Safety Light Corporation
4150A Old Berwick Road
Bloomsburg, Pa. 17815
Phone: 570-784-4344
Fax: 570-784-1402
E-Mail: lharmon@safetylight.com

RECEIVED
REGION 1

2000 OCT 27 PM 2:21

October 26, 2000

United States Nuclear Regulatory Agency
475 Allendale Road
King of Prussia, Pa. 19406
Attention: Marie Miller

030-05980

Re: Decommissioning Plan and Cost Estimate for license no. 37-00030-02

Dear Ms. Miller,

The above enclosures satisfies condition 13 of license 37-00030-02.

If you have any further questions, please do not hesitate to contact me.

Very Truly Yours,

Larry Harmon

Larry Harmon,
Plant Manager

NMSS/RGN MATERIALS-002

Safety Light Corporation
4150A Old Berwick Road
Bloomsburg, Pa. 17815
Phone: 570-784-4344
Fax: 570-784-1402
E-Mail: lharmon@safetylight.com

October 26, 2000

United States Nuclear Regulatory Agency
475 Allendale Road
King of Prussia, Pa. 19406
Attention: Marie Miller

Re: Decommissioning Plan and Cost Estimate for license no. 37-00030-02

Dear Ms. Miller,

Safety Light Corporation, by this letter, authorizes the Nuclear Regulatory Commission the right to copy the Decommissioning Cost Estimate and releases the NRC entirely from the Proprietary Statement on Page I of the document.

If you have any further questions, please contact me.
Very Truly Yours,

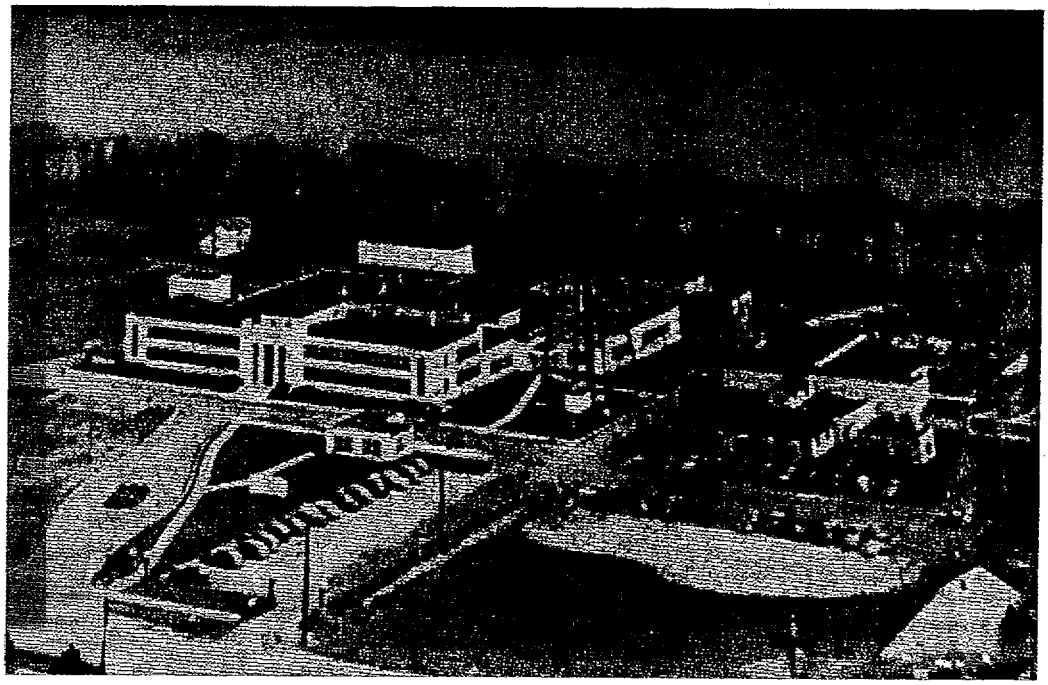
Larry Harmon

Larry Harmon,
Plant Manager

Decommissioning Plan for Safety Light Corporation Bloomsburg, Pennsylvania

NRC Material License No. 37-00030-02

October 2000



Prepared for:
Safety Light Corporation
Bloomsburg, Pennsylvania

Prepared by:
GTS Duratek

Radioactive Solutions
Commercial Services
628 Gallaher Rd.
Kingston, TN 37763

DECOMMISSIONING PLAN

for

SAFETY LIGHT CORPORATION
BLOOMSBURG, PENNSYLVANIA

Revision 0
October 2000

Prepared By: Sandra M Beeler — 10/13/00
Sandra M. Beeler — Date
Senior Radiological Engineer

Reviewed By: Paul C Ely — 10/13/00
Paul Ely — Date
Senior Radiological Engineer

Approved By: Robert Hornbeck — 10/18/00
Robert Hornbeck — Date
Operations Manager
Commercial Services

Prepared By:

GTS Duratek, Inc.
628 Gallaher Road
Kingston, TN 37763

TABLE OF CONTENTS

1 GENERAL INFORMATION 1-1
1.1 Introduction 1-1
1.2 License Information 1-2
1.3 Objective of the Decommissioning Plan 1-2
1.4 Site Description 1-2
1.5 Recent Regulatory Activities 1-4
1.6 References for Section 1 1-8
2 DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES 2-1
2.1 Decommissioning Criteria 2-1
2.2 Current Assessment of Residual Radioactive Contamination 2-3
2.3 Decommissioning Organization and Responsibilities 2-3
2.3.1 Decommissioning Project Quality Assurance Plan 2-4
2.3.2 Training 2-4
2.3.2.1 General Employee Training 2-4
2.3.2.2 Radiation Worker Training 2-5
2.3.2.3 Respiratory Protection Training 2-5
2.3.3 Procedures 2-6
2.3.4 Contractor Assistance 2-6
2.4 Decommissioning Activities and Tasks 2-6
2.4.1 Task Descriptions 2-6
2.4.1.1 Remove Specified Rooms from the Etching Building 2-6
2.4.1.2 Remove Soil North of Well 5 2-9
2.4.1.3 Remove Ion Exchange Building and Surrounding Soil 2-9
2.4.1.4 Remediate Old Garage Foundation and Surrounding Soil 2-9
2.4.1.5 Remove Soil Under Old Loading Dock 2-10
2.4.1.6 Remove 8 x 8 Block Building and Surrounding Soil 2-10
2.4.1.7 Remove Old Radium Vault and Surrounding Soil 2-10
2.4.1.8 Remove Soil Around Machine Shop 2-10
2.4.1.9 Remove Above-Ground Silo and Surrounding Soil 2-11
2.4.1.10 Decontaminate Remaining Site Buildings and Structures 2-11
2.4.1.11 Decontaminate Remaining Site Soils 2-14
2.4.2 Remediation Methods 2-19
2.4.2.1 General Surface Decontamination 2-19
2.4.2.2 Concrete Surface Removal 2-22
2.4.2.3 Metal Surface Removal 2-23
2.4.2.4 Demolition and Component Removal 2-23
2.4.2.5 Soil Excavation 2-25
2.4.3 Decommissioning Accident Analysis 2-26
2.4.3.1 Potential Radiological Accidents 2-27
2.4.3.2 Evaluation of Public Impact from Accident Scenarios 2-29
2.4.3.3 Evaluation of Worker Exposure from Accident Scenarios 2-33
2.4.3.4 Conclusions 2-33
2.4.4 Decommissioning Schedule 2-34
2.5 References for Section 2 2-34

3 DESCRIPTION OF METHODS USED FOR PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY 3-1
3.1 Facility Radiological History Information 3-1
3.2 Radiological Concerns During Decommissioning Activities 3-2
3.3 Radiation Safety Program 3-2
3.3.1 As Low As Reasonably Achievable (ALARA) Policy 3-3
3.3.1.1 Administrative Limits 3-4
3.3.1.2 Radiological Performance Goals 3-4
3.3.1.3 ALARA Plans and Procedures 3-4
3.3.1.4 Radiological Work Planning 3-4
3.3.2 Radiation Work Permits 3-5
3.3.3 Contamination Control 3-6
3.3.3.1 Protective Clothing 3-6
3.3.3.2 Access Control 3-6
3.3.3.3 Ventilation 3-7
3.3.3.4 Air Sampling Program 3-7
3.3.4 Handling and Storage of Radioactive Material 3-7
3.3.5 Instrument Program 3-7
3.3.6 Radiation Survey Program 3-8
3.3.7 Personnel Monitoring 3-9
3.3.7.1 External Exposure Monitoring 3-9
3.3.7.2 Internal Exposure Monitoring 3-10
3.3.8 Respiratory Protection 3-10
3.3.8.1 Respirator User Qualification 3-10
3.3.8.2 Respiratory Protection Equipment Description and Selection 3-11
3.3.8.3 Equipment Inspection and Maintenance 3-11
3.3.9 Quality Assurance 3-11
3.4 Contractor Personnel 3-12
3.5 Radioactive Waste Management 3-12
3.5.1 Estimated Volumes of Radioactive Waste 3-13
3.5.2 Radioactive Waste Handling and Storage 3-13
3.5.3 Radioactive Waste Transportation 3-14
3.5.4 Radioactive Waste Processing 3-14
3.5.5 Waste Classification 3-15
3.5.6 Disposal of Radioactive Waste 3-15
3.5.7 Disposal of Non-Radioactive Waste 3-15
3.6 References for Section 3 3-16
4 PLANNED FINAL RADIATION SURVEY 4-1
4.1 Data Quality Objectives Process 4-1
4.2 Survey Design 4-3
4.2.1 Calculations 4-3
4.2.1.1 Classify Survey Units 4-3
4.2.1.2 Specify the Acceptable Decision Error 4-9
4.2.1.3 Calculate the Relative Shift 4-9
4.2.1.4 Obtain the Number of Samples Per Survey Unit 4-10
4.2.1.5 Estimate the Sample Grid Spacing 4-10

4.2.1.6 Perform Evaluation for Small Areas with Elevated Radioactivity 4-10
4.2.1.7 Optimize the Design 4-10
4.2.2 Measurements and Sampling 4-11
4.2.2.1 Scanning 4-11
4.2.2.2 Direct Measurements 4-11
4.2.2.3 Sampling and Laboratory Analysis 4-11
4.3 Background Radiation Data 4-12
4.4 Instruments 4-12
4.5 Data Analysis 4-12
4.6 Documentation 4-13
4.7 References for Section 4 4-13
5 FUNDING 5-1
5.1 Decommissioning Cost Estimate 5-1
5.2 Funding Plan 5-1
5.3 References for Section 5 5-1
6 PHYSICAL SECURITY PLAN AND MATERIAL CONTROL AND ACCOUNTING PLAN PROVISIONS IN PLACE DURING DECOMMISSIONING. 6-1
6.1 References for Section 6 6-1

LIST OF TABLES

Table 2-1, DCGLs for Surfaces and Soil 2-2
Table 2-2, Rooms in Etching Building Scheduled for Early Removal 2-7
Table 2-3, Maximum Activity in Soil Samples North of Well 5 2-9
Table 2-4, Equipment Selection Matrix 2-20
Table 2-5, Estimated Radioactive Material Inventories 2-28
Table 2-6, TEDE Calculation Table for Scenario 1: Contaminated Soil Release 2-31
Table 2-7, TEDE Calculation Table for Scenario 2: Release from Fire During Remediation Activities 2-32
Table 2-8, TEDE Calculation Table for Scenario 3: Release During Building Demolition 2-33
Table 4-1, Survey Unit Classification 4-4

LIST OF FIGURES

Figure 1-1, Map of 10 Mile Radius Surrounding the Safety Light Site 1-5
Figure 1-2, Map of 1-Mile Radius Surrounding Safety Light Site 1-6
Figure 1-3, Map of Safety Light Site 1-7
Figure 2-1, Etching Building 2-8
Figure 4-1, Data Quality Objectives Process Steps 4-2
Figure 4-2, Inputs to Final Status Survey Plan 4-2

Attachment 1, Safety Light Dose Assessment
Attachment 2, Derivation Of χ/Q

1 GENERAL INFORMATION

1.1 Introduction

NRC License No. 37-00030-02 Amendment 51 requires Safety Light to provide to the Region 1 Office, for review and approval, "a schedule and plan for continuation of site remediation, including any necessary additional characterization. The schedule shall include the licensee's strategies for meeting the requirements of 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination," including any revised decommissioning cost estimates." This decommissioning plan has been prepared to meet this license condition.

This document consists of six main sections:

- Section 1, General Information, provides information on the license to be terminated, the overall objectives of this decommissioning plan, and a description of the site and recent regulatory activities.

- Section 2, Description of Planned Decommissioning Activities, covers the decommissioning criteria, current assessment of residual radioactive contamination, decontamination organization and responsibilities, decommissioning activities and tasks, and decommissioning accident analyses. This section includes a proposed decommissioning schedule.

- Section 3, Description of Methods Used for Protection of Occupational and Public Health and Safety, discusses the radiological history of the facility and special radiological concerns during decommissioning activities. It describes the radiation protection and radioactive waste management programs that will be implemented during decommissioning operations.

- Section 4, Planned Final Radiation Survey, discusses survey design, use of background radiation data, instrument selection and use, and data analysis and documentation.

- Section 5, Funding, provides an updated decommissioning cost estimate and describes the current funding plan approved by the NRC.

- Section 6, Physical Security Plan and Material Control and Accounting Plan Provisions in Place During Decommissioning, describes the physical security plan that will be in place during decommissioning operations.

1.2 License Information

License:	37-00030-02
Docket Number:	030-05980
Location of Use:	Safety Light Corporation 4150-A Old Berwick Road Bloomsburg, PA 17815
Licensee of Use:	Safety Light Corporation 4150-A Old Berwick Road Bloomsburg, PA 17815
Licensee Contact:	Mr. Larry Harmon Safety Light Corporation 4150-A Old Berwick Road Bloomsburg, PA 17815

1.3 Objective of the Decommissioning Plan

The objective of this document is to provide plans for decommissioning retired facilities, equipment, and land at the Safety Light site that were used for previous operations involving radioactive material. This document does not address the decommissioning of the tritium facility and support buildings, which are currently operating under NRC License 37-00030-08. This decommissioning plan was prepared using the guidelines provided in NRC Regulatory Guide 3.65, *Standard Format and Content of Decommissioning Plans for Licensees Under 10 CFR Parts 30, 40, and 70* (NRC 1989). It includes a decommissioning schedule and cost estimate. This plan also incorporates draft guidance contained in Regulatory Guide DG-4006, *Demonstrating Compliance with the Radiological Criteria for License Termination* (NRC 1998).

The overall objective of the decommissioning is to remediate retired facilities and soil areas to a condition which corresponds to a calculated dose to the public of less than 25 mrem/year from applicable pathways. The facilities and land may then be released for unrestricted use. In addition, analysis must demonstrate that levels of residual radioactivity are as low as reasonably achievable (ALARA). Section 2 of this plan provides a more detailed discussion of decommissioning objectives.

1.4 Site Description

The Safety Light Corporation site is located in South Centre Township of Columbia County in central Pennsylvania, about 6 miles east of Bloomsburg and 6 miles west of

Berwick. The north site boundary is the Old Berwick Road and the south site boundary is the Susquehanna River. Safety Light Corporation owns the Vance-Walton property located along the southeast corner of the site. Other residential tracts of land are adjacent to the east and west boundaries of the site.

Figure 1-1, Map of 10 Mile Radius Surrounding the Safety Light Site, and Figure 1-2, Map of 1-Mile Radius Surrounding Safety Light Site show the location of the Safety Light site. The site is located on the United States Geological Survey (U.S.G.S.) Bloomsburg, Pennsylvania quadrangle topographical map at North 40° 00' 55" latitude and West 76° 22' 35" longitude.

Safety Light Corporation occupies approximately 2 acres of the 10-acre site. The current product line includes a variety of tritium-based products:

- self-luminous safety devices for use in commercial/military aircraft, commercial buildings and marking of aircraft and helicopter landing areas;
- research and industrial applications;
- titanium tritide-coated rods and pins for use in military and industrial type electron tubes; and
- tritium targets for use in neutron-generating devices.

Safety Light leases the remaining 8 acres of the site to USR Metals, Inc., who conducts non-radioactive operations involving the manufacture of dials, nameplates, and other specialty products used in a variety of industrial and military applications. Their operations also involve anodizing of aluminum products, and application of specialty protective films to the surfaces of various metal items.

Figure 1-3, Map of Safety Light Site, shows existing buildings and features. This decommissioning plan will not address the nuclear building (No. 12), machine shop (No. 11), solid waste building (No. 10), and liquid waste building (No. 8), which support current operations under NRC License 37-00030-08.

A hydrogeological investigation (Meiser and Earl 1979) confirmed the presence of shallow, unconfined groundwater in a highly conductive, alluvial aquifer beneath the site. Water level measurements indicated a relatively flat hydraulic gradient from north to south toward the river. The measurements also indicated riverbank storage and groundwater mounding in the area immediately south of the disposal pits and lagoons. These transient variations in the water table elevation may divert groundwater flow beneath the site toward the east or west. The Susquehanna River is the only natural surface water on or adjacent to the site. Because of the large average flow rate, discharge of contaminated groundwater should not cause significant increases above background levels of radionuclide concentrations in the river.

The *Characterization Survey of Safety Light Corporation Site at Bloomsburg, Pennsylvania U.S.A* (Monserco 1996) documents available information on site history

from the 1940's through 1995. Section 3.1 of this decommissioning plan summarizes the most relevant historical information for purposes of decommissioning planning.

1.5 Recent Regulatory Activities

Monserco Limited of Ontario, Canada performed a site characterization in 1995 (Monserco 1996) using a characterization working document (Monserco 1994) approved by the NRC. On November 5, 1998, Safety Light submitted a site decommissioning plan to the NRC for review that summarized decommissioning activities that would be undertaken to remediate the buildings, soil, and underground silos contaminated with radioactive material from past operations.

Amendment 51 to NRC License No. 37-00030-02, dated December 28, 1999 requires Safety Light to provide to the Region 1 Office, for review and approval, "a schedule and plan for continuation of site remediation, including any necessary additional characterization. The schedule shall include the licensee's strategies for meeting the requirements of 10 CFR Part 20, Subpart E, "Radiological Criteria for License Termination," including any revised decommissioning cost estimates."

