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Current Practices for Maintaining Occupational Exposures ALARA at Low-Level Waste Disposal Sites

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Pacific Northwest Laboratory
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U.S. Nuclear Regulatory
Commission

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Washington, D.C. 20555

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Current Practices for Maintaining Occupational Exposures ALARA at Low-Level Waste Disposal Sites

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SUMMARY

The United States Nuclear Regulatory Commission contracted with Pacific Northwest Laboratory (PNL) to provide technical assistance in establishing operational guidelines, with respect to radiation control programs and methods of minimizing occupational radiation exposure, at Low-Level Waste (LLW) disposal sites. The PNL, through site visits, evaluated operations at LLW disposal sites to determine the adequacy of current practices in maintaining occupational exposures as low as is reasonably achievable (ALARA). The data sought included the specifics of: ALARA programs, training programs, external exposure control, internal exposure control, respiratory protection, surveillance, radioactive waste management, facilities and equipment, and external dose analysis. The results of the study indicated the following:

The Radiation Protection and ALARA programs at the three commercial LLW disposal sites were observed to be adequate in scope and content compared to similar programs at other types of nuclear facilities. However, it should be noted that there were many areas that could be improved upon to help ensure the health and safety of the occupationally exposed individuals.

SUMMARY

The United States Nuclear Regulatory Commission contracted with Pacific Northwest Laboratory (PNL) to provide technical assistance in establishing operational guidelines, with respect to radiation control programs and methods of minimizing occupational radiation exposure, at low-level waste (LLW) disposal sites. The PNL, through site visits, evaluated operations at LLW disposal sites to determine the adequacy of current practices in maintaining occupational exposures as low as is reasonably achievable (ALARA). The data sought included the specifics of: ALARA programs, training programs, external exposure control, internal exposure control, respiratory protection, surveillance, radioactive waste management, facilities and equipment, and external dose analysis. The results of the study indicated the following:

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CONTENTS

SUMMARY	iii
1.0 INTRODUCTION.....	1
2.0 ALARA PROGRAM.....	3
2.1 POLICY STATEMENT.....	3
2.2 RESPONSIBILITY.....	3
2.3 AUTHORITY.....	3
2.4 WRITTEN PROCEDURES.....	4
2.5 TRAINING.....	4
2.6 ALARA COMMITTEE.....	4
2.7 PROGRAM FEEDBACK.....	5
2.8 TREND ANALYSIS.....	5
2.9 GUIDANCE.....	6
3.0 TRAINING PROGRAM.....	7
3.1 PERSONNEL SELECTION AND QUALIFICATION.....	7
3.2 GENERAL ORIENTATION TRAINING.....	7
3.3 RADIATION WORKER TRAINING.....	8
3.4 RADIATION PROTECTION STAFF TRAINING.....	10
3.5 HANDS-ON TRAINING.....	10
3.6 SUMMARY.....	11
4.0 EXTERNAL EXPOSURE CONTROL.....	13
4.1 RESPONSIBILITY.....	13
4.2 RADIATION ENVIRONMENT.....	13
4.3 DOSIMETRY DEVICES.....	14
4.4 SPECIAL DOSIMETRY.....	15

4.5	PERSONNEL MONITORED.....	15
4.6	QUALITY ASSURANCE.....	16
4.7	DOSE REVIEW AND RECORD SYSTEM.....	17
4.8	EXPOSURE CONTROLS AND REDUCTION METHODS.....	18
5.0	INTERNAL EXPOSURE CONTROL.....	21
5.1	RESPONSIBILITY.....	21
5.2	INTERNAL DOSIMETRY.....	21
5.3	EXPOSURE REVIEW PROGRAM.....	24
5.4	EXPOSURE LIMITATIONS PROGRAMS.....	24
5.5	QUALITY ASSURANCE PROGRAM.....	24
6.0	RESPIRATORY PROTECTION.....	27
6.1	PROGRAM ESTABLISHMENT.....	27
6.2	ENGINEERING CONTROLS.....	29
6.3	TRAINING AND RETRAINING.....	31
6.4	MAINTENANCE.....	32
6.5	SUPPLIED-AIR RESPIRATORS.....	33
7.0	SURVEILLANCE.....	35
7.1	SCOPE.....	35
7.2	TYPES OF RADIATION.....	36
7.3	INSTRUMENTATION.....	44
7.4	ANALYTICAL INSTRUMENTATION.....	46
8.0	RADIOACTIVE WASTE MANAGEMENT.....	49
8.1	RESPONSIBILITY.....	49
8.2	TYPES OF RADIOACTIVE WASTES.....	49
8.3	ACCEPTANCE OF WASTES.....	49

8.4	MOVEMENT OF WASTES.....	51
8.5	BURIAL OF WASTES.....	52
8.6	VOLUME REDUCTION.....	57
8.7	WASTE INSPECTION.....	57
8.8	SITE GENERATED WASTES.....	57
8.9	STORAGE OF WATES.....	58
8.10	RELEASE RATES.....	58
9.0	FACILITIES AND EQUIPMENT.....	59
9.1	COUNTING ROOMS.....	59
9.2	PERSONNEL DECONTAMINATION.....	59
9.3	EQUIPMENT DECONTAMINATION.....	63
9.4	CHANGE ROOMS.....	63
9.5	LAUNDRY.....	63
9.6	EXTERNAL DOSIMETRY.....	63
9.7	INTERNAL DOSIMETRY.....	64
9.8	INSTRUMENT STORAGE.....	64
9.9	CALIBRATION FACILITY.....	64
9.10	MEDICAL AID.....	65
9.11	OFFICES.....	65
9.12	ACCESS CONTROL.....	65
9.13	TRAINING FACILITY.....	66
9.14	OFFSITE FACILITIES.....	66
9.15	PROTECTIVE EQUIPMENT.....	66
9.16	ALARA IMPACT.....	67

10.0	EXTERNAL DOSE ANALYSIS.....	69
10.1	DOSIMETRY DATA ANALYSIS.....	69
10.2	SUMMARY AND CONCLUSIONS.....	84
11.0	CONCLUSIONS.....	85
	BIBLIOGRAPHY	87

8.9	STORAGE OF WASTE.....	58
8.10	RELEASE RATES.....	58
9.0	FACILITIES AND EQUIPMENT.....	59
9.1	COUNTING ROOMS.....	59
9.2	PERSONNEL DECONTAMINATION.....	59
9.3	EQUIPMENT DECONTAMINATION.....	63
9.4	CHANGE ROOMS.....	63
9.5	LAUNDRY.....	63
9.6	EXTERNAL DOSIMETRY.....	63
9.7	INTERNAL DOSIMETRY.....	64
9.8	INSTRUMENT STORAGE.....	64
9.9	CALIBRATION FACILITY.....	64
9.10	MEDICAL WARD.....	65
9.11	OFFICES.....	65
9.12	ACCESS CONTROL.....	65
9.13	TRAINING FACILITY.....	66
9.14	OFFSITE FACILITIES.....	66
9.15	PROTECTIVE EQUIPMENT.....	66
9.16	ALARM IMPACT.....	67

FIGURES

1	Projection of Dose Reduction Goals.....	17
2	Chair Counting System Using Multiple Detectors.....	22
3	Examples of Respiratory Protection Used at Low-Level Waste Sites.....	30
4	Container Opening Facility With Overhead Fume Hood.....	31
5	Survey of Incoming Shipment.....	37
6	Monitoring of Personnel During Preparation for Cask Unloading.....	38
7	Monitoring of Personnel During Unloading of Barrels.....	39
8	Exposure Rate Monitoring at Burial Trench Edge.....	39
9	Relative Position of Personnel to Cask During Unloading Operation	40
10	Relative Position of Crane Operator to Contents of the Trench During Burial Operation.....	40
11	Relative Position of Forklift Operator to Waste Container During Unloading.....	41
12	Personnel Frisking Station.....	42
13	Air Monitoring Station.....	43
14	Radioactive Waste Receiving Area.....	50
15	Vehicle Radiological Survey.....	51
16	Cask Preparation and Equipment Decontamination Facility.....	52
17	Final Cask Preparation Prior to Unloading at Trench.....	53
18	Waste Removal Platform on Trench Floor.....	54
19	Waste Handling Equipment.....	54
20	Waste Removal Platform on Trench Edge.....	55
21	Random Placement of Barrels.....	55
22	Selected Placement of Barrels.....	56

23	Radwaste Supervisor and Health Physicist Directing Cask Placement Operations.....	56
24	Typical Disposal Site Layout.....	60
25	Site Access Building.....	61
26	Gas Flow Proportional Counter, Multichannel Analyzer, and Single Sample Holder.....	62
27	Gas Flow Proportional Counter and 100 Specimen Sample Changer.....	62
28	Whole-Body Counting Chair.....	64
29	Thyroid Counting Equipment.....	65
30	Dose, Activity, and Volume Graph; Site A 1981.....	74
31	Dose, Activity, and Volume Graph; Site B 1981.....	75
32	Dose/Activity vs. Activity, Site A 1981.....	77
33	Dose/Activity vs. Activity, Site B 1981.....	78
34	Dose/Volume vs. Volume, Site A 1981.....	79
35	Dose/Volume vs. Volume, Site B 1981.....	80
36	Activity Concentration of Wastes, Sites A and B 1981.....	81

TABLES

1	Training Summary.....	11
2	Whole-Body Dosimetry at the LLW Disposal Sites.....	15
3	10 CFR 20.101 Whole-Body Dose Limits.....	18
4	Predominant Radionuclides.....	45
5	Cumulative and Average Dose Figures, Sites A and B 1981.....	70
6	Waste Activity Received and Dose per Unit Activity, Sites A and B 1981.....	71
7	Waste Volume Received and Dose per Unit Volume, Sites A and B 1981.....	71
8	Site Dose Comparison, Sites A and B 1981.....	73
9	Doses to Radiation Worker Groups, Site A 1981.....	82
10	Doses to Radiation Worker Groups, Site C 1981.....	82
11	Doses (Rem) to Waste Handling Group; Sites A, B and C 1981.....	83
12	Annual Man-Rem Figures, Sites A and B 1981.....	83
13	Summary of Average Monthly Doses; Sites A, B and C 1981.....	84

TABLES

1	Training Summary.....	11
2	Whole-Body Dosimetry at the LW Disposal Sites.....	15
3	10 CFR 20.101 Whole-Body Dose Limits.....	18
4	Predominant Radionuclides.....	45
5	Cumulative and Average Dose Figures, Sites A and B 1981.....	70
6	Waste Activity Received and Dose per Unit Activity, Sites A and B 1981.....	71
7	Waste Volume Received and Dose per Unit Volume, Sites A and B 1981.....	71
8	Site Dose Comparison, Sites A and B 1981.....	73
9	Doses to Radiation Worker Groups, Site A 1981.....	82
10	Doses to Radiation Worker Groups, Site C 1981.....	82
11	Doses (Rem) to Waste Handling Group, Sites A, B and C 1981.....	83
12	Annual Man-Rem Figures, Sites A and B 1981.....	83
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1.0 INTRODUCTION

Pacific Northwest Laboratory (PNL) has been contracted by the Office of Nuclear Material Safety and Safeguards (NMSS) to provide technical assistance to the Nuclear Regulatory Commission (NRC). The objective is to establish operational guidelines, with respect to radiation control programs and methods of minimizing occupational radiation exposure, at low-level waste (LLW) disposal sites. This has been accomplished, in part, by characterizing operations at LLW disposal sites through site visits by PNL to determine the status of current practices in maintaining occupational radiation exposures as low as reasonably achievable (ALARA).

The radiation control practices of the three commercial LLW disposal sites at Hanford, Washington; Beatty, Nevada; and Barnwell, South Carolina were characterized by this research program. A team of three to four PNL Health Physics staff members visited each site for two days during the period March to May, 1982. Characterization of each of the three licensee programs was accomplished by the use of a modified Management and Oversight Risk Tree (MORT)^(a) analysis. The MORT approach was based on the use of analytical logic methodologies with analytical trees and applicable questions for each tree. Specific areas evaluated by the use of the MORT system were:

- ALARA program
- training program
- external exposure control
- internal exposure control
- respiratory protection
- surveillance
- radioactive waste management
- facilities and equipment.

These eight areas are discussed in Chapters 2.0 through 9.0 of this document.

The site visits were also used by PNL to gather occupational dose data from the three LLW disposal sites. Individual and collective dose information is summarized in Chapter 10.0 of this document. Information of concern includes: cumulative exposure, monthly exposure, exposure per volume buried and exposure per activity buried.

(a) Developed by the Idaho National Engineering Laboratory for the Department of Energy (1978). Modified by PNL for Health Physics Appraisals of Commercial Nuclear Power Plants, NRC Licensed Uranium Mills and Commercial LLW Disposal Sites.

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Pacific Northwest Laboratory (PWL) has been contracted by the Office of Nuclear Material Safety and Safeguards (NMSS) to provide technical assistance to the Nuclear Regulatory Commission (NRC). The objective is to establish operational guidelines with respect to radiation control programs and methods of minimizing occupational radiation exposure, at low-level waste (LLW) disposal sites. This has been accomplished, in part, by characterizing operations at LLW disposal sites through site visits by PWL to determine the status of current practices in maintaining occupational radiation exposures as low as reasonably achievable (ALARA).

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2.0 ALARA PROGRAM

Reducing radiation exposure to "As Low As Reasonably Achievable (ALARA)" should be the goal of radiation protection programs. The ALARA programs of three low-level waste (LLW) disposal sites (Hanford, Washington; Beatty, Nevada; and Barnwell, South Carolina) were reviewed based on the implementation of the ALARA concept. It was observed that the programs were formalized and generally adequate in detail and scope.

The ALARA programs at the three licensees consisted of the following elements: policy statement, program responsibility, authority, procedures, training, ALARA committee, program feedback and trend analysis.

2.1 POLICY STATEMENT

It was observed that a formalized written ALARA policy statement existed at the three licensees. The policy statements were issued by corporate management and were reaffirmed by management on a routine basis. A site-wide ALARA effort was favored by the policy statements as opposed to only a radiation protection department effort. In other words, ensuring that exposures were maintained ALARA was not only the responsibility of the Radiation Protection Department but was also the responsibility of each individual radiation worker. Personnel that were expected to receive greater than 500 mrem per calendar year (occupational exposure) were categorized as radiation workers and were issued personnel monitoring devices (as discussed in Chapter 4.0). Therefore, exposures were maintained ALARA both by individual radiation workers through the mechanism of training and by the Radiation Protection Department through observance and control of the overall program. The policy statements covered the areas of ALARA program responsibility, authority of staff members and feedback mechanisms.

