposures using the relationship that either the ALI for radon or the WLM limits for radon daughters is equivalent to a TCEDE of 5 rem (see 10 CFR [20.1001-20.240-1], App B).

In order to determine which of the two limits (TEDE of 5 rem/year or sum of the deep-dose equivalent and the CDE to any organ of 50 rem) are the most restrictive for the particular mix of radionuclides, the TEDE and the CDE to the maximally exposed organs are calculated as described above. The DDE is added to the CDE and compared to the 50 rem organ dose limit; the TEDE is compared to the 5 rem annual limit. The calculations will be made for all employees according to the requirements in 10 CFR 20.1202.

The radionuclide mix will either be determined by estimating the volume-weighted radionuclide mix using waste characterization data or by a laboratory analysis of composites of work area or personnel monitor air filters.

The tables above normally use the maximum dose equivalent per unit intake. When uranium tailings are being handled, dose equivalent values for the

lung clearance class "D" will be used for uranium and "Y" for thorium since this is standard practice based on industry data. Other modifications to the parameter values will be made when the information is available.

17.4.4.3 Calculation of Internal Dose from Oral Ingestion

The ingestion of radionuclides at the Envirocare site is controlled primarily by restricting eating and drinking to monitored clean areas. In addition, the use of respiratory protection in the most highly contaminated areas minimizes the potential for contaminating the face and transfer of material from other parts of the body to the mouth.

While it is unlikely, the internal dose will be calculated and included in the employees total dose assessment should Envirocare be made aware of such occurrence. An assessment of the radionuclide intake will be made and the respective Committed Dose Equivalent per Unit Intake via Ingestion factor will be use to calculate the CDE and CEDE (see ICRP Tables or Table 13.24.21 in the Health Physics and Radiological Health Handbook).

17.4.4.4 Calculation of Internal Dose from Intake through Wounds or Skin Absorption

Employees at the Envirocare site are normally protected from intake through wounds and skin absorption by wearing protective clothing. Should an accident result in an open wound, the CRSO or Site RSO will inform the attending physician of the fact for his guidance in effecting removal or reduction of the amount of radioactive material remaining in the wound. The CRSO will perform an investigation and estimate the intake using data from wound monitoring or other available information.

The CDE to any organ will be estimated using methods similar to those used in NCRP Report 111, Developing Radiation Emergency Plans for Academic, Medical or Industrial Facilities. August, 1991. Table 4.2 provides values of maximum committed dose equivalent to any organ for adults per unit intake. These were derived by taking the ICRP Publication 30 values for ingestion and dividing by the gut transfer factor f_1. Envirocare will use a similar approach by estimating the radionuclide mixture and intake for each radionuclide, and calculating the CDE to each organ using appropriate f_1 values and CDE per unit intake for each radionuclide of significance via the ingestion pathway.

Calculated CDE's will be compared to the standards of 10 CFR 20.1201. Additional efforts at reducing dose will be based on total CDE and the potential for reducing the CDE through available means.

17.4.4.5 Bioassay

All permanent employees working at the site will be required to participate in a urine bioassay program to assist in evaluating internal deposition of radionuclides. A baseline urine sample will be collected upon employment and annually thereafter. Samples will be routinely analyzed for gross beta (minus K-40), Ra-226, isotopic thorium, and total uranium.

An increase above baseline levels equal to three standard deviations of the baseline values for the entrance bioassay will trigger an investigation of the work activities, an increased frequency of sampling, and a more detailed analysis to estimate the intake and resultant dose equivalent.

For those personnel working directly with the waste, a quarterly sampling program will be instituted. At this time it is anticipated that most waste will be similar in physical and chemical composition to uranium mill tailings. A urine bioassay action level of 1.5 mg/l has been derived for natural uranium (U-nat) at which time better controls on intake must be instituted. This action level was derived (see below) assuming chronic exposure to airborne tailings where the quarterly intake is equal to ten percent of the TEDE. A similar derivation for other radiological mixes may be required and a different action level used when large quantities of other 11e.(2) materials are being handled.

Based on experience at Envirocare's NORM disposal facility, it is unlikely that any employee's bioassay results will be above the action level. If any result does exceed the action level, the causes for such a level will be investigated and steps will be taken to reduce the employee's future exposure to inhaled or ingested radioactive materials.
A special bioassay sampling will be done for all personnel involved in an incident determined by the CRSO as having a potential for a significant intake of radionuclides. Twenty-four hour fecal and urine samples will be collected on a periodic basis until activities are below the minimum detectable levels or a determination is made that continued monitoring is not necessary. If the waste contained high Th-232 concentrations, lung or whole-body counting techniques may be employed to measure deposition in the body.