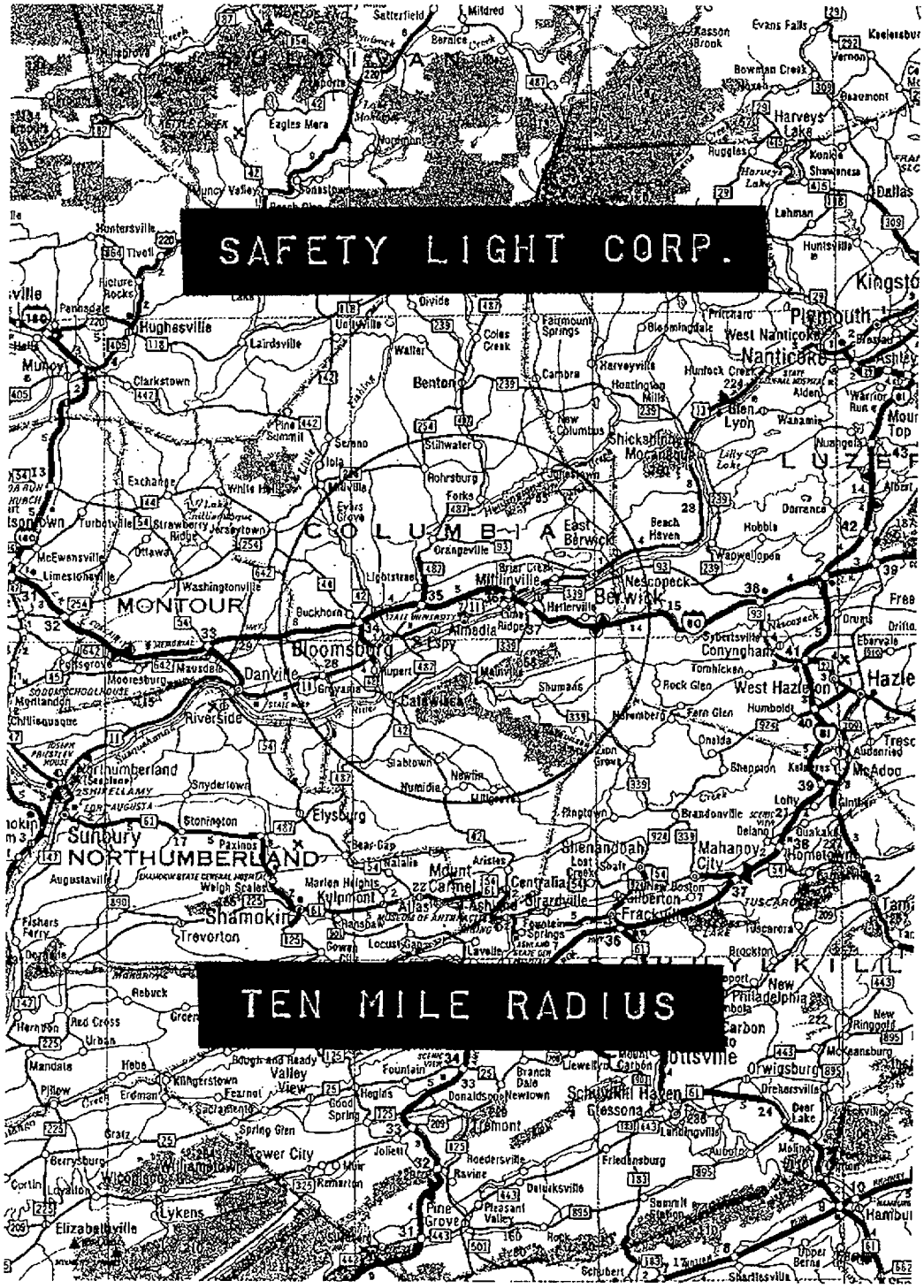


Figure 1-1, Map of 10 Mile Radius Surrounding the Safety Light Site

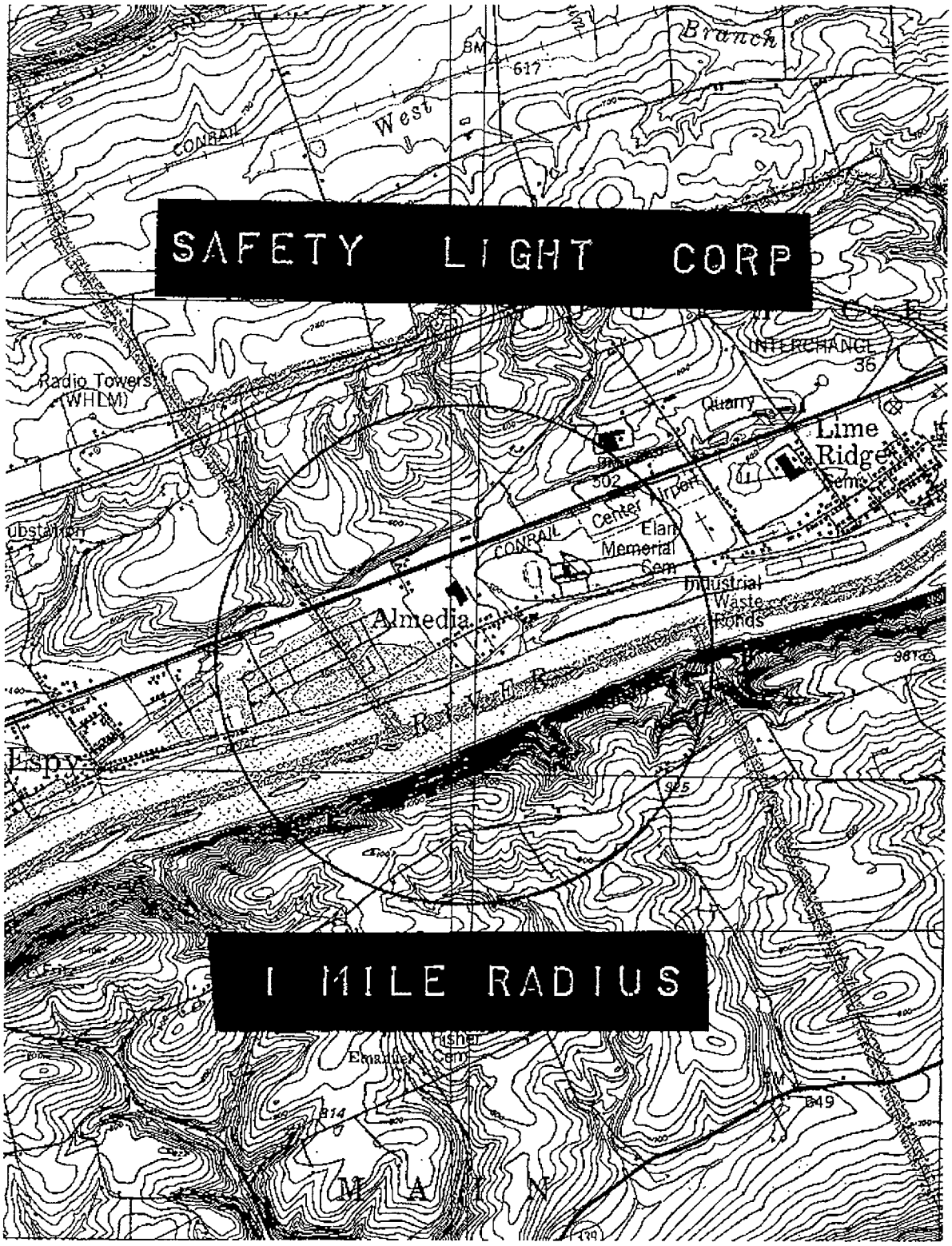


Figure 1-2, Map of 1-Mile Radius Surrounding Safety Light Site

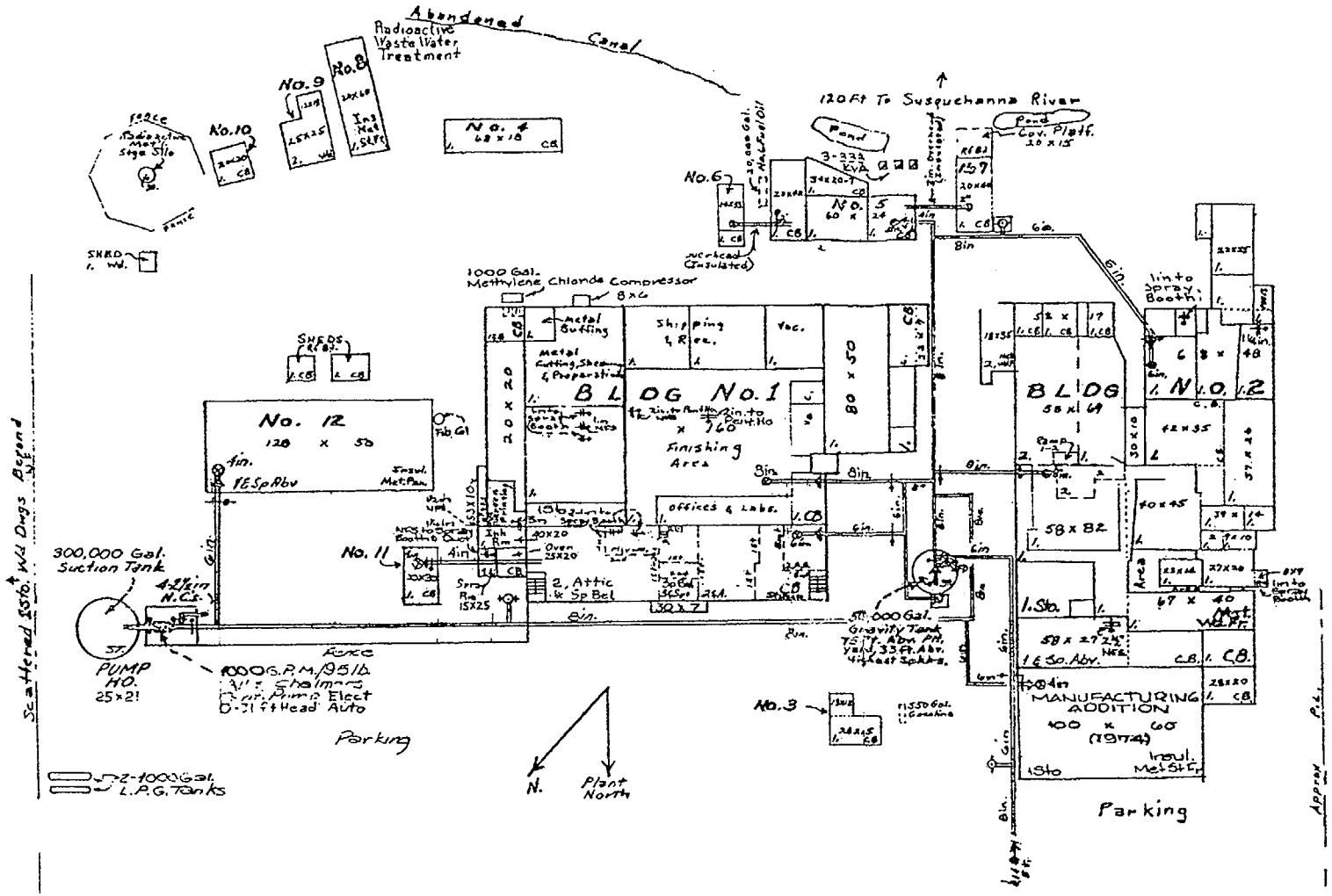


Figure 1-3, Map of Safety Light Site

1.6 References for Section 1

Meiser and Earl, 1979. *Hydrogeological Investigation of Alluvial Ground Water System, U.S. Corporation, Bloomsburg, Pennsylvania.*

Monserco Limited, 1994. *Characterization Plan for Safety Light Corporation Site, Bloomsburg, Pennsylvania U.S.A.*

Monserco Limited, 1996. *Characterization Survey of Safety Light Corporation Site at Bloomsburg, Pennsylvania U.S.A.*

NRC, 1989. *Standard Format and Content of Decommissioning Plans for Licensees Under 10 CFR Parts 30, 40, and 70.* Regulatory Guide 3.65.

NRC, 1998. *Demonstrating Compliance with the Radiological Criteria for License Termination.* Draft Regulatory Guide DG-4006.

NRC Materials License 37-00030-02 Amendment 51,12/28/99.

2 DESCRIPTION OF PLANNED DECOMMISSIONING ACTIVITIES

2.1 Decommissioning Criteria

The overall objective of the Safety Light decommissioning is to remediate retired facilities and soil areas to a condition that corresponds to a calculated dose to the public of less than 25 mrem/year from applicable pathways. The facilities and land may then be released for unrestricted use. This dose limit appears in 10 CFR 20.1402, *Radiological Criteria for Unrestricted Use* (NRC 2000a).

The Derived Concentration Guideline Level (DCGL) is defined in MARSSIM (USEPA et. al. 1997) as the radionuclide-specific concentration within a survey unit corresponding to the release criterion. The DCGL is dependent upon several factors including the radionuclides of interest, applicable dose pathways, area occupancy and the future use of the facility. DCGLs assume a uniform level of residual radioactivity across the survey unit.

For site-specific analysis, licensees may use models other than the DandD code to demonstrate compliance with the 25 mrem criteria, provided they can demonstrate that the model and parameters used in that model are appropriate for the site (NRC 2000b).

The proposed DCGLs for surface contamination at the Safety Light facility are those published in the 1998 Federal Register Notice, *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination* (Table 1) (NRC 1998), supplemented by RESRAD-BUILD calculations for radium-226 and americium-241. (Refer to Attachment 1, Safety Light Dose Assessment, for these calculations and the associated ALARA analysis.) Buildings with surface contamination below these levels will be deemed acceptable for release for unrestricted use provided that (NRC 2000b)

- residual radioactivity has been reduced to levels that are "as low as is reasonably achievable" (ALARA),
- the residual radioactivity is contained in the top layer of the building surface (i.e., there is no volumetric contamination), and
- the fraction of removable surface contamination does not exceed 0.1.

The proposed DCGLs for surface soil (top 15 centimeters) contamination were calculated using RESRAD. (Refer to Attachment 1, Safety Light Dose Assessment, for these calculations and the associated ALARA analysis). These levels will be deemed acceptable for release for unrestricted use provided that (NRC 2000b)

- residual radioactivity has been reduced to levels that are ALARA,
- the residual radioactivity is contained in the top layer of the surface soil (i.e., a thickness of approximately 15 cm),
- the unsaturated zone and the groundwater are initially free of radiological contamination, and
- the vertical saturated hydraulic conductivity at the specific site is greater than the infiltration rate.

Additional subsurface soil analyses were required for the Safety Light site because there is activity below the surface soil layer. Attachment 1, Safety Light Dose Assessment, describes the RESRAD runs used to assign DCGLs for subsurface soil contamination, and the associated ALARA analyis. The numerical release criterion proposed for demonstrating that the dose criterion has been met for soils will be that the sum-of-fractions (quotients of concentrations and DCGLs) of contributing radionuclides for both surface and subsurface soil shall be less than unity. If a survey unit fails to meet this numerical release criterion, the need for additional sampling or remediation will be evaluated.

The following table lists the proposed DCGLs for surfaces and soil.

Table 2-1, DCGLs for Surfaces and Soil

Radionuclide	Surface Contamination DCGL ($dpm/100cm^2$)	Surface Soil Concentration (to 15 cm) DCGL (pCi/g)	Subsurface Soil Concentration 15cm to 2 m deep DCGL (pCi/g)
H-3	1.2 E+08	110	696
Sr-90	8.7 E+03	30.1	6.1
Cs-137	2.8 E+04	14.1	43
Ra-226 + C	3.5 E+03	4	1.8
Am-241	1.8 E+02	10.3	1.1

+C indicates the value for radium-226 with its decay progeny in equilibrium. This value is the concentration of the parent radionuclide, but accounts for contributions from the complete chain of progeny in equilibrium with the parent radionuclide.

2.2 Current Assessment of Residual Radioactive Contamination

The 1995 site characterization (Monserco 1996) included direct measurements of accessible structures and equipment and soil and water sampling. The principle radionuclides of interest are cesium-137, radium-226, strontium-90 and americium-241.

2.3 Decommissioning Organization and Responsibilities

The remediation organization will be integrated into the existing Safety Light organization, and will comply with the existing license and applicable regulatory requirements.

The direct responsibility for operational oversight of activities conducted under NRC License No. 37-00030-02 rests with the Plant Manager. This individual has overall responsibility for operations, engineering, industrial hygiene, safety, security, environmental compliance, radiation protection and quality assurance.

Reporting to the Plant Manager is the Radiation Safety Officer (RSO). This individual is responsible for:

- developing and executing the Health and Safety Program (Safety Light 1998),
- developing and updating the Safety Light Corporation Radiological Contingency Plan (Safety Light 1999),
- developing and executing acceptable onsite and offsite environmental programs, and
- acting as liaison to the U.S. Nuclear Regulatory Commission and the Pennsylvania Department of Environmental Protection.

The plant radiation safety staff reports to the RSO.

The Radiation Safety Committee is responsible for approving the Health and Safety Program and Radiological Contingency Plan, approving new or revised production processes, and reviewing significant radiological incidents involving exposure of individuals or emissions of radioactive material. This committee is comprised of the Plant Manager, Radiation Safety Officer, Radiation Safety Consultant, and Production Manager.

The Decommissioning Contractor Project Manager will report to the Safety Light Plant Manager. This individual will coordinate the activities of the primary decommissioning contractor and any subcontractors.

2.3.1 Decommissioning Project Quality Assurance Plan

A Decommissioning Project Quality Assurance Plan will be developed to identify applicable procedures and requirements based on government and regulatory requirements, contractual commitments, and supplemental quality standards.

2.3.2 Training

Individuals (employees, contractors, and visitors) who require access to a radiologically restricted area associated with the remediation effort shall receive appropriate training commensurate with the potential hazards to which they may be exposed.

Radiation protection training will be provided to personnel who will be performing remediation work in radiological areas or handling radioactive materials. The training will ensure that decommissioning project personnel have sufficient knowledge to perform work activities in accordance with the requirements of the Health and Safety Program and accomplish ALARA goals and objectives. The principal objective of the training program is to ensure personnel understand the responsibilities and the required techniques for safe handling of radioactive materials and for minimizing exposure to radiation.

Records of training will be maintained, which will include the trainee's name, date of training, type of training, test results, authorization for protective equipment use, and instructor's name. Radiation protection training will provide necessary information for workers to implement sound radiation protection practices. The following are examples of the training programs that will be applicable to decommissioning activities.

2.3.2.1 General Employee Training

A general training program designed to provide orientation to project personnel and meet the requirements of 10 CFR 19 will be implemented. General Employee Training (GET) will be required for all personnel assigned on a regular basis to the decommissioning project. This training will include:

- project orientation/access control,
- Introduction to radiation protection,
- quality assurance,
- industrial safety, and
- emergency procedures.

Forty-hour Hazwoper training will also be required for all personnel assigned on a regular basis to the decommissioning project.

2.3.2.2 Radiation Worker Training

Radiation Worker Training (RWT) will be required for decommissioning project personnel working in restricted areas and will be commensurate with the duties and responsibilities being performed. Personnel completing RWT will be required to pass a written examination on the material presented. Completion of this training will qualify an individual for unescorted access to radiologically controlled areas. RWT will include

- fundamentals of radiation;
- biological effects of radiation;
- external and internal radiation exposure limits and controls;
- contamination limits and controls;
- management and control of radioactive waste, including waste minimization practices;
- response to emergencies; and
- worker rights and responsibilities.

In addition to a presentation of the topics identified above, participants will be required to participate in the following demonstrations:

- proper procedures for donning and removing a complete set of protective clothing (excluding respiratory protection equipment);
- reading and interpreting self-reading and/or electronic dosimeters;
- proper procedures for entering and exiting a contaminated area, including use of proper frisking techniques; and
- use of a Radiation Work Permit (RWP).

Personnel who have documented equivalent RWT from another site may be waived from taking training except for Safety Light site-specific training, and will be required to pass the written examination and demonstration exercises.

2.3.2.3 Respiratory Protection Training

Individuals whose work assignments require the use of respiratory protection devices will receive respiratory protection training in the devices and techniques they will be required to use. The training will comply with the requirements of 10 CFR 20, Subpart H, Regulatory Guide 8.15, NUREG-0041, and 29 CFR 1910.134. Training will consist of a lecture session and a simulated work session. Personnel who have documented equivalent respiratory protection training may be waived from this training.

2.3.3 Procedures

Safety Light Corporation will conduct decommissioning activities in accordance with written procedures approved by Safety Light management. Procedures will be prepared, reviewed, revised, approved, controlled, and implemented to ensure that operations are performed in a safe manner.

2.3.4 Contractor Assistance

Contractors and subcontractors performing work under this Decommissioning Plan shall comply with applicable Safety Light Corporation procedures.

2.4 Decommissioning Activities and Tasks

Decommissioning activities under Material License No. 37-00030-02 will be performed while production operations under Material License No. 37-00030-08 are ongoing. A stipulated amount is set aside from operating funds each month for site decommissioning. Additionally, Safety Light currently leases portions of the facility to USR Metals, Inc. for operations that do not involve use of radioactive materials. Therefore, the proposed order and schedule for decommissioning activities and tasks takes into account impact on current operations and budgetary considerations.

2.4.1 Task Descriptions

For purposes of decommissioning planning and cost estimating, the activities that will be required to decommission the site consist of the discrete tasks described below. This section presents a brief description of each facility and soil area (refer to Figure 1-3, Map of Safety Light Site), summarizes relevant characterization data, and discusses the planned method of remediation. Section 3 discusses radiological controls that will be in place during decommissioning activities.

