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Technology, Safety and Costs of Decommissioning Reference Nuclear Fuel Cycle and Non-Fuel Cycle Facilities Following Postulated Accidents

Main Report

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Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

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Main Report

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FOREWORD
BY
NUCLEAR REGULATORY COMMISSION STAFF

The NRC staff is reappraising its regulatory position relative to the decommissioning of nuclear facilities.⁽¹⁾ As a part of this activity, the NRC has initiated two series of studies through technical assistance contracts. These contracts are being undertaken to develop information to support the preparation of new standards covering decommissioning.

The basic series of studies covers the technology, safety, and costs of decommissioning reference nuclear facilities. Light water reactors and fuel-cycle and non-fuel-cycle facilities are included. Facilities of current design on typical sites are selected for the studies. Separate reports are prepared as the studies of the various facilities are completed.

The first report in this series covers a fuel reprocessing plant;⁽²⁾ the second addresses a pressurized water reactor;⁽³⁾ and the third deals with a small mixed-oxide fuel fabrication plant.⁽⁴⁾ The fourth report, an addendum to the pressurized water reactor report,⁽⁵⁾ examines the relationship between reactor size and decommissioning cost, the cost of entombment, and the sensitivity of cost to radiation levels, contractual arrangements, and disposal site charges. The fifth report in this series deals with a low-level waste burial ground;⁽⁶⁾ the sixth covers a large boiling water reactor power station;⁽⁷⁾ and the seventh examines a uranium fuel-fabrication plant.⁽⁸⁾ The eighth report covers non-fuel-cycle nuclear facilities.⁽⁹⁾ The ninth report, an addendum to the low-level waste burial ground report,⁽¹⁰⁾ supplements the description of environmental radiological surveillance programs used in the parent document. The tenth report deals with a uranium hexafluoride conversion plant.⁽¹¹⁾ The eleventh report addresses the decommissioning of nuclear reactors at multiple-reactor power stations.⁽¹²⁾ The twelfth report covers nuclear research and test reactors.⁽¹³⁾ The thirteenth report examines the decommissioning of reference light water reactors that have been involved in serious accidents.⁽¹⁴⁾ The fourteenth and fifteenth reports are addendums to the pressurized water reactor report and the boiling water reactor report, respectively, and examine the impacts on decommissioning of both of these plant types of a temporary inability to dispose of waste offsite at the time of decommissioning.^(15,16) The sixteenth report, an addendum to the nuclear research and test reactors report, addresses the sensitivity of decommissioning radiation exposure and costs to selected parameters at nuclear research and test reactor facilities.⁽¹⁷⁾ The seventeenth report deals with the decommissioning of independent spent fuel storage installations.⁽¹⁸⁾ Additional addendums to the pressurized and boiling water reactor report examine the radioactive wastes expected to result from decommissioning the reference BWR and classify those

wastes in accordance with 10 CFR 61.^(19,20) An addendum to the Multiple Reactor Station Report examines the impact on multiple reactor station decommissioning of an interim inability to dispose of wastes offsite.⁽²¹⁾ This report conceptually evaluates factors of technical requirements, cost, and safety for the post-accident cleanup and decommissioning of fuel-cycle and non-fuel-cycle facilities that have experienced a significant accident.

The second series of studies covers supporting information on the decommissioning of nuclear facilities. Four reports have been issued in the second series. The first consists of an annotated bibliography on the decommissioning of nuclear facilities.⁽²²⁾ The second is a review and analysis of current decommissioning regulations.⁽²³⁾ The third covers the facilitation of the decommissioning of light water reactors.⁽²⁴⁾ The fourth report covers the establishment of an information base concerning monitoring for compliance with decommissioning survey criteria.⁽²⁵⁾ The fifth report addresses the technology and cost of termination surveys associated with decommissioning of nuclear facilities.⁽²⁶⁾

The information provided in this report, including any comments, will be included in the record for consideration by the Commission in establishing criteria and new standards for decommissioning. Comments on this report should be mailed to

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ABSTRACT

Technical requirements, costs, and safety are conceptually evaluated for the post-accident cleanup and decommissioning of fuel cycle and non-fuel cycle facilities that have experienced a significant accident.

The initial effort following the accident is to bring the plant under control and to prevent further accident consequences. Stabilization of the facility is followed by accident cleanup and by decommissioning or refurbishment. This study provides an analysis of accident cleanup and decommissioning activities for accident scenarios that are considered to be credible with respect to the initiating events for each facility.

Accident cleanup is postulated to include 1) initial decontamination of building surfaces to reduce the subsequent occupational dose to cleanup and decommissioning workers, and 2) management of the resulting wastes. For the reference MOX plant, accident cleanup is estimated to require 1.5 years and cost \$4.6 million. For the reference U-Fab plant, accident cleanup is estimated to require 1.0 year and cost \$2.2 million. All costs are in 1981 dollars and include planning and preparation as well as the actual cleanup activities.

Decommissioning is assumed to follow accident cleanup. DECON at the reference MOX plant is estimated to cost \$11.5 million following cleanup after the reference accident. DECON at the reference U-Fab plant is estimated to cost \$5.3 million following cleanup after the reference accident. DECON of the reference non-fuel cycle facility laboratory is estimated to cost \$74 thousand following the reference accident, and DECON of the reference ore processing facility is estimated to cost \$157 thousand.

One of the differences between post-accident cleanup and decommissioning and normal-shutdown decommissioning is the higher radiation exposure to workers during post-accident operations. For accident cleanup and decommissioning following the reference accident at the reference MOX plant, total occupational radiation doses (external doses from gamma radiation) are estimated to be about 33 man-rem for accident cleanup and about 70 man-rem for DECON following accident cleanup. This compares with an occupational radiation dose of about 70 man-rem for normal-shutdown DECON at the reference MOX plant. For accident cleanup and decommissioning following the reference accident at the U-Fab plant, total occupational radiation doses are estimated to be about 0.4 man-rem for accident cleanup and about 15 man-rem for DECON following accident cleanup. This compares with an occupational radiation dose of about 15 man-rem for normal shutdown DECON at the reference U-Fab plant. In order to ensure that worker doses are ALARA, careful planning and rehearsal of cleanup operations and the use of remote and semi-remote cleaning techniques are required to reduce occupancy times in high-radiation areas and to minimize occupational exposures during accident cleanup.

The public safety impacts of post-accident cleanup and decommissioning are greater than the corresponding impacts of normal-shutdown decommissioning. However, radiation doses to the public from routine accident cleanup and decommissioning operations are below permissible radiation dose levels in unrestricted areas and are well within the range of annual radiation doses from normal background.

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1.0 INTRODUCTION

This report contains the results of a study sponsored by the Nuclear Regulatory Commission (NRC) to conceptually evaluate post-accident decommissioning of reference nuclear fuel cycle and non-fuel cycle facilities. Two nuclear fuel cycle facilities are covered: a mixed oxide fuel fabrication (MOX) plant and a uranium fuel fabrication (U-Fab) plant. The non-fuel cycle facilities are a pharmaceutical manufacturing plant and an ore processing facility. The primary purpose of this study is to provide information on the available technology, the safety considerations, and the probable costs of post-accident decommissioning and of accident cleanup activities that precede the decommissioning for the reference facilities. Information from this study is intended as background data and to form a basis for the modification of existing regulations and the development of new regulations pertaining to decommissioning activities.

A post-operations activities flow sheet showing the sequence of operations leading to decommissioning and release of a facility for unrestricted use is shown in Figure 1.0-1. As can be seen from the figure, the activities that follow shutdown of a facility that has been involved in an accident are somewhat different from the activities that follow normal shutdown. Post-accident activities include stabilization, accident cleanup, and active decommissioning. Stabilization is the period during which the accident is brought under control and the plant is returned to a stabilized condition. Activities during this period include bringing the plant to a safe shutdown condition, the restoration of essential systems and services required to maintain the plant, and preliminary surveys to determine the extent of damage resulting from the accident and to assess the radiological condition of the plant. Once the situation is stabilized, accident cleanup can begin. Accident cleanup is considered to be those activities resulting in cleanup of contamination, and processing and disposal of wastes generated by the accident. As shown in Figure 1.0-1, accident cleanup could either be followed by recovery of the facility for restart or by decommissioning. If it is decided to retire the facility from service, decommissioning activities are considered to begin following completion of the accident cleanup.