2.2 RESPONSIBILITY

The ultimate responsibility for the ALARA program was given to the site manager at all three LLW sites; however, the everyday responsibilities were delegated to the radiation safety officers (RSOs) by the site managers. These responsibilities were outlined in the written policy statements and ALARA procedures. The ALARA procedures as defined here, and in subsequent discussions, are those formalized written procedures that pertain directly to the institution of ALARA concepts. Practices which are not based on formalized written procedures, but are used in the field, will be termed ALARA practices as opposed to ALARA procedures.

2.3 AUTHORITY

The RSOs at the three sites were observed to have the authority to implement all aspects of their respective ALARA programs. In addition, the radiation protection organizations had the authority to ensure that the programs were implemented and to suspend unsafe operations when observed. These

authorities were outlined in the written policy statements and ALARA procedures. Specific authorities included:

- approval of plans for construction or modification of facilities
- issuance and approval of radiation work permits (as discussed in section 7.1.1 of this document)
- determination of operational protective measures to ensure ALARA
- training and qualification of radiation workers.

2.4 WRITTEN PROCEDURES

The three LLW disposal sites had made a commitment to the principle of maintaining radiation exposures ALARA. This was evidenced by the generation of detailed ALARA procedures, the assignment of responsibility and the use of hands-on training. ALARA procedures were observed to be formalized, up-to-date and site specific. The LLW disposal sites were observed to have ALARA procedures for specific subject areas including: (1) ALARA Program Establishment and Responsibilities, (2) ALARA Training, (3) ALARA Committees, (4) ALARA Exposure Goals and Limits, (5) Pregnant Female Exposure, (6) Special Maintenance, (7) Facility Design Changes, (8) Equipment Decontamination and (9) Exposure and Surveillance Records. The procedures were reviewed by the RSOs and the ALARA committees on a routine basis. This was performed to ensure that they were realistic, cost effective and met the ALARA philosophy.

2.5 TRAINING

Training applicable to ALARA was observed to be given not only on a site-wide level (Chapter 3.0 of this document) but also to individual working groups. The latter was accomplished by working group foremen and/or supervisors. The training given on a site-wide basis was found to be up-to-date (given on a routine basis), formalized and documented at all three sites; however, the working group training was not documented at any of the LLW disposal sites. The majority of ALARA training, whether site-wide or working group, was accomplished by hands-on instruction. ALARA training was based on operational considerations (such as manpower requirements, cost and feasibility), in addition to radiation protection concerns.

2.6 ALARA COMMITTEE

ALARA committees existed at all three licensees. At one site the committee was informal and its membership varied depending upon what jobs or practices were being reviewed. The RSO was always an integral member of the committee. The second site had a formal ALARA committee which consisted of the site manager, the RSO and the chemistry supervisor. Additional committee members from selected working groups were added as necessary. The ALARA committee for the third site consisted of thirteen permanent members from site management, radiation protection, quality assurance, maintenance and waste handling. Their activities included:

1. review of ALARA and related procedures
2. review and investigation of over-exposures or other abnormal radiation exposure occurrences
3. review of facility design changes
4. verification and review of ALARA training
5. investigation of employee complaints related to ALARA
6. performance of audits
 - procedures
 - records
 - radiation protection program
7. verification that site management was responsive to ALARA philosophy
8. made recommendations to the operations and/or radiation protection departments based upon the above activities.

The ALARA committee at this site met quarterly at a minimum. A formalized record of activities and recommendations was maintained on file for each meeting.

2.7 PROGRAM FEEDBACK

A formal feedback program was a part of the ALARA program at only one site. It was instituted to communicate the effectiveness of the ALARA program to site and corporate management. The feedback program consisted of the investigation of employee complaints/suggestions relating to ALARA. This was accomplished by the establishment of a formalized communication system between employees and management. One of the communication links at this site was a "safety suggestion program." Monetary awards were given based on potential cost or radiation exposure savings. It was observed that workers had pride in their work and this alone could be an incentive for the development of an effective ALARA program.

2.8 TREND ANALYSIS

Trend analysis was observed to be an integral part of the radiation protection programs at all three sites (as discussed in Chapter 3.0 of this document). In addition, it was used as a valuable tool for the overall ALARA programs of the licensees. At one site, trend analysis was performed on the following data sets:

1. collective exposure
 - site total
 - site workers

- operators
- radiation protection technicians
- maintenance personnel
- supervisors

2. monthly exposure
3. exposure versus volume buried
4. exposure versus activity buried
5. activity received per time interval
6. activity concentration received per time interval
7. waste volume received per time interval

Further details on the above data sets are given in Chapter 10.0 of this document.

2.9 GUIDANCE

ALARA-related guidance, currently in the possession of and used by the three commercial LLW disposal sites, was observed to be:

- Code of Federal Regulations, Title 10-Energy, Part 20 - Standards for Protection Against Radiation (10 CFR 20)
- Code of Federal Regulations (Proposed), Title 10-Energy, Part 61 - Licensing Requirements for Land Disposal of Radioactive Waste (10 CFR 61)
- U.S. Nuclear Regulatory Commission, Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable, Regulatory Guide 8.10
- U.S. Nuclear Regulatory Commission, Instruction Concerning Prenatal Radiation Exposure, Regulatory Guide 8.13
- U.S. Department of Energy, A Guide to Reducing Radiation Exposure to As Low As Reasonably Achievable (ALARA), DOE/EV/1830-15.

3.0 TRAINING PROGRAM

The radiation protection training programs of the three low-level waste (LLW) disposal sites were reviewed. It was observed that the programs were formalized and generally adequate in detail and scope.

3.1 PERSONNEL SELECTION AND QUALIFICATION

Radiation protection duties at commercial LLW sites were observed to be the responsibility of a select group of staff members. At two of the sites this group was small and consisted of a Radiation Safety Officer (RSO) and from zero to five Radiation Protection Technicians. The radiation protection group at the third site was composed of a Regulatory Affairs Manager, to which the RSO was responsible, and six (at a minimum) Radiation Protection Technicians. In addition, the Radiation Protection staff provided support for related duties such as Industrial Safety, Environmental Monitoring, etc.

The responsibilities and duties for licensee radiation protection staff were outlined in formalized job descriptions; however, the selection and qualification criteria were observed not to be formalized as outlined in ANSI 3.1-1978. In particular, the academic background and previous radiation protection experience of potential RSOs and the criteria for advancement of technicians are of major importance but were not formally addressed. It was observed that academic background and experience were formally addressed for radiation protection instructors.

3.2 GENERAL ORIENTATION TRAINING

The training programs at the three licensees consisted of general orientation training (given initially to all site personnel), radiation worker training, radiation protection staff training and hands-on training. The RSO at two of the sites and the "Training Coordinator" at the other site were formally given the overall responsibility for the training programs.

General orientation training was given to all new employees at the three sites within the first week of employment. In addition, visitor and contractor personnel were also given this training. The orientation training consisted of class room instruction, supplemented by other training methods. In particular, one site handed out reference material and showed two forty-five minute films followed by a written test. The tests and reference material was generated by site staff members; whereas, the films were obtained from a commercial vendor. Course material covered the areas of site security, radiation protection fundamentals, emergency plans and other site specific subjects. Personnel who were only given this training, such as clerical or support staff, were retrained every two years at two sites and yearly at the third site. In addition, specialized training for biological risks of radiation exposure to the unborn child was observed to be given to female employees.

Formalized orientation training lesson plans and procedures existed at all three LLW disposal sites and were observed to be reviewed routinely to ensure that they were consistent with site-specific procedures and methods. Training was conducted by the RSO (or designate) at two sites and by the training coordinator at the third site. Orientation training records consisted of a sign-off sheet (instructor's signature) and the student's exam which were placed in each individual's personnel file. At two sites, the individual's training records and external/internal exposure records were kept in the same file. Training records were maintained for a minimum of two years at all three sites.

The minimum qualification criteria for the instructors (RSOs) at two sites included an associate degree (engineering/science) and three years of professional radiation protection experience. The training coordinator at the third site held a B.S. degree (education) and was a certified teacher with four and one-half years of teaching experience.

3.3 RADIATION WORKER TRAINING

Radiation worker training was given to new employees from selected job categories within one month of hire at the three LLW disposal sites. Personnel from the following job categories received this training: managers/supervisors, quality assurance, radiation protection, equipment operators and radiation waste handlers (maintenance and riggers).

Radiation worker training at two sites covered a two-day classroom period in which the following subjects were covered:

1. radiation exposure control

- limits (regulatory and administrative)
- stay time
- posting
- sources of radiation
- dosimetry
- biological risks

2. contamination control

- sources of contamination
- control of contamination
- protective clothing
- controlled area exiting
- airborne contamination
- respiratory protection equipment

3. Department of Transportation (DOT) labeling

4. radiological emergencies.

This training was followed by a written test with 25 to 30 questions. Passing grades on the tests were 70%. A demonstration of hands-on skills, for such subjects as personnel dosimeter reading, donning/removing protective clothing and respiratory protection equipment, personal contamination surveys (frisking), recovery from "spills" of radioactive material and exiting controlled areas, was also required. The successful completion of the written test and hands-on evaluation were documented by instructor signatures on sign-off sheets. The exams of the students were included in the individual's file along with the sign-off documentation. Retraining was performed every two years using the same course outline. The training methodology was developed by the site Radiation Protection staff.

Radiation worker training at the third site consisted of a sixteen packet self-study course that was developed by the onsite radiation protection staff. The following subjects were covered:

- nature of matter
- radioactivity and units
- biological effects
- prenatal exposures
- monitoring and control of external radiation exposure
- contamination control
- radiation safety calculations
- Department of Transportation (DOT) requirements
- land burial for LLW
- radioactive material accountability
- quality assurance
- emergency preparedness
- role of health physics technician
- occupational safety and health
- first aid
- fire protection.

This training was followed by a written test for each packet with a grade of 70% required to pass. The duration of the self-study course was dictated by the learning ability of the student. Successful completion of the exams were documented in each individual's record file. Retraining was performed on an annual basis with the course materials distributed throughout the year. This essentially provided continuous retraining.

Formalized training lesson plans and procedures existed at all three LLW disposal sites and were observed to be up-to-date. At one site the procedures for all types of radiation protection training had the following format:

- scope of training
- applicable references
- definitions
- training program responsibilities
- lesson plan outlines
- retraining/recertification criteria
- record maintenance.

3.4 RADIATION PROTECTION STAFF TRAINING

Radiation protection staff training consisted of the training given to the radiation protection technicians by the RSOs. In addition, at two sites the RSOs received training from corporate management every two years. The radiation protection technician training was observed to be an ongoing function at all three sites. Both types of training were based on Radiation Protection lesson plans developed by corporate staff members for two sites and by the onsite radiation protection staff for the third site. The areas of concern were:

- units of radiation and radioactivity
- types of radiation
- shielding
- radiation detection
- biological effects
- sources of radiation
- counting statistics
- radiation surveys
- airborne surveys
- contamination surveys
- protective clothing
- respiratory protection
- contamination control
- decontamination
- external exposure control
- radioactive waste disposal
- environmental monitoring
- emergencies.

Upon completion of the initial training, a written exam was given to the technician with a grade of 70% required to pass. At two sites, the technician was then given an oral exam by a corporate panel of midlevel management who had previous LLW radiation protection experience. An oral exam grade of 70% was required for certification as a radiation protection technician. Reviews were conducted annually at these two sites on the work performance of each certified technician. If, in the opinion of the reviewer, a technician's performance was unsatisfactory in any area, another review was conducted within one month. If three areas were considered unsatisfactory, the technician was removed from work until recertified. These evaluations were supplemented by refresher exams with passing grades of 60% in individual areas and 70% overall. RSO certification, annual evaluation and recertification was performed by corporate management. Recertification was observed to be required for any radiation protection technician or RSO that had not performed in their respective capacities for a period of three months or longer.

3.5 HANDS-ON TRAINING

Hands-on training was observed to exist for all workers at the three LLW disposal sites. At two of the sites, this training was formally documented. Where feasible, mock ups and preplanning were used as part of the hands-on

training. Hands-on training included instrument use, survey techniques, use of protective clothing and respiratory protection, and decontamination of equipment and personnel, at a minimum.

3.6 SUMMARY

Table 1 summarizes the types of training, who receives the training and on what frequency.

TABLE 1. Training Summary

<u>Training</u>	<u>Employees Receiving Training</u>	<u>Initial Training</u>	<u>Site</u>	<u>Frequency of Retraining</u>
General Orientation	All Site Personnel	Within One Week of Employment	A	2 years
			B	2 years
			C	1 year
Radiation Worker	Radiation Workers	Within One Month of Employment	A	2 years
			B	2 years
			C	1 year
Radiation Protection Staff	RSOs	Prior to Advancement to RSO(a)	A	2 years
			B	2 years
			C	None
	Radiation Protec- tion Technicians	Prior to Advancement to Technician(a)	A	On Going
			C	On Going
Hands-On	Radiation Workers	As Needed	A	As Needed
			B	As Needed
			C	As Needed

(a) May be waived for new employees based upon previous job experience or education.

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Training	Employees Receiving Training	Initial Training	Site	Frequency of Receiving
General Orientation	All Site Personnel	Within One Week of Employment	A B C	2 years 2 years 1 year
Radiation Worker	Radiation Workers	Within One Month of Employment	A B C	2 years 2 years 1 year
Radiation Protection Staff	RPS	Prior to Advancement to (RPS)	A B C	2 years 2 years None
Radiation Protection Technicians		Prior to Advancement to Technicians	A B C	On going On going On going
Hands-on	Radiation Workers	As needed	A B C	As needed As needed As needed

(a) may be waived for new employees based upon previous job experience or education.

4.0 EXTERNAL EXPOSURE CONTROL

The methods and devices used to control external radiation exposures were characterized at the three LLW disposal sites.

4.1 RESPONSIBILITY

At two of the sites visited, the Radiological Safety Officer (RSO) was responsible for control of external radiation exposures; a Health Physics Supervisor (of which there was one) was responsible at the remaining site. The Health Physics Supervisor acted in the capacity of a site RSO.

Exposure control appeared to be assigned at a sufficient level. The RSOs at two sites report directly to the site manager, thus allowing Health Physics concerns to be expressed directly to management. The RSO at the remaining site was the site manager. A dual role, such as this, is not normally desirable; however, at present, the work requirements of both site manager and RSO should not cause a lack of attention to the ALARA program. This may not hold true if the future work requirements of either position are increased.