Excretion models will be used along with waste characterization data, bioassay data, and operational data to estimate the radionuclide intake and the resultant dose to the organs. Methods recommended in NCRP Report No. 87, "Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition" will be used. The guidance of 10 CFR 20.1201 will be followed in cases where significant organ doses or TEDE's are found.

<u>Derivation of Action Level for Uranium Tailings</u>

The worker exposure pathway for radionuclides under normal operations is via the inhalation pathway. Routine chronic exposure to radionuclides is limited by dust control measures and use of respiratory protection. However, to check the adequacy of these measures, *in vivo* or *in vitro* methods may be employed periodically, as determined by the CRSO, to assure that intakes are a small fraction of the regulatory limits.

No single method exists that will adequately detect intakes of potential 11e.(2) radionuclides at levels near the allowable limit of intake (ALI). Bioassay methods work well for the normally soluble uranium isotopes but fail to detect the insoluble thorium isotopes. Similarly, whole-body counting or lung counting methods may detect levels of Th-232 and Ra-226 (Radon daughters) at or near the ALI, depending upon the distribution in the body but fail to detect Th-230, Ra-228 or other alpha or beta emitting radionuclides. For acute intakes, analysis of the feces is normally more sensitive than other methods, while for chronic intakes it is not a viable method.

Section 17.1.1 presents a review of potential wastes for disposal at Clive. Most of those wastes are expected to contain significant weight percentages of uranium which may be used as an indicator to estimate other radionuclide intakes within the mixture.

This method is presently being used at the UMTRA sites and is described in Reif, 1992. Calculations similar to the approach in that reference will be used to develop an action level for Clive for the case where wastes similar to uranium mill tailings are being handled. Changes will be made to reflect the recent NRC regulatory requirement to limit the TEDE to 5 rem per year.

Ref Reif, R. H., Turner, J. B. and D. S. Carlson. "Uranium in Vitro Bioassay Action Level Used to Screen Workers for Chronic Inhalation Intakes of Uranium Mill Tailings", Health Physics Vol. 63, No. 4 (1992) p398.

Reif(1992) develops a radionuclide mix for mill tailings based upon actual data from the UMTRA sites. This radionuclide mix will also be assumed in this analysis and is presented in Table 17.4.1.

Presented in Table 17.4.1 are the TEDE factors per unit uptake for each of the radionuclides that contribute more than 1 percent of the TEDE.

Table 17.4.1 TEDE Per Unit Intake for Uranium Mill Tailings

Radionuclide	Lung Class	Relative Activity	TEDE (mrem/mCi)
U-nat	D	2	2.6 E+3
Th-230	Y	13	2.6 E+5
Ra-226	W	13	8.6 E+3
Pb-210	D	13	1.4 E+4
Po-210	D	13	9.4 E+3

Using the radionuclide mix in the above table, the U-nat intake equal to a TEDE of 500 mrem for the mixture was calculated to be 260 pCi. The next step is to estimate the quantity that will be transferred to the blood and eliminated via the urine.

ICRP Publication 30 uses a fractional transfer factor of inhaled activity to blood for long-lived Class D radionuclides as:

Fraction = $0.48 + 0.15\ f_1$, where f_1 is the fraction entering the blood via the GI tract. For Class D uranium, f_1 is equal to 0.05.

The concentration of U-nat in the urine at the end of a 90-day chronic exposure period, is approximately equal to the product of the daily intake rate and the intake retention fraction divided by the daily urine volume. Within the accuracy of the model, it will be assumed that all of the uranium in the blood is eliminated via the urine and thus the retention fraction is equal to 0.49.

If we assume a three-month chronic exposure at which the employee received an intake of tailings equal to 10 % of the allowable annual TEDE, the uranium concentration in a 24-hour voiding urine sample can be calculated as follows:

> (260 pCi/90 days)(0.49)/(1.4 liters/day) = 1.0 pCi/liter, where the 1.4 l/day is the daily urine produced by standard man, 90 days is the exposure time, and 0.49 is the intake retention fraction.

This concentration of U-nat in urine is equivalent to 1.5 mg/l, a level easily detectible using fluorimetry analysis.