This study considers the technology, safety, and costs of accident cleanup and decommissioning. Details of the stabilization period and of recovery of the facility for restart are not analyzed. Because the overall accident cleanup activities would be similar whether the facility was ultimately restarted or decommissioned, the accident cleanup analysis is considered to be a good representation of this period, independent of the ultimate use of the plant. Accident cleanup requirements are also shown to be essentially independent of the decommissioning alternative chosen.

FIGURE 1.0-1. Post-Operations Activities Flow Sheet

Although many of the considerations involved with normal decommissioning also apply to cleanup and decommissioning following an accident, there are important differences in some of the specific requirements. To facilitate comparisons between the requirements and costs of post-accident and normal-shutdown decommissioning and to provide consistency with earlier decommissioning studies, previously published reports of facility decommissioning following normal shutdown^(1,2,3) are used for the reference facility descriptions and basic decommissioning information needed for this study. The reference MOX plant is the Cimarron, Oklahoma, plant of Kerr-McGee.⁽¹⁾ The U-Fab plant is

the Wilmington, North Carolina, plant of General Electric Company⁽²⁾ and the non-fuel cycle facility is based on the New England Nuclear Corporation facility.⁽³⁾ The use of these plants as reference facilities for this study should not be construed as implying anything about their reliability and/or safety relative to other plants in operation. Their use merely facilitates comparisons with the earlier, non-accident, decommissioning studies.

Requirements for post-accident decommissioning and for accident cleanup preceding the decommissioning are analyzed for three accident scenarios. The scenarios are defined in terms of their consequences (e.g., radioactive contamination levels, radiation exposure rates, and damage to buildings and equipment). The accident scenarios are believed to be credible on the basis of reviews of accident literature, safety analysis reports, and NRC safety evaluation reports. Minor accidents (i.e. spills, etc.) that would not result in a significant accident cleanup effort or that would not greatly impact the requirements and costs of decommissioning are not considered. Only accidents that result in relatively severe contamination and/or relatively large-scale mechanical damage implying considerable difficulty in implementing accident cleanup and decommissioning procedures are considered. It is not intended that the accident scenarios covered in this study be inclusive of all possible accidents, but they should provide a boundary condition for consideration of post-accident decommissioning.

As discussed above, following the postulated accidents and stabilization periods, accident cleanup procedures begin. These accident cleanup procedures are designed to reduce the initial high levels of radioactive contamination present in the facility, thereby reducing the radiation dose received by cleanup and decommissioning workers. The overall tasks that must be performed during accident cleanup are expected to be relatively independent of whether the plant is refurbished and restarted or decommissioned. However, the methods used to complete certain cleanup tasks may vary, depending on whether the decision is to restart or to decommission and, in the latter case, on the decommissioning alternative chosen.

It is beyond the scope of this study to evaluate the alternatives of permanently shutting down or restarting a facility that has been involved in one of the accident scenarios described above. If the facility is to be permanently shut down, actual decommissioning activities are considered to begin following the completion of accident cleanup. Three alternative approaches to decommissioning have been defined:⁽⁴⁾

- DECON - The immediate removal from the facility of all material with residual radioactivity levels greater than those permitted for unrestricted use of the property. DECON meets the requirements for termination of the facility license and renders the facility and site available for unrestricted use within a finite time period.

- SAFSTOR - Activities designed to place (preparations for safe storage) and maintain (safe storage) a radioactive facility in such a condition that the risk to public safety is within acceptable bounds. At the conclusion of the safe storage period, the facility must be decontaminated to levels that permit its release for unrestricted use (deferred decontamination).
- ENTOMB - Cleanup and decontamination, to a lesser extent than DECON, is coupled with the confinement of the remaining contaminated components in a strong and structurally long-lived material to assure retention until the radioactivity decays to levels that permit unrestricted release of the property.

The accident cleanup activities that precede decommissioning result in the initial decontamination of building surfaces and equipment. At the conclusion of the cleanup activities, many of the initial effects of an accident are eliminated or reduced in magnitude. Therefore, the requirements and costs of decommissioning following accident cleanup are not strongly affected by the specific condition of the plant immediately following an accident. One factor that can influence the decommissioning requirements and costs is the residual contamination remaining on building surfaces and equipment after accident cleanup activities are completed.

Since the SAFSTOR and ENTOMB alternatives are not likely to be used for the facilities considered in this study, decommissioning is analyzed in detail only for DECON. Estimates are made of manpower requirements, occupational radiation exposures, and costs for decommissioning following the reference accident.

Sets of work plans are developed for the conceptual accident cleanup of the reference facilities for decommissioning via the DECON alternative. From these work plans, estimates are made of manpower requirements, major resource and equipment needs, volumes of contaminated material packaged for disposal, costs of accomplishing the work, and exposure of workers to radiation as a result of the accident cleanup and decommissioning efforts. These work plans and estimates of airborne releases of radioactive materials are used to evaluate the impacts of accident cleanup and decommissioning operations on the general public. The plans and techniques used in this study represent operations that would be required and could, therefore, be used to safely decontaminate and decommission a nuclear facility that has been involved in an accident.

The safety impacts and estimated costs developed in this study are sensitive to the specifics of the reference plants and the postulated accidents. Such specifics include the levels of radioactive contamination and extent of physical damage associated with the accident, the operating history of the plant, the location of the plant relative to waste disposal sites, and the

structural and mechanical details of the plant. The safety impacts and costs are also sensitive to the assumptions made in this study about the techniques and procedures used for accident cleanup and decommissioning. The bases and assumptions on which the study is based should be carefully examined before the study result are used.

The results of the reference analysis are summarized in Chapters 8 through 11. Sufficient detail is provided in these chapters and their corresponding appendices to justify the results that are presented and to enable the reader to trace the logic used in arriving at these results.

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2. H. K. Elder and D. E. Blahnik, Technology, Safety and Costs of Decommissioning a Reference Uranium Fuel Fabrication Plant, NUREG/CR-1266, Vols. 1 and 2, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, October 1980.
3. E. S. Murphy, Technology, Safety and Costs of Decommissioning Reference Non-Fuel-Cycle Nuclear Facilities, NUREG/CR-1754, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, February 1981.
4. Draft Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, NUREG-0586, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C., January 1981.

2.0 SUMMARY

This chapter summarizes the results of a study sponsored by the Nuclear Regulatory Commission (NRC) to conceptually evaluate post-accident decommissioning of fuel cycle and non-fuel cycle facilities and the accident cleanup that precedes the decommissioning. The purpose of the study is to provide information on the available technology, the safety considerations, and the probable costs of cleanup and decommissioning of fuel cycle facilities that have experienced a significant accident. A summary of the results is given in the following sections.

2.1 STUDY APPROACH

Post-accident decommissioning and the accident cleanup activities are analyzed for accident scenarios chosen to illustrate technological requirements, occupational radiation doses, and cleanup and decommissioning costs that are substantially greater than those estimated for decommissioning following normal shutdown. The parameters characterizing the reference accident scenarios that are important to accident cleanup and decommissioning activities are the resulting radioactive contamination levels, radiation exposure rates, and damage to the structures and equipment.

The activities that follow shutdown of a facility that has been involved in an accident are somewhat different from the activities that follow normal shutdown. Post-accident activities include stabilization, accident cleanup, and active decommissioning or restart. Stabilization activities may include the efforts specifically designed to shut down the plant and maintain it in a safe shutdown condition. They may also include efforts to stabilize the plant following the accident by isolating and containing accident contamination until cleanup facilities are available. In addition, stabilization activities may include the restoration of essential systems and services required to maintain the plant in a safe shutdown condition. The specific nature of these stabilization activities can be very diverse depending on facility design and the nature of the accident. Because of this, and because the study primarily deals with decommissioning and accident cleanup, this study does not analyze the technology, safety, and costs during the stabilization period.

Once the situation is stabilized, accident cleanup can begin. Accident cleanup is considered to be those activities leading to cleanup of contamination and processing and disposal of wastes generated by the accident. Accident cleanup can either be followed by recovery of the facility for restart or by decommissioning. This study includes only an analysis of the activities related to decommissioning and does not analyze the activities related to refurbishment following accident cleanup.