4.2 RADIATION ENVIRONMENT

Health Physicists at the three commercial waste sites have defined the anticipated radiological conditions of their respective sites. The identification of these conditions was apparently obtained through operating experience rather than through a study or a safety analysis report.

Radioisotopes causing the greatest external radiation hazards in the work areas were high energy, gamma-emitting fission products and by-product material. Of these, Cobalt-60, Cesium-137, and a variety of radioiodines have been identified by the sites to be of major concern even though container integrity is still intact. This is due to the fact that containers offer little shielding towards attenuating high energy gammas.

Four primary sources of radiation exposure were identified during the site visits. These included incoming trucks with wastes, deactivated trenches, active trenches, and individual containers such as 55 gal barrels, wooden crates casks, and other waste items. These sources of exposure are discussed further in Chapter 8.0, but a brief comment here may be helpful.

4.2.1 Incoming Trucks

Trucks loaded with wastes and parked in the receiving area were the major source of exposure outside the controlled area. The exposure rates emanating from the trucks were directly controlled by Department of Transportation Regulations (Title 49) and were not observed to exceed more than a few mR/hr at one meter from the outside surface of the truck. Waste check-in personnel were the employees primarily exposed to this source.

4.2.2. Deactivated Trenches

Deactivated trenches did not contribute significantly to occupational doses because workers seldom came into close contact with them. In addition, dose rates on top of the covered trenches seldom exceeded 1 to 2 mrem/hr.

4.2.3 Active Trenches

The active trench area was one of the two largest sources of exposure to radiation workers. Personnel in and around the trench were exposed to many truck loads of waste containing barrels, crates and casks. Further discussion of this area is found in Chapter 8.0.

4.2.4 Individual Containers

Exposure from individual containers in off-loading operations was the other major source of occupational dose. Because personnel were required to be in close contact with waste containers during this operation, the potential for higher doses was greater than for other operations.

4.3 DDSIMETRY DEVICES

All sites used commercial vendors to supply and process personnel dosimeters; therefore, no processing equipment or methods were available for study. Two sites used film dosimeters and one site used thermoluminescent dosimeters (TLD) for measuring radiation dose. The film dosimeters contained at least two different types of metallic filters. The dosimeters were designed so that the filters, or a combination of the filters, shielded various portions of the film. This allowed for a determination of the penetrating and nonpenetrating radiation ratios.

The TLD contained two lithium fluoride chips, with one chip being filtered. This filtering configuration allowed a determination of the penetrating and nonpenetrating radiation ratios in a manner similar to the film dosimeters. Because site operators anticipated only beta and gamma exposures, no neutron dosimetry was worn. However, provisions for obtaining neutron dosimeters were in place.

The dosimetry devices in use appeared to respond well to the radiation energies mentioned previously, as well as to the intensities of the radiation fields encountered (leading to accumulated doses of 10 to 10,000 mrem). This judgment was based solely on a review of vendor specifications - no site has tested the accuracy of the vendor's claims, except in the area of reporting integrated doses. See Table 2 for a comparison of whole-body dosimetry devices in use by the three LLW disposal sites.

TABLE 2. Whole-Body Dosimetry at the LLW Disposal Sites

<u>Dosimeter</u>	<u>Radiations Detected</u>	<u>Useful Range</u>	<u>Period of Use</u>
Film	Beta greater than 1.5 MeV; gamma 18 KeV to 20 MeV	10 mrem to >500 rem	Monthly or bimonthly
TLD	Beta greater than 1.5 MeV; gamma 18 KeV to 20 MeV	10 mrem to >500 rem	Monthly or quarterly
Pencil Dosimeter	Gamma 30 KeV to 2 MeV	Variable - 0 to 200 mrem; 0 to 1 rem; 0 to 5 rem	Recharge daily

4.4 SPECIAL DOSIMETRY

Extremity monitoring was available to designated employees at all the sites. These were not worn routinely, however. Two sites used the following criteria for requiring extremity monitoring: ring dosimeters, wrist dosimeters, etc. were required when the dose rate at two feet from the trunk of the body exceeded 1.5 times the dose rate at the trunk. The remaining site gave the option of wearing extremity monitoring devices to the individuals, except when required by the RSO. The extremity dosimeters observed had the same response characteristics as those listed for the whole-body TLD badge.

Each site claimed its dosimeters could measure skin doses accurately. The whole-body dosimeters in use had a 7 to 10 mg/cm² filter which simulates the outer layer of nonreproducing skin. The ability of the dosimeters to detect or measure skin doses was not demonstrated by the licensees. No site had recorded a skin dose different from that of the whole-body dose.

Persons not employed at the sites, but who visit controlled areas, were issued pencil dosimeters (pocket-ion chambers) for temporary monitoring. The readings were recorded at the beginning and end of each day's visit by either a health physicist or a security guard responsible for site access. The difference was the day's accumulated exposure. TLDs or film dosimeters were issued to individuals on a temporary basis if the visitor stayed more than a week or worked in a high radiation area.

4.5 PERSONNEL MONITORED

At all sites, it was determined that only those employees with a reasonable potential for radiation exposure were to be provided with dosimetry. This essentially included all site workers, even clerical and other support

personnel. At one site with a large number of clerical personnel, a random sampling of workers in this category wore dosimeters to backup dose assumptions for this group.

A large variation in the number of radiation workers was observed between the sites. One site monitored approximately 8 persons, while approximately 23 were monitored at the second and 35 at the third site. Only approximate figures were available, because of the fluctuation in personnel numbers due to employee turnover and changing job requirements.

4.6 QUALITY ASSURANCE

Quality Assurance (QA), as defined by ANSI and as used in this document, is "all those planned or systematic actions necessary to provide adequate confidence that an item or a facility will perform satisfactorily in service." Because attainment of the ALARA goals depend heavily upon the quality of radiation measurements and work performed, the QA aspects of each site's operations were studied.

Quality assurance varied widely between the sites. One site ran a program with all the requirements of a good QA program. All outside vendors were required to be cleared by a department dedicated to QA prior to the purchase and use of any service. Of special interest was the QA of personnel dosimeters and radiation measurement instruments.

The site with the comprehensive QA program spiked test dosimeters and included them in each batch of personnel dosimeters returned to the vendor for processing. A 100 mCi Cesium-137 sealed source was used to produce a known radiation field for the tests. Test dosimeters were spiked to the dose levels of interest to the licensee, primarily in the range from minimum detectable to 10 rem. The other sites were hindered in testing dosimeters because they did not possess a reliable test source. However, dosimeters were spiked occasionally with instrument check sources at these sites.

In one program, both a vendor calibration tag and a QA test tag must be attached to portable instruments before they can be used. The additional QA tag ensured instrument operability upon return from the vendor. The remaining sites relied more heavily on the vendor's calibration and gave the instruments only an occasional response check with a sub-microcurie source.

Only one site had established a formal management review of its QA program. The remaining two sites performed infrequent and informal reviews. All sites eventually reviewed procedures, ALARA evaluations, training programs, and work recently performed. The management of all sites appeared to be responsive to adverse findings of the QA program, as required by procedures. In short, two sites had a somewhat less than desirable QA program. This may be attributed to either a smaller operation or to a lighter commitment on the licensee's part. These two sites did not have a QA department or employees dedicated to QA functions.

4.7 DOSE REVIEW AND RECORD SYSTEM

Incoming dose data was reviewed by the Radiation Safety Officer as it was received from the vendor. Dosimetry reports were then passed on to the designated management personnel with the RSO's comments. Reports were reviewed at the same frequency as they were received from the vendor, usually on a monthly or quarterly basis. Trends were plotted at two of the sites and used for feedback in the ALARA review. The third site was similar, but the data was only tabulated, not plotted. One site reported inconsistencies in the dose plot and included reasons for higher or lower than expected values. Recommendations for future reduction of doses were also included. Another site plotted the doses on a superimposed graph of anticipated doses (see Figure 1) that reflected reductions towards its ALARA goals. At all three sites, the dosimetry reports were evaluated against 10 CFR 20 limits, administrative limits, and ALARA goals.

FIGURE 1. Projection of Dose Reduction Goals

Exposure discrepancies between whole-body dosimetry devices (i.e., between pencil dosimeters and TLDs) were reviewed by the RSO in all cases. This review was performed to ensure that different dosimetry methods were in fact reasonably agreeable for identical exposure conditions. There was a distinct trend for the licensee to suspect a pencil dosimeter when its readings conflicted with a film or TLD. This can probably be attributed to previous experience involving the shortcomings of the pencil dosimeter, such as continuous charge leakage or complete discharge due to improper handling.

The RSO was responsible for dosimetry records at two sites, while the remaining site employed a dedicated records clerk and several assistants. All of the dosimetry records observed were complete and up-to-date. Again, from what was observed, this information is disseminated in a timely manner. Procedures for records handling (e.g., record pathway, required review, required signatures, filing, etc.) were observed at only one site.

4.8 EXPOSURE CONTROLS AND REDUCTION METHODS

Formal written procedures at the sites included administrative action levels and the appropriate response action when employees approached or exceeded preset dose levels. Administrative action levels were limits placed by management on the radiation doses that individuals could receive over a period of time. Radiation doses were set lower than the regulatory limits to ensure that they would not be exceeded. The regulatory limits, as used by the State licensing authorities, were the same as those given in 10 CFR 20 and are listed in Table 3. The basis for these numbers are also given in 10 CFR 20.

Two sites set action levels for whole-body doses that were somewhat less than the regulatory limits of 3 rem per quarter. These were 200 and 400 mrem per month. The remaining site incorporated the regulatory limits of 10 CFR 20 into its administrative action levels. Past experience indicated that the administrative action levels were rarely, if ever, exceeded at any site. No case was observed where 10 CFR 20 dose limits for whole body, skin, or extremities were exceeded.

TABLE 3. 10 CFR 20.101(a) Whole-Body Dose Limits

Quarterly limit	1.25 rem or 3 rem, as long as lifetime limit not exceeded
Lifetime limit	5(N-18) rem ^(b)

(a) 10 CFR dated January 1, 1982.

(b) "N" equals the individual's age in years at his/her last birthday.

Response actions for employees receiving higher than normal doses were rather consistent from site to site. Cases that approached action levels were

simply noted for future reference. Employees who exceeded action levels faced two possible alternatives, dependent upon the severity of the dose: 1) rearrangement of work assignments within the employee's work group so that higher dose rate areas were avoided, or 2) reassignment of the individual to another area or job function where no radiation exposures were expected. The action of reassigning a worker usually triggered an investigation into what was causing the higher doses.

To some extent, all sites attempted to equalize doses to persons within each job category, while still trying to decrease the average dose. The most accepted method to achieve this result was through work assignments. Persons collecting a significantly higher dose than the group-dose-average during the first month or two of a quarter were reassigned to different low-exposure rate jobs for the remainder of the quarter. The use of this method in conjunction with ALARA methodologies has apparently contributed to the success of exposure control at all three sites. In general, procedures reflected existing regulations and an honest attempt to fulfill the ALARA concept was apparent.

All sites reported effective access controls to prevent unnecessary exposures onsite and intrusion into the site. Two sites displayed rather lax security at the site boundaries by not having any surveillance or patrol; however, because of their isolated location this has not been a problem. All sites were surrounded by a chain link fence of at least eight feet in height.

These three sites had only one access point for workers to enter and leave the controlled areas (see Chapter 9.0). This method allowed strict control over personnel who were permitted to enter the site and gave assurance that exiting personnel would pass by or through a monitoring station. Only one of the sites visited had a full time guard on duty at the site access point. However, because of site design and the commitment observed in isolating waste from the work environment, contamination control has not been a problem at the sites.

When handling radioactive wastes, all sites used shielding and remote handling tools, where possible. The heavy equipment used in hauling and placing waste containers appeared to offer excellent shielding to operators and support personnel. Temporary shielding, such as lead sheets, were available but used infrequently. The reason given was that placing the shields would lengthen the time of operation and increase the number of persons exposed. Licensees stated that the use of lead sheets may result in a lower average exposure, but the man-rem figure would be increased.

When abnormally high hazard wastes (e.g., spent resin wastes in casks or higher level TMI cleanup wastes) were scheduled to arrive, two sites performed rehearsals on radwaste movement and handling. Arrival of higher than normal activity wastes requiring special precautions appeared to average about one shipment every month or two. The rehearsal allowed problems to be worked out in a safe environment and also gave the Health Physicist a good estimate of

the time and distances involved in the operation. From this, he could calculate dose commitments for the operation and determine what shielding and time limits were needed.

Most waste disposal operations were controlled by standard operating procedures. Those operations that were not, were covered by a Radiation Work Permit at two of the sites. At a minimum, this permit contained information on the work to be performed, the radiation fields encountered, time limits if needed, shielding, protective clothing, and the appropriate approvals. The third site did not use Radiation Work Permits; however, work performed outside the scope of procedures was under the supervision of Health Physicists.

5.0 INTERNAL EXPOSURE CONTROL

The internal exposure control programs of the three low-level waste (LLW) disposal sites were reviewed. It was observed that the programs were well established and generally adequate in scope.

The predominate mechanism for the receipt of internal exposure was observed to be the inhalation of airborne radioactive material. Generation of airborne radioactivity could result as a consequence of two different conditions. The first would be the receipt of a shipping container with loose surface contamination on its exterior. The other condition would result from the rupture or leakage of a shipping container, while being handled at the LLW disposal site. This in turn, could result from either dropping the container during emplacement in the waste trench, or by the buildup of pressure inside the container due to the effect of heat (particularly on free standing liquids).

The potential for personnel receiving internal exposure, from the mechanisms of ingestion and absorption (through the skin), was determined to be low unless waste containers were opened for regulatory inspections. This low potential was based on the fact that the opening of waste containers was performed infrequently. In addition, at one site a facility with a HEPA filtered fume hood had been built especially for the opening and inspection of waste container contents (see Chapter 6.0).

It should be mentioned at this point that even though the potential for inhalation was greater than for ingestion or absorption, the overall potential for receipt of internal exposure at LLW disposal sites was very low. This is based on the distance of the workers from the waste, the large dilution factor resulting from the outdoor operations, and the packaging of LLW to prevent release of radioactive material.

5.1 RESPONSIBILITY

The internal radiation exposure control programs at the three licensees consisted of the dosimetry, exposure review, exposure limitations, and quality assurance programs as they related to internal exposure control. The Radiation Safety Officer (RSO) at the licensees was formally given the overall responsibility for internal exposure control. However, it was observed at one facility that this responsibility had been turned over to the chemistry laboratory supervisor on a day-to-day basis.