17.4.5 Assessing Dose Equivalent from External Radiation Sources

All personnel entering the restricted area are required to wear radiation dosimeters at all times.

17.4.5.1 Permanent Employees

Permanent employees are issued a thermoluminescent dosimeter (TLD) badge provided by Envirocare. These badges are exchanged on a quarterly basis or read as soon as practical upon termination of employment. Badges are selected that measure the skin dose equivalent (shallow dose) as well as the deep dose equivalent for compliance with 10 CFR 20.1203 and 10 CFR 20.1502 and are worn in the proper place as instructed by the RSO. All badges, along with control badges, are maintained at the manned access control point when the employee is not at work.

Processing is done by a dosimetry processor holding accreditation from the National Voluntary Laboratory Accreditation Program of the National Institute of Standards and Technology appropriate for the radiation fields at the Envirocare site.

It is not anticipated that the measurement of the shallow dose equivalent will be difficult since the very soft beta radiations will be absorbed by the protective clothing of the employees as well as the relative large thickness of the air between the personnel and the waste. A periodic review of the appropriateness of the TLD program will be made by the CRSO with necessary measurements to document the findings. The use of thin window ion chambers or other methods will be used to measure the ratio of total dose rate to penetrating dose rate for each waste type at the worker's point of maximum exposure. This will be compared to the shallow and deep dose equivalent measured by the worker's personal dosimeter.

Should the CRSO determine that it is necessary to measure the shallow dose rather than use TLD devices, Envirocare will implement a procedure to calculate the shallow dose by applying a correction factor to the TLD reading(s). All exposures will be recorded when received from the dosimetry vendor to demonstrate compliance with the standards. In the event that an individual loses the personal TLD, the SRSO or his designee will investigate the potential exposure conditions and provide an estimate of the exposure.

All employees will notify their supervisor immediately upon discovery that a TLD has been lost. A new dosimeter will be issued immediately.

At this time, it is not anticipated that extremity monitoring will be necessary. However, the SRSO will monitor the work activity and if extremity monitoring is warranted, appropriate dosimeters will be obtained from the dosimetry vendor.

NRC Regulatory Guide 8.30 discusses the concern for measuring the shallow dose from yellow cake where the contact dose rate is approximately 150 mrad/hour and the dose at 30 cm is approximately 1 mrad/hour. While Envirocare understands this concern, we do not believe that the beta dose will be significant in the 11e.(2) wastes received at the site. Disposal of 11e.(2) material will normally be depleted in uranium isotopes and the disposal of separated uranium will be limited by the concentration limits in the waste acceptance criteria which is small compared to the approximately 600,000 pCi/g in yellowcake or other uranium compounds. During the waste handling operations at Envirocare, direct contact with the waste is normally not made and the combination of low activities, large distances, and protective clothing will limit the shallow dose equivalent to acceptable levels for the wastes containing uranium compounds.

Because of the low radionuclide activities in the waste, there is little potential for a significant penetrating or non-penetrating external radiation dose from airborne radioactive material. The deep dose equivalent component of this small dose, will be included in the employee's personal dosimeter reading.

17.4.5.2 Visitors and Temporary Employees

Individuals who are visiting the site on a limited basis will be issued a pocket dosimeter to record exposure. The dosimeter is read upon exiting the controlled area and recorded on the Access Log. In the case of individuals visiting as a group, one pocket dosimeter may be used providing they stay together.

17.4.6 Radiation Monitoring

17.4.6.1 Equipment, Instrumentation, and Facilities

Health Physics instrumentation selected for this program includes the portable and laboratory equipment described below.

a. Berthold Model 1043AS hand and foot monitor - 1 each. Selected as a sensitive personnel portal monitor capable of measuring alpha and

beta contamination levels simultaneously and independently and providing both a direct printer record of each survey and a computer record for each individual using coded identification badges.

b. Ludlum Model 19 Micro-R Survey Meters - 3 each. Selected as the basic survey meter for gamma exposure rates for area surveys and incoming shipments. Due to the low exposure rates encountered, a scintillation survey meter capable of performing accurate measurements in the range of background is required. The selected meters are rugged, dependable, easy to use, and feature a range of 0 to 5,000 mR/h over 5 ranges.

c. Berthold Model 122 contamination survey meter - 3 each. This meter measures alpha and beta surface contamination independently and provides a direct readout of area contamination levels. It operates over a wider range of temperature conditions than other survey meters and is well suited for field use in meeting the release standards presented in Section 17.4.7.1.

d. Ludlum Model 177 Ratemeter with Model 44-9 Pancake G-M Detector - 3 each. Selected as a portal frisker for personnel surveys due to the high sensitivity of the pancake detector and alarm-ratemeter capability of the ratemeter.