2.4.1.1 Remove Specified Rooms from the Etching Building

Under this task, rooms 21, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, and 45 of the Etching Building (Refer to Figure 2-1) would be dismantled. These rooms are slated for early demolition due to their poor material condition. They are structurally independent from the rooms that will remain. Many of these rooms are not safe to enter, and therefore were not surveyed during the 1995 site characterization, as indicated in the following table:

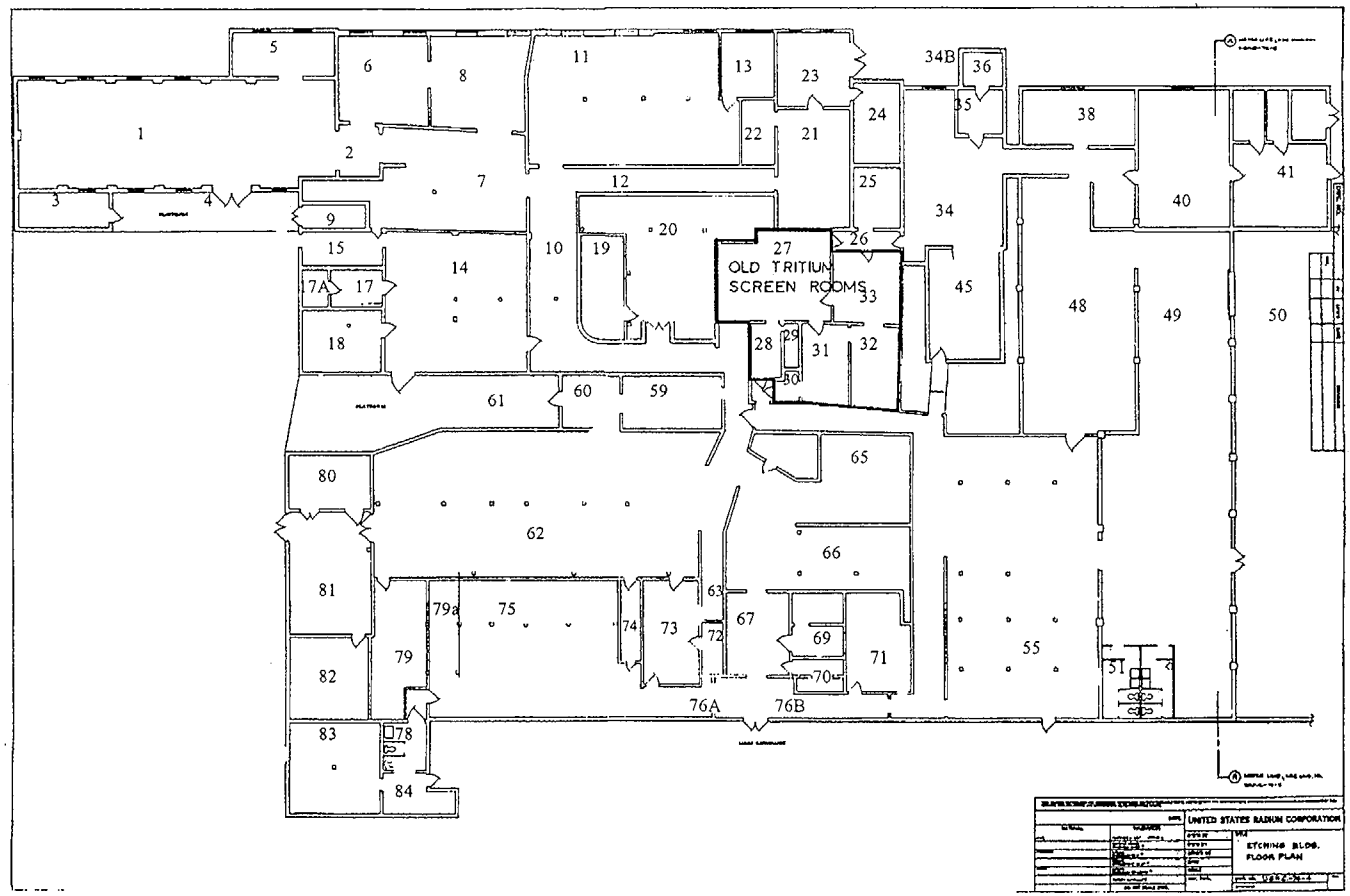
Table 2-2, Rooms in Etching Building Scheduled for Early Removal

Room No.	Description	Characterized
21	Unknown	No
23	Unknown	No
24	Unknown	No
25	Unknown	No
26	Hallway between tritium screening rooms	No
27	Old tritium screening room	No
28	Old tritium screening room	No
29	Old tritium screening room	No
30	Old tritium screening room	No
31	Old tritium screening room	No
32	Old tritium screening room	No
33	Old tritium screening room	No
34	Laboratory	Yes
35	Spray booth	Yes
36	Unknown	Yes
45	Joe Sharpless machine shop	No

The exhaust ducts, absolute filter bank, blower, and discharge stack for the tritium screen rooms are intact.

These rooms will be demolished and removed from the site for processing or disposal at a licensed radioactive waste disposal facility. Following building demolition, the surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

Figure 2-1, Etching Building



2.4.1.2 Remove Soil North of Well 5

Table 2-3, Maximum Activity in Soil Samples North of Well 5, summarizes the maximum radioactivity concentrations reported in surface soil samples from grids 250-256 and 228-234 during the characterization study. Three wells were also drilled during the characterization study within these grids: Well M3 in grid 233, Well M4 in grid 229, and Well M5 in grid 250. The water level at Well M3 was about 1.2 meters. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

Table 2-3, Maximum Activity in Soil Samples North of Well 5

Radionuclide	Maximum Activity (pCi/g)
Cesium-137	142
Bismuth-214	294
Lead-214	305
Radium-226	421
Beta (based on Sr-90)	191

2.4.1.3 Remove Ion Exchange Building and Surrounding Soil

This small room, originally used for cesium wastewater treatment, no longer contains any equipment or material. The characterization survey indicated all grids had fixed contamination and hot spots. The building will be demolished and removed from the site for processing or disposal at a licensed radioactive waste disposal facility. Following demolition, the surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.4 Remediate Old Garage Foundation and Surrounding Soil

The old garage was used to store radioactive materials beginning in the 1940s. A cesium-137 source reportedly exploded in the garage. The structure has been removed leaving only the cement foundation. Elevated cesium-137 levels were found in this area (grids 128, 129, 150, 151) during the characterization survey. The garage foundation will be decontaminated by scabbling, or completely removed if necessary. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.5 Remove Soil Under Old Loading Dock

A surface soil sample taken from the 10 by 10-meter grid 116 during the characterization study had 3,335 pCi/g radium-226. The surface and subsurface soil will be surveyed, and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.6 Remove 8 x 8 Block Building and Surrounding Soil

This building was previously used to store strontium-90 deck markers, and is currently used to store tritium-contaminated equipment. The results of surface soil sample analyses from grid 110 during the characterization study were 7,265 pCi/g cesium-137, 90 pCi/g bismuth-214 (a radium-226 daughter), and approximately 900 pCi/g beta (based on strontium-90). The building will be demolished, and the foundation will be decontaminated by scabbling, or completely removed if necessary. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.7 Remove Old Radium Vault and Surrounding Soil

This structure was originally used for pouring lead and handling and storing radium bromide, radium foil, and radium radiation sources. The radioactive materials have been removed from the building. This building is unsafe to enter due to its poor physical condition, and therefore the interior walls and floor were not characterized. The NRC reported high levels of alpha contamination in this building in 1970. The building will be demolished, and the foundation will be decontaminated by scabbling, or completely removed if necessary. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.8 Remove Soil Around Machine Shop

Surface soil samples from grids 44, 45, 46, 47, 18, 19 and 20 north of the machine shop had elevated radium 226 levels. The maximum radium-226 concentration was 891 pCi/g in grid 45. The maximum beta activity (based on strontium-90) was 481 pCi/g in grid 46. This area of radium-226 contaminated soil may be the result of traffic from the old radium laboratory. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.9 Remove Above-Ground Silo and Surrounding Soil

This structure, located in grid 173, was originally used to store radium-226 ionitrons. Tritium-contaminated waste is currently stored in the silo. The only elevated activity in grid 173 during the characterization survey was 21 pCi/g cesium-137. The building will be demolished, and the foundation, which is degrading, will be decontaminated by scabbling, or completely removed if necessary. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.10 Decontaminate Remaining Site Buildings and Structures

This decommissioning plan and associated cost estimate does not include decommissioning of the nuclear building (No. 12), machine shop (No. 11), solid waste building (No. 10) and liquid waste building (No. 8). These buildings are currently used for operations under Material License No. 37-00030-08.

Before this phase of decommissioning begins, Safety Light will determine the desired disposition of the remaining building contents (recycle/reuse or dispose). Based on cost considerations, contents may either be surveyed for potential release onsite or packaged and shipped to a licensed radioactive waste processor for survey/processing.

In addition to the buildings mentioned above that are currently used for tritium operations, this plan assumes the main building, etching building, and USR Metals liquid waste building will remain after decommissioning. All other buildings will be demolished as part of the decommissioning effort. In general, for the buildings left standing, hot spots and elevated grids identified in the characterization survey will be decontaminated using appropriate techniques as described in later sections.

2.4.1.10.1 Main Building (No. 1)

This building occupies 30,000 square feet, and contains 73 rooms, three basements, and two loading docks. The original 13,600 square foot building was built in 1945, with a one-story 14,000 square foot addition constructed in 1948-49. Later a 2,000 square foot one-story extension was added. In 1968, the radium and tritium hand application area on the second floor was partially decontaminated, but contaminated ductwork remains. Contamination may also be present between the floor of this area and the ceiling below. The main roof was decontaminated in 1969. The main building is currently used by both Safety Light and USR Metals. The first floor houses administrative offices for both companies, and is also used for storage and non-radiological operations. The second floor is used for storage. The third floor "art room" is currently not accessed due to pigeon infestation. A sewer grate behind the building is also contaminated.

Due to personnel occupation in the building, preliminary remediation efforts will require special care and scheduling. Temporary containment tents and ventilation systems will be used to minimize the potential for airborne dust and radiological contaminants affecting occupied areas. Remediation will include equipment removal, scabbling of contaminated concrete floor surfaces, and decontamination of wall surfaces by surface cleaning or surface removal methods.

2.4.1.10.2 Etching Building (No. 2)

The original etching building was built during the 1940s and had a floor area of approximately 16,000 square feet. The building was expanded to 32,000 square feet between 1949 and 1976. A 6,000 square foot manufacturing addition was constructed in 1974. The current building contains 84 rooms and 3 attics. Sixteen rooms in the tritium screening area that are in poor physical condition are scheduled for early removal. The primary radiological production activities that took place in this building were assembly and manufacture of radium and tritium instruments and dials. A 12" concrete floor in the maintenance wire enclosure area covers an old drain contaminated with radium. USR Metals currently leases approximately 25 percent of this building, and parts of the building are used by Safety Light for storage and assembly of nonradioactive components for exit signs. Safety Light also uses portions of the building, including the attic, for document storage.

Due to personnel occupation in the building, preliminary remediation efforts will require special care and scheduling. Temporary containment tents and ventilation systems will be used to minimize the potential for airborne dust and radiological contaminants affecting occupied areas. Remediation will include equipment removal, scabbling of contaminated concrete floor surfaces, and decontamination of wall surfaces by surface cleaning or surface removal methods.

2.4.1.10.3 Personnel Office Building (No. 3)

This building was originally a nurse's station. It was subsequently used for administrative office space and storage of radium-226 and strontium-90 screening machines and strontium chloride. The building is in poor structural condition and is used for storage of miscellaneous items. A dry well in the basement may have been used for radioactive waste disposal. The well cover has high levels of contamination, but the well itself was not characterized. After the personnel office building is demolished, the dry well will be characterized and excavated as needed. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.10.4 Lacquer Storage Building (No. 4)

This building is used to store solvent drums, and has no history of radioactive materials use. Following final survey, this building will be demolished. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.10.5 USR Metals Liquid Waste Building (No. 5)

The carpenter shop at the rear of this building was used to store radium in the late 1940s. A strontium-90 source reportedly exploded on the east wall. This part of the building will be decontaminated by scabbling and surface cleaning.

The main liquid waste building currently includes a boiler room, a waste room where waste metals are treated, and a compressor room. No operations involving radioactive materials were reportedly performed in this building. Remediation will include equipment removal, scabbling of contaminated concrete floor surfaces, and decontamination of wall surfaces by surface cleaning or surface removal methods. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.10.6 Well House (No. 6)

The well house consists of three rooms, including the north and south adhesives rooms. The old water supply well is probably contaminated with radium-226. The south adhesives room was decontaminated in 1958. Following decontamination, this building will be demolished and the well grouted. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.10.7 Pipe Shop (No. 7)

The pipe shop was constructed in the 1940s over the portion of the abandoned canal that served as a disposal area for radium-contaminated ductwork from the United States Radium Corporation facility in Brooklyn, New York. The building was originally used for maintenance and lead melting. The primary radionuclides used in the building were radium and tritium. It is now used to store tritium screening machines, painting tables, and lead melting pots. It is posted as an airborne radioactivity area and ventilated to reduce radon concentrations. This building will be demolished. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be

excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.10.8 Old House (No. 9)

The 2-story old house, built in the 1800s, is in poor structural condition. A fire in 1998 destroyed the roof and attic. The first floor of the old house is currently used to store plastic-covered pipes and ductwork removed from the tritium processing building. The second floor is not used. The building has a dug out earth basement. Runoff water sampled after the fire contained 1×10^5 pCi/l tritium. This building will be demolished. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.10.9 Strontium-90 Sources Vault

This building was used to store strontium-90 solutions. It is currently used to store non-contaminated equipment. Surface contamination levels on the floor of this building during the Monserco characterization were as high as 1.2×10^6 dpm/100 cm^2. This building will be demolished. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11 Decontaminate Remaining Site Soils

The grids with higher levels of contamination will be remediated first. This will lower the background for direct measurements, and minimize the potential for cross-contamination. The primary radionuclides of interest in soil based on the results of the characterization are cesium-137, radium-226, americium-241, and their daughter products.

2.4.1.11.1 Subsurface Soil Surrounding Underground Silo Area

During the 1950s, these silos were used for disposal of miscellaneous solid radioactive waste. Silo remediation was completed in 2000. The base of the silos remains, as well as surrounding contaminated soil.

Wells M9 and M10 were drilled south of the west and east silos, respectively, during the characterization survey. The maximum gross beta counts (based on strontium-90) in soil samples collected during well drilling were 55.5 pCi/g for well M9 at 3.66 meters and 249.5 pCi/g for well M10 at 4.88 meters. The beta activity in well M10 samples peaked between 3.66 and 5.49 meters below ground surface. The soil from well M10 contained a

maximum of 82 pCi/g cesium-137 at a depth of 0.61 meters. The maximum cesium-137 concentration in soil from well M7, drilled northwest of the silos, was 4 pCi/g at 2.44 meters, with no detectable gross beta activity.

The area will be excavated and the contaminated subsurface silo remains and contaminated soils will be removed. The surface and subsurface soil in this area will be surveyed and soils in excess of the DCGL limits will be excavated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.2 Abandoned Canal Area

The rock-lined canal extends along the southern boundary of the Safety Light site parallel to the Susquehanna River. The onsite portion of the canal covers approximately 37,500 square feet. Up to seven lagoons were constructed in the canal using earthen dams as partitions. The canal is now filled with earth with the exception of two lagoons. Between 1948 and 1949, a portion of the canal may have been used to dispose of radium-226 contaminated ductwork from United States Radium Corporation in Brooklyn, New York. Liquid waste from production activities was routed to the open portions of the canal before 1960. During the 1960s, the water in the three eastern lagoons was treated and the two easternmost lagoons were backfilled. The third easternmost lagoon was backfilled in the late 1970s. The surface and subsurface soil in this area will be surveyed, and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.3 East and West Lagoon Area

From 1948-1954, the existing east lagoon, which covers approximately 250 square feet, was used to dispose of sewage and process wastewater from the radium laboratory in the main building. In the 1960s, the east lagoon contents were pumped to the west lagoon. The west lagoon, which covers approximately 750 square feet, was also used to dispose of silver plating waste and anodizing solutions. The lagoons flooded in 1972, causing contamination of the surrounding soil.

Wells M4 and M5 were drilled east and southeast of the east lagoon area, respectively, during the characterization survey. The maximum gross beta counts (based on strontium-90) in soil samples collected during well drilling were 37.4 pCi/g for well M4 at 4.27 meters and 46.2 pCi/g for well M5 at 0.61 meters. The maximum radium-226 concentration in soil samples collected during well drilling was greater than 30 pCi/g for well M4 at 1.22 meters and 215 pCi/g for well M5 at 0.61 meters.

The lagoons will be pumped out, and the recovered water sampled and routed to the river. The surface and subsurface soil in the lagoon area will be surveyed and soils in excess of

the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.4 East Plant Dump

The east plant dump is located on the bed of the abandoned canal near the pipe shop, and may contain strontium-90 contamination. Contaminated ductwork from the Brooklyn, New York site disposed of in this area may be causing the elevated radium--226 concentrations noted during the characterization survey. The surface and subsurface soil in this area will be surveyed and soils and any ductwork found in this area will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.5 West Plant Dump

The west dump may have been used for disposal of solid waste in the 1940's. In the early 1970s, 12,000 pounds (78 drums) of contaminated soil were removed from this area and shipped offsite for disposal. A ground penetrating radar survey performed during the site characterization showed small metallic objects, possibly radium dials, buried at the west dump. This survey did not indicate the presence of buried drums.

Well M13 was drilled in grid 170 in the west dump area during the characterization survey. The maximum concentration of radium-226 in soil samples obtained during well drilling was greater than 10 pCi/g at a depth of 0.61 meters.

The surface and subsurface soil in this area will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.6 Drain Lines

A cement trough with sewer grates behind the main building may have radium-226 and strontium-90 contamination. This was part of the conveyance system that transported process water from the main building to the east lagoon. Information on the location of all drain lines and potential contamination is not available. The cement trough will be surveyed and removed if contaminated. The drain lines will be located via excavation or other means, surveyed, and removed if contaminated. All radioactive materials will be removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.7 Drywell Tanks

The drywell below the personnel building is discussed in Section 2.4.1.10.3.

2.4.1.11.8 Soil by Old Berwick Rd.

A soil pile deposited near the northwest corner of the site during construction work has radium-226 contamination. The pile is 30 by 15 feet, and 2 to 5 feet high. This soil pile will be removed. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.9 Soil from Vance-Walton Property

Soil contaminated with radium-226 and cesium-137 found on the Vance-Walton property was transferred inside the fenced area south of the above-ground silo. The soil mound is 16 by 10 feet and 2 to 4 feet high. This soil pile will be removed. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.10 Soil North of Lacquer Storage Building

This area may have been used for solvent disposal. The surface and subsurface soil will be surveyed for radiological and chemical materials. Soils in excess of hazardous material limits or DCGL limits will be excavated and removed from the site for disposal at an appropriate waste disposal facility.

2.4.1.11.11 Employee Parking Area

The graveled employee parking areas include an east section, approximately 125 by 125 feet (included in grids 10, 11, 18, and 19), and a west section, approximately 80 by 125 feet (included in grids 6,7,15, and 16). Based on the results of the characterization survey, some areas with elevated levels of radium-226 may be present in both parking lots. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.12 Sidewalk Areas

The sidewalks will be surveyed and decontaminated by scabbling or removed as necessary. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.13 Site Paved Roads

The roads will be surveyed and decontaminated by scabbling or removed as necessary. The surface and subsurface soil will be surveyed and soils in excess of the DCGL limits will be excavated and removed from the site for disposal at a licensed radioactive waste disposal facility.

2.4.1.11.14 All Other Land Inside Fenced Area

Thirteen wells were drilled and subsurface soil samples were taken during the site characterization. In addition to the results previously discussed in relation to identified remediation areas, the following results were documented:

- Grid 180 north of the lacquer storage building had no detectable activity
- Grid 240 near the southeast corner of the property had 48 pCi/g cesium-137 at 1.83 meters
- Grid 233 near the southwest corner of the property near the effluent stream had a maximum of 8.8 pCi/g gross beta (based on strontium-90) at 1.22 meters and greater than 48 pCi/g cesium-137 at 1.22 meters.
- Grid 244 south of the liquid waste building at the fence line had a maximum of 36.8 pCi/g gross beta (based on strontium-90) at 0.61 meters and 102 pCi/g radium-226 at 0.61 meters.
- Grid 178 at the southwest corner of the liquid waste building had a maximum of 2.98 pCi/g gross beta (based on strontium-90) at 3.66 meters and greater than 4 pCi/g radium-226 at 0.61 meters.
- Grid 203 immediately south of the lacquer storage building had a maximum of 9.6 pCi/g gross beta (based on strontium-90) at 2.58 meters and greater than 28 pCi/g radium-226 at 2.58 meters.
- Grid 223 to the south and between the liquid waste building and lacquer storage building near the fence line had greater than 10 pCi/g radium-226 at 0.61 meters and no detectable beta activity.

Contaminated soil was found to a maximum depth of 5.49 meters (grid 161 southeast of the well house) during the well drilling operation.

Radioactivity above NRC guidelines was found in seven onsite wells.

2.4.2 Remediation Methods

Remediation methods are dependent on the facility to be remediated, type and extent of the radionuclides, and the remediation objective. Contaminated facilities and soils will fall into one or more of the following categories for remediation:

- Remove and dispose of as radioactive waste
- Decontaminate for unrestricted release
- No action since radiological conditions meet acceptance criteria

Each facility and area has been evaluated to determine the best method for remediation. Criteria that were used in the evaluation include: availability of a burial facility, the cost of decontamination versus the cost of burial, radiological and occupational hazards involved, and site operations in progress or planned.

Structures, equipment, and components will be removed using proven mechanical/thermal cutting and demolition equipment. Mechanical methods include saw cutting, concrete scabbling, the use of jackhammers, and tractor loader. Thermal methods include metal cutting with an oxy-acetylene torch, plasma arc cutting, and oxy-lance cutting methods. Contaminated soils will be excavated.

The following sections describe commercially available methods of decontamination that may be used during the Safety Light decommissioning project. Table 2-4, Equipment Selection Matrix, summarizes the most suitable decontamination methods based on type and location of contamination.

2.4.2.1 General Surface Decontamination

2.4.2.1.1 Strippable Coatings

Strippable coatings are useful for removing loose contamination from large surface areas and protecting areas that may become contaminated during work activities. If the coating is used to protect an uncontaminated surface, the coating can be stripped away if it becomes contaminated, or the contamination can be sealed in place with a second layer of coating and then stripped away. Strippable coatings are applied using airless sprayers, paint rollers, or brushes.