The goals of accident cleanup are to 1) reduce the initial high levels of radioactive contamination present in the facility, thereby reducing the radiation dose received by cleanup and decommissioning workers; and 2) collect, package for disposal, and dispose of the readily dispersible radioactivity in the plant. To achieve these goals, accident cleanup is postulated to include the following tasks:

- processing of contaminated liquid released during the accident
- initial decontamination of building surfaces and decontamination or disposal of some equipment
- solidification and packaging of wastes from the above accident cleanup operations.

These accident cleanup activities are necessary and would be approximately the same whether the plant is ultimately refurbished or decommissioned, and, if it is decommissioned, they would be independent of whichever decommissioning alternative is chosen. The rationale for this is discussed in detail in Section D.1 of Appendix D. Decontamination during the accident cleanup period (whether for eventual restart or decommissioning) cannot be too chemically corrosive or destructive, since this could compromise the integrity of systems that must remain intact during cleanup and decommissioning. The work required to complete specific cleanup tasks is, of course, determined by the severity of the accident.

If it is decided that a facility is to be permanently shut down, actual decommissioning activities are considered to begin following the completion of the accident cleanup activities listed above. The alternatives for decommissioning are DECDN (immediate decontamination to unrestricted release), SAFSTOR (safe storage with deferred decontamination to unrestricted release), and ENTOMB (entombment of radioactive materials with decay to unrestricted release). In this study, decommissioning is analyzed in detail only for DECON since the other alternatives are not likely to be used for non-reactor nuclear facilities.

Work plans developed for the conceptual accident cleanup and for decommissioning provide the bases for estimates of manpower requirements, major resource and equipment needs, volumes of radioactive materials requiring disposal, exposure of workers to radiation as a result of cleanup and decommissioning efforts, and costs of accomplishing the work. Estimates of releases of radioactive material during accident cleanup and decommissioning operations are used to evaluate the impacts of these operations on the general public.

The results of the study are dependent on assumptions made regarding facility design and location, accident scenarios, and resultant contamination. The results also depend on assumptions concerning capability to proceed with accident cleanup and eventual decommissioning and on work plans and techniques used to achieve the desired decommissioned condition. The choices of plans and techniques in the study are believed to be representative of the state-of-the-art and to represent operations that would be required and could therefore be used to safely decontaminate and decommission non-reactor nuclear facilities that had been involved in an accident. In using the results of this study, particularly as they apply to accident cleanup, consideration should be given to uncertainties in estimating contamination levels, plant damage, and other factors that can affect the situation. Some of these factors are discussed in Section 2.2. Application of the study results to situations different from those assumed in the study could produce erroneous conclusions.

2.2 KEY STUDY BASES

The major study bases are listed below.

- The study must yield realistic and up-to-date results.
- Accident scenarios illustrate a range of accident cleanup and decommissioning requirements and are in the range of scenarios considered as design bases in NRC safety evaluations.
- To facilitate comparisons with the earlier non-accident decommissioning studies, the same reference plants are used for this post-accident study.
- The postulated activities are conducted within the framework of existing regulations and regulatory guidance.
- Current technology and techniques are used in descriptions of accident cleanup and decommissioning procedures.
- The postulated activities conform to the ALARA occupational exposure philosophy.
- Unrestricted release of the decontaminated facility is predicated on decontamination to residual levels of radioactivity as specified by present regulatory guidance.
- Work schedules include allowances for inefficiencies associated with work in high-radiation areas.

- Manpower requirements are adjusted so that average individual radiation doses to workers do not exceed 5 rem/year with the intention of keeping occupational exposures ALARA.
- Costs of accident cleanup and decommissioning are in 1981 dollars.
- Sufficient funding is available to carry out the accident cleanup and decommissioning without significant delays.
- No scheduling or funding allowances are made for research and development activities except those related to procurement of the special tools and equipment needed for accident cleanup.
- Based on 10 CFR Part 61 criteria, low-level radioactive wastes are assumed to be disposed of by shallow-land burial. For other wastes that are unsuitable for shallow-land burial, criteria are not yet well defined. Hence, these wastes, specifically highly radioactive and/or transuranic wastes, are assumed to be sent to a federal repository.

2.3 CLEANUP AND DECOMMISSIONING EXPERIENCE AT ACCIDENT-DAMAGED FACILITIES

A few accidents at non-reactor nuclear facilities have necessitated extensive post-accident cleanup operations; however, no accidents have resulted in a requirement to decommission commercial non-reactor facilities. Most of the techniques and procedures used to decontaminate or decommission a plant following an accident are similar to those used for decommissioning following normal shutdown, although consideration must be given to the problems of working in higher than normal radiation areas.

Some accidents in fuel cycle and non-fuel-cycle facilities have resulted in high levels of radioactive contamination on building surfaces and equipment and in high radiation exposure rates to accident cleanup personnel. In all cases where contamination has occurred, methods and procedures have been devised to safely remove the contamination with only modest total radiation doses to decontamination workers.

2.4 REFERENCE FACILITIES AND SITE

The reference MOX plant for this study is the Cimarron, Oklahoma, plant of Kerr McGee. It is the same plant that was used as the reference plant for a previous study (NUREG/CR-0129)⁽⁴⁾ of MOX plant decommissioning following normal shutdown. The reference U-Fab plant is the Wilmington, North Carolina, plant of General Electric Company and is the same plant that was used as the

reference plant for a previous study (NUREG/CR-1266)⁽⁵⁾ of U-Fab decommissioning following normal shutdown. The reference non-fuel cycle plant and ore processing site are taken from NUREG/CR-1754⁽⁶⁾ which described normal decommissioning of the reference facilities. The use of these plants as reference facilities should not be construed as implying anything about their reliability and/or safety relative to other facilities in operation or under construction. Their use facilitates comparisons with the earlier, non-accident, decommissioning studies.

The reference site used in these analyses is typical of a midwestern or middle southeastern river site. This site has been developed for use in the series of decommissioning studies that is being performed for the NRC by Pacific Northwest Laboratory. Sufficient descriptive information is presented for both the facility and the site to permit the development of the detailed work plans, the cost estimates, and the safety assessments that are the results of this study.

2.5 REFERENCE ACCIDENT SCENARIOS AND RESULTANT CONTAMINATION LEVELS

The reference accident scenarios provide the bases for the post-accident cleanup and decommissioning cost and safety estimates given in this report. From the viewpoint of this study, which deals with accident cleanup and decommissioning, the consequences of an accident (i.e., the radiological and physical condition of the plant following an accident) are much more important than the sequence of events that occur during the accident. Therefore, detailed descriptions of accident sequences are not given. The reference accident scenarios provide information about radioactive contamination, radiation exposure rates, and damage to structures and equipment. The consequence scenarios chosen for this study are believed to be credible with respect to initiating circumstances and are in the range of scenarios currently as design bases by the NRC in safety evaluations. A summary of the accident scenarios used for the reference facilities which form the basis of this study is listed in Table 2.5-1.

2.6 ACCEPTABLE RESIDUAL RADIOACTIVE CONTAMINATION LEVELS

Unrestricted release of a nuclear facility following decommissioning requires that the radioactivity remaining in the facility and on the site be reduced to levels that are considered acceptable for unrestricted access and subsequent NRC license termination. Criteria that currently exist or are being developed for the unrestricted release of a decommissioned facility are

TABLE 2.5-1. Summary of Accident Scenarios

<u>Accident Parameter</u>	<u>Small MOX Plant Accident Description</u>
• Accident	Explosion followed by major facility fire
• Inventory at Risk	125 kg Pu
• Release Mechanism	1. Blending tank failure 100% liquid release 2. 4% aerosol release due to fire
• Radioactivity Released	
- Inside Building	35 kg Pu
- Outside Building	5 g Pu
• Physical Damage to Building	Fire damage to interior, building exterior not breached

<u>Accident Parameter</u>	<u>U-Fab Plant Accident Description</u>	
• Accident Description	UF ₆ cylinder rupture (2 cylinders damaged)	UF ₆ cylinder rupture
• Inventory at Risk	2160 kg UF ₆	4400 kg UF ₆
• Release Mechanism	UF ₆ vaporization 1/3 of inventory released	UF ₆ vaporization 1/3 of inventory released
• Radioactivity Released		
- Inside Building	700 kg UF (440 kg U ₆)	None
- Outside Building	2 kg UF ₆ (~1 kg U)	1260 kg of UF ₆
• Physical Damage to Building	None	None