   The thin-window GM detector is sensitive to alpha, beta, and gamma radiation. The radiation types can be determined by selective use of shielding.

e. Ludlum Model 9 Ion Chamber Survey Meter - 2 each. Selected to provide a wide range of exposure rate measurements with little dependence on gamma energy. This instrument is rugged and reliable, and has a range of 0 - 5 R/h over 4 ranges.

f. Self-Reading Dosimeters (Victoreen 541R or equivalent or Bicron Model PDM-207 or equivalent). Selected to provide detection capability of approximately 1 mR over a scale of 0 - 200 mR. Used to record exposures to visitors and temporary employees while in the controlled area.

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

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

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

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

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

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

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

All instruments will be efficiency checked or source checked prior to use on a daily basis. Alpha and beta laboratory counters will be efficiency checked each day that they are in use. Portal monitors will be source checked at the beginning of each day using a source that is adequate to indicate an alarm. The response of hand-held radiation detection instruments will be compared to known sources prior to each use.

The respiratory protection equipment and protective clothing are located in the change room in the Administration building. Portable radiation instruments and laboratory instruments are located in the radiological laboratory in the Administration Building.

17.4.6.2 Area Radiation Surveys

Routine external gamma surveys using a gamma scintillation survey meter will be conducted and documented in areas involving disposal material in accordance with the type, frequency, and location(s) listed in Table 17.8. Addition area gamma surveys will be performed during daily operations as considered necessary by health physics personnel.

Routine wipe surveys for surface contamination will be conducted as listed in Table 17.8. The wipes will be analyzed for gross alpha contamination using a Ludlum Model 1000 Scaler or equal with a Model 43-10 alpha scintillation probe or equal. They will also be analyzed for gross beta contamination using a Ludlum Model 2200 scaler or equivalent and a Model 120 gas flow proportional counter or equivalent.

17.4.6.3 Airborne Particulate Radioactivity Monitoring

Work areas and boundary areas will be monitored for airborne radioactive particulates. The continuous airborne particulate samplers operated on site as part of the environmental monitoring program (see Section 7) will provide an overall average of the concentrations of airborne radioactivity. In addition to the fixed-location environmental stations, work-place samples will be collected to better assess potential exposure to employees.

On-site air particulate samples will be collected by means of F & J Specialty Products, CO. Model FJ-28B Low Volume Air Sampler, or equivalent, operating at 60 liters per minute (lpm) with a 2-inch diameter glass fiber filter. This sampler was selected on the basis of its demonstrated reliability, continuous flow control, and ability to collect sufficient sample during the weekly sample period to meet the sensitivity requirements set forth in Section 7.3.1. The sampling locations, shown in Figure 7.1, were selected to monitor airborne particulate radioactivity at site boundary locations as well as near on-site operational areas such as the rollover, disposal cell and haulways.

Work area samples will be collected with FJ-HV-1 high volume air samplers, or equivalent. The FJ-HV-1 sampler collects samples at 120 lpm and is used as a moveable sampler to collect airborne particulates at locations where a.c. power is available, or by means of a portable generator. For locations where a.c. power is not available, battery-powered portable samplers capable of collecting at least 20 lpm will be used.

Both samplers were selected to collect sufficient sample on a 2-inch glass fiber filter to permit detection levels comparable to Table 1 of 10 CFR {20.1001-2401], Appendix B, making estimation of potential exposures sufficiently sensitive for occupational exposures.

The a.c.-powered samplers will be used at locations such as the rollover, along haul ways, or near excavation and disposal activities to collect 8-hour, work-day samples. Samples will be collected daily at two locations during periods of high work activity and a minimum of twice each week during periods of low work activity. During the winter months when disposal work has been terminated, no measurements will be made. Sample collection data will include a short statement of weather conditions during collection so that results may be compared to prevailing conditions.
At the end of the sampling period, air particulate samples will be stored in envelopes and marked with the pertinent information. After a delay of seven days, air filters will be counted for gross alpha and beta levels. Gross alpha activity levels will be compared to the DAC for Th-232 of 5

E-13 mCi/ml; gross beta act vity levels will be compared to the DAC for Pb-210 of 1E-10 mCi/ml.