Table 2-4, Equipment Selection Matrix

Decontamination Method	Loose Surface Contamination – Large Surface Area	Loose Surface Contamination – Small Surface Area	Loose Surface Contamination – w/o Water Treatment or Containment	Fixed Contamination – Floors/Walls – Large Areas	Fixed Contamination – Floors/Walls – Isolated Areas	Fixed Contamination – Concrete Removal Required	Compressed Air Required for Operation	Electrical Service Required for Operation	HEPA Vacuum Required for Operation	Water Source Required for Operation
Blastrac Unit				✓				✓	✓	
Sand Blaster	✓		✓	✓			✓			
Pentek Corner Cutter					✓		✓		✓	
McDonald HF 1-Piston Scabbler					✓		✓			
McDonald 3WCD 3-Piston Scabbler					✓		✓		✓	
McDonald U-5 5-Piston Scabbler				✓			✓		✓	
Pentek Squirrel III Scabbler					✓		✓		✓	
EDCO Floor Plane				✓					✓	
Power Washer 1-5K psi	✓							✓		✓
Hydrolaser 5-10K psi	✓							✓		✓
ULTRA High Pressure Unit 35-40K psi	✓			✓	✓			✓		✓
Jack Hammer						✓	✓			
Bristar Expandable Grout						✓				
Hands on Decontamination		✓	✓							
Strippable Coatings	✓	✓	✓				✓			

2.4.2.1.2 Vacuuming/Scrubbing/Wiping

These techniques are appropriate when gross loose surface contamination is visible on the targeted surface. Vacuum operations use systems equipped with a HEPA filter. If a wet vacuum is required for liquid retrieval, the vacuum system also includes an automatic water shut-off system to prevent destruction of the HEPA filter when the unit is full. Scrubbing and wiping techniques are appropriate where access is limited. Vacuums can be used on any type of surface, but scrubbing/wiping is normally used on smooth, nonporous surfaces.

2.4.2.1.3 Pressurized Water

Pressurized water spraying may be used for general area decontamination or decontamination of items in a confined space. This method would only be used in areas where the spent water can be directed into a drain, sump, or some other available means of collection. The contaminated water would be treated before discharge. Pressure spray methods include:

- Low Pressure Spray (Power Wash)

  Water is sprayed on the surface to be decontaminated with the objective of removing loosely adhered contamination. This technique is effective on coated surfaces that allow the contamination to be easily removed. Water pressure is generally in the 1,500 to 5,000 psi range with water consumption typically 3 to 6 gallons per minute.

- High Pressure Spray (Hydrolaser)

  A powerful stream of water is applied to the surface in a side to side, top to bottom fashion. This method is appropriate for large surfaces and complex structures or equipment. This technique is effective on coated and uncoated contaminated surfaces. The water can be applied at various temperature ranges and the addition of chemical agents may increase the overall decontamination factor. High-pressure sprays typically operate over a pressure range of 5,000 to 20,000 psi, with a water consumption rate of about 5 gallons per minute.

- Ultra High Pressure

  Ultra High Pressure water can be used two ways. The first method is to direct a precise stream of water from a multi-jet rotating nozzle at the target surface. This method is capable of removing loose and fixed contamination from the surface. Ultra High-Pressure water is capable of removing ¼ to ½ inch of concrete surface per pass with this configuration. The second method uses a single nozzle that is capable of cutting material from the surface. Water consumption varies with nozzle selection;

rates from 2 to 6 gallons per minute are typical with water pressure approaching 40,000 psi.

- Steam Lance

  Saturated steam is directed on the surface to be decontaminated. Crystalline materials can be made soluble and particulates removed using this technique. This method may be useful for surfaces coated with sludge or oil where chemicals and degreasers are not permitted.

2.4.2.2 Concrete Surface Removal

2.4.2.2.1 Scabbling

Scabblers remove the surface by impacting the area with air driven tungsten carbide tipped bits. Scabbler sizes range from single-piston units suited for small constricted or isolated areas, to multi-piston units designed for operation in large open areas. Surface removal can vary from a light single pass removing 1/16 inch to multiple drawn out activities removing 1 inch or more. HEPA filter vacuum units are attached to shrouds around the scabbler to control airborne radioactivity, when necessary.

2.4.2.2.2 Scarification

Scarifiers also remove a surface layer of material from concrete floor slabs or similar surfaces. This equipment is most suitable for wide-open contaminated areas. The scarifier is a mechanically powered (electric, gas, or propane) device that removes surface layers of material with a rotating drum equipped with tungsten carbide-tipped cutters. During operation, the bits are forced against the surface at a predetermined depth and lateral speed. A HEPA filtered vacuum system operated in conjunction with a vacuum shroud attached to the scarifier controls airborne activity during operation. Typical surface removal depths vary between 1/16 to ¼ inch per pass. On a smaller scale, hand held grinders can also be used to remove surfaces from concrete.

2.4.2.2.3 Needle Gun

A needle gun operates by pneumatically driving specially hardened needles into the surface to be removed. It is designed to remove surface material from small areas or restricted spaces. The process takes place within a vacuum shroud, preventing the escape of dust, debris, and airborne contamination. The vacuum shroud is connected to a HEPA filtered vacuum system, which provides the negative pressure required.

2.4.2.2.4 Abrasive blasting

Abrasive blasting (refer to Section 2.4.2.3.1) is usually used for metal surface removal, but it can also be used effectively for concrete surface removal. Blastrac (refer to Section 2.4.2.3.2) is commonly used for decontamination of concrete floors.

2.4.2.3 Metal Surface Removal

2.4.2.3.1 Abrasive Blasting

Abrasive blasting removes loose and fixed contamination from surfaces with a high production rate. This technique uses non-hazardous abrasive material (e.g., sand, steel, walnut shells, and plastic) suspended in a medium (air or water) that is propelled against the targeted surface. Supplemental HEPA filter ventilation may be used to control airborne dust and contamination. Abrasive blasting is well suited to overhead and vertical surfaces. Depending on the equipment used and contamination levels encountered, the blasting medium may be reused.

2.4.2.3.2 Recycled Abrasive Blast (Blastrac)

Shot blasting is an airless method for stripping and cleaning surfaces. Selecting the proper shot size and residence time controls the rate of surface removal. The shot is propelled at the surface using a centrifugal blast wheel. As the wheel spins, the abrasive is hurled from the blades, blasting the surface with the media. The abrasive is continuously recycled using a vacuum system in conjunction with a separation system. Supplemental HEPA filter ventilation is required to control airborne dust and contamination.

2.4.2.4 Demolition and Component Removal

Decontamination of floors, walls, and other structures may require complete or partial removal of these items. Methods used for concrete removal include jackhammers, expandable grout, concrete saws, and diamond wire saws.

2.4.2.4.1 Jackhammer

Jackhammers are the most commonly used tool to remove concrete. Jackhammers range in size from hand-held units to large hoe rams mounted on tracked excavators. The concrete is degraded through constant pneumatic impact of a chisel-pointed bit. This method works well but is noisy and produces large quantities of dust and debris.

Typically a containment tent is constructed over the area and supplemental HEPA filter ventilation controls airborne contamination. Where large areas of concrete must be removed, it may be cost effective to decontaminate the concrete to acceptable levels by other means and then use large hoe rams to remove the concrete. The work area is constantly sprayed down with water to control airborne dust and contamination

2.4.2.4.2 Expandable Grout

Expandable grout may be used for demolition of concrete structures or removal of predetermined layers of concrete from structures. Holes are systematically drilled into the concrete. The grout is then mixed with the appropriate quantity of water and poured into the pre-drilled holes. As the grout hardens, it expands and cracks the concrete apart. The advantages of this method include demolition without noise, vibration, airborne releases, or disruption to nearby activities.

2.4.2.4.3 Diamond Wire Cutting

Diamond wire cutting techniques can remove large segments of concrete. A diamond-studded cable is circulated by a hydraulic pulley drive system through the concrete, cutting through concrete, steels rebar, and other steel members in the concrete. Hydraulic cylinders control the tension of the cable. Holes are drilled through the concrete to enable stringing the cable into cutting target areas that would otherwise be inaccessible. Water applied to cool and lubricate the cable also aids in control of airborne dust and contamination. A slurry collection system collects contaminated cutting slurry, decants the slurry, and recycles the water.

2.4.2.4.4 Tractor Loader

A tractor loader or similar equipment can selectively demolish buildings and load the resulting debris into containers for shipping and disposal.

2.4.2.4.5 Other Cutting Equipment

Steel pipes are generally removed using mechanical or thermal cutting methods, such as hand-held band or reciprocating saws or an oxy-acetylene cutting torch. Oxy-lance and plasma arc cutting methods are also commercially available for cutting larger items such as steel tanks.

2.4.2.5 Soil Excavation

2.4.2.5.1 Stormwater Management and Erosion Control

The areas to be excavated will be isolated from potential run-on and run-off. In general, berms and stormwater diversion structures/ditches will isolate disturbed areas. Excavations will be maintained free of water when possible. Accumulated water will be removed, tested, and released or treated if required. Excavated material will be covered or containerized in designated restricted areas to prevent erosion before final disposition.

2.4.2.5.2 Equipment

The following conventional earthwork equipment may be used for excavation at the Safety Light site: long-reach trackhoe, backhoe, bobcat loader, gradall or similar equipment, all terrain forklift, and soil conditioning and compacting equipment.

2.4.2.5.3 Radiological Control/Equipment Decontamination

Equipment will be checked for contamination before exiting the designated restricted area. Equipment will be decontaminated in the restricted area if necessary. The restricted area will be marked by flags, physical barricades, or fencing as appropriate.

2.4.2.5.4 Occupational Safety

Shoring and bracing will be installed as required to protect personnel, structures, and utilities.

2.4.2.5.5 Excavation procedures

Whenever feasible, the excavation equipment will be located outside of the excavation area to minimize the potential for contamination. Normally, earthwork equipment used within the excavation will have rubber tires. The selection/size of the equipment will vary with the depth and extent of the planned excavation. Areas adjacent to the excavations will be covered with polyethylene or similar material to catch spills from the excavation equipment. The excavation operation will be monitored and guided using appropriate field radiation survey instruments.

2.4.2.5.6 Loading/Shipping

Loading of shipping containers and hauling equipment will be controlled to minimize the potential for contamination on external surfaces. Containers/loads will be secured or covered. Material designated for offsite disposal will be removed from the excavation, placed in packaging that meets DOT requirements, and staged in a secured area that precludes inadvertent removal from the site, pending shipment.

2.4.2.5.7 Staging/Segregation

Additional soils removed to gain access to the areas of elevated concentrations of radioactive material or excavated in conjunction with the removal of structures will be staged, segregated, and stockpiled based on levels of radioactivity present, to the extent practical. Stockpiles will be identified and covered to prevent erosion, fugitive emissions, and inadvertent removal.

2.4.2.5.8 Backfill

The excavations will be backfilled as soon as possible with segregated material or clean borrow from onsite or offsite. Excavation locations will be temporarily marked such that the NRC verification survey contractor can easily locate these areas. Backfill material will be sampled and tested to verify that it meets the specified radiological, chemical, and geotechnical criteria, dependent on the future use of the area. These criteria will be called out in a specification.

2.4.2.5.9 Contaminated Soil Segregation

An area will be designated and prepared for the segregation of potentially contaminated soils. Excavated material will be spread over the prepared surface and composite samples will be prepared and analyzed. Soils will be segregated based on activity level to the extent practical. Stockpiles of contaminated soils will be located within secure areas to prevent inadvertent removal or disturbance, identified, and covered with polyethylene or similar material to prevent erosion by wind or rain. Stockpiled segregated soils will then be dispositioned for backfill of excavations or proper packaging for transport to a licensed disposal facility.

2.4.3 Decommissioning Accident Analysis

This section addresses the potential radiological accidents that could occur during decommissioning of the Safety Light facility and affect public or occupational health and safety. The accident analyses show that the doses to the public from potential accidents

are below the U.S. EPA Protective Action Guides (PAGs) developed by the EPA to protect members of the public from the consequences of accidents (USEPA 1992). In addition, doses to workers from potential accidents are below the permitted annual exposure limits. Therefore, no new protective measures are required to protect public or occupational health and safety.

A screening analysis of potential accidents at a level of detail consistent with existing information about the radiological hazards at the Safety Light facility was performed. A screening analysis approach is appropriate for this accident analysis because the radioactive inventories at the Safety Light facility are very small compared to those in operating reactors and in fuel cycle facilities subject to NRC regulation. The screening analysis for the Safety Light facility identified and analyzed plausible accident scenarios that could occur during decommissioning activities. Section 2.4.3.1 identifies potential radiological accidents at the Safety Light facility based on decommissioning activities and radiological hazards. Section 2.4.3.2 contains an analysis of potential accident scenarios to estimate the total effective dose equivalent (TEDE) to a member of the public at the site boundary. Section 2.4.3.3 presents bounding estimates of worker exposure from worst-case accident scenarios.

2.4.3.1 Potential Radiological Accidents

The task of identifying potential accident scenarios included:

- evaluating Safety Light facility areas that contain the highest inventories of radioactive material,
- describing energy sources and external events,
- reviewing proposed activities, and
- considering combinations of these elements that could lead to a release of radioactive material.

Because of the limited inventory, the evaluation of accident scenarios conservatively assumed that no design or procedural controls would be available to prevent or mitigate accidental releases, although such controls will be implemented during decommissioning activities. This assumption resulted in a worst-case accident analysis.

2.4.3.1.1 Highest Radionuclide Inventories at the Safety Light Facility

Radionuclide inventories for various areas of the Safety Light facility were estimated based on data from the Monserco characterization study (Monserco 1996). (Refer to Table 2-5, Estimated Radioactive Material Inventories.) For buildings, these estimates involved calculating a conservative surface contamination level (in dpm/100cm^2) based upon average hot spot data, to be applied to the building's entire interior surface area. Room 88 of the main building has the highest estimated inventory of radium-226, 0.007

curies. Radium-226 has the highest dose conversion factors, and hence will result in the highest doses to the workers and public. The rooms slated for early demolition in the etching building have the highest estimated inventory of the structures to be demolished as part of the decommissioning, approximately 0.004 curies.

The other buildings and structures at the Safety Light facility have lower estimated radionuclide inventories. No accidents were identified for these other buildings and structures that differ from those discussed in Section 2.4.3.1.2. Therefore, the results of accident analyses conducted for room 88 of the main building and the rooms slated for demolition in the etching building bound the potential impacts of inside accidents during the Safety Light decommissioning.

Accidents could also occur outside buildings in areas with identified soil contamination. Most of the area between the major structures and the river is contaminated. This area also has the highest concentration of radionuclides, and contains an estimated 16 curies of cesium-137 and 7.4 curies of radium-226. (Refer to Table 2-5, Estimated Radioactive Material Inventories.) Therefore, the results of the accident analyses conducted for this area bound the potential impacts of outside accidents during the Safety Light decommissioning.

Table 2-5, Estimated Radioactive Material Inventories

	Curies					
Building or Room	**Cs-137**	**Ra-226**	**Sr-90**	**Am-241**	**H-3**	**Total**
Etching Building Rooms for Demolition	2.59E-03	1.39E-03	4.47E-04	4.47E-05	0.00E+00	4.47E-03
Carpenter Shop	0.00E+00	4.47E-03	4.47E-03	0.00E+00	0.00E+00	8.95E-03
Cesium Ion Exchange Hut	3.45E-04	1.84E-04	5.94E-05	5.94E-06	0.00E+00	5.94E-04
Above-Ground Silo	4.68E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.68E-04
Old House	4.37E-04	2.34E-04	7.54E-05	7.54E-06	3.95E-04	1.15E-03
8 by 8 Building	1.31E-03	7.00E-04	2.26E-04	2.26E-05	1.54E-06	2.26E-03
Strontium-90 Sources Vault	5.86E-04	3.13E-04	1.01E-04	1.01E-05	1.00E-06	1.01E-03
Radium Vault	1.86E-06	3.43E-05	0.00E+00	0.00E+00	0.00E+00	3.62E-05
Well House	0.00E+00	1.47E-04	0.00E+00	0.00E+00	0.00E+00	1.47E-04
Main Building Room 88 (Radium Laboratory)	1.36E-04	6.89E-03	0.00E+00	0.00E+00	0.00E+00	7.03E-03
Main Building Room 209 (Hallway)	1.42E-06	7.24E-05	0.00E+00	0.00E+00	0.00E+00	7.38E-05
Main Building Room 214 (Dial Processing Office)	2.21E-06	1.13E-04	0.00E+00	0.00E+00	0.00E+00	1.15E-04
Main Building Room 215 (North Washroom)	2.02E-06	1.03E-04	0.00E+00	0.00E+00	0.00E+00	1.05E-04
Main Building Room 218 (Radium Dial Production)	7.89E-06	4.01E-04	0.00E+00	0.00E+00	0.00E+00	4.09E-04
Main Building Room 301A (Third Floor Stairwell)	1.14E-05	5.78E-04	0.00E+00	0.00E+00	0.00E+00	5.90E-04
Soil Area 1 Adjacent to Old Berwick Road	0.00E+00	4.56E-02	0.00E+00	0.00E+00	0.00E+00	4.56E-02
Soil Area 2 North of Main Building	0.00E+00	2.23E-01	0.00E+00	0.00E+00	0.00E+00	2.23E-01
Soil Area 3 North of Machine Shop	5.47E-04	9.33E-02	0.00E+00	5.19-03	0.00E+00	9.90E-02

Building or Room	Curies					
	Cs-137	Ra-226	Sr-90	Am-241	H-3	Total
Soil Area 4 South of Emergency Water Tank	3.89E-03	1.07E-03	0.00E+00	0.00E+00	0.00E+00	4.96E-03
Soil Area 5 West of Water Tower	0.00E+00	3.12E-04	0.00E+00	0.00E+00	0.00E+00	3.12E-04
Soil Area 6 Next to Main Building at South Side	0.00E+00	2.15E-03	0.00E+00	0.00E+00	0.00E+00	2.15E-03
Soil Area 7 Southern Site to River	1.68E+01	7.70E+00	0.00E+00	1.66E-01	0.00E+00	2.47E+01

2.4.3.1.2 Potential Accident Scenarios

Considering the planned decommissioning activities, accident scenarios that could result in releasing radioactive material as airborne particles small enough to be respirable were evaluated. Such releases could occur during structural demolition operations, a fire during remediation operations, or dropping of a container of radioactively contaminated soil. Because all Safety Light facility buildings are outside of the 500-year floodplain, and releases from soil areas near the river will be minimal based on historical erosion information, extreme precipitation events are not expected to cause offsite radiological impacts. The potential onsite and offsite impacts of accidents will be mitigated by the implementation of Safety Light emergency procedures contained in the Health and Safety Program (Safety Light 1998) and Radiological Contingency Plan (Safety Light 1999).

Based on the decommissioning activities outlined in Section 2.4.1 and the radiological inventories identified in Section 2.4.3.1.1, the following accident scenarios were evaluated:

- During excavation of contaminated soil in the area between the major structures and the river, a container filled with contaminated soil drops, producing airborne radioactive particles.
- During remediation in Room 88, a fire produces airborne radioactive particles.
- During demolition of the designated etching building area, high winds and aggressive demolition methods result in the release of fine dust containing radioactive materials.

2.4.3.2 Evaluation of Public Impact from Accident Scenarios

This section further develops and analyzes the accidents having the greatest potential for offsite impacts to estimate the total effective dose equivalent (TEDE) to a member of the public.

2.4.3.2.1 Assumptions

The following assumptions apply to all of the accident analyses:

- The radionuclide inventories were based on the data provided in the 1996 site characterization (Monserco 1996).

- To be conservative, unfavorable weather conditions for atmospheric dispersion were assumed. For the purposes of this analysis, atmospheric stability class F with a wind speed of 2 m/s (6.6 feet/second) was assumed. This represents a situation with minimal dispersion of a potential radioactive plume. In addition, the radioactive material was assumed to be released at ground level and to remain airborne as it travels downwind.

2.4.3.2.2 Method for Calculating Total Effective Dose Equivalent (TEDE)

The potential consequences of the postulated accidents were quantified by calculating the TEDE to a member of the public at the site boundary using Equation 2-1.[1] Then the calculated TEDE was compared to the U.S. EPA's lower PAG of 1 rem (1000 mrem), to determine whether or not the calculated exposure is acceptable.

Equation 2-1

$TEDE_i = CEDE_i + Ext_i$

Where:

TEDE = total effective dose equivalent
CEDE = committed effective dose equivalent
Ext = contribution from external irradiation
i = radionuclide.