TABLE 2.5-1. (contd)

Accident Parameter	Non-Fuel Cycle Facility	
	Accident Description	
• Accident	Major facility fire	Fire in storage area
• Inventory at Risk	Max. inventory in each laboratory	Max. inventory in plant storage area
• Release Mechanism	100% of ^3H , ^{14}C , ^{125}I 1% of ^{137}Cs and ^{241}Am	100% of ^3H , ^{14}C , ^{125}I 1% of ^{137}Cs and ^{241}Am
• Radioactivity Released		
- Inside Building	2000 Ci ^3H , 10 Ci ^{14}C 1 Ci ^{125}I , 0.5 Ci ^{137}Cs , 1 Ci ^{241}Am	4000 Ci ^3H , 15 Ci ^{14}C , 4 Ci ^{125}I , 1 Ci ^{241}Am
- Outside Building	Negligible	Negligible
• Physical Damage to Building	Fire Damage to Building	Fire Damage to Building

Accident Parameter	Ore Processing Facility Accident Description
• Accident	Tailings dam failure
• Inventory at risk	8.2×10^4 kg of U_3O_8 2.05×10^5 kg ThO_2
• Release mechanism	Waterborne solids released from tailings pond
• Radioactivity Released	Uranium and daughters

summarized in Chapter 7. Criteria under development will base allowable residual radioactivity levels for facility release on the dose anticipated to be received by individuals who use the facility or site after the license is terminated.

2.7 MOX AND U-FAB PLANT CLEANUP AND DECOMMISSIONING ACTIVITIES AND COSTS

Post-stabilization activities at an accident-damaged nuclear facility include: 1) preparations for accident cleanup, 2) accident cleanup of the

process building and of other buildings as necessary, and, if the plant is to be permanently shut down, 3) decommissioning of the facility. This section briefly summarizes some requirements and procedures of these activities and costs.

2.7.1 Preparations for Accident Cleanup

Preparations for accident cleanup include the following activities (not necessarily in the order listed):

- contaminated area entry and data acquisition
- preparation of documentation for regulatory agencies
- design, fabrication, and installation of special equipment
- development of detailed work plans and procedures
- selection and training of accident cleanup staff.

Initial entries into the contaminated and damaged areas are made for the purpose of obtaining data on the radiological and physical condition of the building. Data obtained during these entries include measurements of contamination levels and radiation exposure rates, estimates of physical damage, and information about the operational status of plant systems and services. These data are needed to provide a basis for planning accident cleanup operations and for preparing documentation for regulatory agencies.

A major task is the preparation by the licensee of the necessary documentation that may be required to obtain regulatory approvals as necessary to proceed with cleanup operations. The time requirement for furnishing information to regulatory agencies, issuing environmental statements and assessments, and securing regulatory approvals to proceed with specific cleanup tasks is a critical factor in determining when actual cleanup operations can begin, and could be a cause of delay in accident cleanup.

Several major facility and equipment items needed for accident cleanup are designed and fabricated during preparations for cleanup. These items include:

- an evaporator/solidification facility to process the decontamination solutions generated
- a volume reduction incinerator to reduce the total quantity of waste requiring disposal
- shielded and unshielded storage facilities for interim storage of radioactive wastes; these may be required if there is difficulty in disposing of wastes because of regulatory or political constraints
- a laundry facility.

Detailed work plans and work procedures for accident cleanup are developed during the preparations phase. Requirements for cleanup of the contaminated plant are based on an evaluation of the condition of the plant following the accident.

The selection and training of key staff for accident cleanup begin during the preparations phase.

2.7.2 Accident Cleanup Activities

1. Decontaminate the building to permit access.
2. Install the evaporator solidification facility.
3. Flush contaminated tanks and pipes and process the contaminated liquid.
4. Continue the decontamination of building surfaces and equipment.
5. Process contaminated liquids from cleanup operations.
6. Replace contaminated filters.
7. Solidify and package wastes from accident cleanup.
8. Treat and dispose of wastes from accident cleanup.
9. Perform radiation survey to determine the extent of residual contamination following accident cleanup and to verify the effectiveness of cleanup operations.

The objective of initial decontamination of surfaces and equipment is to reduce surface contamination levels and resultant radiation exposure rates to permit reasonable occupancy times for workers engaged in cleanup operations.

Radioactive wastes from accident cleanup operations include:

1. Solid Materials - Dry radioactive wastes generated from decontamination. These materials consist of trash, contaminated equipment and material, and HEPA and roughing filters.
2. Process Solids - Contaminated sludges and process solid wastes that arise from the treatment of water and decontamination liquids. These solid wastes include filter cartridge assemblies and evaporator bottoms.

3. Chemical Decontamination Solutions - Liquid decontamination wastes that have not been treated to generate process solids. These wastes are immobilized by incorporation in cement or in vinyl ester styrene.

2.7.3 Decommissioning Activities

Following the completion of the accident cleanup activities, the actual decommissioning activities begin. As a result of the efforts during accident cleanup, the decommissioning activities are considered not to be affected greatly by the condition of the plant immediately following the accident. Hence decommissioning can be carried out in a more stable environment than accident cleanup.

Many decommissioning tasks are common to both post-accident and normal-shutdown decommissioning. However, changes in the physical and radiological condition of the plant that still remain following accident cleanup can result in substantial changes in the time and manpower required for post-accident decommissioning. Radiation doses to workers during post-accident decommissioning are higher than those following normal shutdown because of contamination on building and equipment surfaces that remains after accident cleanup activities are completed. Physical damage to the plant may compromise some systems and equipment needed for the performance of decommissioning tasks, thus necessitating repairs or substitutions and increasing the time and cost of post-accident decommissioning.

In this study, post-accident decommissioning is analyzed for the DECON alternative. DECON is the decommissioning alternative that leads to the earliest release of the facility and site for unrestricted use and to the earliest termination of the facility's nuclear license. The decontamination and dismantlement activities during post-accident DECON are similar to activities during DECON following normal shutdown, described in References 4, 5 and 6. These activities include:

- decontamination of the surfaces of process systems and equipment
- disassembly and disposal of contaminated equipment, including ductwork and piping
- removal of contaminated concrete
- packaging and shipment of radioactive wastes to a waste disposal site
- a final radiation survey.

Some of these activities are initiated during accident cleanup. However, the bulk of this work is carried out during DECON, particularly the removal of large equipment components and of contaminated structural material. Accident-generated contamination results in a somewhat greater level of effort and a greater volume of radioactive waste material for post-accident DECON than for DECON following normal shutdown.

2.7.4 Costs of Accident Cleanup and of Decommissioning

Estimated costs of accident cleanup and of decommissioning at the reference MOX plant and U-Fab plant are summarized in this section. All costs are in 1981 dollars and include a 25% contingency.

Costs of MOX Plant Accident Cleanup

Accident cleanup costs at the reference MOX plant are summarized in this section. These costs are based on the key study basis assumptions listed in Section 4.2 of Chapter 4, and on the accident cleanup activities described for the reference MOX plant in Chapter 8.

Total estimated costs requirements for accident cleanup at the reference MOX plant are shown in Table 2.7-1. Accident cleanup following the reference accident is estimated to cost 4.6 million and to require 1.5 years for completion. These costs and times include the requirements for planning and preparation as well as for the actual cleanup activities.

TABLE 2.7-1. Summary of Estimated Costs of Accident Cleanup
at the Reference MOX Plant

<u>Cost Category</u>	<u>Cost in Millions of 1981 Dollars^(a)</u>	<u>Percent of Total</u>
Manpower		
Planning and Preparation	0.328	8.9
Accident Cleanup	1.916	51.7
Equipment and Supplies	0.526	14.2
Disposal of Radioactive Material	0.360	9.7
Miscellaneous Owner Expense	0.536	14.5
Specialty Contractors	<u>0.037</u>	<u>1.0</u>
Subtotal	3.703	100.0
25% Contingency	<u>0.926</u>	
Total Costs of Accident Cleanup	4.629	

(a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

Costs of MOX Plant Decommissioning

Total estimated costs of decommissioning following accident cleanup are summarized in Table 2.7-2 for the reference MOX plant. Decommissioning cost details are presented in Chapter 8. Decommissioning following accident cleanup is estimated to cost \$11.5 million and to require 1.75 years for completion.