5.2 INTERNAL DOSIMETRY

The internal dosimetry programs included biosurveillance, timekeeping and exposure records systems.

5.2.1 Biosurveillance

The biosurveillance methods used by the licensees were gamma lung scans, thyroid counting and urinalysis. Baseline gamma lung scans were performed at all sites within three months of employee hire, annually thereafter and upon employee termination. It was observed that all radiation workers and selected support personnel were covered by this program at the three sites. At two of the sites, the lung scans were performed by a vendor using a mobile unit onsite which was capable of both collecting and analyzing data. The other licensee had an onsite facility which consisted of a counting chair and multiple detector arrangement as shown in Figure 2. This facility was capable of performing both gamma lung scans and thyroid counting. The subject to be monitored was required to sit in the chair which was then moved forward so that the detectors and collimations would be in the appropriate counting positions. At the time of the site visit by PNL, one of the two sites that used vendor services had just completed constructing an onsite facility having these capabilities.

FIGURE 2. Chair Counting System Using Multiple Detectors

A calibration phantom had been used routinely at one site for the facility shown in Figure 2. However, the licensee had not documented the results. It is not known whether the two vendor facilities were calibrated with a phantom or any radioisotope sources. The radioisotopes that represented a major portion of the radioactive material inventory in the waste and were most readily detectable were ^{58}Co , ^{60}Co and ^{137}Cs .

Thyroid counting was observed to be performed at only two sites. Both sites had baseline surveys performed within three months of employee hire. The frequency of survey was monthly at one site and annually at the other. The major isotope of concern was ^{125}I . Thyroid counting at employee termination was observed to exist in some cases, but was not consistently performed.

Normal counting times ranged from 5 to 10 minutes each for lung scans and thyroid monitoring. The minimum detection levels for individual radioisotopes were observed to be about 1% of a maximum permissible organ burden (MPOB). Action levels requiring investigation by the Radiation Protection Department, at all three LLW sites, ranged from 5 to 10% of a MPOB.

Bioassay analyses may be performed using urine, fecal or blood samples. The three licensees had only urinalysis performed on a routine basis. This analysis was done by a separate vendor for each site. In all cases, a baseline was performed within one week of employee hire for tritium, gross alpha and gross beta (excluding ^{40}K). The frequency of survey, thereafter, was performed on an annual basis at two sites and quarterly at the third site. All three sites performed urinalysis upon employee termination. Normally reported urinalysis results ranged from nondetectable to 25 dpm/ml for tritium, gross beta and gross alpha. One vendor also used fluorometric analysis (uranium) on urine samples, with the typical levels showing nondetectable ($<2 \mu\text{g}/\text{liter}$).

5.2.2 Timekeeping

To supplement biosurveillance, the LLW disposal sites planned to use timekeeping in conjunction with air sample results to calculate maximum permissible concentrations (MPC-hours). This was accomplished by dividing the air sample concentration data by the appropriate value given in Appendix B of the Code of Federal Regulations, Title 10 - Energy, Part 20 - Standards for Protection Against Radiation (10 CFR 20). The resulting number was then multiplied by the exposure time in hours to yield an MPC-hour value. The results of these calculations would be used to determine the need for additional biosurveillance. However, it must be noted that the air sample results measured at all three LLW sites were normally less than background values. This was based on the location of the air samplers (discussed in section 7.2.3) used to monitor the workers unloading waste shipments. Thus, the air sample data did not provide effective input to the planned time-keeping system.

5.2.3 Exposure Records

An individual exposure record file existed for each staff member at all three licensees. The files contained external exposure data, the results of all biosurveillance, and any calculations used to determine MPC-hours for suspected inhalations or ingestions. Standardized NRC-4 and NRC-5 forms, personnel decontamination reports, investigation reports and baseline data were also contained in the individual's file. The records were found to be up-to-date, the data formalized and potential overexposures well documented with detailed investigations.

The routine training programs (as discussed in Chapter 3.0) established by the three sites, for the workers evaluated by the internal dosimetry program, were found to be adequate. The basis for various types of sampling, frequency of sampling and biological pathways were discussed during routine training. In addition, the proper collection techniques, especially for urinalysis, were also covered with the workers.

5.3 EXPOSURE REVIEW PROGRAM

The internal exposure review program consisted of written procedures establishing licensee policy at the three LLW disposal sites to ensure that exposures were ALARA. It was observed that all the sites had up-to-date, formalized procedures for the internal exposure control program. The procedures encompassed sample collection and analysis, frequency of biosurveillance, record maintenance, action limits and handling of contaminated or irradiated injured personnel. However, the computation of dose commitments from activity or concentration data was not outlined in the procedures nor was the calculation performed. Data evaluation stopped short of applying dose models as referenced in NRC regulatory guides, ICRP publications and NCRP reports. The licensees were observed to have the above applicable reference documents in their possession.

The RSOs at all three licensees reviewed the exposure records on a routine basis against 10 CFR 20 limits. All suspected inhalations or ingestions were investigated by the RSO and resampling (by additional biosurveillance methods) was performed upon the RSOs advice. Trend analysis was an important tool in the LLW external exposure control programs to maintain exposures ALARA. This was observed not to be the case for internal exposure. Correlations between the following items were not evaluated routinely by any of the licensees:

- work activities
- air sample data
- bioassay data

5.4 EXPOSURE LIMITATIONS PROGRAMS

The internal exposure limitations program consisted of radiological posting, uptake limits, engineering controls, access controls and protective clothing including respiratory protection. Because of the limited area of potential internal exposure (trench area), the three licensees maintained an adequate limitations program by supplying health physics coverage almost continuously. In addition, detailed procedures and training enhanced the program. More detail on training and respiratory protection can be found in this document in Chapters 3.0 and 6.0, respectively.

5.5 QUALITY ASSURANCE PROGRAM

The internal exposure quality assurance program consisted of the calibration of portable survey instruments and lung/thyroid counters and reliability checks on data received from vendors. The discussion of portable survey

instruments is presented in Chapter 7.0 of the document. As indicated earlier, one of the three sites is now calibrating their own lung/thyroid counter with a radioisotope phantom. The sources used in this calibration are traceable to the National Bureau of Standards (NBS). The other two sites, however, have accepted the vendors data on faith. In addition, all three licensees accept the vendors urinalysis data without performing reliability checks, such as: radioisotope phantom checks on licensee and vendor lung/thyroid counters, radioisotope spiking of urine samples or the splitting of individual urine samples to separate analysis laboratories.

Instrument is presented in Chapter 7.0 of the document. As indicated earlier, one of the three sites is now calibrating their own (non-thyroid) counter with a radiostandard phantom. The sources used in this calibration are traceable to the National Bureau of Standards (NBS). The other two sites, however, have accepted the vendor's data without performing reliability checks, such as: radiostandard phantom checks on licensed and vendor (un) licensed counters, radiostandard spiking of urine samples or the spiking of individual urine samples to separate analysis laboratories.

6.0 RESPIRATORY PROTECTION

One of the major pathways for the uptake of radionuclides by individuals is by inhalation (see Chapter 5.0). An effective method for minimizing inhalation of radionuclides by workers, where airborne radioactive material is present, is by the use of respiratory protection.

The potential for high concentrations of airborne radionuclides at commercial low-level waste sites is low. Dispersion of radioactive gases and particulates, resulting from container rupture, would be expected to show significantly reduced concentrations of airborne radionuclides in an open air environment. Historical data from air sampling and biosurveillance programs at the sites supported this opinion; however, all three sites had chosen to use respiratory protection as a precautionary measure, which is consistent with sound ALARA principles.

6.1 PROGRAM ESTABLISHMENT

Respiratory protection programs were reviewed at the three commercial low-level waste sites. Each site appeared to have adequate provisions for the protection of personnel against the inhalation of radioactive gases and particulate material. Normal operations at the sites did not require the use of respiratory protection; therefore, the degree of readiness to cope with abnormal circumstances, such as ruptured or suspected leaking containers, was of primary interest. Respiratory protection was being worn when containers were opened for inspection and were also available for use at all active burial trenches. Personnel performing unloading operations had been informed to use these respirators immediately, if a leak was suspected or when instructed by health physics (HP) personnel. The respirators were to be used while evacuating the immediate problem area and for the duration of potential inhalation of radionuclides. According to discussions with HP personnel at all sites, specific circumstances would dictate whether immediate evacuation was necessary. Respiratory equipment requirements were to be ascertained by the RSO or his delegated representative; however, the use of respirators could be authorized by HP personnel at all three sites.

6.1.1 Responsibility

It was noted that the ultimate responsibility for ensuring an adequate respiratory protection program rested with the RSO at two of the sites and with the manager of the Health and Safety Department at the third; although, responsibility for the preparation and implementation of procedures remained with the Health Physics Department.

6.1.2 Written Procedures

Although respiratory protection programs were in place at all three sites only one had a formalized written procedure dedicated specifically to respiratory protection. This procedure provided for instruction in the use, fitting, and maintenance of each type of respirator that was used onsite. A

certificate from a medical doctor who was approved by site management (see section 6.1.4) was a prerequisite for wearing respirators, as was the training in their proper use. Facial seal, maximum continuous wearing time, entrainment curves and action levels were also addressed in the procedure. Maximum continuous wearing times can be found in section 6.1.4. Entrainment curves were keyed to air sample data through maximum permissible concentration formulae. Action levels were keyed to 10 CFR 20, Appendix B, Table 1, Column 1 and a corresponding state regulation.

The two sites that did not have formalized written procedures specific to respiratory protection referred to respirator usage in various radiation work procedures. The need for specific procedures was recognized by site management who indicated that revisions were presently being generated to encompass this area.

6.1.3 Program Effectiveness

The effectiveness of a respiratory protection program is keyed to historical data. The air sampling records which were reviewed revealed no case where airborne radionuclide levels approached the concentration values listed in Appendix B, Table 1, Column 1 of 10 CFR 20. The maximum permissible concentrations in air for a restricted area were compared to air sample results for the isotopes of major concern (e.g., ^{125}I , ^{137}Cs , ^{60}Co). Combined isotope (mixture) values were also compared. Biosurveillance records indicated that no instances were found where the measured organ or body burden approached the MPOB or MPBB values listed in the Tables from ICRP 2. Recordkeeping was found to be very good. One site had records dating back approximately 20 years.

The frequencies of routine biological sampling are discussed in section 5.1 of this document. Personnel at each site who were affected by a suspected airborne release of radioactivity were to receive additional whole-body, lung or thyroid counts, as well as bioassay surveillance.

6.1.4 Wearer Requirements

Onsite, hands-on training and retraining in the wearing and use of respirators were being done at the three sites. Training in the use of respirators was required at all sites before workers could enter a radiation zone. Retraining frequencies ranged from 1-2 years based upon procedural requirements, but records indicated that retraining was being conducted much more frequently, averaging about every 6 months. Methods for checking respirator fit varied at the three sites. Negative pressure tests were common at all sites and two sites also performed smoke tests before respirator use. One site required an annual qualitative test, in which individuals entered a sealed chamber wearing respirators that had provisions for measurement of stannic chloride levels, both inside and outside of the respirator. From the ratio of the measurement levels, protection factors could be assigned to individuals while wearing that type of respirator.

Smoke tests consisted of breaking a smoke capsule near the respirator. If the wearer could smell the smoke, while wearing the respirator, the fit was unsatisfactory. One site required medical certification of the wearer's physical condition, as to lung and heart impairments, which might be adversely effected by wearing respirators. One site specified a wearing time restricted to four continuous hours or six total hours per day. Wearing times were not specified at two sites.

6.1.5 NIOSH Approved Respirators

Respiratory protection observed at the three commercial low-level waste sites (see Figure 3) consisted of:

- half-masks
- full-face canister masks
- full-face supplied-air masks.

The half-masks were being used at one site for protection against blowing sand and dust rather than for protection against airborne radionuclides. The typical full-face respirators, which were observed, accepted screw-on type canisters which, according to manufacturers specifications, were 99.97% efficient against 0.3 micron particulates. Canisters designed for protection against chemicals and vapors could be interchanged when performing related tasks. The full-face canister type was the most commonly available respirator at all sites. "Scott Air Pacs," consisting of a full-face respirator that draws breathing air from a back-pack supply tank, were available in limited quantities at each site. These systems were to be used when airborne radionuclide concentrations were high. All respirators, that were observed, were approved by the National Institute of Occupational Safety and Health (NIOSH).

6.2 ENGINEERING CONTROLS

Engineering controls applicable to respiratory protection at LLW sites are discussed in this section.

6.2.1 Ventilation and Containments

Engineering controls, such as fume hoods, are not generally applicable to outdoor operations. Therefore, few controls should be expected at the low level waste sites, except where work such as opening containers would be performed. One of the three commercial sites had a new facility which was constructed for this purpose. This system was designed to exhaust air through HEPA filters and contained facilities for remote "off-gas" sampling and counting within a closed system (Figure 4).

6.2.2 Alarm Systems

An alarm to indicate high radiation levels was being used at one site. The detector was positioned at the trench lip near the burial operation. The alarm was preset at approximately 400 mR/hr. No alarms to warn of high

Full-Face Respirator

Half-Mask

FIGURE 3. Examples of Respiratory Protection Used at Low-Level Waste Sites

FIGURE 4. Container Opening Facility With Overhead Fume Hood

airborne radionuclide concentrations were observed. While alarm systems offer an early warning mechanism, their use at the burial trenches was complicated by the continual movement of the work area.

6.3 TRAINING AND RETRAINING

After reviewing the information which was supplied concerning respiratory protection programs and interviewing various operational personnel, it was determined that adequate training programs were in place at all sites. Further, if an abnormal circumstance should arise, the field personnel would receive additional guidance by the Health Physics Departments.

6.3.1 Instructor Qualifications

Radiation Safety Officers were instructing the training sessions at all sites. Interviews with the RSOs indicated that each had a high degree of knowledge in the area of respiratory protection. It was noted that the instructors personally demonstrated the proper wearing, use and care of each respirator type.

6.3.2 Retraining

All sites had ongoing training and retraining programs. One site held four-hour weekly training sessions which were diversified to cover all

operational areas. The proper use of respirators was one element of the program. Training and retraining appeared to be adequate at all sites (see section 6.1.4).