The committed effective dose equivalent (CEDE) is the dose contribution from inhalation as the cloud passes by the receptor. Consistent with the lung model developed by the International Commission on Radiological Protection (ICRP 1979), the CEDE is found by:

Equation 2-2

$CEDE_i = Q_i (\chi/Q) \times B \times D_i$

Where:

Q_i = the total released activity of nuclide i, in Ci

χ/Q = the airborne dosage (concentration integrated over the duration of

[1] This estimate of the TEDE neglects any contribution from gamma rays emitted by radionuclides deposited on the ground. Such doses build up relatively slowly and, if necessary, can be controlled by various countermeasures.

cloud passage) per unit activity released, in s/m³. The derivation of χ/Q presented in Attachment 2 shows that for a distance of 100 meters (0.06 mi) in atmospheric stability class F with a wind speed of 2 m/s, $\chi/Q = 4.52 \times 10^{-3}$ s/m^3.

B = the breathing rate, typically 3.3×10^{-4} m^3/s. (This is the breathing rate for adults during light activity [ICRP 1979]).

D_i = the factor that converts the amount of activity inhaled into the CEDE. Values of D_i are given in Federal Guidance Report No. 11 (USEPA 1988).

The dose contribution from external irradiation is found by:

Equation 2-3

$Ext_i = Q_i\ (\chi/Q)\ F_i$

Where:

F_i = the dose coefficient for air submersion. Values of F_i are given in Federal Guidance Report No. 12 (USEPA 1993).

2.4.3.2.3 Scenario 1: Contaminated Soil Release

The area of contaminated soil at the Safety Light facility having the highest concentration of radionuclides is the area between the major structures and the Susquehanna River. The estimated concentrations of radionuclides are 1,686 pCi/g radium-226, 3,674 pCi/g cesium-137 and 43 pCi/g americium-241. If 20 percent of a 55-gallon drum of this contaminated soil became airborne **in respirable form** (i.e., approximately 5.3×10^4 g, assuming a soil density of 1.28 g/cm^3), the airborne quantities of radionuclides would be 8.99×10^{-5} curies of radium-226, 1.96×10^{-4} curies of cesium-137, and 2.30×10^{-6} curies of americium-241. Using the values of χ/Q and B given in Section 2.4.3.2.2 in Equation 2-1 through Equation 2-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 2-6, TEDE Calculation Table for Scenario 1: Contaminated Soil Release.

Table 2-6, TEDE Calculation Table for Scenario 1: Contaminated Soil Release

Nuclide *i*	Q_I (Ci)	D_I (mrem/Ci)	F_i ([mrem/s]/[Ci/m^3])	CEDE_i (mrem)	Ext$_i$ (mrem)	TEDE_i (mrem)
Cs-137	1.96E-04	3.19E+07	0.0287	0.0093	2.54E-08	0.0093
Ra-226	8.99E-05	8.58E+11	1.1667	115.0	4.74E-07	115.3
Am-241	1.94E-06	4.44E+11	3.0296	1.520	3.14E-08	1.520
Total	2.88E-04	--	--	116.3	5.31E-07	116.3

As shown in Table 2-6, the calculated TEDE is 116 mrem, to which the external dose is a negligible contributor. The TEDE of 116 mrem, derived using conservative assumptions, is less than the U.S. EPA's lower PAG of 1 rem (1000 mrem) (USEPA 1992).

2.4.3.2.4 Scenario 2: Release from Fire During Remediation Activities

The radium laboratory, room 88 of the main building, has the highest estimated radium-226 inventory. Radium-226 has the highest dose conversion factors (D_i and F_i), and hence is the radionuclide of greatest significance for purposes of dose calculations. Assuming the entire area is contaminated at a level corresponding to an average hot spot activity of 279,501 dpm/100cm^2, the estimated inventory of this area totals 0.007 curies, 98% of which is assumed to be radium-226. The construction in this area of the building is concrete block. A fire is assumed to start in and be confined to this room during remediation activities. Combustion of the surface area of the room is assumed to release approximately 10% of the contamination, or 1.41×10^{-5} curies of cesium-137, and 6.89×10^{-4} curies of radium-226. Using the values of χ/Q and B given in Section 2.4.3.2.2 in Equation 2-1 through Equation 2-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 2-7, TEDE Calculation Table for Scenario 2: Release from Fire During Remediation.

Table 2-7, TEDE Calculation Table for Scenario 2: Release from Fire During Remediation Activities

	Q_i	D_i	F_i	$CEDE_i$	Ext_i	$TEDE_i$
Nuclide i	**(Ci)**	**(mrem/Ci)**	**[(mrem/s)/(Ci/m^3)]**	**(mrem)**	**(mrem)**	**(mrem)**
Cs-137	1.41E-05	3.19E+07	0.0287	0.0007	1.82E-09	0.0007
Ra-226	6.89E-04	8.58E+11	1.167	882.042	3.63E-06	882.042
Sr-90	0.00E+00	2.39E+08	0.0279	0.00000	0.00E+00	0.00000
Am-241	0.00E+00	4.44E+11	3.0296	0.000	0.00E+00	0.000
Total	7.03E-04	--	--	882.042	3.63E-06	882.042

As shown in Table 2-7, the calculated TEDE is 882 mrem, to which the external dose is a negligible contributor. The TEDE of 882 mrem, derived using conservative conditions, is less than the U.S. EPA's lower PAG of 1 rem (1000 mrem) (USEPA 1992).

2.4.3.2.5 Scenario 3: Release During Building Demolition

The demolition of the etching building area could result in a release of material as a dust puff from a collapsing building. Collapse of these rooms is assumed to release approximately 20% of the contamination. Thus, collapse of the entire area would release 5.19×10^{-4} curies of cesium-137, 2.77×10^{-4} curies of radium-226, 8.94×10^{-5} curies of strontium-90, and 8.94×10^{-6} curies of americium-241. Using the values of χ/Q and B

given in Section 2.4.3.2.2 in Equation 2-1 through Equation 2-3, the CEDE, the contribution from external irradiation (Ext), and the TEDE were calculated as shown in Table 2-8, TEDE Calculation Table for Scenario 3: Release During Building Demolition.

Table 2-8, TEDE Calculation Table for Scenario 3: Release During Building Demolition

Nuclide i	Q_I (Ci)	D_I (mrem/Ci)	F_i ([mrem/s]/[Ci/m^3])	CEDE_i (mrem)	Ext$_i$ (mrem)	TEDE_i (mrem)
Cs-137	5.19×10^{-4}	3.19×10^{7}	0.02867	0.02470	$6.72\ 10^{-8}$	0.02470
Ra-226	2.77×10^{-4}	8.58×10^{11}	1.167	354.8	1.46×10^{-6}	354.8
Sr-90	8.94×10^{-5}	2.39×10^{8}	0.02789	0.03192	1.13×10^{-8}	0.03192
Am-241	8.94×10^{-6}	4.44×10^{11}	3.030	5.921	1.22×10^{-7}	5.921
Total	--	--	--	360.8	1.66×10^{-6}	360.8

As shown in Table 2-8, the calculated TEDE is 361 mrem, to which the external dose is a negligible contributor. The TEDE of 361 mrem, derived using conservative conditions, is less than the U.S. EPA's lower PAG of 1 rem (1000 mrem) (USEPA 1992).

2.4.3.3 Evaluation of Worker Exposure from Accident Scenarios

The accident scenario that would result in the greatest potential release during decommissioning is Scenario 2 (Section 2.4.3.2.4, Scenario 2: Release from Fire During Remediation Activities.) This accident could result in 6.89 x 10^{-4} curies of radium-226 becoming airborne. If a worker inhales a fraction (i.e., 1×10^{-6} [one millionth]), of the radioactive material released during the fire, the worker would inhale 6.89×10^{-10} curies. Using the dose conversion factor, D_i, for radium-226, 8.58×10^{11} mrem/curie, the dose to the worker would be (6.89×10^{-10} curies) (8.58×10^{11} mrem/curie) = 591 mrem. This dose is well below the allowable annual exposure for workers, 5 rem (5000 mrem) (NRC 2000a).

The accidents discussed in Sections 2.4.3.2.3 and 2.4.3.2.5 would result in even less severe consequences than the 591 mrem calculated above. Therefore, it is unlikely that an accident could occur where a worker would accumulate a significant fraction of the 5-rem annual exposure limit. As described in Section 3.3, the radiation protection program will include worker protection and approved work control permits and procedures.

2.4.3.4 Conclusions

The accident analysis shows that the postulated accident scenarios would result in TEDEs to a member of the public at the site boundary that are less than the U.S. EPA's lower PAG of 1 rem (1000 mrem) (USEPA 1992). In addition, doses that workers could

receive from an accident are less than the allowable annual exposure for workers, 5 rem (5000 mrem) (NRC 2000a).

Since the results of the accident analysis show that offsite consequences from accidents are below the U.S. EPA's PAGs, offsite emergency plans are not needed.

2.4.4 Decommissioning Schedule

Section 2.4.1 presents the discrete decommissioning tasks in their recommended sequence for completion. However, the preferred order for completion may be revised based upon the costs of each task and the monies available at any given time as agreed upon by the NRC and Safety Light management.

2.5 References for Section 2

Beyeler, W. E., et al. 1998. *Review of Parameter Data for the NUREG/CR-5512 Building Occupancy Scenario and Probability Distributions for the DandD parameter Analysis, Letter Report for NRC Project JCN W6227.*

ICRP (International Commission on Radiological Protection), 1979. *Limits for Intake of Radionuclides by Workers.* ICRP Publication No. 30, Pergamon Press, Oxford, England.

Monserco Limited, 1996. *Characterization Survey of Safety Light Corporation Site at Bloomsburg, Pennsylvania U.S.A.*

NRC, 1976a. *Manual of Respiratory Protection Against Airborne Radioactive Materials.* NUREG-0041.

NRC, 1976b. *Acceptable Programs for Respiratory Protection.* Regulatory Guide 8.15.

NRC, 1998. *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination.* Federal Register: November 18, 1998 (Volume 63, Number 222).

NRC, 1999. *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination.* Federal Register: December 7, 1999 (Volume 64, Number 234).

NRC, 2000a. 10 CFR 20. *Standards for Protection Against Radiation.*

NRC, 2000b. *Use of Screening Values to Demonstrate Compliance with the Final Rule on Radiological Criteria for License Termination.* Federal Register: June 13, 2000 (Volume 65, Number 114).

29 CFR 1910. *Occupational Safety and Health Standards.*

Safety Light Corporation, 1998. *Health and Safety Program.* Revision 11.

Safety Light Corporation, 1999. Radiological Contingency Plan.

USEPA (U.S. Environmental Protection Agency), 1988. *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion,* Federal Guidance Report No. 11, EPA-520/1-88-020, Office of Radiation Programs, Washington, DC.

USEPA, 1992. *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, EPA-400-R-92-01, Office of Radiation Programs, Washington, DC.

USEPA, 1993. *External Exposure to Radionuclides in Air, Water and Soil*, Federal Guidance Report No. 12, EPA-402-R-93-081, Office of Radiation and Indoor Air, Washington, DC.

USEPA, U.S. NRC, U.S. DOE, and U.S. DOD, 1997. *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, Final, NUREG-1575, EPA 402-R-97-016.

3 DESCRIPTION OF METHODS USED FOR PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

3.1 Facility Radiological History Information

The *Characterization Survey of Safety Light Corporation Site at Bloomsburg, Pennsylvania U.S.A* (Monserco 1996) documents available information on site history from the 1940's through 1995. This section summarizes the most relevant historical information for purposes of decommissioning planning.

In the 1940s, United States Radium Corporation manufactured self-illuminating watch and instrument dials, and other articles containing radioactive materials. These activities mainly involved radium-226 and minor amounts of polonium-210. Waste disposal areas included the west dump (solid waste), canal and pipe shop (contaminated ductwork), and east lagoon (sewage and process waste water). Radioactive materials were stored in the garage.

In the 1950s, United States Radium Corporation primary production activities used cesium-137 and strontium-90. United States Radium Corporation also manufactured radium products and neutron sources. Other radionuclides used include tritium, carbon-14, thallium-204, krypton-85, and nickel-63. During the 1950s, waste was placed in two buried underground silos, which were remediated in 2000. Strontium-90 was also discharged directly into a dry well onsite. This well may have been located in the basement of the former personnel building.

The AEC issued license No. 37-00030-02 to United States Radium Corporation in 1956. This license authorized use and distribution of products containing a variety of radionuclides including carbon-14, iron-55, cobalt-60, nickel-63, zinc-65, strontium-90, cesium-137, pollonium-210, neptunium-237, uranium-238, and americium-241.

In the 1960s, United States Radium Corporation discontinued the practice of disposing of liquid waste in the lagoon, and constructed a holding tank and evaporator in the location of the current liquid waste building. Contaminated lagoon sediments were excavated and disposed of, and the two easternmost lagoons were backfilled.

In 1969, United States Radium Corporation sold all their radioisotope business except for tritium production. The AEC amended License No. 37-00030-02 to authorize possession and use of byproduct material for site characterization and decommissioning of facilities, equipment, and land from past operations. License No. 37-00030-08 was issued for manufacture of certain devices containing tritium as well as research and development activities. United States Radium Corporation erected the Nuclear Building to house the tritium operations.

The Susquehanna River flooded the site in 1972, destroying the holding tank and evaporator and possibly causing soil contamination. United States Radium Corporation backfilled these structures and constructed the current liquid waste building at that location. Also during the 1970s, radium-contaminated soil from the west dump area was shipped offsite for disposal, the third easternmost lagoon was backfilled, and diesel fuel spilled near Well #11. In 1974, United States Radium Corporation expanded the Etching Building.

In 1981, Oak Ridge Associated Universities (ORAU) surveyed the site and found elevated levels of radium-226, cesium-137, and strontium-90 in the soil and elevated levels of tritium and strontium-90 in the groundwater. United States Radium Corporation changed its name to Safety Light in 1982. During the late 1980s, Safety Light purchased the neighboring Vance-Walton property. Contaminated soil subsequently found on this property was transferred to the original Safety Light property.

The NRC issued an Order Modifying Licenses and Demand for information in 1989. Safety Light responded by submitting a Site Characterization Plan.

In 1990, Chem Nuclear Systems Inc. performed a hydrogeologic and radiological evaluation of the site. The groundwater flow appeared to be from the northern part of the site south toward the Susquehanna River. This study detected elevated tritium in soil and groundwater, and identified the underground silos as the source of strontium-90 contamination.

Monserco Limited of Ontario, Canada performed a site characterization in 1995 (Monserco 1996) using a Characterization Working Document (Monserco 1994) approved by the NRC.

Safety Light recently completed remediation of the east and west underground silos. The waste recovered from these silos is packaged for shipment to a waste disposal site. A concrete pad and bottom tank ring remain in place at the former location of each silo.

3.2 Radiological Concerns During Decommissioning Activities

The primary radiological hazard during decommissioning activities will be inhalation of airborne radioactive material generated by operations such as scabbling, demolition, and excavation. Section 3.3 describes the radiation safety program, including measures that will be taken to maintain internal exposure to radioactive material ALARA.

3.3 Radiation Safety Program

The existing Safety Light Health and Safety Program (Safety Light 1998) will form the basis for the radiation safety program during decommissioning. The information

provided in the following sections will supplement this program as necessary. The radiation safety program will ensure that all radiological activities conducted during the decommissioning of the Safety Light site comply with regulatory requirements. Radiological hazards will be monitored and evaluated routinely to maintain radiation exposures and the release of radioactive materials to unrestricted areas as far below specified limits as reasonably achievable. The radiation safety program will be integrated into all decommissioning project work activities, and each element of the program will be defined and implemented by approved policies, procedures, and guidelines.

Implementing procedures will provide guidance for performing specific tasks and methods used to maintain a radiologically safe working environment. Examples of radiation safety program implementing procedures include

- radiation, surface, and airborne radioactive material surveys;
- identification and posting of radiation, high radiation, surface and airborne radioactivity areas;
- access controls for radiation, high radiation, surface, and airborne radioactivity areas;
- protective clothing selection, issue, donning and removal;
- personnel radioactivity monitoring and decontamination; and
- radiological protection incidents and reports.

Responsibility for the decommissioning project radiation safety program rests with the Safety Light Radiation Safety Officer and Radiation Safety Committee as currently established in the Safety Light Health and Safety Program.

3.3.1 As Low As Reasonably Achievable (ALARA) Policy

Safety Light management is committed to incorporating ALARA principles into all radiological work activities. This commitment will ensure individual and collective occupational radiation doses, the release of radioactive effluents, and doses to the general public from decommissioning operations are ALARA, taking social, technical and economic factors into consideration.

The Safety Light Plant Manager and Radiation Safety Officer are responsible for radiation protection and maintaining occupational exposure ALARA during decommissioning. Contractor management will be held responsible for strictly adhering to the ALARA policy. The Safety Light Radiation Safety Committee will review and participate in implementation of the ALARA program by reviewing ALARA job reviews when prescribed by procedure, and ensuring the program complies with conditions of the license. All decommissioning project personnel will be made aware of management's commitment and instructed on their responsibility to execute project activities in accordance with the ALARA policy.

3.3.1.1 Administrative Limits

Administrative exposure limits will be used to ensure that personnel do not exceed the exposure limits specified in 10 CFR 20. The administrative limits will also serve as a management tool to ensure that individual and collective doses are maintained ALARA. Administrative exposure limits will be established such that increasing exposure levels require increasing levels of management approval.

Internal radiation exposure will be minimized by establishing exposure limits and administrative exposure controls, identifying and controlling sources or potential sources of airborne radioactivity, and using engineering controls. Respiratory protection may be used when engineering controls may not adequately protect the worker.

3.3.1.2 Radiological Performance Goals

Radiological performance goals will be established for the decommissioning project. The Radiation Safety Committee will review these goals to ensure they are challenging, yet achievable through proper implementation of ALARA measures. Decommissioning project radiological performance goals may include

- collective exposure for the entire decommissioning project;
- collective exposure for individual decommissioning tasks;
- collective exposure for decommissioning project organizations (e.g., contractor personnel, radiation safety personnel);
- maximum annual individual exposure; and
- number of radiological occurrences.

3.3.1.3 ALARA Plans and Procedures

Implementing procedures will address the organization, responsibilities and operation of the ALARA program. In addition to the ALARA policy statement, procedures and/or guidance documents will cover responsibility for and performance of ALARA job reviews and the Radiation Safety Committee charter.

3.3.1.4 Radiological Work Planning

Decommissioning activities inside the restricted area will be conducted according the radiation protection procedures to control workers' exposure and the spread of radioactive material. Work involving significant radiation exposure will be planned as far in advance as practical. During planning, unnecessary work steps will be deleted,

radiation and radioactivity levels in the work area will be determined, and collective exposure estimated. The individual planning the work will

- determine needed tools, parts, and equipment before work begins and stage them to minimize delays;
- coordinate efforts of different groups, such as decontamination, construction, and radiation safety, so work can proceed in a systematic and efficient manner;
- minimize the number of workers assigned to a particular job;
- coordinate work by area so that work is not duplicated for multiple tasks to be performed in the same area;
- perform as much work outside of radiation areas as possible;
- remove or shield sources of radiation to allow work to proceed in lower exposure rate areas;
- identify required procedures and facilitate review and concurrence before work may proceed; and
- identify special tools (remote handling equipment, video monitors, etc.) and temporary services (auxiliary lighting, power, communications), and ensure availability and operational status at the work location.

All work will be conducted according to the specific requirements of an RWP (refer to Section 3.3.2).

3.3.2 Radiation Work Permits

A Radiation Work Permit (RWP) system will be used for the administrative control of personnel entering or working in areas that have radiological hazards present. Work techniques will be specified in such a manner that personnel exposures, both individual and collective, are maintained ALARA. RWPs do not replace work procedures, but act as a supplement to procedures. Radiation work practices will be considered when procedures are developed for work that will take place in a radiologically controlled area.

Each RWP will describe the job, specify protective clothing and equipment, and provide personnel monitoring requirements. RWPs will also specify any special instructions or cautions pertinent to radiation hazards in the area, and list radiological hazards present; area dose rates; and the presence and intensity of hot spots, loose surface radioactivity, and other hazards, as appropriate. The radiation safety organization will perform radiation, surface radioactivity, and airborne surveys as required to define and document the radiological conditions for each job.

RWPs for jobs with low dose and minimal hazards will be approved by the radiation safety technician. The Radiation Safety Officer will approve RWPs for jobs with potentially high dose or significant radiological hazards. Collective RWPs for the completion of specific work evolutions with high estimated collective or individual

exposures or in high hazard work locations may also require review and approval by the Safety Light Radiation Safety Committee. The following topics will be covered by RWP implementing procedures:

- requirements, classifications and scope for RWPs;
- criteria for performing a formal ALARA job review;
- initiating, preparing and using RWPs;
- extending expiration dates of an RWP;
- terminating RWPs.