After counting, filters will be stcred in closed containers for future analysis. If warranted by calculations of probable exposure, the composite filters will be analyzed for Th-230, Th-232, Pb-210, Ra-228, and Ra-226, to provide precise data on radionuclide concentrations in the work environment and potential levels of internal exposure. Results of the isotopic analyses will be compared to limits provided in 10 CFR [20.1001-20.2401], Appendix B.

Gross alpha concentrations of 5 E-13 mCi/ml or gross beta concentrations of 1 E-10 mCi/ml on individual air filters are considered "action levels", and will trigger the following response by the Site Radiation Safety Officer:

1. The SRSO will evaluate site conditions to determine whether additional dust suppression methods are needed, whether posting for airborne radioactivity (20 CFR 10.1902) is required, and whether respiratory protection requirements are adequate.
2. The sample will be analyzed by gamma spectrometry and, if necessary, by radiochemical separation and laboratory analysis to determine the activities of the radionuclides present.
3. If it is confirmed that any employees exceeded the concentration limits of 10 CFR [20.1001-20.2401], Appendix B, Table 1, considering any respiratory protection devices, special urine/or fecal samples may be collected from the most significantly exposed employee to determine the extent of radionuclide uptake due to inhalation of dust. The situation will be investigated to determine the cause for such concentrations and the means of reducing such exposures in the future.

Air sampling results for airborne particulates and radon will be used to calculate internal doses to employees. Those employees in assignments most likely to receive exposure to higher concentrations of airborne particulates will be required to routinely wear respirators.

17.4.6.4 Personnel Contamination Monitoring

The use of protective clothing should minimize the potential for skin contamination. However, all personnel working in the restricted areas will be required to be monitored before leaving the access control area and must meet the release standards of Table 17.6. A hand and foot monitor sensitive to both alpha and beta contamination will be used for routine monitoring for contamination of personnel.

Workers involved in handling material will be required to wash exposed skin (hands and face) before they leave the site. In addition, showers are provided in the change area for use by all workers, as may be required by individual conditions, when exiting the site.

Workers are advised to consider any measurable contamination on their person as excessive and the goal is to keep such contamination below detectable levels.

Personnel will be expected to accomplish this by washing exposed areas of the skin with soap and water. If this does not reduce the levels below the standards of Table 17.6, the SRSO will be notified and other attempts will be made. Special radiation decontamination cleansers will be used to reduce skin contamination levels. Personnel with skin contamination will not be allowed to leave the site without approval of the CRSO.

All personal contaminated clothing or personal articles that cannot be decontaminated below the limits of Table 17.6 will be retained at the site and managed as radioactive waste.

All personnel contamination events will be documented.

The accident evaluation of Section 17.1.4 and the routine worker evaluation of Appendix A show that it is extremely unlikely that any employee could receive a lung burden of radioactivity which would require any action. If such an event did happen, the individual involved would be transported to a facility to receive a whole-body count to evaluate the potential dose. Subsequent actions, such as reassignment to a function not involving radiation exposure would be considered.

A worker might be injured in an accident that would result in the impaction of radioactive waste into a wound. Envirocare policy is to attempt to monitor injured employees before they are transported to medical care. In any case, the treating physician is informed that the injury involves possible radioactive contamination. Because the radionuclides involved are relatively insoluble, normal cleansing of the wound should remove most, if not all, of the radioactivity. A radiation survey will be used to estimate the remaining radioactivity and potential doses calculated as described in 17.4.4.4. The need for additional treatment would be based on the results of the monitoring.

Bioassay samples will be used, as necessary to help determine the body burden of any radioactivity which might have resulted from an unusual inhalation situation or wound.

Any employees who are believed to have received a TEDE of greater than 200 mrem from any source in one quarter will be notified and will assist in determining the source of the exposure and in finding a way to reduce future exposures.

17.4.6.5 Occupational Radon and Radon Daughter Monitoring

The handling of large quantities of Ra-226 and Th-232 bearing materials is expected to release Rn-222 (radon) and Rn-220 (thoron). The concentrations will vary depending upon the type of waste handled.