6.3.3 Scope/Contents

Documented training sessions covered a variety of subjects. Simulated field training was covered, along with the operation of the respiratory devices and interchanging of the cartridges for different purposes. Minor repair, cleaning, and disinfection of respirators was also addressed. No specific weakness was found in the respirator training and retraining programs at any of the three sites. Much of the training was "hands-on." Information concerning the training session content was obtained from interviews with the site RSOs and/or managers. Documentation containing training dates and training subjects was observed at all three sites.

6.3.4 Fitting

Typical field checks for proper respirator fit were covered in the training at all three sites. Full-face canister masks were field tested primarily by sealing the filter cartridge's air entrance with the hand, thus blocking the wearer's air intake. This test was quick and simple and could be conducted without additional equipment (e.g., smoke tubes), in case of the inadvertent need for respiratory protection. Facial hair, such as heavy beards, will many times hinder the proper fit of respirators. This aspect was considered at all three sites. Smoke tests for supplied-air respirator fit were being conducted as part of the training programs, and before field use. Fitting tests were repeated on an annual basis at a minimum.

6.4 MAINTENANCE

The emphasis of the maintenance programs for respiratory protective equipment was on the inspection and repair of the devices with subsequent documentation and issuance.

6.4.1 Inspection and Repair

Respirators which were new or had been reprocessed by a commercial reprocessor received an operational inspection and an inspection tag. Before field use, respirators were visually inspected for such things as cracked viewing shields, proper canister attachment, deteriorated component parts, and to ensure proper canister usage. Supplied-air units were checked for proper air flow and adequate air supplies. Scott Air Pacs typically had either 45- or 60-minute time duration supply tanks. One MSA forced air unit was observed; this unit had a 30-minute wearing time limit due to the life of the battery used to supply power to the air pump.

At all sites, a single full-face mask was used for not more than one full shift. It was then reprocessed (i.e., surveyed for radioactive contamination, cleaned and sanitized). All sites reprocessed full-face respirators through

an offsite commercial facility; however, two sites first attempted onsite cleaning and sanitizing of respirators by using electric dishwashers and/or a commercial product known as "MSA Sanitizer." If this effort was not successful, commercial reprocessing was then done.

6.4.2 Records

Documentation of respirator maintenance and use was being done at all sites, when they were worn for protection against radionuclides. One site allowed the use of half-masks for protection against blowing dust and dirt. In these cases, documentation was not done but individuals were responsible for maintaining the respirators after use.

6.4.3 Issuance

Issuance of respirators was under the authority of health physics supervisors or radiation safety officers at all sites. Authority was delegated to the operational health physics technicians to require personnel to wear respirators if needed during burial operations. Contact was maintained via radio between the RSO/HP supervisor and the field HP at all sites.

One site had a very firm policy concerning respirator usage and return after use. Personnel who did not conform to policy were liable for time off without pay.

6.5 SUPPLIED-AIR RESPIRATORS

All sites had supplied-air respirators available for emergencies. Scott Air-Pacs were predominate for this purpose. Limited quantities of supplied-air respirators at two sites caused some concern in case of an unusual circumstance. No systems were observed which supplied air to respirators from remote air pumps or centralized systems.

an effort to convert facility; however, two sites first attempted onsite cleaning and sanitizing of respirators by using electric dishwashers and/or commercial product known as "USA Sanitizer". If this effort was not successful, commercial reprocessing was then done.

6.4.2 Records

Documentation of respirator maintenance and use was being done at all sites when they were worn for protection against radioactive dust. The site allowed the use of half-masks for protection against blowing dust and dirt. In these cases, documentation was not done but individuals were responsible for maintaining the respirators after use.

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7.0 SURVEILLANCE

Ongoing monitoring of radiological conditions at the three low-level waste sites was the key to maintaining personnel exposure ALARA. Trends had been established from historical data, which offered insight into the need for improved operational techniques, thus resulting in the reduction of personnel exposure. For example, dosimetry records at one low-level waste site had indicated that the fork lift operators, who unloaded waste barrels, had consistently received relatively high penetrating doses. Lead blanket shielding was positioned on the fork lift resulting in reduced exposure to the operator. Measurements from portable radiation detection instruments frequently reflect a need to reduce personnel exposure while performing certain tasks, especially where ongoing surveillance records show consistently high radiation levels.

Time, distance and shielding were found to be the primary dose reduction methods used at all sites, which is consistent with ALARA principles. Routine portable instrument surveys for detecting surface contamination in work areas, on clothing, tools and equipment were also being performed to help control or prevent the possible spread of radioactive contaminants which may eventually come in contact with personnel.

Monitoring programs aimed toward providing protection against the inhalation and ingestion of airborne radionuclides included air sampling and biosurveillance (e.g., as bioassay, lung, thyroid and whole-body counts) at all sites. These methods provide indicators that are after-the-fact of an exposure; thus, the data could only be used for dose assessment and to aid in evaluating operating techniques. Preventive measures were best provided by the use of respiratory protection. The implementation of this ALARA principle relied heavily upon the judgment and foresight of field monitoring personnel.

Other ongoing monitoring programs, such as the sampling of ground water, soil and vegetation, and environmental air samples were found to be only indirectly linked to ALARA programs for site personnel.

7.1 SCOPE

Surveillance programs at all sites were found to be adequate in scope. Some portions of the monitoring program may be improved, however, by a more intensified effort (e.g., placement of burial trench air samplers to better represent breathing zones of workers).

7.1.1 Procedures

Procedures keyed to the various ongoing activities were reviewed at each site. All sites operated under a set of procedures which had been drafted by the respective corporate offices to meet the intent of 10 CFR, 49 CFR, and state regulations. Additional site specific procedures were also in place. The site specific procedures addressed day-to-day operating techniques, while the corporate procedures referenced the federal and state regulations from

which dose and contamination limits were taken (mainly 10 CFR 20, 49 CFR Parts 100-199 and South Carolina Regulation 61-63, Title A).

It was noted that procedural updates were documented and were being done as needed. Procedures were found to be adequate for current operations at one site. Corporate procedures were being updated at the other two sites to include a specific respiratory protection procedure which was lacking; these procedures should then be adequate.

7.1.2 Training and Retraining

The general training programs for the site radiation workers are covered in Chapter 3.0 of this document. More specifically, the training and retraining of radiation technicians or radiation monitors were found to be extensive at all sites. Two of the sites required a written exam and an additional oral exam (the oral exam was administrated by the corporate management and managers of the company's other sites).

One site had initiated a unique program to train radiation workers, not normally assigned as radiation monitors, to qualify as "routine monitors." Upon qualifying as a routine monitor, these personnel could perform radiation monitoring duties on a routine basis as long as specified exposure and contamination levels were not exceeded. These limits were:

Exposure rates not exceeding

- 500 mrem/hr, whole body
- 1000 mrem/hr, skin
- 5000 mrem/hr, extremities

Contamination levels not exceeding

- 5,000 dpm/100 cm², alpha
- 50,000 dpm/100 cm², beta-gamma

7.1.3 Records

Record keeping was found to be a strong point in the surveillance programs at all three sites. Detailed records which included routine and special surveys, air sample data, personnel exposure, smear sample data, trench inspection and daily dose levels, instrument calibration, cask inspections, shipments, and many other categories were reviewed.

7.2 TYPES OF RADIATION

Radiation types as discussed in this section refer to three categories:

- external exposure
- surface contamination
- airborne contamination.

7.2.1 External Exposure

Discussion in this category is limited to exposure from external sources of ionizing radiation. At the low-level waste sites, common external sources of ionizing radiation included single and composite barrels (drums), wooden crates and shipping casks; all contained radioactive waste. The principal thrust of a radiation protection program should be aimed toward maintaining personnel exposure ALARA. Protective measures leading to this involve many parameters. As mentioned in Chapter 7.0, the key factors in reducing radiation exposure to personnel are time, distance, and shielding. Job planning, prejob rehearsal of high exposure tasks, teamwork, experience, attitude, and safety are other important dose reduction tools. In general, each site was found to be using sound radiation protection techniques and practices. Exposure limits per time interval which were allowable at each site were set at or below the values contained in 10 CFR 20.101 limits. Attempts were being made to distribute dose among personnel where practical.

The sequence of events from the time of shipment arrival until the shipment was deposited in the burial trench was much the same at all three sites, although the barrel unloading techniques varied. Surveys of arriving shipments were made almost immediately when arrival occurred during normal operating hours (Figure 5).

FIGURE 5. Survey of Incoming Shipment

The initial survey information was used to evaluate both exposure rates and contamination levels. Each shipment was followed by radiation protection technicians, as unloading operations proceeded (Figure 6). Typically,

FIGURE 6. Monitoring of Personnel During Preparation for Cask Unloading

constant monitoring was being done as the barrels or casks were placed into the trenches (Figure 7).

Exposure rates at the edge of the burial trenches were frequently measured (Figure 8). Cask unloading required riggers and radiation protection technicians to be in close proximity to the cask for periods of several minutes (Figure 9).

Typically, the crane operators (classified as trench workers at one site) received a high amount of exposure because they were close to, and not shielded from, the contents of the trench (Figure 10). Where cranes were not used for unloading, forklift operators were in a vulnerable position for receiving exposure (Figure 11).

7.2.2 Surface Contamination

Historically, the records show that contamination from radionuclides has not presented a major problem at the three waste sites. The potential for radioactive contamination is the over-riding factor and surveillance programs were noted as being pointed toward the potential hazards. Never-the-less, routine surveys of personnel, equipment, and work areas are the indicators of a contamination problem and this surveillance was being done extensively at all sites.

FIGURE 7. Monitoring of Personnel During Unloading of Barrels

FIGURE 8. Exposure Rate Monitoring at Burial Trench Edge

FIGURE 9. Relative Position of Personnel to Cask During Unloading Operation

FIGURE 10. Relative Position of Crane Operator to Contents of the Trench During Burial Operation

FIGURE 11. Relative Position of Forklift Operator to Waste Container During Unloading

Routine surveys of personnel were conducted periodically as a job progressed (where low radiation background shelters were available) and always when exiting a radiation (controlled) zone. Equipment surveys were handled in much the same manner. Surveys of work areas were being done on a periodic frequency, based on each area's potential for contamination (e.g., office areas were being surveyed every 6 to 12 months, while buildings within the controlled area were surveyed monthly).

Contamination surveys followed the guidelines of NRC NUREG 1.86 and posting of restricted areas followed 10 CFR 20.203 at all sites. Because most vehicle and cargo surveys were DOT related, the prescribed contamination limits of 49 CFR 173.397 were followed at all sites. Survey records indicated that shipments were being monitored for both contamination and exposure rates upon arrival and the trailers were again surveyed after unloading. Continuous monitoring was being done at the operational trenches. Personnel, clothing, tools, respirators, etc., were surveyed within shielded hutments near the trench work areas (Figure 12), because of the elevated radiation background levels at the trenches. This shielding reduced background levels to allow meaningful surveys. "Swipes" or smear samples of equipment in the high exposure areas could also be checked remotely within the shielded hutments. Personnel, tools, and equipment were again surveyed when exiting the restricted or controlled areas. These surveys were performed with portable Geiger-Mueller instruments and portable scintillation type alpha meters. Contamination surveillance was well covered in the procedures at all sites.

FIGURE 12. Personnel Frisking Station

7.2.3 Airborne Contamination

There exists a potential for airborne contamination from radionuclides at low-level waste sites due to the possible rupture or degradation of the integrity of transport/burial media. Although air sampling equipment was being used at all sites (Figure 13), placement of the collection media (i.e., sample filters) at representative breathing air locations was found to be deficient. It should be noted that placement of the filters at a point which would be representative of all potentially affected personnel is virtually impossible; therefore, the "worst case" or closest human access point should be monitored, where possible. In addition, the level of the sample media should closely resemble that of the breathing zone of a person working in the area. Physical restrictions, which may alter free air flow to the sample media, must also be avoided (e.g., enclosures, walls).

FIGURE 13. Air Monitoring Station

Air sampling equipment which was observed was typically located on the edge of the burial trench and down wind to prevailing wind patterns. Arrangement of the sample media ranged from levels approximately 18 inches from the ground to a height of about 5 feet; however, the sample head was positioned inside of the encasement hutch housing the system. Frequent movement of the sample apparatus dictates portability, however care should be taken to collect representative samples. Samples collected at 18 inches above the ground are not representative of the human breathing zone, nor are samples collected within an enclosure. Collection time should cover the total time period of the operation being monitored and flow rates through the sampling system should correspond favorably to the breathing rate of a standard man while working (i.e., 20 to 30 ℓ /min, ICRP Publication Number 23). Flow rates of air samplers which were observed were typically set at about 30 ℓ /min and covered the total time of operation.

The perimeter of one site was well surrounded by continuous environmental air sample collectors. These were placed at representative breathing zone levels.

Individual, lapel-type air sampling units were on hand at one site but these units are not practical for constant usage at this type of operation, except for providing time/motion information. High volume, portable sampling equipment was also available for use at all sites.

7.3 INSTRUMENTATION

The radionuclides expected to be most prevalent at the low-level waste sites were cobalt-58, cobalt-60, cesium-137 and iodine-125. Table 4 lists these isotopes, the principal radiations associated with each, their energies and the percent of total decay events. Also listed are the types of portable instrumentation used for detecting each.

A variety of radiation detection instruments were observed at the three commercial low-level waste sites. Portable, semifixed, and fixed position instruments were being used as conditions warranted. No deficiency was apparent in the types of instruments needed to accomplish the various measurements. Energy-compensated Geiger-Mueller (GM) detectors were favored at two sites for exposure rate meters, while one site seemed to favor the ionization type instrument. The ruggedness of the GM detector was opted for rather than the more fragile ionization chamber instrument for this purpose. It should be noted that careful attention should be given to the energy response limitations of GM type units for establishing exposure rates. This is because unshielded GM tubes typically over-respond to photon energies below 0.2 to 0.3 MeV. To compensate for this over-response, a material of sufficient thickness and density is added (surrounding the GM tube) to flatten the response curve in the low-energy region while maintaining unity between the actual exposure rate and the instrument's response in a known upper energy radiation field. An under-response to photon energies below a few hundred KeV may then result. Typical photon calibration sources are cesium-137 (0.662 MeV) and cobalt-60 (1.250 MeV).

7.3.1 Inventory

Inventories of instruments appeared to be adequate at all three sites. Several models of portable instruments were available for use, as were backup instruments of each model. A total count of portable instruments was not determined, but the numbers ranged from two or three of each model at two sites up to several of each at the third site. The instruments included alpha scintillation, GM, ionization chambers, and neutron detectors. High level gamma detectors, as well as instruments which respond to micro-R levels of photon radiation, were also on hand at all three sites. Inventories appeared to provide adequate instrumentation at the sites even during periods of reduction due to calibration and repair.