TABLE 2.7-2. Summary of Estimated Costs of Post-Accident DECON at the Reference MOX Plant^(a)

<u>Cost/Category</u>	<u>Cost in Millions of 1981 Dollars^(b)</u>	<u>Percent of Total</u>
Staff Labor		
Planning and Preparation	0.946	10.3
Decommissioning	5.269	57.5
Equipment and Supplies	1.217	13.3
Disposal of Radioactive Material	0.820	9.0
Miscellaneous Owner Expense	0.670	7.3
Specialty Contractors	0.238	2.6
Subtotal	9.160	100.0
25% Contingency	2.290	
Total Decommissioning Costs	11.450	

(a) Summary does not include the cost of packaging and disposal to shallow-land burial of the sludge from the sewage lagoons that would add an additional \$0.75 million in waste disposal and contingency costs.

(b) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

Costs of U-Fab Plant Accident Cleanup

The estimated costs of accident cleanup at the reference U-Fab plant are summarized in Table 2.7-3. These activities are estimated to require about 0.5 year (plus 0.5 year for planning and preparations), at a cost of approximately \$2.2 million.

Manpower represents about 53% of the total cost of accident cleanup. Manpower costs include both support staff and accident cleanup worker labor.

Costs of U-Fab Plant Decommissioning

Total estimated costs of decommissioning following accident cleanup are summarized in Table 2.7-4 for the reference U-Fab plant. Decommissioning cost

TABLE 2.7-3. Summary of Estimated Costs of Accident Cleanup at the Reference U-Fab Plant

<u>Cost/Category</u>	<u>Cost in Millions of 1981 Dollars^(a)</u>	<u>Percent of Total</u>
Manpower		
Planning and Preparation	0.266	15.3
Accident Cleanup	0.657	37.9
Equipment and Supplies	0.230	13.3
Disposal of Radioactive Materials	0.240	13.8
Miscellaneous Owner Expense	0.302	17.4
Specialty Contractors	<u>0.040</u>	<u>2.3</u>
Subtotal	1.735	100.0
25% Contingency	<u>0.434</u>	
Total Costs of Accident Cleanup	2.169	

(a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

TABLE 2.7-4. Summary of Estimated Costs of Post-Accident DECON at the Reference U-Fab Plant

<u>Cost/Category</u>	<u>Cost in Millions of 1981 Dollars^(a)</u>	<u>Percent of Total</u>
Manpower		
Planning and Preparation	0.398	9.3
Decommissioning	2.334	54.6
Equipment and Supplies	0.174	4.1
Disposal of Radioactive Material	0.273	6.4
Miscellaneous Owner Expense	1.011	23.6
Specialty Contractors	<u>0.084</u>	<u>2.0</u>
Subtotal	4.274	100.0
25% Contingency	<u>1.073</u>	
Total Decommissioning	5.347	

(a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

details are presented in Chapter 9. Decommissioning following accident cleanup is estimated to cost \$5.3 million and to require one year for completion.

2.8 DECOMMISSIONING OF REFERENCE NON-FUEL CYCLE FACILITY

Estimates of time and manpower requirements and total costs are made for DECON following an accident at the reference laboratory described in Chapter 5. The decommissioning analyses for this laboratory use cost data for the decommissioning of facility components. Costs of planning and preparation and of a final radiation survey of the decommissioned facility are added to the basic decontamination costs of the individual components. Decommissioning requirements and costs following an accident for the reference laboratory are summarized in Table 2.8-1.

TABLE 2.8-1. Summary of Estimated Requirements and Costs for DECON at the Reference Laboratory Following an Accident

	Requirement or Cost for Reference Laboratory ^(a) ²⁴¹ Am Laboratory
Time (days)	81
Manpower (man-days)	336
Occupational Dose ^(b) (man-rem)	40
Costs (\$ thousands) ^(c)	
Staff Labor	49.1
Equipment	2.5
Supplies	7.0
Waste Management	15.3
Totals	74

(a) The listed value represents the requirement or cost for both planning and preparation and the actual decommissioning of the laboratory.

(b) Estimated on the assumption that workers do not use protective respiratory equipment. Doses could be reduced by 1 or 2 orders of magnitude through the use of this equipment. This is a likely alternative for the ²⁴¹Am laboratory.

(c) Costs are in 1981 dollars and include a 25% contingency.

Decommissioning following an accident is preceded by a period of planning and preparation that includes activities to ensure that decommissioning is performed in a safe and cost-effective manner in accordance with all applicable federal, state, and local regulations. Planning and preparation activities include the preparation of documentation for regulatory agencies, an initial radiation survey to determine the radiological condition of the laboratory, and the development of detailed work plans.

DECON options postulated for the components of the reference laboratories are arbitrary but are believed to represent reasonable approaches to the decommissioning of particular components. Some fume hoods and glove boxes are assumed to be decontaminated to unrestricted release levels, while other hoods and glove boxes are packaged for disposal, depending on the magnitude and type of surface contamination. Laboratory workbenches and other components such as refrigerators and storage cabinets are assumed to be decontaminated to unrestricted release levels. Sinks are decontaminated, and drain lines are packaged for disposal. Ventilation ductwork is sectioned and packaged for disposal. Building surfaces are generally assumed to be decontaminated to unrestricted use levels.

The final decommissioning activity is a comprehensive radiological survey to document levels of radioactivity remaining in the facility after DECON is completed and to certify that these levels are less than those specified for unrestricted release.

Decommissioning is assumed to be performed by employees of the owners or operators of the laboratories. The basic decommissioning work crew includes a foreman and three technicians, assisted by a health physicist. Craftsmen (electricians, pipefitters, etc.) are added to this crew on a part-time basis to perform specific tasks. Staff labor costs are postulated to include the salary of a supervisor on a half-time basis.

Costs for decommissioning the reference laboratory include the costs of staff labor, equipment, supplies, and waste management. Costs are estimated for planning and preparation, for the actual decommissioning, and for the termination survey. Total costs listed in Table 2.8-1 are the sum of all of these costs. All costs are expressed in 1981 dollars and include a 25% contingency.

2.9 DECOMMISSIONING OF REFERENCE ORE PROCESSING FACILITY SITE

Estimates are made of time and manpower requirements and total costs of decommissioning the reference site briefly described in Chapter 5. For the tailings pile containing uranium and thorium residues, estimates are made of requirements and costs for both the site stabilization and the removal options. Decommissioning requirements and costs for the reference site are summarized in Table 2.9-1.

TABLE 2.9-1. Summary of Estimated Manpower Requirements, Costs, and Radiation Doses for Decommissioning the Ore Processing Site Following an Accident

Site	Requirement or Cost			
	Time (days)	Manpower (man-days)	Cost ^(a) (\$ thousands)	Occupational Radiation Dose (man-rem)
Tailings Pile				
Stabilization Option	32	174	157	0.08
Removal Option	139	1660	8380	1.0

(a) Costs are in 1981 dollars and include a 25% contingency.

Because concentrations of radioactivity are low and inhalation of resuspended particulates either is not a serious consideration or can be protected against by the use of respiratory equipment, removal of the waste and contaminated soil is accomplished with standard earthmoving equipment. Radioactive material is packaged in plastic-lined plywood boxes for shipment to a shallow-land disposal site.

For the site with a contaminated tailings pile, site stabilization is assumed to include the following procedures. The pile is covered with a 50-mm-thick layer of asphalt. This asphalt layer is then covered with 1 meter of soil. The soil is mounded slightly at the center of the pile to allow water to drain from the soil cover and to prevent the accumulation of runoff from rainfall or snowmelt. After compaction and contouring of the soil cover, the area is seeded with grass.

Decommissioning activities include a radiological survey to assess the condition of the site before site stabilization or removal operations begin and restoration of the site by backfilling and planting vegetation after waste removal is completed. A final radiation survey to verify that the radioactivity remaining on the site is less than release limits is performed prior to release of a site for unrestricted use. Decommissioning is assumed to be performed by a contractor hired by the owner or operator of the site.