The occupational limit for radon daughter exposure is four (4) WLM while the limit for thoron daughter exposure is 12 WLM.
The occupational exposure limit for radon without daughters present is 4,000 pCi/l while for radon with all daughters present (100 % equilibrium) is 30 pCi/l. The exposure limit for thoron without daughters is 7,000 pCi/l and 9 pCi/l with daughters in equilibrium.

All work areas, including the administration building, will be monitored for radon and thoron using pairs of E-Perm ion chambers. One chamber responds to radon and thoron, the other responds primarily to radon. The readings along with the difference in the readings are used to calculate the radon and thoron concentrations. The minimum detectable concentration varies with the mixture of radon and thoron. If only radon is present, the MDC is approximately 500 pCi/liter-hours, or 0.75 pCi/l-month, where a month is considered continuous exposure for 4 weeks. If only thoron is present, the MDC is approximately 3.6 pCi/l-month. Detectors will be placed in the work areas and read weekly. While the measured average concentrations will be for 24 hours/day rather than the average for the work day, the results should be conservative in that the meteorology of the site is expected to enhance the levels at night.

Due to the long exposure times for the E-Perms, other measurements of the work area environment will be made to assess the workers exposure to radon and thoron and their daughter products. The E-Perm results of the radon and thoron measurements will be supplemented by grab samples for radon and thoron concentration and grab samples for radon and thoron WL determinations. If exposures are likely to exceed 10 percent of the allowable limits over a 40 hour exposure period, the grab sample results will be used to estimate the radon daughter equilibrium and the E-Perm radon concentration

results will be used to calculate a monthly average WL for radon and thoron. The radon and thoron WL results will then be used in determining the internal dose equivalents for the workers.

The occupational limit for radon daughter exposure is four (4) working months (WLM) per year, which is equivalent to a DAC of 30 pCi/l of Rn-222 in equilibrium with its daughters.

Instant WL Monitors or grab sample techniques will be used to monitor the work area on a weekly basis during periods of calm winds. For work areas routinely falling below 10 percent of the WL limits for radon and thoron daughters(0.03 WL and 0.1 WL for radon and thoron, respectively), the exposure will not be considered in the dosimetry program, provided there are no minors or declared pregnant women in the area (see 10 CFR 20.1205 (g)).

If grab samples are taken, the Ogden method, [Ogden, T.L. (1974). "*A method for measuring the working-level values of mixed radon and thoron daughters in coal mine air.*" Ann Occ. Hyg. **17**, 23.] [Ogden, T.L. (1977). "*Radon and thoron daughter working levels from ordinary industrial hygiene samples*" Ann. Occ. Hyg. **20**, 49.] will be used to measure radon and thoron daughter - WL concentrations with sample collection volumes and counting times sufficient to provide a lower limit of detection (sensitivity) of better than 0.03 WL (See NRC Regulatory Guide 8.30, "Health Physics Surveys in Uranium Mills" and the references cited therein). Instant WL meters or continuous WL monitors will be used only if the equivalent sensitivity can be achieved.

17.4.6.6 Environmental Monitoring Program

The environmental monitoring program is presented in Section 7.

17.4.7 Personnel Protection and Contamination Control

17.4.7.1 Access Control

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

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

1. Permanent employees will be issued a thermoluminescent dosimeter (TLD) badge provided by Envirocare. These dosimeters will be exchanged and

returned to the vendor on a quarterly basis. Permanent employees will pick up and turn in their dosimeters at the beginning and end of their working day at the manned access control point.

2. Individuals who are visiting the site on a limited basis will be issued a pocket dosimeter to record exposure. Visitors will pick up and turn in their pocket dosimeters at the manned access control point when they enter and exit the site. The dosimeters will be read as the individual leaves the site and recorded in the Access Log.
3. A group of visitors may all use the exposure from either one TLD or one pocket dosimeter in a situation where the entire group is to stay in the same vicinity while in the restricted area.

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

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

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

The quarterly dosimeter results from the quarterly exchange of dosimeters will be promptly recorded by the Site Radiation Safety Officer (SRSO), or his designee. The data will then be reviewed by the SRSO. Higher than expected personnel exposures will be further investigated by the Corporate Radiation Safety Officer (CRSO) and/or a contractor consultant.

All exiting employees must be surveyed for contamination using an alpha sensitive instrument. Records are maintained of the number of employees found to be contaminated and the level of contamination.