7.3.2 Calibration

Records and calibration stickers at all sites revealed that no portable instruments were overdue for primary calibration. Primary calibration frequencies of six months (quarterly for some models) were standard at all sites. These calibrations were being conducted by independent calibration laboratories, who followed ANSI N323-1978 criteria.

Onsite response checks were being conducted daily or before instrument use by using check sources of known activities. Each site had procedures in place for this task.

TABLE 4. Predominant Radionuclides

Isotopes ^(a)	Major Radiations	Energies (MeV)	Percent of Total Decay Events	Portable Detection Instrumentation
Cobalt-58	β^+	0.474 maximum		GM
	γ	0.511	30%	GM--Ionization Chamber
		0.818	99%	GM--Ionization Chamber
		0.865	1.4%	GM--Ionization Chamber
		1.67 (Fe x-rays)	0.6%	GM--Ionization Chamber
Cobalt-60	β^-	1.48	0.12%	GM--Ionization Chamber
		0.314	99+%	Not Detectable ^(b)
	γ	1.173	100%	GM--Ionization Chamber
		1.332	100%	GM--Ionization Chamber
Cesium-137	β^-	1.176	7%	GM--Ionization Chamber
		0.514 maximum		GM
	e^-	0.624, 0.656		GM--Ionization Chamber
	γ	0.662 (Ba x-rays)	85%	GM--Ionization Chamber
Iodine-125	γ	0.035 (Te x-rays)	7%	GM--Ionization Chamber
	e^-	0.004, 0.030		Not Detectable ^(b)

(a) Isotopic information from "Table of Isotopes," by C.M. Lederer, J. M. Hollander, and I. Perlman, 1967.

(b) Energies are below the levels detectable by thin window GM tubes and ionization chambers used for field measurements.

7.3.3 Radiation (Exposure Rate) Instruments

As mentioned in section 7.3, beta-gamma exposure rate instruments were of the ionization-chamber and energy-compensated GM types. Area radiation monitors were not being used. All sites had portable instruments available that were capable of measuring exposure rates from beta, gamma, and neutron sources. No areas of deficiency were noted.

7.3.4 Airborne Radiation Detection Instruments

Portable air sampling pumps were being used at all sites to collect air samples. Fixed position continuous air samples were being collected at all sites, in addition to those being taken close to the burial operations. Section 7.2.3 mentions the air sample stations on the perimeter of one site which operated continuously. One of the other sites had a fixed position continuous air sampler at a point downwind (to prevailing wind) and within a few hundred feet of the trench operation point. The third site had two continuous air samplers, one in the maintenance shop and one near the liquid waste tanks. Sample collection periods of 24 hours were common at all sites for continuous air samples.

Constant Air Monitors (CAMs), which sample the air continuously while monitoring the radiation level of the sample, were not being used at the sites. All sites did have capabilities to perform onsite isotopic analysis of air samples and each site had agreements with offsite independent laboratories to provide backup service, if needed.

7.3.5 Contamination (Count Rate) Instruments

Portable count-rate meters for performing contamination surveys included beta-gamma and alpha detectors. Pancake-type GM probes were predominately used for beta-gamma; alpha probes were mainly the scintillation-type.

"Friskers" or instrumentation used to perform personnel surveys consisted of semifixed (bench-type) electronic packages with shielded (open window) pancake GM probes attached as the detector. A portal monitor was being used in addition to the frisking unit at one site. The ability of this unit to detect contaminated spots on personnel is largely dependent on the source to detector geometry. Thus, while they are helpful in detecting grossly contaminated individuals, they may not detect a single radioactive particle on a person. This unit detected only penetrating radiation.

Another site used a hand-and-shoe counter in conjunction with the frisking unit. This unit detected alpha at a sensitivity of 30 picocuries (pCi) in 12 seconds and beta-gamma at 300 pCi in 12 seconds.

When portal monitors or hand-and-shoe counters are used in conjunction with frisking units, the probability of detection of contamination is high. When used alone, they are useful primarily as indicators and follow-up surveys with portable instruments are necessary to define specific areas of contamination.

7.4 ANALYTICAL INSTRUMENTATION

Germanium-lithium and sodium-iodide detectors interfaced with multi-channel analyzers and computers provided isotopic determination capabilities. Gas proportional counters were used for counting air samples. Two sites had liquid-scintillation counters to count low-energy beta emitters such as carbon-14 and tritium.

The above instrumentation is considered to be state-of-the-art equipment for applications at the low-level waste sites where information is needed relatively quickly. Sensitivity levels and counting efficiencies varied with each system and sometimes between instrument checks. It was noted, however, that documentation of these changes was being recorded at each site.

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8.0 RADIOACTIVE WASTE MANAGEMENT

Management of radioactive wastes was the primary function of all the waste sites visited. Since waste operations led to the major portion of employee contact with radioactivity, dose saving efforts have the potential of being most successful in this area. Throughout this chapter, refer to Figure 24 (p. 60) for details on the general site layout.

8.1 RESPONSIBILITY

In all cases, the site manager was ultimately responsible for waste management. His approval was normally required before major facility, equipment, or procedural changes were implemented by the person responsible for the ALARA program. Communication and cooperation between the two positions appeared to be adequate at all sites. This judgment was based on conversations with appropriate persons and a review of previous dose reduction efforts.

8.2 TYPES OF RADIOACTIVE WASTES

No site accepted radioactive waste in a free-standing liquid form. State licenses required liquids to be either contained in an absorbent medium or solidified with a maximum of 0.5% to 1% (by volume) free-standing liquid. Vermiculite and diatomaceous earth are examples of absorbent materials allowed by license conditions; cement and asphalt were acceptable solidification media.

Sites were authorized to receive some radioactive gases. Krypton-85 and Xenon-133 were accepted at two sites after strict licensing criteria were met. License conditions specifically call for 1) gases to be contained in a U.S. Department of Transportation approved cylinder, 2) an internal pressure not to exceed 1.5 atmospheres, and 3) an activity limit of 100 Ci per cylinder. The remaining site could also receive gaseous tritium.

With respect to radioactive waste as a solid, special nuclear material (SNM) could be received at one site but was specifically excluded at the other two. Transuranic elements (TRU), excluding smoke detector materials, were not accepted unless the specific activity was less than 10 nCi per gram. All other solids, including source material, which met site volume, activity, weight and packaging restrictions were accepted for burial. Sites occasionally applied for variances to the above limitations.

8.3 ACCEPTANCE OF WASTES

One- to two-weeks advance site notification for incoming waste was the first step in acceptance of the waste. This allowed the licensee to schedule disposal operations and gave the Health Physicist time to work out any special procedure or shielding configuration which may be required.

All sites possessed a receiving area where waste shipments were parked until checked (Figure 14). Trucks were parked at a minimum of twenty meters from the nearest occupied building and were far removed from highways or other public access areas. Shipments arriving during nonworking hours could park in these areas; however, the drivers were held responsible for the shipment until it was accepted by the site.

All sites provided a prompt radiological survey for each arriving shipment. The survey consisted of recording the surface dose rate on all four trailer sides (or container if not enclosed), below the trailer, and at the cab in the vicinity of the driver. Figure 15 shows a survey underway. Wipe samples were taken on the outside surface of the trailer and on a few accessible barrels or the waste cask, whichever was applicable.

State licensing conditions specified the radiological limits for acceptance of waste. In all three cases, the limits were the same as those established in Department of Transportation regulations (Title 49 - Code of Federal Regulations 173.397). Specifically, fixed contamination was limited to 0.5 mR/hr at the surface, and removable contamination limits were 2200 disintegrations per minute (dpm) per 100 cm² for beta-gamma and 220 dpm per 100 cm² for alpha activity. External exposure rate maximums were 10 mR/hr at the surface and 2 mR/hr at three feet.

One site followed the radiological survey with a formal QA check. The condition of the load and the truck were judged against site acceptance

FIGURE 14. Radioactive Waste Receiving Area

FIGURE 15. Vehicle Radiological Survey

standards. Each shipment had to be approved by the QA department prior to acceptance. The other two sites included an informal QA inspection of arriving wastes as part of the radiological survey.

Documentation of each shipment was then carefully checked at all sites. At a minimum, the shipment number, activity, material, radiological survey, and waste packaging must match site documentation if the load were to be accepted. Procedures called for prompt notification of the licensing agency if documentation problems arose.

Each shipment to the sites required additional approval by a State Inspector prior to acceptance. This inspection was similar to that performed by the licensee and appeared to be a double-check of the acceptance procedure. Two states maintained resident inspectors at the sites and the third state scheduled inspector visits to coincide with the arrival of wastes.

8.4 MOVEMENT OF WASTES

At one site, the waste was taken to the trench immediately after acceptance. This was possible because of the infrequent arrival of waste shipments, which were mostly in 55-gallon steel drums.

A second site, one which received waste primarily in 55-gallon steel drums, had a trailer parking area near the controlled area. Drivers left loaded trailers there and picked up empty ones before departure. A licensee-

owned tractor would then transport the trailers into the isolation area when the trench crew called for more waste. From one to six trailers were parked in this area at all times, but none were observed to be parked longer than two days.

The remaining site received frequent shipments of radioactive wastes in casks. Cask shipments were driven to a preparation facility where a majority of the cask bolts and tie-downs were removed (Figure 16 shows the cask preparation facility which also included the equipment decontamination facility). By removing the cask bolts and tie-downs in an isolated facility, work was avoided in which personnel would be openly exposed to surrounding shipments and/or waste in the trench. The casks were then transported to the trench when called for. Figure 17 shows a radiation waste technician performing final cask preparations prior to unloading at the edge of the trench. Other shipments were transported directly from the parking area to the trench. This licensee was observed to maintain four loaded trailers near the trench. However, areas occupied by personnel were at least 40 meters from the loaded trucks.

8.5 BURIAL OF WASTES

Three different methods were employed for waste burial at the commercial sites.

One site drove loaded trailers into the trench for unloading. A small platform at the appropriate height provided a base for forklift removal of

FIGURE 16. Cask Preparation and Equipment Decontamination Facility

FIGURE 17. Final Cask Preparation Prior to Unloading at Trench

55-gallon drums (Figure 18). The drums were then placed in the trench with a light-duty crane (Cherry Picker as shown in Figure 19). Wooden crate containers were handled similarly, except that the crates were neatly stacked and the barrels were placed in random positions behind them. The crates usually presented lower external dose rate fields and thus were placed between the barrels and the operators to provide extra shielding. At this site and at the others visited, waste was required to be covered with fill dirt when either external dose rates exceeded preset limits (100 mR/hr at the trench edge) or uncovered waste extended more than 100 feet beyond the backfilled portion of the trench.

A second method utilized a flatbed trailer on the upper edge of the trench as shown in Figure 20. Barrels were removed two at a time by a forklift and placed on the open trailer. A medium-duty crane then placed the barrels, also two at a time, into the trench in random positions behind stacked wooden crates (Figure 21). Wooden crates, which were stacked neatly in the trench, required personnel to be in close contact with waste and thus allowed for increased exposures.

The third site unloaded crates and palletized barrels by a forklift inside the trench. These containers were then stacked neatly as shown in Figure 22. Each wooden pallet held four or six barrels. Cask liners and nonpalletized barrels were removed on the upper edge of the trench by a medium-to heavy-duty crane. They were then placed in back of the trench between the backfill and other wastes. Figure 23 shows a waste supervisor directing cask placements

FIGURE 18. Waste Removal Platform on Trench Floor

FIGURE 19. Waste Handling Equipment

FIGURE 20. Waste Removal Platform on Trench Edge

FIGURE 21. Random Placement of Barrels

FIGURE 22. Selected Placement of Barrels

FIGURE 23. Radwaste Supervisor and Health Physicist
Directing Cask Placement Operations

as a Health Physicist controls his exposure. License conditions call for wastes to be covered with fill dirt when either the external dose rate at the trench lip approached 100 mRem/hr, the waste height approached the height of the line drawn on the trench side (Figure 22), or uncovered wastes extended more than 100 feet beyond the backfilled portion of the trench. In actual practice, no continuous work was performed at the trench edge and licensees seldom allowed dose rates to approach 100 mR/hr at this point.

8.6 VOLUME REDUCTION

No licensee was involved in waste volume reduction operations such as compaction or incineration. Apparently, a growing number of waste shippers were providing this service by compacting or incinerating wastes.

8.7 WASTE INSPECTION

One or two barrels per week were opened at each of two sites for the sole purpose of compliance inspections. State inspectors requested these inspections to check containers which were suspected of containing liquids, TRU, or other license excluded wastes. One site had a special facility with a large overhead hood for this purpose. A strong air flow kept volatile material concentrations to a minimum in employee breathing zones. This facility is discussed further in Chapters 6.0 and 9.0. Procedures for the operation of this new facility were being developed at the time of the visit.

The second site opened barrels for inspection in the trench just prior to placement. No procedures or radiation work permits were used for this operation. A practice such as this did not appear to be consistent with ALARA, since no advance planning was done to assess and avoid external exposure from the barrel, external exposure from the trench wastes, or internal exposure from volatile or particulate materials in the barrel.

The remaining site did not open barrels for any reason.

8.8 SITE GENERATED WASTES

Very little radioactive waste was generated at the sites with the practices then in operation. One facility had an equipment sandblast decontamination facility which generated about two barrels per calendar quarter of slightly contaminated sand. These barrels were placed in the trench and buried as waste. In fact, all wastes generated at the sites were disposed of in this manner. The reason given was that all waste accepted by the site was licensed to be buried by the site.

The remaining two sites allowed decontamination water to be disposed of as normal waste water. Since all waste water was held in underground storage tanks, which were located inside the site boundary of the two isolated desert

sites, this practice should not present any problems. In addition, equipment decontamination was rarely necessary and usually involved microcurie levels of contaminants.

8.9 STORAGE OF WASTES

There were no formal provisions for the long term storage of radioactive wastes. State licenses limited site storage of wastes to a time period not greater than six months. However, two sites had stored materials or slightly contaminated equipment for months, or even years, with no definite plans for disposal. One site had a fenced and posted area where contaminated equipment and some liquid waste was stored. The second site had a large collection of slightly contaminated equipment and machinery covered with plywood and a tarpaulin. These items were being stored until decisions were made concerning possible decontamination and salvage operations. This practice is questionable since it is not a specified license condition and thus, no guidance for proper storage was given.