3.3.3 Contamination Control

The contamination control program is designed to reduce and minimize contaminated areas, tools, and equipment; prevent the spread of radioactivity, and maintain releases of radioactive materials ALARA. Major components of the program include the appropriate use of protective clothing; identification, posting, and control of contaminated areas; appropriate use of ventilation and other engineered controls; dust control measures; decontamination of tools, equipment and areas; and surveys and monitoring. Procedures will provide guidance for identifying contaminated areas, release criteria, and decontamination of components, tools, equipment, and materials. Procedures will specify the requirements for the overall control and accountability of byproduct materials used as sources including material receipt, leak testing, inventory, safe handling, and disposal.

3.3.3.1 Protective Clothing

Procedures and/or RWPs will provide guidance on use of protection clothing to reduce skin exposure and the potential for contamination.

3.3.3.2 Access Control

Areas at Safety Light that present a radiological hazard will be posted in such a manner that personnel are made aware of the presence and extent of the hazards in the area. Areas will be posted and access controlled per 10 CFR Part 20 requirements based on hazard evaluation.

3.3.3.3 Ventilation

Local HEPA filtration systems will be used during activities that could result in significant airborne radioactive particulate activity, such as cutting and grinding. If the filtered air exhausts directly to an unrestricted area, appropriate air sampling will be performed at the point of discharge.

3.3.3.4 Air Sampling Program

Area and breathing zone air sampling will be the principle means for determining intakes of radioactive material. Air samples will be taken

- during any work known or suspected to cause airborne radioactivity,
- during breach of a radioactive system,
- to demonstrate the adequacy of prescribed respiratory protection measures,
- to determine posting requirements,
- when DAC hour-tracking is used to control internal exposure,
- during or before initial entry into any area known or suspected to contain airborne radioactivity concentrations in excess of 0.25 DAC or loose surface contamination in excess of 100,000 dpm/100 cm^2, and
- at the exhaust of HEPA ventilation.

3.3.4 Handling and Storage of Radioactive Material

Procedures will provide requirements for radioactive material control, including the handling and storage of radioactive material. Radioactive material will be stored in specially designated restricted areas. These areas may contain reusable equipment and tools, waste staged for processing, waste or other materials prepared for shipment, or equipment and tools awaiting decontamination or reuse. Procedures will describe posting requirements, access controls, survey requirements, and controls placed on the movement of equipment/materials to and from the storage area.

3.3.5 Instrument Program

Sufficient facilities, equipment, and instruments will be available to permit the radiation

protection staff to function effectively. The facilities, and types and quantities of instruments will be adequate to meet project needs. Equipment will include sample counting instruments and portable survey instruments.

Areas will be provided for the storage and issue of project instruments. The operation, repair, calibration, and issue of instruments will be performed according to ANSI standards and manufacturer's recommendations as detailed in procedures or manufacturer's instructions. These procedures describe the proper techniques and the limitations for the specific piece of equipment. Calibrations will be appropriate for the anticipated radiation fields and will be traceable to the National Institute of Standards and Technology (NIST). Certificates of calibration will be maintained in project records. Field instruments will be response checked daily or prior to use.

3.3.6 Radiation Survey Program

Radiological area surveys will be performed to monitor and record the radiological conditions of the work area and adjacent unrestricted areas, and to meet the site administrative guidelines and the requirements of 10 CFR 20. These surveys will identify and measure direct radiation levels, airborne radioactivity, and surface radioactivity.

Results will be documented on survey forms or recorded in logs. This information will be available to personnel entering the radiological area. The information on the survey form may include a sketch or map of the area, contact and general area dose rates, radioactivity levels, identification of specific hazards such as hot spots, and the location of radiological boundaries.

A supervisor will review all evaluations to ensure that they are adequate to assess the radiological hazards in the area, and that all information is properly recorded. The supervisor reviewing the survey will confirm that the results are consistent with those anticipated and if not, will determine the reason for the variance.

Surveys will be posted to inform workers of the radiological hazards associated with the work area.

Survey frequencies will be established by the Radiation Safety Officer based on the hazard that may be encountered, the potential for changing radiological conditions, and the frequency of occupation. Evaluations will provide positive verification that radioactive materials are being adequately controlled and are not spreading to unrestricted areas.

3.3.7 Personnel Monitoring

3.3.7.1 External Exposure Monitoring

Personnel monitoring devices will be required for all personnel meeting the exposure conditions specified in 10 CFR 20 and in the administrative radiation safety procedures. Specifically, personnel will be required to wear external monitoring devices whenever work assignments require access to radiologically controlled areas. External radiation will be monitored using primary and secondary dosimeters, such as thermoluminescent dosimeters (TLDs), self reading dosimeters (SRDs), and electronic dosimeters. The official record of accumulated external exposure will normally be obtained from the primary TLD. Secondary dosimeters will serve as a backup to primary dosimeters and as a means for tracking exposure between processing periods. Primary and secondary dosimeters will be worn on the trunk of the body between the neck and waist in close proximity to each other unless the RWP specifies otherwise.

Multiple whole body monitoring may be required for work in a non-uniform radiation field, when the portion of the body receiving the highest exposure is not easily determined or is subject to change. RWPs will be used to specify multiple dosimeter requirements (location and position) for personnel working in these areas.

Extremity monitoring devices will be worn when exposure conditions justify their use. Specific criteria for determining the need for extremity monitoring and for determining the extremity dose will be identified in procedures or provided on the RWP.

Primary dosimeters will be processed by a facility that has current accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP). Secondary dosimeters will be calibrated semi-annually using NIST-traceable radiation fields.

Dose information from other sources may replace or supplement primary dosimeter results if the primary dosimeter results are unavailable due to loss or damage or if the results are suspect. In these cases, the actions taken and the justification for such actions will be documented and approved by the Radiation Safety Officer according to procedures for evaluating exposure when dosimeters are lost.

The skin of the whole body will normally be monitored using whole body dosimeters. Procedures and/or the RWP will provide instructions on the proper methods for wearing dosimeters in order to properly measure skin dose. If whole body dosimeters are not adequate to accurately measure skin dose, skin dose calculations may be performed using applicable procedures or accepted industry computational models.

3.3.7.2 Internal Exposure Monitoring

Breathing zone air samples and/or bioassay measurements will be used to determine intakes of radionuclides. Breathing zone air samples will be the principle means for determining the amount of intake of radioactive material.

The bioassay program will provide a quality control check for the internal exposure control program. Bioassay may include whole body counting (in-vivo) and/or analysis of excreta (in-vitro). Bioassay results will be used to evaluate potential intakes, estimate the magnitude of intakes, and provide data necessary to assess the committed effective dose equivalent.

Procedures will include criteria for performing bioassay, as well as methods for data analyses, interpretation, and dose assessment. The methods and techniques prescribed by these procedures will follow applicable guidance documents (NRC 1993 and NRC 1980).

3.3.8 Respiratory Protection

The respiratory protection program will be established in accordance with 10 CFR 20 Subpart H and 29 CFR 1910.134, and will be based on the guidance provided in ANSI Z88.2 (ANSI 1992), NRC Regulatory Guide 8.15 (NRC 1976b) and NUREG-0041 (NRC 1976a).

Elements of the respiratory protection program will include

- written standard operating procedures and policy statement;
- proper selection of equipment, based on the hazard;
- proper training and instruction of users;
- proper fitting, use, cleaning, storage, inspection, quality assurance, and maintenance of equipment;
- appropriate surveillance of work conditions, degree of employee exposure to stress;
- regular inspection and evaluation to determine the continued program effectiveness;
- program responsibility vested in one qualified individual;
- an adequate medical surveillance program for respirator users;
- use of only Bureau of Mines/National Institute of Occupational Safety and Health (NIOSH) certified equipment; and
- maintenance of a bioassay program.

3.3.8.1 Respirator User Qualification

Respirator users will be screened and/or examined to establish physical and psychological capabilities necessary to perform tasks using a respirator. Personnel will

be trained before using respiratory protection devices. Personnel will be fit tested before the first use of respirators requiring a face piece-to-face seal, and annually thereafter.

3.3.8.2 Respiratory Protection Equipment Description and Selection

The requirement for and selection of respiratory protection equipment for radiological purposes will normally be determined and approved as part of the RWP process. Project supervisors will be responsible for worker compliance with RWP requirements. Only National Institute of Safety and Occupational Health (NIOSH) certified respiratory protection equipment will be used. Routine and emergency use respirators will be issued per applicable procedures. Breathing air may be supplied to respirators from compressed breathing air cylinders or air compressors. All sources of supplied air will meet the requirements for Grade D or better breathing air.

3.3.8.3 Equipment Inspection and Maintenance

Inspection and maintenance of respiratory protection equipment will be performed according to manufacturers' and regulatory requirements. Respirators will be cleaned, sanitized, inspected, and maintained per approved procedures. Qualified personnel will repair respirators using parts designed for the respirator. Respirators ready for use will be stored to protect against dust, sunlight, heat, extreme cold, excessive moisture, and damaging chemicals.

3.3.9 Quality Assurance

The essential elements of the quality assurance and quality control (QA/QC) program for the Safety Light decommissioning will be

- establishment/implementation of plans, procedures, and protocols for the field operations,
- actions to assure that the procedures are understood and followed by the implementing staff, and
- documentation of the data collected.

The specific QA/QC program components are
- selection of personnel,
- establishment of project plans and procedures,
- documentation of survey data and sample analyses,
- personnel training,
- selection of appropriate instruments to perform the surveys,
- proper instrument calibration and daily functional checks, and

- management oversight of remedial action activities relative to the adherence to procedures, protocols, and documentation requirements.

Details of these elements will be set forth in the project plans and procedures. The remedial actions and the associated data acquisition and recording will be guided and conducted in compliance with these QA/QC requirements.

3.4 Contractor Personnel

Contractors and subcontractors performing work under this Decommissioning Plan shall comply with applicable Safety Light Corporation procedures. Contractor management will be held responsible for implementing the radiation protection program outlined in this plan, including the ALARA policy, during work activities. Contractor personnel who require access to a radiologically restricted area associated with the remediation effort shall receive appropriate training commensurate with the potential hazards to which they may be exposed.

3.5 Radioactive Waste Management

This section addresses the equipment to be used and technologies and procedures to be implemented for the management of radioactive waste during the Safety Light decommissioning project. The technical approaches described in this section are based on experience and address facets of planning, decontamination, packaging, storage, transportation, volume reduction or beneficial reuse, and final disposition of material, while minimizing secondary wastes and radiation exposure.

The radioactive waste management program will be based on the following elements:

- location and availability of disposal sites;
- potential for offsite release during decommissioning operations;
- preventing contamination of uncontaminated areas;
- use of existing facilities to support the waste packaging operations;
- methods of approach related to waste type, waste class, and impact on safety;
- cost effectiveness;
- logistical approach to decommissioning operations;
- ensuring that occupational exposures are maintained ALARA;
- minimizing the impact on the health and safety of the general public;
- maintaining flexibility for waste management to allow for unexpected wastes and changes in available technology;
- minimization of radioactive waste; and
- quality control.

3.5.1 Estimated Volumes of Radioactive Waste

An estimated 172,205 cubic feet of soil and radioactive waste from site buildings and their contents will be generated during decommissioning of the Safety Light site. The *Decommissioning Cost Estimate* prepared concurrently with this decommissioning plan provides greater detail on waste volumes. Minimization of the quantity of radioactive waste requiring disposal is a high priority for the Safety Light decommissioning project. Project management will incorporate the following radioactive waste volume minimization practices into work procedures.

- Radiation worker training will identify policies and practices to prevent unnecessary generation of mixed or radioactive wastes.
- Unnecessary generation of mixed or radioactive wastes will be minimized by controlling chemicals brought onsite, and preventing unnecessary packaging, tools, and equipment from entering radiologically controlled areas.
- Some materials may be reused during the decommissioning project, including contaminated tools, equipment, and clothing.
- The volume of radioactive waste will be minimized by decontaminating areas and equipment where practical, and by segregating waste as radioactive and non-radioactive, where practical.
- Decontamination activities will be planned to minimize the generation of secondary waste volumes during decontamination processes.
- Bulky material will be decontaminated or cut up to reduce volume. Some metal materials, incinerables, and compactible waste may be sent offsite for processing.
- Waste containers for direct burial will be packaged to minimize void space. Space around large bulky objects will be filled with small items and debris.

3.5.2 Radioactive Waste Handling and Storage

Waste will be packaged at the point of generation or at a designated location onsite. Packaging may include approved disposal containers. Other forms of packaging may include radioactivity control measures, such as bagging and/or wrapping. Heavily contaminated items may require further radioactivity control measures, such as the application of a fixative, in addition to bagging or wrapping.

Excavated piles of soil may temporarily be stored in berms and piles until sampling and analysis determines if the soil is a radioactive waste. Berms and soil piles will be protected from weather, and runoff will be controlled until soils are determined to be acceptable for use as backfill or packaged as waste.

After packaging, the waste will be transported to an onsite staging area to be secured and prepared for shipment. If necessary, the waste container will be placed in a storage area. Waste storage facilities that may be used during decommissioning activities include

- trailers and sea/land containers to house dry and solid waste,
- selected yard areas for short term storage of packaged waste staged for transport, and
- temporary storage areas for building rubble and soil.

After all waste is removed from storage, the storage areas will be surveyed and decontaminated, if necessary, and temporary structures will be removed.

3.5.3 Radioactive Waste Transportation

All radioactive material and radioactive waste shipments shall be performed according to DOT and other applicable federal regulations, as well as burial site requirements. The decommissioning contractor will have the appropriate DOT certification. Radioactive waste shall only be shipped to a licensed waste processing or disposal facility. Where applicable, a state user's permit may be required. All required documents shall be complete and legible and shall meet the requirements of 49 CFR, 10 CFR 20, and the receiving facility license requirements.

Before waste is shipped from Safety Light, each package will be inspected to ensure it meets all applicable design requirements and the container is not damaged or impaired. The quality control program for waste packaging and shipping will provide assurance and verification of compliance with radioactive waste shipping regulations.

Most shipments are expected to be low specific activity (LSA) and shipped in exclusive use vehicles. Radioactive material and waste will be transported by truck or rail, depending on the volume of material, vehicle availability and packaging requirements. The actual routing of shipments may vary with weather and highway conditions. Additionally, local and state restrictions pertaining to radioactive material transport may affect some route selections, particularly in congested metropolitan areas. The carrier is responsible for selecting the appropriate route, which must conform to applicable federal state, and local shipping regulations and requirements.

3.5.4 Radioactive Waste Processing

The project will result in the accumulation of contaminated material and debris. To minimize the volume of material for disposal, licensed waste processors, capable of reducing this volume to the maximum extent practical, will be used when cost effective. A cost benefit analysis will be performed for each waste stream and/or waste category.

Contaminated water may be generated as a result of draining, decontamination, and cutting processes. The contaminated liquids will be processed in a temporary treatment system (i.e., ion exchange and filtration system), solidified, or sent to a licensed waste processor. All liquid wastes will be processed according to approved procedures for

waste collection and discharge. Liquids released from the site will be monitored and controlled to ensure all releases of radioactivity to the environment are as low as reasonably achievable and meet established administrative controls and regulatory limits.

3.5.5 Waste Classification

Waste will be classified for disposal based on federal regulations and disposal site criteria. Waste classification procedures will ensure that a realistic representation of the distribution of radionuclides in waste is known and that waste classification is performed in a consistent manner. Any of the following basic methods, used individually or in combination, will be used to achieve this goal: materials accountability (process knowledge), classification by source, gross radioactivity measurements, and measurements of specific radionuclides.

Waste characterization will be performed on samples from each distinct area, building, or process, as applicable. Individual waste stream designations will be established for areas and processes that have similar radionuclide distributions and physical properties (dry active waste, liquids, etc.)

Appropriate instruments will be used to determine the type and quantity of radioactive material in each waste stream. Samples to determine radionuclide distribution will be obtained from wastes, whenever practicable. The curie content of each package can be calculated when radionuclide concentrations are determined, or by using a dose rate to curie conversion factor. Characterization will be performed by monitoring and/or sampling before packaging, and the activity of each radionuclide present in the mixture will then be used to estimate the activity in the final package. Radioactive waste will be classified according to 10 CFR 61.

3.5.6 Disposal of Radioactive Waste

Radioactive waste will be disposed of at licensed disposal facilities, as required. A facility or facilities for final disposition will be selected based upon waste characterization results and individual site criteria.

3.5.7 Disposal of Non-Radioactive Waste

No chemicals or other substances will be used during decommissioning operations that may become hazardous wastes or result in mixed waste. To reduce or avoid the generation of mixed wastes, project management will control the use of any chemical or other substance that could create a mixed waste concern.

A limited number of soil samples were analyzed for volatile organics, hydrocarbons, and heavy metals during the characterization. The only sample with positive results for volatile organics was from the loading dock area. One sample from behind the lacquer

storage building had a positive result for hydrocarbons. The highest concentrations of heavy metals were also in the loading dock sample.

Volatile organics were also detected by PID during drilling of wells M1 (north of the lacquer storage building) and M7 (southeast of the well house). Hydrocarbon odors were reported during the drilling of wells M1 (north of lacquer storage building), M8 (southwest corner of liquid waste building), M9 (south of west silo), and M13 (west dump area). Volatile organics were detected in well water from 11 wells, and heavy metals were detected in well water from all 13 wells.

If hazardous materials containing radioactive material are identified during decommissioning, they will be classified and stored onsite until declassified or approved for disposition. These materials will be managed according to Subtitle C of RCRA to the extent it is not inconsistent with NRC handling, storage and transportation regulations.

3.6 References for Section 3

ANSI Z88.2, 1992. *American National Standard for Respiratory Protection.*

Monserco Limited, 1994. *Characterization Plan for Safety Light Corporation Site, Bloomsburg, Pennsylvania U.S.A.*

Monserco Limited, 1996. *Characterization Survey of Safety Light Corporation Site at Bloomsburg, Pennsylvania U.S.A.*

NRC NUREG-0041, 1976a. *Manual of Respiratory Protection Against Airborne Radioactive Materials.*

NRC Regulatory Guide 8.15, 1976b. *Acceptable Programs for Respiratory Protection.*

NRC Regulatory Guide 8.26, 1980. *Application of Bioassay for Fission and Activation Products.*

NRC Regulatory Guide 8.9, 1993. *Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program.*

Safety Light Corporation, 1998. *Health and Safety Program.* Revision 11.
10 CFR 20. *Standards for Protection Against Radiation.*

10 CFR 61. *Licensing Requirements for Land Disposal of Radioactive Waste.*

29 CFR 1910. *Occupational Safety and Health Standards.*

49 CFR. *Transportation.*

4 PLANNED FINAL RADIATION SURVEY

The final radiation survey plan for the Safety Light site will be developed using the guidance provided in Draft Regulatory Guide DG-4006, *Demonstrating Compliance with the Radiological Criteria for License Termination* (NRC 1998) and NUREG-1575, *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)* (USEPA et. al. 1997). The purpose of the final radiation survey is to demonstrate that radiological conditions in each survey unit at the Safety Light site are at or below the established release criteria (refer to Section 2.1), to support termination of the NRC license and release of the site for unrestricted use.

4.1 Data Quality Objectives Process

Consistent with the referenced guidance, the final radiation survey plan will apply the Data Quality Objectives (DQO) process. EPA developed this process as a series of planning steps to ensure that the type, quantity, and quality of environmental data used in decision-making are appropriate for the intended application (USEPA 1994). Figure 4-1, Data Quality Objectives Process Steps, depicts the seven steps in the DQO process.

The DQO process is iterative; additional information is applied as it is revised or collected. This process ensures that the type and quality of radiological data needed to support license termination are considered early in the decommissioning process. Figure 4-2, Inputs to Final Status Survey Plan, illustrates the type of information that will be available when the final radiation survey plan is prepared, and the additional information that should be available when the plan is implemented.