Personnel or materials leaving the restricted area will be required to meet the conditions of the following table (see Section 16.3 for equipment/vehicle decontamination procedures):

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

Column I	Column II	Column III	
Nuclide[a]	Average[b,d,f]	Maximum[b,d,f]	Removable[b,e,f]
U-nat,U-235,U-238, and associated decay products	5,000 dpm alpha/100cm^2	15,000 dpm alpha/100cm^2	1,000 dpm alpha/100cm^2
Transuranics, Ra-226, Ra-228,Th-230,Th-228, Pa-231,Ac-227,I-125, I-129	100 dpm/ 100 cm^2	300 dpm/ 100 cm^2	20 dpm/ 100 cm^2
Th-nat,Th-232,Sr-90 Ra-223,Ra-224,U-232 I-126,I-131, I-133	1,000 dpm/ 100 cm^2	3,000 dpm/ 100 cm^2	200 dpm/ 100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except SR-90 and others noted above	5,000 dpm beta-gamma/100 cm^2	15,000 dpm beta-gamma/100 cm^2	1,000 dpm beta-gamma/100 cm^2

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

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

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

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

e. The amount of removable radioactive material per 100 cm^2 of surface area should be determined by wiping the area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of know efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

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

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

There will be no high or very high radiation areas on site due to the concentration limitations in the waste acceptance criteria. As shown in Section 17.1.4, even with

wastes as high as 15,000 pCi/g of each radionuclide the external gamma exposure rate would not exceed 50 mR/h. Therefore, no special access control procedures as required in 10 CFR [20.1601-20.1602] will be developed.

17.4.7.2 Protective Clothing and Change Facilities

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

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

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

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

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

17.4.7.3 Respiratory Protection Program

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

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

recordkeeping. The CRSO is responsible for administering the respiratory protection program.

17.4.7.4 Dust Control Measures

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

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

   a) Date of application
   b) Number of treatments
   c) Rainfall received
   d) Time of day treatments were made

17.4.7.5 Envirocare Site Regulations

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

17.4.8 Health and Safety Training

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

persons to enter the restricted area is related to the activities for which the person will enter the restricted area.

There are three categories of restricted-area functions:

(1) Permanent Employee
(2) Temporary Worker
(3) Visitor

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

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

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

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

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

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

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

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

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

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

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

The training is meant to educate the employees in the fundamentals of handling radioactive materials, to provide information on the ways and means of minimizing exposure, and to inform employees of practices and programs aimed at preventing possible spread of contamination.

The semi-annual refresher sessions for permanent employees will be provided to keep the employees aware of the nature of the material with which they have daily contact. The semi-annual refresher course will be a one-hour review of the topics discussed in the 3-hour training.

The Restricted Area Entrance Training will be given on site by the CRSO or SRSO, or any Envirocare Health Physics Specialist II. During this training, procedures and precautions will be explained and the trainees will be required to read and sign either the release form or a training roster form. The training records will be maintained by the SRSO.

In addition to the above training all Envirocare site employees will be required to attend at least 20 hours of training annually taught by qualified personnel. This training will be tailored to the specific employees needs and duties and will cover such topics as general

occupational safety, radiological safety, and training on any specific items such as new procedures or safety deficiencies.

17.4.9 Staffing and Personnel

17.4.9.1 Responsibilities

The Corporate Radiation Safety Officer (CRSO) is responsible for assuring that the environmental health and safety requirements at the site are being met and, in particular, the operations at the site are in compliance with Nuclear Regulatory Commission License Requirements. All health and safety related procedural changes are approved by the CRSO.

The Site Radiation Safety Officer (SRSO) has the day-to-day radiation safety responsibilities and reports to the CRSO while working very closely with the Site Manager. Assisting the SRSO are Access Control Technicians, Health Physics Specialists II, and an Environmental Coordinator . The Environmental Coordinator is responsible for conducting the routine environmental monitoring program and performing certain laboratory analyses.

17.4.9.2 Certification for Access Control Technician and Health Physic Specialist

All personnel must be certified before they can be classified as either an Access Control Technician or a Health Physics Specialist . This certification will include training and testing beyond that given in the restricted-area training program. Specific training and experience requirements for the positions, entrance training, on-the-job training, and examinations are listed in the Procedures Manual, Appendix B. The following is a summary of requirements for certification in those areas:

Access Control Technician

1. 20 classroom hours of training in areas of chemistry, physics, radiation safety, construction safety, operation of equipment and site operations.
2. Pass a written exam designed specifically for access Control Technician.
3. Pass, to the satisfaction of the Site Radiation Safety Officer, a practical test designed to assure that candidate possesses knowledge for all equipment is being handled properly and all duties can be performed effectively.