Decommissioning costs include the costs of staff labor, equipment, supplies, soil sample analyses, waste management, and a contractor's fee. Total costs shown in Table 2.9-1 are the sum of planning and preparation, actual decommissioning, and termination survey costs. All costs are expressed in 1981 dollars and include a 25% contingency. Approximately 90% of the cost of the

removal option for decommissioning a tailings pile is related to waste management (i.e., the packaging, transportation, and disposal of soil and waste exhumed from the sites).

2.10 OCCUPATIONAL AND PUBLIC SAFETY

Radiological and nonradiological safety impacts from routine activities and from potential industrial and transportation accidents are identified and evaluated for post-accident cleanup and decommissioning at the reference non-reactor nuclear facilities. Safety impacts are calculated for accident cleanup following each of the reference accident scenarios for each of the reference facilities. Decommissioning safety impacts are calculated for the DECON alternative following the reference accident. The safety evaluation includes consideration of radiation dose to the public from routine activities and from postulated industrial accidents. The evaluation employs current data and methodology, along with engineering judgment when necessary, to estimate the required input information and the resulting safety impacts. The approach used to evaluate all the safety impacts is believed to be conservative.

The results of the safety evaluations of post-accident cleanup activities are summarized in Table 2.10-1 for the reference MOX and U-Fab plants. Details of the safety evaluation of post-accident cleanup activities are given in Chapter 12. The principal source of radiation dose to the public is the atmospheric release of radionuclides from the facility during routine activities. Potential lost-time injuries to workers are primarily due to accident cleanup activities, although about one-third of the injuries are estimated to result from transportation tasks. Essentially no fatalities are predicted to occur during accident cleanup. Occupational radiation doses during accident cleanup result almost entirely from routine onsite activities.

The results of the safety evaluations of post-accident decommissioning activities are summarized in Table 2.10-2 for the reference MOX and U-Fab plants. Details of the safety evaluations of post-accident decommissioning activities are given in Chapter 12.

The principal source of radiation dose to the public is the atmospheric release of radionuclides from decommissioning activities. Occupational radiation doses during decommissioning result primarily from routine onsite activities.

For the reference non-fuel cycle facility laboratory, the occupational radiation dose is estimated to be 42 man-rem. Estimates of occupational radiation dose are made by multiplying worker dose rates shown in Chapter 12 by the estimated man days required to decommission the facility taken from Chapter 10.

TABLE 2.10-1. Summary of Safety Analysis for Accident Cleanup at the Reference MOX and U-Fab Plants

Type of Safety Concern	Source of Safety Concern	Units	MOX Plant	U-Fab Plant
<u>Public Safety^(a)</u>				
Radiation Dose	Accident Cleanup Activities ^(a)	man-rem	2.3	0.06
	Transportation ^(c)	man-rem	1.9	0.53
<u>Occupational Safety</u>				
Serious Lost-Injuries	Accident Cleanup Activities	total no.	1.3	0.46
	Transportation	total no.	0.59	0.03
Fatalities	Accident Cleanup Activities	total no.	0.0083	0.003
	Transportation	total no.	0.035	0.002
Radiation Dose	Accident Cleanup Activities	man-rem	33.0	0.81
	Transportation	man-rem	0.4	--

(a) Radiation doses from atmospheric releases during normal cleanup activities. Doses resulting from industrial accidents are not included.

(b) 50-yr committed dose equivalent to the bone, for the total population within 80 km of the site.

(c) 50-yr committed dose equivalent to the total body, for the population along the transport route.

The estimated occupational exposure is based on the assumption that decommissioning workers do not use protective respiratory equipment. The dose could be reduced by one or two orders of magnitude if respiratory equipment is used. For the ore processing site, occupational dose estimates are made by multiplying the man days required to decommission a site by the average radiation dose rate. For the reference site the total occupational dose is 0.08 man-rem.

2.11 CONCLUSIONS AND RECOMMENDATIONS

Cleanup and decommissioning of a non-reactor nuclear facility that has been involved in a serious accident is technically feasible. Many of the techniques and procedures used to decontaminate or decommission a facility

TABLE 2.10-2. Summary of Safety Analysis for Decommissioning at the Reference MOX and U-Fab Plants Following the Reference Accident

Type of Safety Concern	Source of Safety Concern	Units	MOX Plant	U-Fab Plant
<u>Public Safety^(a)</u>				
Radiation Dose	Accident Cleanup Activities ^(a)	man-rem	2.2	0.06
	Transportation ^(c)	man-rem	1.5	0.53
<u>Occupational Safety</u>				
Serious Lost-Time Injuries	Accident Cleanup Activities	total no.	1.0	0.42
	Transportation	total no.	0.57	0.03
Fatalities	Accident Cleanup Activities	total no.	0.007	0.003
	Transportation	total no.	0.034	0.002
Radiation Dose	Accident Cleanup Activities	man-rem	70.2	15.7
	Transportation	man-rem	6.4	2.6

(a) Radiation doses from postulated industrial accidents are not included.

(b) 50-yr committed dose equivalent to the bone, for the total population within 80 km of the site.

(c) 50-yr committed dose equivalent to the total body, for the population along the transport route.

(d) Includes only preparations for safe storage.

(e) neg. = negligible. Impacts of continuing care expected to be negligible compared to those of decommissioning activities.

following an accident are similar to those used for decontamination or decommissioning following normal shutdown and may be used with proper consideration of problems involved in working in areas with higher radiation levels.

Existing regulations can be used to cover post-accident cleanup and decommissioning. However, some modifications and/or additions that deal specifically with waste management requirements should be developed. Radioactive wastes from accident cleanup and decommissioning may range from low-specific-activity trash and rubbish to high-specific-activity wastes, including evaporator bottoms. Some of these wastes may not be suitable for disposal by shallow-land burial. Regulatory attention should be given to defining waste

disposal criteria that will minimize the impacts of management of high-specific-activity wastes on the costs and occupational exposures for accident cleanup and decommissioning.

This study shows that the costs of accident cleanup and decommissioning of a nuclear facility following a serious accident can be greater than the costs of decommissioning following normal shutdown. In addition, uncertainties in the plant conditions following an accident, need for construction of special facilities and equipment, regulatory requirements, problems in obtaining adequate financing, and the delays resulting from all of these factors can significantly increase accident cleanup costs to amounts greater than the engineering costs estimated in this study. Property damage insurance appears to be a suitable mechanism for ensuring the availability of funds for post-accident cleanup and the increased costs of decommissioning above those estimated for normal decommissioning, and efforts should continue to be made to assure that an adequate level of property damage insurance coverage is carried by operators of non-reactor nuclear facilities.

Accidents can result in high levels of radioactive contamination on building surfaces and equipment and in high radiation exposure rates to personnel involved in accident cleanup operations. In accidents that have occurred to date, procedures have been devised to remove the radioactive contamination with only modest radiation doses to individual workers. These procedures have included the use of shielding and of remotely operated cleaning equipment and the careful planning and rehearsal of cleanup operations to limit the times that individual workers spend in high-radiation areas.

The public safety impacts of post-accident cleanup and decommissioning are estimated to be only slightly greater than the corresponding impacts from normal-shutdown decommissioning. The primary contribution to public dose is the airborne release of particulate radioactivity during accident cleanup. In all cases, radiation doses to the public from routine accident cleanup and decommissioning operations at the reference facilities are estimated to be below permissible radiation dose levels in unrestricted areas.

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3.0 REVIEW OF ACCIDENTS AT NUCLEAR FUEL CYCLE AND NON-FUEL CYCLE FACILITIES

A review of literature available on accidents that have occurred at nuclear fuel cycle and non-fuel cycle facilities is presented in this chapter. This information provides data on the different types of accidents that have occurred at these facilities and also provides a review of available post-accident decontamination or decommissioning activities that have taken place as a result of accident recovery efforts.

The nuclear industry in the United States has an excellent safety record. The accidents that have occurred have not resulted in significant release of radioactive contamination to inhabited areas. Most of the accidents in fuel cycle and non-fuel cycle facilities have occurred in research facilities, which have a higher incidence of accidents than commercial plants. In several instances, accidents have resulted in significant levels of radioactive contamination on building surfaces and equipment. In all cases where contamination has occurred, methods and procedures have been devised to remove the contamination with only relatively small total radiation doses to decontamination workers.