8.10 RELEASE RATES

None of the site procedures addressed release rates of radioactive materials originating from site operations. They relied upon their onsite environmental monitoring programs to detect radiation levels above background that would indicate an operational deficiency. The sites felt that keeping releases below measurable levels satisfied the goals of ALARA. Since modern instrumentation can detect contaminant levels which translate to submillirem dose levels to the occupational worker, this practice does appear reasonable. The parameters which were monitored at the sites were air, soil, vegetation, groundwater, and subsurface water from wells.

9.0 FACILITIES AND EQUIPMENT

Site facilities and equipment which contribute to the operation of an ALARA program were evaluated. Refer to Figures 24 and 25 for a layout of a typical low-level waste disposal site and site access building.

9.1 COUNTING ROOMS

Each site had one room located near the truck check-in area where radiological samples (i.e., contamination wipe samples) were prepared and analyzed. Two of the sites had permanent structures for count rooms, while the remaining site made use of a temporary area during construction of a permanent structure. Analytical instruments in use were gas-flow proportional counters, well-scintillation counters, Geiger-Muller counters, and liquid-scintillation counters. See Figures 26 and 27 for specific examples. All count rooms were fairly well isolated from the trench areas to eliminate interference from background radiations. In general, the permanent structure count rooms were considered adequate; the temporary count room was considered inadequate due to its small size, lack of lighting and poor ventilation.

To analyze onsite environmental samples such as vegetation, soil, and well water, two sites chose an offsite vendor, and thus had no onsite laboratory. The remaining site maintained a well-equipped laboratory dedicated to analysis of onsite environmental samples. The lab was located about 100 meters outside the controlled area and was shielded from the trench area by an earthen levee. Instrumentation observed in the lab included a gas-flow proportional counter, a liquid-scintillation counter, and a well-scintillation counter.

9.2 PERSONNEL DECONTAMINATION

All sites maintained personnel decontamination facilities near waste handling operations. They were located near the access points to the controlled areas as seen in Figure 25. In general, personnel decontamination facilities consisted of one or two showers, which were separate from showers located in the locker room area. This prevented contaminated and noncontaminated persons from utilizing the same shower areas.

As shown in Figure 25, workers entering the site access building from the control area removed work clothes before frisking. Contamination on individuals should have been found during the frisk before the step-off pad and decontamination (decon) showers were passed. Contaminated individuals then showered to remove the contaminants. Progress of the decon efforts at one site was checked by a health physics technician with a dedicated frisker located near the step-off pad. When count rates decreased below the contaminated level, the individual proceeded across the step-off pad. The two sites without a frisker in the decon area provided a health physics technician to aid in personnel decontamination operations.

FIGURE 24. Typical Disposal Site Layout

FIGURE 25. Site Access Building

FIGURE 26. Gas Flow Proportional Counter, Multichannel Analyzer, and Single Sample Holder

FIGURE 27. Gas Flow Proportional Counter and 100 Specimen Sample Changer

The decontamination showers were rarely used at the sites since personnel contamination rarely occurred. When needed, however, the showers were observed to be functional with a plentiful supply of soaps and other decontamination solutions, such as potassium permanganate (KMNO_4). Decontamination procedures were observed to rely heavily on washing the affected areas with warm water, soap, and/or other decon solutions. Contaminated individuals who could not be handled onsite were referred to a physician.

9.3 EQUIPMENT DECONTAMINATION

Only one facility dedicated to equipment (i.e., trucks and tools) decontamination was observed (Figure 16). This, as previously mentioned, was a sand-blasting decon facility, and was observed to be a very effective method for removing surface contamination. The sand-blasting decon facility consisted of a corrugated steel building with doors on each end that were large enough to accommodate tractor-trailer rigs and their loads. Sand-blasting equipment was located on the inside. The building used an air cleaning system to control dust and contaminant levels. Infrequent decon operations generated about two barrels of slightly contaminated sand per calendar quarter.

9.4 CHANGE ROOMS

Since all sites required personnel to wear protective clothing near the trenches, change rooms were essential. Except for one temporary change room, they were adequate in that they were heated, cooled, orderly, and well equipped with lockers, benches, etc. Stocks of clothing appeared to be sufficient to last for several days of normal operations. These rooms were located near decontamination facilities and access points.

The temporary change room facility at one site was part of a vehicle maintenance building. The facility was considered inadequate because ventilation was poor, space and equipment were lacking, and it was located approximately 200 meters from the decon facility and access point.

9.5 LAUNDRY

Two licensees laundered their protective work clothes on site. Clothes were monitored prior to washing and those which exceeded preset count rates or exposure rates (any detectable rates at one site, 0.2 mR/hr at the other site) were disposed of as radioactive wastes. These two sites drained the waste laundry water into underground disposal tanks as previously discussed. The third site contracted its laundry to an outside vendor.

9.6 EXTERNAL DOSIMETRY

Since vendors were used to supply and process personnel dosimeters, no facilities existed for this purpose. There were adequate areas (controlled environment and low radiation background) to store and disseminate dosimeters as needed.

9.7 INTERNAL DOSIMETRY

Two licensees had internal dosimetry instrumentation on site. One maintained an underground whole-body counting chair shown in Figure 28. It is more accurately described as a lung and thyroid counter. The chair to detector distance was variable. A 3 inch by 3 inch NaI lung detector and 1 inch by 1 inch NaI thyroid detector were connected to a multichannel analyzer. The third site used a 2 mm by 37 mm NaI detector for thyroid counting. Figure 29 shows the detector and foam pad configuration that allowed reproducible geometry over a wide range of neck sizes. The detector was attached to a scaler, and a multichannel analyzer was also available.

9.8 INSTRUMENT STORAGE

Storage of portable instruments was generally adequate. Portable instruments at two sites were stored in a controlled environment, but one of these areas lacked shelf space. The third site used an interim storage building in which air temperature and humidity were not controlled.

9.9 CALIBRATION FACILITY

All portable instruments were calibrated by offsite vendors. No onsite calibration facilities existed; therefore, none were observed.

FIGURE 28. Whole-Body Counting Chair

FIGURE 29. Thyroid Counting Equipment

9.10 MEDICAL AID

First aid was the only medical treatment provided at the sites. Medical facilities did not exist at the sites. Passive first aid stations were generally located near the decon areas. The kits at these stations contained general first aid supplies such as pads, gauze, tape, splints, etc. Only informal agreements for medical treatment at offsite hospitals were in place. The sites felt it extremely unlikely that an individual would be too contaminated to be admitted to an emergency room.

9.11 OFFICES

Two sites used manufactured housing conversions as offices. These appeared to have adequate work room and environmental controls. The third site used permanent buildings for offices. Special ventilation was not deemed necessary for any of the offices because no radioactive waste was allowed into the buildings.

9.12 ACCESS CONTROL

All site boundaries were restricted with chain-link fences of at least two meters in height. Only one access point per site was observed. The access point consisted of one large gate to allow passage of trucks and equipment and one smaller personnel gate. Only one site employed a full time guard to limit personnel access. This was also the only site where exiting employees were required to pass through a portal monitor, thus ensuring

personnel were checked for contaminants. The other sites relied on employees frisking themselves with hand-held G-M probes. Licensees expressed no problems with intrusion from offsite. Security around the outside perimeters of two sites was observed to be rather lax. Any encouragement to intrude, because of this, was apparently offset by the isolated setting of these disposal sites.

9.13 TRAINING FACILITY

Only one dedicated training facility was observed. The training room, which could seat about 15 persons, was orderly and, in general, conducive to learning. Training equipment included a slide projector, overhead viewgraph, and a videotape recorder. The other two sites used large offices or lunchrooms during training sessions and instruction was limited to lectures and informal discussions.

9.14 OFFSITE FACILITIES

Formal agreements for the use of offsite facilities were almost nonexistent. States where two sites were located maintained chartered radiological emergency assistance teams which could supply Health Physicists, light supplies and equipment during an emergency. The members of these teams were volunteers from several licensed facilities within the respective states and included some members of the respective state health departments. No facilities were available through these groups, however.

All three sites were located near Federal Government Reservations which had radiological support facilities for their own operations. Due to security and other considerations, these facilities were not available to the commercial low-level sites. Informal verbal agreements to supply some personnel and supplies in the event of an emergency were mentioned.

9.15 PROTECTIVE EQUIPMENT

The quantities and types of protective clothing in stock appeared to be adequate for both normal and abnormal operations. Personnel working at or near the trenches were required to wear white cloth overgarments of which a plentiful supply of various sizes was observed. A plentiful supply of disposable hoods, gloves, and boots were also on hand; although, they were seldom worn with the coveralls. Heavy-duty work gloves were worn in place of rubber gloves by waste handlers to protect hands from injuries. Past experience has shown no need to wear head and shoe covers.

Temporary shielding was available at all sites in the form of lead blankets, bricks, and sheets. Shielding appeared readily available to the Health Physicist on duty. The Health Physicists used general operational guidelines and professional judgment as a guide when deciding upon shield configurations. Shielding was used when abnormally high activity waste was handled and calculations showed an overall dose savings through the use of shielding.

9.16 ALARA IMPACT

ALARA programs were instituted by waste disposal site operators. These programs were developed to minimize occupational exposures while considering economic and technical constraints. At the time of the site visits, the licensees were still striving for and obtaining significant decreases in worker doses. This trend was expected to continue for a time due to the short time period the programs have been operated. Much of the success in dose control was directly attributed to recent design and use changes in site facilities and equipment.

The efficient use of facilities and equipment was observed to have a positive impact upon dose control efforts. Facilities and equipment, in general, were observed to be adequate to protect the radiation worker from existing radiation hazards.

10.0 EXTERNAL DOSE ANALYSIS

External radiation dose data was gathered for analysis during visits to the three low-level waste (LLW) disposal sites. In particular, information was collected that would depict the effects of instituting an ALARA program at LLW sites.

Three common methods of determining occupational doses to workers were available to the sites. They were the use of personnel dosimetry devices, calculation of doses through exposure models, and projection of doses from survey data. The licensees' data were based entirely upon vendor personnel dosimetry reports.

Regrettably, only one full year of data (1981) had been accumulated since all sites began formal ALARA programs. A formal ALARA program is one which is based upon written procedures and dose reduction methodologies which are supported by site management. Two sites initiated programs in January of 1981 while the third began its program a few months prior to that date. Many aspects of the ALARA programs were in operation before these dates, but others, such as the ALARA management structure, were not formalized. Since portions of records prior to the year 1981 were found to be unusable, missing, or unavailable to the researchers, dose data collected from the sites were limited to the period in which all sites had formally established programs.

Sufficient reliable data did exist, however, to allow determination of several general trends. The following section focuses on the presentation and analysis of available data. Findings are discussed and tabulated, and generalizations are offered based on observed dose patterns.

10.1 DOSIMETRY DATA ANALYSIS

Dosimetry data was collected from commercial low-level waste disposal sites. The purpose of data collection, and the subsequent analysis, was to characterize occupational dose levels received at the site and to project future expected dose levels.

10.1.1 Dosimetry Data

Two sites possessed raw dose data in a usable format; the third site supplied previously calculated numbers. For the purpose of this research, usable data was defined as data one step removed from the vendor's dosimetry report. For example, two licensees supplied the summation of site worker dose per month (man-rem), the number of persons monitored, and the activity and volume of waste handled. This data was then analyzed by researchers under known and stated assumptions. However, numbers supplied by the third site must be used with caution, because the assumptions and methods employed in the calculations were unknown.

As stated in Chapter 4.0, the sites monitored approximately 35 persons at the largest site, 23 at the next largest, and 8 at the smallest. These personnel, designated as radiation workers, were monitored by the use of dosimetry devices. The number of monitored personnel fluctuated significantly throughout the year because of employee turnover and changing job requirements. Caution in using these numbers should be exercised for this reason and because of the fact that calculations performed with small groups of numbers may allow large errors to be introduced into final calculations.

Data supplied to the researchers were reported in rem. This analysis assumes that the reported doses in rems were actually absorbed whole-body doses with all pertinent correction factors applied by the vendor.

The present analysis relied heavily upon examination of the raw data supplied by two sites. The waste volume received by these two licensees were substantially different and thus allowed a comparison of dose trends characteristic to both large and small radwaste disposal operations. The two sites characterized in this analysis were designated Site A and Site B. Data from the third site, C, is included in some cases. These designations were used for book-keeping purposes and in no way denote ranking of sites with respect to size, operation, or quality of program.

The 1981 dose data gathered from the sites are listed in Tables 5, 6, and 7. Sites supplied raw monthly data in the form of accumulated dose (man-rem),

TABLE 5. Cumulative and Average Dose Figures, Sites A and B 1981

	Site A		Site B	
	Man-Rem	Individual Average (Rem)	Man-Rem	Individual Average (Rem)
January	0.45	0.020	0.15	0.019
February	0.44	0.019	0.01	0.001
March	0.81	0.035	0.47	0.059
April	3.09	0.134	0.16	0.020
May	6.26	0.272	0.10	0.013
June	2.31	0.100	0.45	0.056
July	1.26	0.055	0.12	0.015
August	.74	0.032	0.12	0.015
September	1.86	0.081	0.13	0.016
October	1.87	0.081	0.01	0.001
November	1.17	0.051	0.04	0.005
December	0.96	0.042	0.07	0.009
Annual Total	21.22	0.923	1.83	0.229
Monthly Avg.	1.77	0.077	0.15	0.019

TABLE 6. Waste Activity Received and Dose per Unit Activity, Sites A and B 1981

	Site A		Site B	
	Activity Received (Ci)	Dose per Activity (Rem/Ci)	Activity Received (Ci)	Dose per Activity (Rem/Ci)
January	568	7.9 E-4	365	4.1 E-4
February	651	6.8 E-4	8,037	1.2 E-6
March	2,612	3.1 E-4	19,599	2.4 E-5
April	1,479	2.1 E-3	9,787	1.6 E-5
May	21,619	2.9 E-4	13,316	7.5 E-6
June	7,719	3.0 E-4	618	7.4 E-4
July	1,029	1.2 E-3	88	1.4 E-3
August	1,965	3.8 E-4	21	5.5 E-3
September	536	3.5 E-3	28	4.6 E-3
October	1,840	1.0 E-3	20	4.9 E-4
November	1,010	1.2 E-3	245	1.6 E-4
December	2,876	3.3 E-4	83	8.4 E-4
Total	43,904	4.8 E-4	52,207	3.5 E-5

TABLE 7. Waste Volume Received and Dose per Unit Volume, Sites A and B 1981

	Site A		Site B	
	Volume Received (Ft ³)	Dose per Volume (Rem/Ft ³)	Volume Received (Ft ³)	Dose per Volume (Rem/Ft ³)
January	94,445	4.8 E-6	21,733	6.9 E-6
February	93,347	4.5 E-6	24,298	4.1 E-7
March	131,832	5.6 E-6	36,977	1.3 E-5
April	136,528	2.1 E-5	2,916	5.5 E-5
May	160,815	3.2 E-5	2,324	4.3 E-5
June	203,204	1.1 E-5	10,701	4.2 E-5
July	77,306	9.0 E-6	2,655	4.5 E-5
August	100,991	5.9 E-6	2,415	5.0 E-5
September	102,228	1.1 E-5	4,651	2.8 E-5
October	116,665	1.0 E-5	3,650	2.7 E-5
November	116,781	9.5 E-6	1,798	2.2 E-4
December	104,138	8.6 E-6	4,379	1.6 E-5
Total	1,438,285	1.2 E-5	118,507	1.5 E-5

activity (curies), and volume (cubic feet) of wastes received for the respective sites. The other values listed in the tables result from calculations performed by the PNL research team.