Figure 4-1, Data Quality Objectives Process Steps (From Figure D.1 of NUREG-1575 (USEPA 1997))

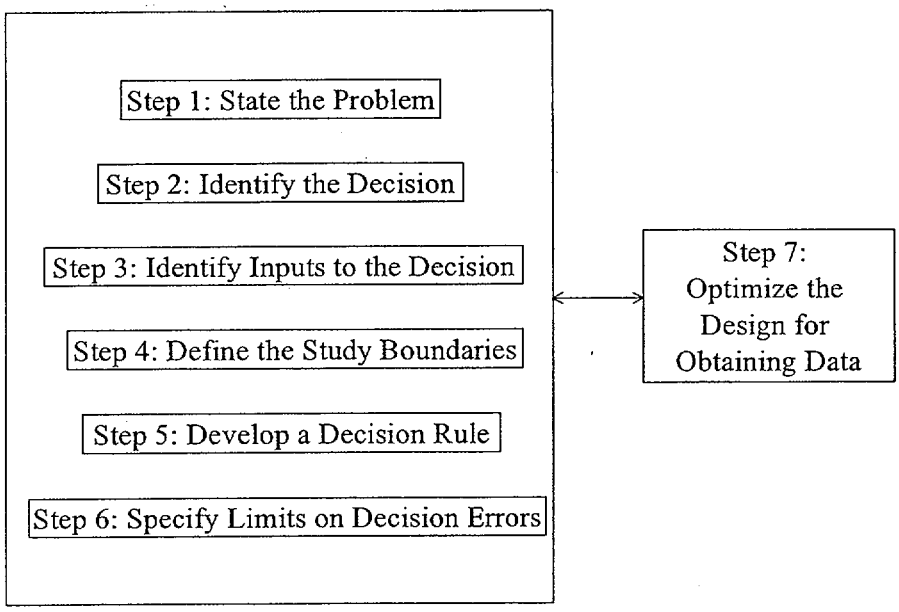
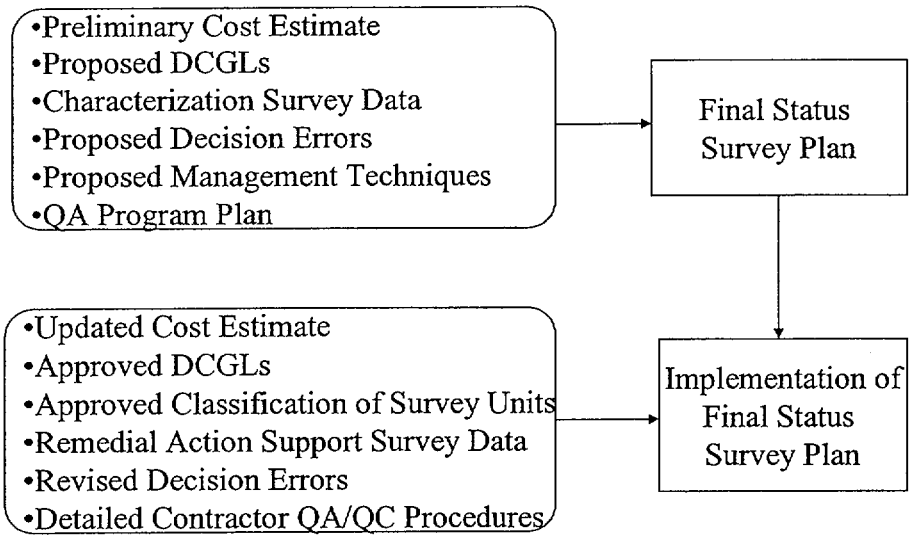


Figure 4-2, Inputs to Final Status Survey Plan



4.2 Survey Design

4.2.1 Calculations

The final radiation survey plan will include instructions for calculating the number of measurements and samples required per MARSSIM guidance. The steps in this process include:

- Classify the survey units
- Specify the decision error
- Calculate the relative shift
- Obtain the number of samples per survey unit
- Estimate the sample grid spacing
- Perform evaluation for small areas with elevated radioactivity

4.2.1.1 Classify Survey Units

The survey units identified for the Safety Light site have been categorized based on the following classifications:

- Class 1 – Impacted areas containing locations that may have exceeded $DCGL_W$ levels before remediation
- Class 2 – Impacted areas containing no locations that may have exceeded $DCGL_W$ levels before remediation
- Class 3 – Impacted areas with low probability of containing any locations with residual radioactivity
- Class 4 – Non-impacted areas where there is no potential for residual radioactivity

Additionally, impacted areas in Classes 1 and 2 have been classified by whether they have significant subsurface (below 15 cm) radioactivity, based on site historical information and available characterization data. Based on the lack of a detailed operational history at the site and the results of previous characterizations, the entire area within the fence will be considered impacted.

Table 4-1, Survey Unit Classification, lists the preliminary classification of survey units for the Safety Light site. Measurements and sampling results obtained during remediation activities will aid final survey design. The reference coordinate system established during the characterization survey will be used to select and relocate measurement and sampling locations, and a diagram of each survey unit will be prepared.

Table 4-1, Survey Unit Classification	
Survey Unit Description	**MARSSIM Classification**
Numbered Buildings	
Office Bldg., #3	Class 1
Lacquer Storage Bldg., #4	Class 1
USR Metals Liquid Waste Bldg., #5	Class 1
Well House, #6	Class 1
Pipe Shop, #7	Class 1
Old House, #9	Class 1
Miscellaneous Buildings	
Radium Vault	Class 1
Sr-90 Sources Vault	Class 1
Tritium Equipment 8x8 Vault	Class 1
Metal Silo	Class 1
Carpenter Shop	Class 1
Cesium IX Bldg.	Class 1
Emergency Water Tank	Class 4
Emergency Water Pump House	Class 4
Water Tower	Class 4
Main Building	
NW Basement Room #1	Class 1
NW Basement Room #2	Class 1
SE Basement, Phosphorous Storage	Class 1
East Loading Dock	Class 1
West Loading Dock	Class 1
Room 85	Class 1
Room 86	Class 1
Room 87	Class 1
Room 88	Class 1
Room 88A	Class 1
Room 88B	Class 1
Room 91	Class 1
Room 92	Class 1
Room 93	Class 1
Room 95	Class 1
Room 96	Class 1
Room 97	Class 1
Room 98	Class 1

Table 4-1, Survey Unit Classification	
Survey Unit Description	**MARSSIM Classification**
Room 99	Class 2
Room 100	Class 1
Room 101	Class 1
Room 102	Class 2
Room 103	Class 1
Room 104	Class 1
Room 105	Class 1
Room 106	Class 2
Room 107	Class 1
Room 108	Class 1
Room 110	Class 1
Room 111	Class 2
Room 112	Class 2
Room 113A	Class 1
Room 113B	Class 1
Room 113C	Class 1
Room 114	Class 1
Room 115	Class 1
Room 116	Class 2
Room 117	Class 2
Room 119	Class 2
Room 120	Class 1
Room 121	Class 1
Room 122	Class 2
Room 123	Class 2
Room 124	Class 2
Room 125	Class 1
Room 126	Class 2
Room 127	Class 1
Room 129	Class 2
Room 130	Class 2
Room 131	Class 2
Room 132	Class 2
Room 135	Class 1
Room 136	Class 2
Room 137	Class 2
Room 139	Class 1
Room 201	Class 1
Room 202	Class 1
Room 203	Class 2
Room 204	Class 1

Table 4-1, Survey Unit Classification	
Survey Unit Description	**MARSSIM Classification**
Room 205	Class 1
Room 206	Class 2
Room 207	Class 3
Room 208	Class 2
Room 209	Class 1
Room 210	Class 2
Room 211	Class 1
Room 212	Class 2
Room 213	Class 2
Room 214	Class 1
Room 215	Class 1
Room 216	Class 1
Room 217	Class 1
Room 218	Class 1
Room 301	Class 1
Etching Building	
Room 1	Class 2
Room 2	Class 1
Room 3	Class 1
Room 4	Class 1
Room 5	Class 1
Room 6	Class 1
Room 7	Class 1
Room 8	Class 1
Room 9	Class 1
Room 10	Class 1
Room 11	Class 1
Room 12	Class 2
Room 13	Class 1
Room 14	Class 2
Room 15	Class 2
Room 16 (17A)	Class 1
Room 17	Class 2
Room 18	Class 2
Room 19	Class 2
Room 20	Class 1
Room 21	Class 1
Room 22	Class 1
Room 23	Class 1
Room 24	Class 1

Table 4-1, Survey Unit Classification	
Survey Unit Description	**MARSSIM Classification**
Room 25	Class 1
Room 26	Class 1
Room 27	Class 1
Room 28	Class 1
Room 29	Class 1
Room 30	Class 1
Room 31	Class 1
Room 32	Class 1
Room 33	Class 1
Room 34	Class 2
Room 35	Class 2
Room 36	Class 1
Room 38	Class 2
Room 40	Class 1
Room 41	Class 1
Room 45	Class 1
Room 48	Class 1
Room 49	Class 1
Room 50	Class 2
Room 51	Class 2
Room 55	Class 1
Room 56	Class 1
Room 58	Class 2
Room 59	Class 1
Room 60	Class 2
Room 61	Class 1
Room 62	Class 1
Room 63	Class 2
Room 65	Class 1
Room 66	Class 2
Room 67	Class 1
Room 69	Class 1
Room 70	Class 1
Room 71	Class 1
Room 72	Class 1
Room 73	Class 2
Room 74	Class 1
Room 75	Class 1
Room 76A	Class 1
Room 76B	Class 1
Room 78	Class 2

Table 4-1, Survey Unit Classification	
Survey Unit Description	**MARSSIM Classification**
Room 79	Class 2
Room 80	Class 1
Room 81	Class 1
Room 82	Class 2
Room 83	Class 1
Room 84	Class 1
Room 201	Class 1
Room 202	Class 1
Room 203	Class 1
Room 204	Class 1
Outdoor Areas, Drains & Sewers	
Pad for Removed Garage	Class 1
Contaminated Soil By Silo	Class 1
Contaminated Soil North of Machine Shop	Class 1
Soil Area Between Canal & River	Class 1, subsurface
Underground Silo Area	Class 1, subsurface
East & West Lagoons (Canal)	Class 1 subsurface
Contaminated Soil Under Loading Dock	Class 1
Sewer Grate Behind Main Bldg.	Class 1
Sidewalk Areas	Class 2
East Plant Dump	Class 1, subsurface
West Plant Dump	Class 1, subsurface
Drain Lines	Class 1
Drywell Tanks	Class 1
Contaminated Soil by Old Berwick Rd.	Class 1
Contaminated Soil from Vance-Walton Property	Class 1
Contaminated Soil North of Lacquer Storage Bldg.	Class 1
Abandoned Canal	Class 1
Contaminated Soil South of Radium Vault	Class 1
Contaminated Soil Southeast of 8x8 Bldg.	Class 1
Employee Parking Area	Class 1
Site Paved Roads	Class 2
All Other Land Inside Fenced Area	Class 2

4.2.1.2 Specify the Acceptable Decision Error

The final radiation survey plan will define the acceptable relationship between the concentration of residual radioactivity in the survey unit and the DCGL so that decisions can be made based on survey results. The weighted DCGL, $DCGL_w$, is the concentration averaged over a survey unit that can be present while still satisfying the criterion. If all measurements in a survey unit are below the $DCGL_w$, the survey unit will meet the release criterion. If the average of the measurements in a survey unit is above the $DCGL_w$, the survey unit will not meet the release criterion. If the average measurement in a survey unit is below the $DCGL_w$, but some individual measurements are above the $DCGL_w$, the statistical tests described in MARSSIM (USEPA et. al. 1997) will be used to determine if the release criterion have been met. The Wilcoxon Rank Sum (WRS) test will be used for contaminants that are also present in background. The Sign test will be used for contaminants that are not present in background.

The following hypotheses are defined for the statistical tests:

- null hypothesis, H_0 – the survey unit does not meet the release criterion
- alternative hypothesis, H_a – the survey unit meets the release criterion

A Type I decision error (alpha) occurs when the null hypothesis is rejected when it is true. This decision error, also called α, represents the probability of determining that a survey unit is below a criterion when it actually above the criterion (false positive). A Type II (beta) error is made when the null hypothesis is accepted when it is false. This decision error, also called β, represents the probability of determining that a survey unit is above a release criterion when it is actually below the criterion (false negative). Typically, both types of error are initially set at 0.05 (5%). The final values may differ depending on data quality objectives.

4.2.1.3 Calculate the Relative Shift

The shift, Δ, is the width of the gray region, which is equal to DCGL - lower bound of the gray region (LBGR). The LBGR is the concentration level below which further remediation is not reasonably achievable. Within the gray region, the consequences of decision errors are relatively minor. The relative shift is denoted by Δ/σ, where σ is an estimate of the standard deviation of the measured values in the survey unit. For final radiation survey design, the measured values will include data from the characterization survey for unremediated survey units, and data from remedial action support surveys for remediated survey units. As recommended in MARSSIM (USEPA et. al. 1997), the LBGR will initially be set to 0.5 $DCGL_w$. The MARSSIM-recommended range for the relative shift is one to three.

4.2.1.4 Obtain the Number of Samples Per Survey Unit

Table 5.3 in MARSSIM gives the minimum number of samples necessary for a given Type 1 error (α), Type 2 error (β) and Δ/σ when the contaminant is present in background. Table 5.4 in MARSSIM gives similar information when the contaminant is not present in background. A similar number of background samples will also be taken in the reference area as applicable.

4.2.1.5 Estimate the Sample Grid Spacing

A triangular grid pattern will be used to establish sampling locations at the Safety Light site, since this method is more efficient for locating small areas of elevated activity. (EPA et. al. 1997) For a triangular grid pattern, the grid spacing, L, is calculated as:

$$L = \sqrt{\frac{A}{0.866\, n_{EA}}}$$

where A = area of the survey unit
n = number of calculated survey locations

4.2.1.6 Perform Evaluation for Small Areas with Elevated Radioactivity

A separate DCGL for elevated measurement comparison, $DCGL_{EMC}$ will be calculated if the residual radioactivity is concentrated in a small area of the survey unit. Results from scanning, direct field measurements, and laboratory analysis of samples may all be used for elevated measurement comparison. Any measurement above the $DCGL_{EMC}$ level will be considered elevated. Further investigation will then be necessary to determine the extent of the elevated area. Additional modeling will determine whether the survey unit meets the release criterion.

4.2.1.7 Optimize the Design

The individuals responsible for finalizing the final radiation survey plan will examine all of the factors that affect decision errors and sample sizes and balance costs and potential risks. This may include further evaluating the $DCGL_w$, the $DCGL_{EMC}$, and the measurement standard deviation. The estimate of the measurement standard deviation will include both the uncertainty in the measurement process and any anticipated spatial and temporal concentration variations.

4.2.2 Measurements and Sampling

4.2.2.1 Scanning

Scanning is performed to locate potential elevated areas of residual activity that require further investigation before performing direct measurements and sampling. Investigation levels will be determined based on Section 6.8.2 of NUREG-1507 (Abelquist et. al. 1997). Building surfaces will receive beta/gamma scans and environs will receive gamma scans. If a scanning response exceeds the investigation level, the area will be investigated, sampled, reclassified, and remediated as needed. If additional remediation is performed, the scan survey will be repeated to demonstrate effectiveness of the remediation. One hundred percent of Class 1 areas will be scanned, and 10-100 percent of Class 2 areas will be scanned. Scanning will be performed as deemed necessary for Class 3 areas.

4.2.2.2 Direct Measurements

Direct measurements will be made at fixed locations to provide a quantitative measure of radioactivity present in surface soils and building surfaces. These measurements will be made using a grid system, with a random start for Class 1 and Class 2 areas and simple random sampling for Class 3 areas. The statistical tests will use the results of these measurements. If these analyses indicate residual activity exceeding guideline values, further remediation will be performed as required, the survey unit will be reclassified if necessary, and scans and direct measurements of the remediated area will be repeated.

4.2.2.3 Sampling and Laboratory Analysis

Sampling and laboratory analysis will be performed to further define the areal extent of residual contamination. For non-gamma-emitting radionuclides, probability sampling and analysis will be performed as described in Section 4.2.2.2 for direct measurements. Samples will be transferred to a radioanalytical laboratory for analyses per documented laboratory-specific standard methods. Per MARSSIM, analytical techniques will provide a minimum detection level of 50 percent of the individual radionuclide $DCGL_w$ (or $DCGL_{EMC}$) values for all primary radionuclides of interest. If these analyses indicate residual activity exceeding guideline values, the survey unit will be reclassified if necessary, further remediation will be performed, as required, and scans and sampling of the remediated area will be repeated.

In areas where the historical site assessment and characterization survey data suggest residual radioactivity may extend deeper than 15 centimeters, sufficient core samples will be taken to the measured depth to meet the applicable statistical test.

4.3 Background Radiation Data

The DCGLs for residual activity represent levels above naturally occurring background radiation. Background radiation includes contributions from natural sources such as

- naturally occurring radionuclides in soils and construction media,
- airborne radioactivity (principally radon and radon progeny) from the radioactive decay of certain naturally occurring radionuclides, and
- cosmic radiation from high-speed particle interactions in the earth's atmosphere.

Background radiation may also include man-made radiation such as atmospheric fission product fallout due to weapons testing and reactor accidents.

As part of the final radiation survey, background levels in non-impacted areas will be established using the guidance in MARSSIM (USEPA et. al.1997) and Regulatory Guide DG-4006 (NRC 1998).

4.4 Instruments

Final radiation survey instruments will be capable of measuring levels less than or equal to 50% of the DCGL for the radionuclides of interest. Instrument sensitivities will be determined following the guidance in NUREG 1507 (Abelquist et. al. 1997) based on nominal literature values, actual instrument response, and site-specific background data. Section 3.3.5 describes the instrument program.

4.5 Data Analysis

The data will initially be reviewed to confirm proper classification of survey units. The concentration of residual radioactivity in each survey unit will then be compared to the $DCGL_w$. If all measurements in a survey unit are below the $DCGL_w$, the survey unit meets the release criterion. If the average of the measurements in a survey unit is above the $DCGL_w$, the survey unit does not meet the release criterion, and additional remediation will be required. If the average measurement in a survey unit is below the $DCGL_w$, but some individual measurements are above the $DCGL_w$, the statistical tests described in MARSSIM (USEPA et. al.1997) will be used to determine if the release criterion have been met. The Wilcoxon Rank Sum (WRS) test will be used for contaminants that are also present in background. The Sign test will be used for contaminants that are not present in background. Sections 4.2.1.2, 4.2.1.3, and 4.2.1.4 define the parameters that will be used with the methods presented in NUREG-1575 (MARSSIM) (USEPA et. al. 1997) and NUREG-1505 (Gogolak et. al. 1998) for determining the number of samples (direct field measurements) that will be necessary for the statistical test to be valid. Class 1 survey units that pass the applicable statistical test but have small areas with concentrations above the $DCGL_W$, and all sample measurements will also be tested using the elevated measurement comparison described in MARSSIM (USEPA et. al. 1997).

4.6 Documentation

The results of all investigations will be documented in the Final Radiation Survey Report, including the results of scan surveys that may have potentially identified areas of elevated direct radiation. The basis for reclassifying any survey units will also be documented.

Records of survey results and calibration results will be maintained for at least 3 years as required by 10 CFR 20.2103.

4.7 References for Section 4

Abelquist, E.W., W. S. Brown, G.E. Powers and A.M. Huffert, 1997. *Minimum Detectable Concentrations with Typical Radiation Survey Instruments for Various Contaminants and Field Conditions*. NUREG-1507.

Gogolak, C.V., G. E. Powers, and A. M. Huffert, 1998. *A Nonparametric Statistical Methodology for the Design and Analysis of Final Status Decommissioning Surveys*. NUREG-1505.

NRC, 1998. *Demonstrating Compliance with the Radiological Criteria for License Termination*. Draft Regulatory Guide DG-4006.

USEPA, 1994. *Guidance for the Data Quality Objectives Process*. EPA/600/R-96/055, EPA QA/G-4.

USEPA, U.S. NRC, U.S. DOE, and U.S. DOD, 1997. *Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)*, Final, NUREG-1575, EPA 402-R-97-016.

5 FUNDING

5.1 Decommissioning Cost Estimate

The *Decommissioning Cost Estimate for Safety Light Corporation* was prepared concurrently with this plan. The total estimated cost for the decommissioning activities described in this plan is $23,508,275 in 2000 dollars. The *Decommissioning Cost Estimate* document provides a breakdown of costs for the individual decommissioning tasks outlined in Section 2.4.1; an explanation of estimation methods; and information on costs for waste disposal, labor, instrument and equipment rental, demolition, and decontamination.

5.2 Funding Plan

Decommissioning funding for the Safety Light site is currently addressed in NRC Radioactive Material License Nos. 37-00030-08 (Condition 20) and 37-00030-02 (Condition 16), as follows:

> Pursuant to 10 CFR 30.11, the licensee is exempted from the provisions of 10 CFR 30.32(h) and 30.35(a) through 30.35(f), provided that the licensee sets aside from operating funds or any other funds, except insurance litigation funds, the following amounts as described in the licensee's letter dated August 3, 1999:
>
> > January 1, 2000 and each month thereafter for 12 months: $7,000.00;
> >
> > January 1, 2001 and each month thereafter for 24 months: $8,000.00;
> >
> > January 1, 2003 and each month thereafter for 24 months: $9,000.00
>
> for a total of $492,000.00.

Any monies obtained from current insurance litigation will also be applied to the decommissioning project.

5.3 References for Section 5

USNRC Materials License 37-00030-02 Amendment 51 dated 12/28/99

USNRC Materials License 37-00030-08 Amendment 13 dated 12/28/99

6 PHYSICAL SECURITY PLAN AND MATERIAL CONTROL AND ACCOUNTING PLAN PROVISIONS IN PLACE DURING DECOMMISSIONING

Safety Light is not required to have an NRC-approved special nuclear material control and accounting plan, since it does not possess a special nuclear material license. The *Health and Safety Program* (Safety Light 1998) includes a "Procedure for Controlling Access to Restricted Industrial Area Owned by Safety Light Corporation". This procedure describes the physical and administrative security features currently in place. Physical security measures during decommissioning will maintain a level of security at least equal to that described in this procedure.