Many of the techniques and procedures employed to decontaminate or decommission a facility following an accident are similar to those used for decontamination or decommissioning following normal shutdown. The removal of radioactive contamination from materials and equipment is a familiar and routine operation in the nuclear field. Decontamination operations of various types have been conducted since the 1940s and the basic technologies of decontamination are well established.⁽¹⁾ The decontamination and decommissioning of fuel cycle and non-fuel cycle facilities following shutdown after normal operations have been described in previous reports in this series.⁽²⁻⁴⁾

Post-accident decontamination requirements will vary with the type of facility and with the nature and severity of the accident. Typical post-accident decontamination activities include the following:

- entry into highly contaminated areas
- removal and processing of any contaminated liquids
- removal of contamination from building and equipment surfaces
- removal of fuel materials and source materials from the plant
- cleanup following radioactive material releases.

In general, the operations required to complete post-accident decontamination differ from those of normal-shutdown decontamination only in magnitude (i.e., level of difficulty of performing the operation) rather than in type of operation.

Details on most of the accidents discussed in this chapter are limited due to the nature of the available information; therefore, to provide as much data as possible, a number of accidents are presented even though the level of detail available varies considerably for the different accident situations. No information is available in the literature on accidents at laboratories that process radioactive material, including radiopharmaceutical and sealed source manufacturers. Post-accident decommissioning experience is presented for each type of facility following the discussion of types of accidents that have occurred.

3.1 PLUTONIUM FUEL FABRICATION

Plutonium fuel fabrication experience is very limited and no plutonium fuel plants are presently operating. However, a number of facilities that process plutonium have been operated for different periods of time and some accidents and cleanup data relating to processing of plutonium have been reported.

3.1.1 Plutonium Facility Accidents

Experimental plutonium-bearing fuel elements have been manufactured at Pacific Northwest Laboratory and at Argonne National Laboratory for years. Plutonium has been used for weapons work at Pacific Northwest Laboratory, Rocky Flats, Mound Laboratory, and Los Alamos. Similarly, there has been a significant amount of experimental chemistry conducted in glove boxes at different laboratories. While this work involved equipment somewhat different from that used in fuel fabrication, the handling of hazardous radioisotopes in various chemical regimes contributes valuable data for safety analyses. Although information available from these operations is probably directly applicable, it should be noted that in many cases this information was compiled in experimental rather than in industrial facilities. The nature of this experimental work leads to a comparatively high accident rate.

Explosion and Fire Accidents Involving Plutonium and Its Compounds

Various incidents ranging from spread of contamination to major fires have occurred involving plutonium and its compounds.⁽⁵⁾ In no case have hazardous quantities of plutonium been released to the environment. Three of the incidents were serious and involved different forms of plutonium. They are discussed below.

In November 1959, an estimated 500 mg of plutonium was blown through the open door and operating holes of a cell during decontamination of an evaporator. The explosion was attributed to an inadvertent use of a nitrate organic cleaning agent. Although nearby buildings, vehicles, the roadway and the

ground were contaminated, air samplers in the area did not indicate air concentrations above acceptable limits. Thus, although air concentrations near the contamination may have exceeded limits for a short time, they did not exceed established limits at the site boundary even under these rigorous conditions.

Detonation of nitrated exchange resin initiated a fire that destroyed a plutonium purification facility in Richland, Washington, in November 1963. Several kilograms of plutonium as nitrate were involved. The integrity of the vessel and glove box was destroyed and material could escape through a partially open door. Although alpha contamination was widespread throughout the facility, air samplers at the site boundaries less than one-quarter mile away indicated no air concentrations exceeding established limits. The alpha contamination in this instance appeared to be associated with soot, and the great number of soot particles generated may have effectively removed a large part of any plutonium that was airborne.

The most serious and significant incident involving plutonium was a fire in a major plutonium fabrication facility at Rocky Flats, Colorado, in May 1969. Products of a fire in one area clogged the exhaust filters of one of three exhaust systems. Flammable vapors passed into other areas. Ultimately, a significant portion of the facility was involved. The supply fans operated during the initial phase of the fire and loss in negative pressure allowed back-diffusion into office areas. Hundreds of kilograms of plutonium as metal and compounds were involved in the accident. Only 200 μCi of airborne material (0.003 g) was released through a damaged exhaust system.⁽⁶⁾ A maximum of 0.5% of the plutonium may have been airborne within the facility.⁽⁵⁾ This value was derived by making the highly conservative assumption that all contamination measured on the ceiling, walls, and floor of the facility and all surfaces outside the enclosure was due to airborne material. The estimate does not include the negligible amounts of plutonium found in the water collected from extinguishment nor the unknown quantities found in the exhaust system. The vast majority of the plutonium used to obtain this estimate was measured as floor contamination in the immediate area of the fire and is probably debris that fell or was washed from the enclosure during extinguishment.

Resin Columns. Four incidents of radioactive releases into nuclear facilities resulting from thermochemical instabilities in ion-exchange processing have been reported.⁽⁷⁾ Plutonium solutions were involved in three and a uranium solution in the other.⁽⁸⁾

Thermal transients, if not properly controlled, can arise from radiolytic heating, from excessively applied heat, from oxidation due to nitration under uncontrolled conditions, or from heating by strong oxidants. During this type of excursion, the resin column heats, becomes pressurized, and ruptures to discharge resin and solution. This type of incident has been successfully modeled

and the operating conditions required to avoid it are well known.⁽⁹⁾ Conscientiously applied administrative controls, reliable monitoring equipment, and adequate safety features would presumably successfully control the process. But there would still exist some finite probability of a simultaneous breakdown of the administrative controls (operator inattention, etc.), monitoring equipment failure (thermocouples and amplifiers) and inoperative safety devices (ruptured disks or check valves).

Prior work,⁽¹⁰⁾ assuming a resin column inventory of about 1400 grams and a glove box rupture, postulated that the resulting plutonium release through the filters to the stack would be about 0.0003 g (30 μ g).

Glove Boxes. Since 1967, four glove box explosions or fires have been reported in which significant amounts of materials were released.⁽⁷⁾ One involved a hood, another an explosion in a vacuum dry box, a third a glove box explosion and fire involving ²³⁸Pu, and the fourth was the Rocky Flats plutonium facility fire that involved plutonium metal and is not representative of a process in a fuel fabrication plant.

The safety problem from a glove box explosion and fire would be explosion of plutonium compounds into the work area and possible dispersal of this plutonium through the exhaust system to the environs.

Several passive and active safety actions are used in a plant to prevent and contain glove box fires. Normal operating procedures specify that:

- flammable solvents having a flash point below 140°F not be permitted inside the boxes.
- inert atmospheres be used in boxes where heat is present
- trash (such as wipes) be kept to a minimum inside boxes and stored in covered metal containers
- PuO₂ and UO₂ not actually in process inside the glove boxes be kept inside metal containers
- dry operations involve PuO₂ and UO₂, which are not flammable materials.

Well-designed glove boxes contain heat detectors and fire extinguishers mounted inside the boxes and have devices for automatically initiated fire suppression. Filters are preceded by fiberglass prefilters, and the final filter bank is protected from hot or burning debris by a temperature-activated deluge system.

The probability of a glove box explosion and fire is low. The accidents analyzed in this study involving a fire and an explosion in a glove box are extreme examples of this type of accident.

Sintering Furnaces. Published operating failure data were not available for sintering furnaces, although a number of sintering furnaces in fuel fabrication facilities are in use. A major vendor, for example, has 30 units in service at uranium plants and one at a plutonium facility, and four vendors furnish over 95% of the large units commercially in use.⁽⁵⁾

Accidents have occurred at sintering furnaces resulting from detonations of explosive mixtures of oxygen and hydrogen, the reducing gas.⁽⁵⁾ The explosive mixtures usually develop from:

1. Incomplete removal of air (oxygen) from a cold furnace at startup before introducing the forming gas containing hydrogen.
2. In-leakage of air (oxygen) into the furnace during operation with the furnace at temperature.
3. In-leakage of air (oxygen) in the line exhausting the forming gas mixture ($H_2 > 12\%$) without first diluting the H_2 content to less than a non-explosive content (about 5% H_2).