The cumulative and average dose figures for 1981 are listed in Table 5. These figures are further subdivided according to Sites A and B. The values listed in the Individual Average Dose column were obtained by dividing the monthly site accumulated dose (Man-Rem) by the number of persons monitored. This provides a monthly average dose per person. In the Man-Rem column, the Annual Total is the accumulated dose for all radiation workers for the year 1981. The Monthly Average in that column is simply the Annual Total divided by 12 months. In the Individual Average column, the Annual Total figure is obtained by dividing the annual man-rem figure by the number of persons monitored. This calculation provides an average yearly dose per person. The Monthly Average figure given in that column is obtained by dividing this number by 12 months.

Table 6 lists the waste activity (curies) and the calculated dose received per unit activity (rem/ft^3). The dose per unit activity is calculated by dividing the accumulated site dose (Man-Rem) for each month by the activity of the waste received during that specific month. The Total Activity Received is the summation of activity received for each month in 1981. The total dose per unit activity is obtained by dividing the Annual Total dose from Table 5 by the total number of curies received for the year.

Table 7 lists the waste volume received (ft^3) and the dose per unit volume (rem/ft^3) of wastes handled in 1981. These values were determined in a similar manner to values in Table 6. Dose per unit volume is obtained by dividing the monthly site dose (man-rem) by the volume of wastes received during that specific month. The total dose per unit volume is calculated by dividing the annual total dose from Table 5 by the total volume of wastes received for the year.

10.1.2 Analysis

The calculated total values in the previous three tables are tabulated in Table 8. This compilation allows a comparison of waste handling operations between Sites A and B by studying the most appropriate parameters. It does not allow a comparison of respective ALARA programs because one of the most important elements, that of trend analysis, can not be established with data from a single year.

The first observation one can make from the values provided in Table 8 is that Sites A and B received approximately equal activities of wastes in 1981. Further, personnel at the two sites collected approximately equal doses of radiation for each unit volume of waste handled. Information collected from Site C included a dose per unit volume figure of $4.39 \times 10^{-5} \text{ rem}/\text{ft}^3$. A higher value such as this is not unexpected for this site because Site C tends to handle more hazardous wastes. Although higher, this figure is still consistent with the other two sites.

TABLE 8. Site Dose Comparison, Sites A and B 1981

	Site A	Site B
Activity Received (Curies)	43,904	52,207
Waste Volume (Ft ³)	1,438,285	118,507
Site Dose (Man-Rem)	21.22	1.83
Dose per Unit Activity (Rem/Ci)	4.83E-4	3.51E-5
Dose per Unit Volume (Rem/Ft ³)	1.5E-5	1.5E-5
Waste Activity Concentration (Ci/Ft ³)	0.031	0.441

The total site dose (man-rem) is observed to differ by more than an order of magnitude between Sites A and B. This is an interesting difference because, as stated, the licensees buried approximately equal activities of waste. Two parameters in Table 8, Waste Volume and Dose per Unit Activity, show that Site A handled about 12 times the waste volume of B and personnel at A received about 13 times the dose per unit of buried activity compared to personnel at B. Despite the difference in values, all the parameters presented so far are consistent, if the waste activity concentration of A is approximately the same as that of B.

Waste activity concentration figures between the sites are not comparable, however. The concentration at Site A is observed to be less than at B by a factor of 14. This tends to point to an apparent shortcoming in dose control methods at Site A. However, one parameter which could account for the discrepancy remains unknown. That is, the composition of the waste itself. Site A could have received a higher percentage of high-energy gamma-emitting radionuclides causing the site to have higher dose rate fields than Site B. Since this parameter remains unknown, no accurate conclusion on the adequacy of dose control programs can be given based on the presented data. However, this data can be manipulated to reveal trends which give an indication as to the origin of the higher Site A doses.

The monthly variations in dose, activity, and volume during 1981 are plotted in Figures 30 and 31. The graphs show that total site dose generally follows both the waste activity and volume trends at each site. No other significant conclusions can be reached in observing these trends.

FIGURE 30. Dose, Activity, and Volume Graph; Site A 1981

FIGURE 31. Dose, Activity, and Volume Graph; Site B 1981

Two important relationships are plotted for each site in the following four figures. The ratios of dose per unit activity are plotted versus activity and shown in Figures 32 and 33. The ratios of dose per unit volume are plotted versus volume as shown in Figures 34 and 35.

Each of these four sets of points were tested for continuous function fit. The formulas used for the test were:

$$y = ax + b \text{ for a linear fit}$$

$$y = a \ln x + b \text{ for a log fit}$$

$$y = ae^{bx} \text{ for an exponential fit}$$

$$y = ax^b \text{ for a power fit}$$

Except for the set of points in Figure 35, a power function proved to be the best fit for the data. Curves for these functions are plotted on the respective graphs.

The data in Figure 35 was not tested because three data points would obviously bias the test. These points, upper right in the graph, are a result of highly abnormal waste characteristics. Specifically, these points coincide with the burial of high activity, large volume waste shipments containing an inventory of young, mixed fission products. A close look at these data points show they originated from the second calendar quarter of 1981, specifically during April, May, and June. During this period, it is observed that Site A received 55% of its annual dose, 70% of its activity for the year (49% in one month), and 35% of the annual waste volume.

Although the possibility of a deficiency in Site A operations cannot be eliminated, it appears more likely now that the difference in performance data between the two sites can be attributed to atypical waste characteristics. Although the quantity of activity is observed to affect the accumulated dose directly, the controlling factor at these two sites appears to be the volume over which the activity is contained. For example, given the same activity waste to handle, the site with the lesser volume will probably receive the lesser radiation dosage due to the decreased amount of time necessary to unload smaller volumes.

The concentration of waste activity (Ci/ft^3) is plotted in Figure 36. The graph shows that the concentration of activity in Site A waste is consistently higher than the waste received at Site B. Note the peak in Site A activity concentration for the second quarter. Not only did the site receive large activities and volumes of waste during these months, but the concentration of radioactive materials in the waste was more than two orders of magnitude higher than the yearly average concentration for that site.

Several other relationships between radiation dose and waste characteristics were explored, but these yielded no useable results. For example, the effects of changing activity concentration on both individual and collective doses was plotted. The data points were so disperse, however, that no pattern was discernible and further analysis was not considered practical.

FIGURE 32. Dose/Activity vs. Activity, Site A 1981

FIGURE 33. Dose/Activity vs. Activity, Site B 1981

FIGURE 34. Dose/Volume vs. Volume, Site A 1981

FIGURE 35. Dose/Volume vs. Volume, Site B 1981

FIGURE 36. Activity Concentration of Wastes, Sites A and B 1981

The breakdown of doses by worker groups gives an indication of how radiation doses vary with job or job description within each site. Data was available from all three sites, but one set was not reliable and thus, not used. Tables 9 and 10 give worker group comparisons using 1981 data. Note

TABLE 9. Doses to Radiation Worker Groups, Site A 1981

	<u>Monthly Man-Rem Average</u>	<u>Group Percent of Site Man-Rem</u>
Maintenance	0.094	5.3
Monitors	0.218	12.3
Operators	0.343	19.4
Supervisors	0.052	3.0
Site Workers	<u>1.062</u>	<u>60.0</u>
Total	1.769	100.0

TABLE 10. Doses to Radiation Worker Groups, Site C 1981

<u>Worker Group</u>	<u>Worker Subgroup</u>	<u>Monthly Man-rem Average</u>	<u>Percent Site Man-rem</u>	<u>Group Percent of Site Man-Rem</u>
Maintenance	Maintenance	0.033	0.72	0.72
Monitors	Health Physics Technicians	0.932	20.45	
	QA	0.209	4.58	
	Health, Safety and Environmental Tech.	0.006	0.13	25.16
Operators	Equipment Ops.	0.969	21.26	24.44
	Truck Drivers	0.145	3.18	
Supervisors	Supervisory/ Management	0.035	.77	1.06
	Contractors	0.013	.29	
Site Workers	Radwaste Techs.	2.216	48.62	48.62
Total		4.558	100.0	

the difference in titles, composition, and number of radiation worker groups used at the two sites.

Obviously, persons handling waste at the trench during offloading and placement operations received a large portion of the licensee's monthly dose. This group, called site workers and radwaste technicians at Sites A and C respectively, appears to receive as much or more radiation dose as all other groups combined. Equipment operators and Health Physics technicians are the next highest groups.

To determine the effectiveness of dose equalization efforts, licensees were asked to provide information relative to the highest single dose to persons working in the waste handling group. Table 11 compares this highest recorded dose to the group's individual monthly average. These two values are within a factor of two for both sites A and C. Site B had one uncharacteristically high dose which makes a comparison questionable.

Annual Man-Rem figures for years prior to 1981 were collected from the sites when available. Two sites had these calculated numbers for two years prior to 1981. Table 12 shows the accumulated dose trends for 1979 through 1981. One observation made is that the annual Man-Rem dose loosely follows that of the site operation activity.

TABLE 11. Doses (Rem) to Waste Handling Group; Sites A, B and C 1981

	<u>Site A</u>	<u>Site B</u>	<u>Site C</u>
Trench Worker Monthly Average	0.137	0.019	0.136
Highest One Month Dose for Trench Worker	0.240	0.130	0.240

TABLE 12. Annual Man-Rem Figures, Sites A and B 1981

	<u>1979</u>	<u>1980</u>	<u>1981</u>
Site A	10.94	4.24	21.22
Site B	2.99	2.73	1.83

10.2 SUMMARY AND CONCLUSIONS

A summary of the previously calculated numbers are found in Table 13. Care must be exercised in the use of the numbers marked with an asterisk. Licensees supplied these numbers and the assumptions and methods used for calculation are unknown. One of these needs mentioning in particular, the average trench worker dose for Site B is the same as the individual average in Table 5. The reason is that all radiation workers are also trench workers.

TABLE 13. Summary of Average Monthly Doses; Sites A, B and C 1981

	<u>Man-Rem Average</u>	<u>Rem/Ci</u>	<u>Rem/Ft³</u>	<u>Trench Worker Average (Rem)</u>
Site A	1.77	4.8 E-4	1.23 E-5	0.137*
Site B	0.15	3.5 E-5	1.54 E-5	0.019*
Site C	4.25*	2.1 E-4*	4.39 E-5*	0.136*

*Figures calculated by licensee.

11.0 CONCLUSIONS

The Radiation Protection and ALARA programs at the three commercial low-level waste disposal sites were observed to be adequate in scope and content and in general very good. However, it should be noted that there were many areas that could be improved upon. Future work should address these particular areas and make appropriate recommendations for program improvements. The program deficiencies of major concern are outlined below.

Formalized ALARA Programs at LLW sites were observed to lack an established communication system between employees and management. The scope of this feedback program would be to investigate employee complaints and suggestions as they relate to ALARA.

It was observed that there were no deficiencies of major concern in the radiation protection training programs at the LLW disposal sites.

The Quality Assurance Programs, for external dosimeter testing, were observed to be inadequate. Dosimeters were neither tested initially to verify the vendor's specifications nor routinely tested to ensure the reliability of the reported exposure data from the vendor.

Quality Assurance on the biosurveillance programs at the LLW disposal sites was observed to be inadequate. Reliability checks on the vendor's lung scan, thyroid and urinalysis data had not been performed routinely or even instituted.

The absence of documented, certified medical examinations for determining physical impairments of respiratory protection wearers, was observed to be a major deficiency in the Respiratory Protection Programs.

Air sampling systems at LLW disposal sites did not adequately provide representative samples of worker breathing zones, due to improper placement of sample collection media.

Formalized or routinely-used procedures for waste inspection were not observed. Onsite above-ground storage of LLW, although specifically excluded in licenses, did exist.

The "temporary" facilities used for count rooms, locker rooms and instrument storage were observed to be inadequate for ensuring the health and safety of the workers.

11.0 CONCLUSIONS

The radiation protection and ALARA programs at the three commercial low-level waste disposal sites were observed to be adequate in scope and content and in general very good. However, it should be noted that there were many areas that could be improved upon. Future work should address these particular areas and make appropriate recommendations for program improvements. The program deficiencies of major concern are outlined below.

Formalized ALARA programs at LLW sites were observed to lack an established communication system between employees and management. The scope of this feedback program would be to investigate employee complaints and suggestions as they relate to ALARA.

It was observed that there were no deficiencies of major concern in the radiation protection training programs at the LLW disposal sites.

The Quality Assurance Program, for external dosimeter testing, were observed to be inadequate. Dosimeters were neither tested initially to verify the vendor's specifications nor routinely tested to ensure the reliability of the reported exposure data from the vendor.

Quality Assurance on the fluoroscopy programs at the LLW disposal sites was observed to be inadequate. Reliability checks on the vendor's lung scan, thyroid and whole-body data had not been performed routinely or even instituted.

The absence of documented, certified medical examinations for determining physical impairments of respiratory protection wearers, was observed to be a major deficiency in the respiratory protection programs.

Air sampling systems at LLW disposal sites did not adequately provide representative samples of worker breathing zones, due to improper placement of sample collection media.

Formalized or routinely-used procedures for waste inspection were not observed. On-site above-ground storage of LLW, although specifically excluded in licenses, did exist.

The "temporary" facilities used for count rooms, locker rooms and instrument storage were observed to be inadequate for ensuring the health and safety of the workers.

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