Health Physics Specialist

1. 40 classroom hours of training in areas of chemistry, physics, radiation safety, construction safety, operation of equipment and site operations.
2. Pass a written exam designed specifically for Health Physics Specialist .
3. Pass a laboratory test designed to assure that all equipment is being handled properly and all duties can be performed effectively.

In addition to the certification, each Access Control Technician and Health Physics Specialist must maintain certification by completing the annual training described in Section 17.4.6.3.

Table 17.1 Characteristics of UMTRA Mill Tailings

Mill Site	Volume (cu m)	Ra−226 (pCi/g)	Vol x Ra (cu m − pCi/g)/Volume
Tuba City	52661	550	8.52
Durango	129323	670	25.48
Grand Junction	249410	665	48.77
Gunnison	119863	315	11.10
Maybell	365781	200	21.51
Naturita	58167	45	0.77
New/Old Rifle	304949	745	66.81
Slick Rock	7513	115	0.25
Lowman	5694	160	0.27
Ambrosia Lake	549525	570	92.11
Shiprock	311346	420	38.45
Bowman/Belfield	21250	50	0.31
Lakeview	83480	110	2.70
Canonsburg	38958	2315	26.52
Falls City	901125	190	50.35
Green River	13774	75	0.30
Mexican Hat	9045	670	1.78
Salt Lake	178730	480	25.23
total volume	3400594		
mean		464	
std deviation		509	
vol wt'd ave concentration			421.25

Figure 17.2 Tailings Characteristics of Operating Mills

Site	Ra−226 Concentration (pCi/g)
Canon City	400
Ambrosia Lake	87
Homestake	300
Panna Maria	198
White Mesa	981
Rio Algom−Lower	420
Shootaring	280
Sherwood	200
Lucky Pile 1−3	153
Shirley Basin	208
Sweetwater	280
Mean Concentration	319
Std Dev	230

Table 17.3 Characteristics of Raffinate Pits at Weldon Spring Site

	Average Radionuclide Concentration (pCi/g)				Volume Weighted Concentration
Radionuclide	Pit 1	Pit 2	Pit 3	Pit 4	
U-238	710	470	520	620	556
U-234	810	560	570	610	598
Th-232	100	120	120	120	118
Th-230	24000	24000	14000	1600	12448
Ra-228	850	200	100	60	157
Ra-226	430	440	460	11	343
Waste Volume(cu m)	13224	13224	98490	42256	167194

Table 17.4 Waste Characteristics at Kerr–McGee West Chicago Site

Characteristic	Sludge Pile	Ponds/Soils	So. Area/soils	Int. Site/ Soils
No. of Samples	11	68	235	4
Ra–226 Concentration	7526	13.4	9.9	1.8
Range	2–31500	0.1–180	0–163	1.2–4.2
U–238 Concentration	135	85.8	127.1	43.3
Range	0–565	0–1860	0–930	14.6–128
Th–232 Concentration	5284	211	169.7	6.6
Range	562–11050	0.1–2110	0.2–4120	0–126

TABLE 17-7

ENVIROCARE DOSE SUMMARY

DOSE mrem	NUMBER OF EMPLOYEES 1988	1989	1990
<10	5	17	10
10	2	4	9
20	1		2
30	2		1
40			1

TABLE 17.8

ROUTINE MONITORING AND SURVEYS

Type	Location	Frequency
A. Gamma Radiation Levels	1. Perimeter of Controlled Area(s) 2. Office Area 3. Lunch/Change Area 4. Transport Vehicles	1. Weekly 2. Weekly 3. Weekly 4. Upon Arrival at Site and before departure.
B. Contamination Wipes	1. Eating Area 2. Change Area 3. Office Areas 4. Railcar rollover and control shack 5. Equipment/Vehicles	1. Weekly 2. Weekly 3. Weekly 4. Weekly 5. Once before release
C. Employee/ Personnel	1. Skin & Personal clothing	1. Prior to exiting controlled area
D. Gamma Exposure	1. Administration Bldg. 2. Security Trailer	1. Quarterly 2. Quarterly
E. Radon Concentration	1. Administration Bldg. 2. Security Trailer	1. Quarterly 2. Quarterly