6.1 References for Section 6

Safety Light Corporation, 1998. *Health and Safety Program*. Revision 11.

ATTACHMENT 1
SAFETY LIGHT DOSE ASSESSMENT

SURFACE CONTAMINATION

Table 1 of the November 18, 1998 Federal Register Notice (NRC 1998a) does not include surface contamination limits for radium-226 or americium-241. RESRAD-BUILD Version 2.37 was used to generate screening values for these radionuclides. In application, the "sum of fractions" rule will be applied to both the RESRAD-BUILD limits and the limits from Table 1 of the November 18, 1998 notice.

Except for the following, the default RESRAD-BUILD parameters were used.

1. The fraction of time spent inside the modeled room was 0.2289, based on a worker scenario of working 50 weeks a year 5 days a week for 8 hours per day.

$$Fraction = \frac{50 weeks}{52 weeks} \times \frac{5 days}{7 days} \times \frac{8 hours}{24 hours} = 0.2289$$

2. The room dimensions were changed to match main building room 88: 10.6 meters by 16.7 meters by 3.6 meters tall.

3. Six surface sources were used that represent the floor, ceiling and four walls. The activity used for each radionuclide of interest was 1 pCi per square meter.

4. The radionuclide lifetime days were increased from 365 days to 4,015 days (11 years). This lifetime was used based on Argonne National Laboratory recommended parameters for *Derivation of Generic Surface Contamination Guidelines for Ra-226, Th-230, Th-232 and Natural Uranium for Building Occupancy and Building Renovation Scenarios* (information provided in an Argonne National Laboratory RESRAD-BUILD workshop June 21, 2000). In addition, the Safety Light facility history demonstrates that the radium material surface contamination has already had a life in excess of 11 years.

5. The inhalation rate was increased from 18 to 28.8 cubic meters per day based on Argonne National Laboratory recommendations (same as item 4 above).

6. The air release fraction was increased from the default value of 0.1 to 0.5 to be more conservative.

RESRAD-BUILD was used to calculate dose rates from 1 pCi/m^2 of radium-226 and americium-241. These values were then used to calculate the activity (in pCi/m^2) required to generate a dose rate of 25 mrem per year. This activity was then converted to dpm/100cm^2 to arrive at the building surface activity required to generate a dose rate of 25 mrem per year. These calculations and results are presented below in Tables 1, 2 and 3.

Table 1, Building Surface Calculated Dose from 1pCi/m^2

Ra-226 Mrem/yr.	Am-241 mrem/yr.
1.56E-04	3.07E-03

Dose Limit: 25 mrem/yr.

Table 2, Building Surface Activity (pCi/m^2) equivalent to 25 mrem/year

Ra-226 pCi/m^2	Am-241 pCi/m^2
1.60E+05	8.14E+03

1 picocurie = 2.22 dpm
1 m^2 = 10,000 cm^2

Table 3, Building Surface Activity (dpm/100 cm^2) equivalent to 25 mrem/yr.

Ra-226 dpm/100cm^2	Am-241 dpm/100cm^2
3,558	181

ALARA CALCULATIONS FOR BUILDING SURFACE DCGL'S

NRC regulations (10 CFR 20.1402) contain two separate and independent requirements for license termination:

- a dose limit must be met, and
- levels of residual radioactivity must be ALARA.

This Safety Light Corporation Decommissioning Plan meets both requirements.

Two different ALARA requirements apply to decommissioning:

- 10 CFR 20.1101(b) requires the licensee to use procedures and engineering controls to maintain doses ALARA during operations (including decommissioning activities), and
- 10 CFR 20.1402 requires that levels of residual radioactivity be ALARA in order to terminate a license.

The latter ALARA requirement is the topic of these calculations.

The method used to demonstrate compliance with the license termination ALARA requirement is the quantitative method in NRC Draft Regulatory Guide DG-4006. (NRC 1998b) The DG-4006 method evaluates the practicality of a cleanup action by quantitatively comparing its costs with the monetary worth of its benefits. A simple equation is used to evaluate the costs versus the benefits of each potential cleanup action.

The DG-4006 method includes the following steps.

- Identify potential cleanup actions that will not be necessary to meet the dose limit.
- For each potential cleanup action, compare the costs vs. the benefits using the equation in DG-4006.
- If a potential cleanup action is cost beneficial, the action must be performed to meet the ALARA requirement.

This method is performed during the remediation planning process before remediation starts. The method uses characterization data to determine if potential cleanup actions not necessary to meet the dose limit are necessary to meet the ALARA requirement.

The Safety Light Corporation site has tritium, strontium-90, cesium-137, radium-226, and americium-241 on building surfaces from licensed operations. SLC proposes to release the site for unrestricted use. Surfaces with activity levels in excess of the DCGLs would be decontaminated to meet these limits.

Table 4: Proposed Safety Light Surface Activity DCGLs

	H-3	Sr-90	Cs-137	Ra-226	Am-241
Surface Activity (dpm/100cm^2)	1.2E+08	8,700	28,000	3,500	180

The ALARA goals proposed by the Safety Light Corporation are the same as the dose-based guideline values in the table above. The DCGLs for the unrestricted area (25 mrem/yr) were obtained from the November 18, 1998 Federal Register notice (NRC 1998a) (for strontium-90 and cesium-137), and calculated using RESRAD-BUILD for an industrial occupancy scenario (for radium-226 and americium-241).

For the simplest case, areas that contain only radium for unrestricted release, Equation 18 from DG-4006 was used to calculate the ALARA concentration.

$$\frac{C_{ALARA}}{DCGL_W} = \frac{Cost_T}{\$2000 \times P_D \times 0.025 \times F \times A} \times \frac{r+\lambda}{1-e^{-(r+\lambda)N}}$$

where:

C_{ALARA} = concentration at which the benefit from removal equals the cost of removal

$Cost_T$ = total cost of remediation

P_D = population density based on occupancy scenario

F = fraction of residual radioactivity removed by remediation action

A = area being evaluated in m^2

R = monetary discount rate in units of yr^{-1}

λ = radiological decay constant for the radionuclide in units of yr^{-1}

N = number of years over which the collective dose will be calculated

The first step was to identify the potential cleanup actions to be evaluated. The actions proposed include:

- decontaminate the surfaces exceeding the DCGL values, and dispose of the waste at Envirocare of Utah; and
- leave the activity in place.

The second step was to estimate the cost of each alternative. The "leave in place" option has zero cost. The cost to decontaminate wall surfaces was evaluated in these calculations because walls are the primary building surface of concern at the Safety Light site. The direct costs of surface decontamination using a scabbler with offsite disposal of all waste over a 100 square meter area with more than 3,500 dpm/100cm^2 was estimated to be $10,396.

Other costs can include the costs of worker accidents and dose during remediation, traffic fatalities, etc. For Safety Light there was no need to estimate other costs because they would not affect the outcome of the analysis.

The third step was to evaluate the other parameters in Equation 18.

Table 5, Equation 18 Parameters

Parameter	Value	Basis
DCGLw	3,500 dpm/100cm^2	RESRAD_BUILD dose modeling
P_D	0.09 person/m^2	NUREG-1496, Vol 2, App. B, Table A.1
F	1	Total Removal

Parameter	Value	Basis
$/person-rem	$2,000	NUREG/BR-0058
A	100 m^2	Site Information
r	0.07/yr	NUREG/BR-0058
N	70 yr	NUREG-1496, Vol 2, App. B, Table A.1
λ	0.000433	radioactive decay data

The fourth step was to calculate the ALARA concentrations using the above values in equation 18. This calculation indicates the $C_{ALARA}/DCGL_W$ ratio is 1.64 or the C_{ALARA} value is larger than the $DCGL_W$ value. For a building surface radium-226 $DCGL_W$ value of 3,500 dpm/100cm^2 the C_{ALARA} value is 5,736 dpm/100cm^2 . Therefore, no building surface with concentrations in excess of 3,500 dpm/100cm^2 radium-226 need be decontaminated to meet the ALARA requirement because the dose limit is more restrictive.

Similar calculations were performed for the other radionuclides of interest at the Safety Light Corporation site. The only Equation 18 values that change for each radionuclide are the $DCGL_W$ value and the λ value. Results from these calculations are presented in the table below.

Table 6, Building Surface ALARA Calculation Results

Radionuclide	λ	$C_{ALARA}/DCGL_W$	Surface $DCGL_W$	Surface C_{ALARA}
H-3	0.0565	2.92	120,000,000	350,000,000
Sr-90	0.025	2.20	8,700	19,122
Cs-137	0.023	2.15	28,000	60,310
Ra-226	0.00043	1.64	3,500	5,736
Am-241	0.0015	1.66	180	299

SURFACE AND SUBSURFACE SOILS

The RESRAD 5.95 computer code was used in assigning surface (soil depth to 15 centimeters) and subsurface (soil depth greater than 15 centimeters) soil DCGLs for the Safety Light site. The "sum of fractions" rule will apply. The surface soil activity will be compared to the RESRAD-generated surface soil guideline values and the subsurface soil activity will be compared to the RESRAD-generated subsurface soil guideline values. The sum of fractions for both surface and subsurface soil must together not exceed 1.0 in order for the survey unit to be released.

Except for the following, the default RESRAD parameters were used.

1. In the surface soil RESRAD calculation, the surface cover layer was set to zero.

2. In the subsurface soil RESRAD calculation, a surface cover of 15 cm was used, which provides for the surface soil (where separate screening levels apply).

3. An average annual precipitation of 1.18 meters was used (46.49 inches). This value was obtained from climatic data for Freeland, Pennsylvania, which is 25 miles east of the Safety Light site.

4. The groundwater at the site averages 3 meters below the surface (see Table 7). In the surface soil RESRAD calculation, it was assumed that the surface contaminated soil was 15 centimeters thick (0.15 meters of surface soil). This leaves a 2.85 meter thick unsaturated layer.

5. In the subsurface soil RESRAD calculation, it was assumed that the surface and subsurface contaminated soil is 2 meters thick: 0.15 meters of surface soil and 1.85 meters of subsurface soil. Some radioactive materials were buried on site at unknown depths and 2 meters deep is appropriate for general disposal. This leaves a 1.0 meter thick unsaturated layer.

6. The soil at the site is primarily a gravelly silt loam (Monserco 1996). Values for soil parameters for this site were obtained based upon this soil description from the RESRAD *Data Collection Handbook (*Yu 1993).

7. The soil density, $\rho = 1.28$, was obtained from Table 2.1 of the *Data Collection Handbook* (Yu 1993).

8. The hydraulic conductivity, K, was obtained from Table 5.2 of the *Data Collection Handbook* (Yu 1993). The saturated hydraulic conductivity was 227 meters/year for *silty loam*. However, the *Data Collection Handbook* indicates that if values in the literature are used in place of actual site data, no more than one significant digit is appropriate. Therefore, a value of 200 meters/year was used for the saturated zone.

9. The *Data Collection Handbook* (Yu 1993) also indicates that in RESRAD, the saturated hydraulic conductivity values related to the contaminated and unsaturated zones of the soil should represent the vertical component of K. Soil with a spatial variability is typical during the geological formation of soils. These soils are called anisotropic and the vertical component of K is typically one or two orders magnitude lower than the horizontal. Therefore, a value of 20 meters/year was used for the contaminated and unsaturated zones.

10. The Soil-Specific Exponential b Parameter obtained from Table 13.1 of the *Data Collection Handbook* (Yu 1993) was 5.3 (same as RESRAD default value) for *silty loam.*

Table 7, Safety Light Corporation Ground Water

Well	Location	Ground Water Depth (m)
M1	North of Lacquer Storage Bldg.	4.27
M2	SE corner of SLC property, southeast of excavation site #2	2.26
M3	Southwest corner of Safety Light property, beside creek	1.04
M4	East of East Lagoon	2.29
M5	Southeast of East Lagoon	1.52
M6	Southwest of excavation site #1	1.66
M7	Southeast of Well House, Northwest of Buried Silos	4.11
M8	South of Liquid Waste Building	3.35
M9	South of West Silo	3.96
M10	South of East Silo	3.96
M11	South of Lacquer Storage Building	3.35
M12	By Cs-137 log hot spot	1.83
M13	West Dump	4.88
	Average	**2.96**

The RESRAD results were:

Table 8, Soil Activity equivalent to 25 mrem/year

Radionuclide	Surface Soil Concentration 15cm deep DCGL (pCi/g)	Subsurface Soil Concentration 15cm to 2 m deep DCGL (pCi/g)
H-3	N/A	696
Sr-90	30.1	6.1
Cs-137	14.1	43
Ra-226 + C	4.0	1.8
Am-241	10.3	1.1

ALARA CALCULATIONS FOR SOIL DCGL'S

NRC regulations (10 CFR 20.1402) contain two separate and independent requirements for license termination:

- a dose limit must be met, and
- levels of residual radioactivity must be ALARA.

This Safety Light Corporation Decommissioning Plan meets both requirements.

Two different ALARA requirements apply to decommissioning:

- 10 CFR 20.1101(b) requires the licensee to use procedures and engineering controls to maintain doses ALARA during operations (including decommissioning activities), and

- 10 CFR 20.1402 requires that levels of residual radioactivity be ALARA in order to terminate a license.

The latter ALARA requirement is the topic of these calculations.

The method used to demonstrate compliance with the license termination ALARA requirement is the quantitative method in NRC Draft Regulatory Guide DG-4006. (NRC 1998b) The DG-4006 method evaluates the practicality of a cleanup action by quantitatively comparing its costs with the monetary worth of its benefits. A simple equation is used to evaluate the costs versus the benefits of each potential cleanup action.

The DG-4006 method includes the following steps.

- Identify potential cleanup actions that will not be necessary to meet the dose limit.

- For each potential cleanup action, compare the costs vs. the benefits using the equation in DG-4006.

- If a potential cleanup action is cost beneficial, the action must be performed to meet the ALARA requirement.

This method is performed during the remediation planning process before remediation starts. The method uses characterization data to determine if potential cleanup actions not necessary to meet the dose limit are necessary to meet the ALARA requirement.

The Safety Light Corporation site has tritium, strontium-90, cesium-137, radium-226, and americium-241 in the soil from licensed operations. The soil with residual radioactivity includes several different areas and varies from 15 centimeters to about 2 meters thick.

SLC proposes to release the site for unrestricted use. Soil with high concentrations would be removed from the site to a licensed off-site disposal facility.

Table 9, Proposed Safety Light Soil DCGLs (pCi/g)

	H-3	Sr-90	Cs-137	Ra-226	Am-241
Surface Soils (15 cm depth)	110	30.1	14.1	4.0	10.3
Subsurface Soil (15-cm to 2-m depth)	696	6.1	43	1.8	1.1

The ALARA goals proposed by the Safety Light Corporation are the same as the dose based guideline values in the table above. The DCGLs for the unrestricted area (25 mrem/yr) were calculated using RESRAD for the resident farmer scenario using site-specific parameters.

For the simplest case, areas that contain only radium for unrestricted release, Equation 18 from DG-4006 was used to calculate the ALARA concentration.

$$\frac{C_{ALARA}}{DCGL_W} = \frac{Cost_T}{\$2000 \times P_D \times 0.025 \times F \times A} \times \frac{r+\lambda}{1-e^{-(r+\lambda)N}}$$

The first step was to identify the potential cleanup actions to be evaluated. The actions proposed include

- dispose of soil at Envirocare of Utah, and
- leave the activity in place.

The second step was to estimate the cost of each alternative. The "leave in place" option has zero cost. The direct cost of offsite disposal of surface soils (15 cm depth) over a 1,000 square meter area with more than 4.0 pCi Ra-226 per gram of soil was estimated to be $424,000 per 5,300 cubic feet (302 tons) of soil.

Other costs can include the costs of worker accidents and dose during remediation, traffic fatalities, etc. For Safety Light there was no need to estimate other costs because they would not affect the outcome of the analysis.

The third step was to evaluate the other parameters in Equation 18.

Table 10, Equation 18 Parameters

Parameter	Value	Basis
DCGLw	4.0 pCi/g	RESRAD Dose modeling
P_D	0.0004 person/m^2	NUREG-1496, Vol 2, App. B, Table A.1
F	1	Total Removal
$/person-rem	$2,000	NUREG/BR-0058
A	1,000 m^2	Site Information
r	0.03/yr	NUREG/BR-0058
N	1,000 yr	NUREG-1496, Vol 2, App. B, Table A.1
λ	0.000433	radioactive decay data

The fourth step was to calculate the ALARA concentrations using the above values in equation 18. This calculation indicates the $C_{ALARA}/DCGL_W$ ratio is 645 or the C_{ALARA} value is over six hundred times larger than the $DCGL_W$ value. For a surface soil radium-226 $DCGL_W$ value of 4.0 pCi/g the C_{ALARA} value is 2,580 pCi/g. Therefore, no surface soil with concentrations in excess of 4.0 pCi/g radium-226 need be shipped for offsite disposal to meet the ALARA requirement because the dose limit is more restrictive. In addition, the removal of subsurface soils is slightly more expensive than for surface soils and the $C_{ALARA}/DCGL_W$ ratio will be even larger. Therefore, no subsurface soils with concentrations in excess of 1.8 pCi Ra-226 per gram need be shipped for offsite disposal to meet the ALARA requirement because the dose limit is more restrictive.

Similar calculations were performed for the other radionuclides of interest at the Safety Light Corporation site. The only Equation 18 values that change for each radionuclide are the $DCGL_W$ value and the λ value. Results from these calculations are presented in the table below.

Table 11, Soil ALARA Calculation Results

Radionuclide	λ	$C_{ALARA}/DCGL_W$	Surface $DCGL_W$	Surface C_{ALARA}	Subsurface $DCGL_W$	Subsurface C_{ALARA}
H-3	0.0565	1,833	110	201,718	696	1,275,768
Sr-90	0.025	1,166	30.1	35,108	6.1	7,115
Cs-137	0.023	1,126	14.1	15,873	43	48,406
Ra-226	0.00043	645	4.0	2,581	1.8	1,161
Am-241	0.0015	668	10.3	6,881	1.1	735

REFERENCES:

Monserco Limited, 1996. *Characterization Survey of Safety Light Corporation Site at Bloomsburg, Pennsylvania U.S.A.*

NRC, 1998a. *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination.* Federal Register: November 18, 1998 (Volume 63, Number 222).

NRC, 1998b. *Demonstrating Compliance with the Radiological Criteria for License Termination.* Draft Regulatory Guide DG-4006.

NRC, 1999. *Supplemental Information on the Implementation of the Final Rule on Radiological Criteria for License Termination.* Federal Register: December 7, 1999 (Volume 64, Number 234).

C. Yu, et. al. 1993. *Data Collection Handbook to Support Modeling the Impacts of Radioactive Material in Soil,* ANL/EAIS-8.

ATTACHMENT 2
DERIVATION OF χ/Q

In the accident analysis presented in Section 2.4.3.2, the quantity χ/Q is used to express the dilution of the released effluent as it travels 100 meters to the site boundary. χ/Q is calculated using the well-established formula for Gaussian dispersion, which is applicable when the effluent is released at such a rate that it does not perturb the existing pattern of turbulent eddies in the atmosphere. This is the expected case for the small releases evaluated in Section 2.4.3 of the decommissioning plan. χ/Q was calculated using the formula:

$$\frac{\chi}{Q} = \frac{1}{U_{10}\pi\Sigma_y\sigma_z}$$

where $\Sigma_y = M\sigma_y$, for distances of 800 meters or less. The value of M is determined from Figure 3 of NRC Regulatory Guide 1.145 (NRC 1983) (M=1 for all cases where the wind speed is 6 meters per second or more). For a wind speed of 2 meters/second and atmospheric stability class F, M=4.

Then for a wind speeds of 2 m/s: $$\frac{\chi}{Q} = \frac{1}{U_{10}\pi M\sigma_y\sigma_z}$$

where:

σ_y = the lateral plume spread in meters (m)
σ_z = the vertical plume spread in meters (m)
U_{10} = the wind speed (meters/second) measured at a height of 10 meters.

The value of σ_y is determined from Figure 1 and σ_z from Figure 2 of NRC Regulatory Guide 1.145 (NRC 1983). The values are a function of the distance from the source to the receptor and the assumed Pasquill's turbulence type. This analysis for the Safety Light facility assumes an average distance of 100 meters (328 feet) from point of release to receptor and category F (Moderately Stable) meteorological conditions. For these assumptions, $\sigma_y = 4.0$ meters and $\sigma_z = 2.2$ meters. Using these values of σ_z and σ_y and a wind speed, u, of 2 m/s, the equation yields:

$$\frac{\chi}{Q} = \frac{1}{(2\ \mathrm{m/sec})\pi(4)(4.0\mathrm{m})(2.2\mathrm{m})} = 4.52\times10^{-3}\ \mathrm{sec/m^3}$$

References:

NRC, 1983. *Regulatory Guide 1.145, Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants.*