Basic safety features on all sintering furnaces include: 1) an automatic nitrogen flush with hydrogen isolation until the flush is complete (timed out) for furnace startup and for protection when the furnace is at temperature, and 2) automatic transfer from H_2 to N_2 make-up if the hydrogen pressure fails or a power failure develops.

Forming gas mixtures are established in some sintering furnaces entirely by valving and monitoring individual gas flow streams. The probability of operator error resulting in a furnace filling with H_2 is finite. When mixing gases automatically at the furnace with a metering device, orifices have become plugged and the gas mixture can suddenly revert to 100% H_2 . Flow alarms are recommended but apparently seldom used. Their reduction of the risk is dependent upon their reliability.

Criticality Accidents

One of the major safety considerations in a fuel fabrication plant is criticality. The occurrence of a criticality event can have severe in-plant consequences if process shielding and emergency procedures are not adequate. To date five criticality events have occurred in fabrication and scrap recovery facilities in the United States.⁽¹¹⁻¹³⁾ All involved fissile solutions entering "unsafe conditions" and all occurred in operating facilities during wet

chemistry operations. Two of the events involved plutonium and three involved highly enriched uranium. No criticality accident has occurred in the United States during the processing of dry material. Nor has such an accident occurred in the processing of wet or dry material of low enrichment. Locations, dates, and personnel exposures resulting from criticality accidents are summarized in Table 3.1-1.

TABLE 3.1-1. Summary of Criticality Accidents

<u>Location</u>	<u>Date</u>	<u>Fissile Material Involved</u>	<u>Form of Fissile Material</u>	<u>Personnel Exposures, No. of rem/indiv. that is exposed</u>	
Y-12 Plant	6/58	U(93)(a)	Solution	461	298
				428	86
				413	86
				341	29
LASL	12/58	Plutonium	Solution	12,000	
				134	
				53	
Hanford	4/62	Plutonium	Solution	110	
				43	
				19	
UNC	7/64	U(93)	Solution	10,000	
				80	
				80	
ICPP	10/78	U	Solution	None	

(a) (U)93 means uranium whose ^{235}U enrichment is 93 wt%.

3.1.2 Cleanup and Decommissioning Experience Following Accidents

Only one document is available in the literature describing cleanup and decommissioning of applicable plutonium facilities following an accident. This document describes the methods, techniques and equipment employed to clean up, dismantle and decommission a plutonium-contaminated facility that was involved in a fire. This study is discussed briefly below.

The reference is:

M. N. Raile, "P-11 Facility Cleanup - Summary Report," ARH-ST-106, Atlantic Richfield Hanford Company, Richland, Washington, December 1974.

The published abstract of this paper follows.

"This document describes methods, techniques, and equipment employed at Hanford for the cleanup, dismantling, and decommissioning of plutonium-contaminated facilities."

The purpose of the decommissioning work described in this document was to restore the site to a natural state that would eliminate potential environmental hazards to people or animals and allow alternative uses of the land area. The report documents the knowledge gained in plutonium contamination control, dismantling and demolition methods and special work techniques for the facility being decommissioned. Time schedules, work plans and procedures, staff organization, auxiliary equipment needs and descriptions, radiation monitoring systems, necessary building services, transportation, waste handling and packaging, and demolition techniques are discussed. A cost summary is provided. The study states that no major injuries or contamination of workers or the environs resulted from the cleanup and decommissioning of this facility.

3.2 URANIUM FUEL FABRICATION

Information specific to uranium fuel fabrication (U-Fab) facilities is limited because only a relatively small number of commercial plants have been constructed. For this reason, applicable information available on other uranium processing facilities is included in this section. These facilities include high-enriched fuel fabrication plants.

3.2.1 Uranium Facility Accidents

A number of accidents of varying degrees of severity have occurred in fuel fabrication plants. Typically, occurrences such as loss of electrical power, pump valve piping leaks, inadequate ventilation of enclosures and spills of UO_2 powder have occurred with minor consequences.⁽¹⁴⁾ These incidents and others (such as filter failures, minor fires, inplant UF_6 releases and inadvertent releases of liquid wastes) have occurred in different plants.⁽¹⁴⁾ These types of accidents have resulted in increased exposure damage to equipment and loss of operating time but have had negligible impact on the environment.

Incidents of the type noted above are a matter of record. Plugged exhaust filters, vent lines, exhaust ducts, or exhaust scrubbers have caused inadequate air flow from process equipment or enclosures.⁽¹⁵⁾ The lack of proper air flow

has resulted in excessive airborne radioactive concentrations in working areas, but has not involved abnormal radioactive releases outside the plant. Similar inplant airborne releases have occurred from gasket failures,⁽¹⁶⁾ improper operating conditions,⁽¹⁷⁾ and improper operating procedures resulting in major spills outside of ventilated enclosures.⁽¹⁸⁾ Higher-than-normal occupational exposures have resulted from operators attempting to repair malfunctioning equipment without supervision by radiation safety personnel.⁽¹⁹⁾

An explosion occurred in an ion exchange resin column in 1967, but did not result in personnel overexposure or release of airborne radioactivity from the plant in excess of 10 CFR 20 limits.⁽²⁰⁾ The explosion was caused by improper blending of concentrated nitric acid entering the column. Improvements in operating parameters and protective systems have significantly decreased the probability of a similar occurrence.

The NRC has received numerous verbal reports of minor UF₆ leaks resulting from such things as plugged transfer lines, improper line connections and fittings, and defective valves. None of these leaks has resulted in significant radioactive releases outside the plant.⁽¹⁴⁾ However, one significant UF₆ release within a plant has been reported.⁽²¹⁾ A valve on a newly installed unit was advertently left open, which resulted in a release of approximately 100 pounds of UF₆ from the process equipment over a 15- to 20-minute period. A cloud of uranyl fluoride was released from the building. The only adverse effect was minor skin burns to an employee who re-entered the vaporization area to determine the source of the leak. No property damage resulted from the incident, and subsequent environmental sampling and testing showed that the release had no discernible effect on the environs.

3.2.2 Cleanup and Decommissioning Experience Following Accidents

The only experience in the open literature with decommissioning of low-enrichment plants following accidents has been the General Electric Co.'s U-Fab plant in San Jose, California. The decommissioning effort was more extensive than should have normally been necessary because, on one occasion, an accident occurred that released a large amount of UF₆ inside the plant.⁽³⁾ This accident contaminated not only all of the building and fixture surfaces in the production areas but also the otherwise clean areas. At shutdown, the area was cleaned to administrative control levels not exceeding 1000 dpm 100 cm² for alpha radiation. Decommissioning was accomplished by dismantling and removing all of the equipment and ventilation system and cleaning the building. Pipes and lighting fixtures were vacuumed or hosed down with water; fluorescent tubes were replaced; ceilings, walls, pipes, and lighting fixtures were damp-wiped; baseboard moldings and tile floors were removed; and concrete floors were vacuumed and mopped. Pump basins that had been formed by constructing concrete berms were cleaned by removing the berms and wet-grinding any hot spots.

3.3 NON-FUEL CYCLE FACILITY EXPERIENCE

A number of non-fuel cycle nuclear facilities that process radioactive material, manufacture radioactive sources, or use radioisotopes for medical or research applications have been decommissioned during the past 30 years.⁽²²⁾ Facilities that have been decommissioned include government, commercial, and institutional installations.

Decommissioned facilities have ranged from small rooms to large buildings and from laboratories in which millicurie amounts of radioactivity were handled within the confines of hoods and glove boxes to major installations in which hot cells were used to process kilocurie amounts of radioactivity. At many large manufacturing and institutional facilities, radioisotope laboratories are frequently decontaminated and remodeled to meet the changing needs of owners and operators of these facilities.

Abstracts of reports on the decontamination of equipment and the decommissioning of laboratories and buildings are found in several bibliographies.⁽²³⁻²⁵⁾ A review of decommissioning activities at government installations is found in the report of a conference on decontamination and decommissioning of Energy Research and Development Administration (ERDA) facilities.⁽²⁶⁾ Several facilities have undergone accidents that have required cleanup or decommissioning.⁽²²⁾

A report by Loysen⁽²⁷⁾ summarizes the economics of building decontamination for sites that were decontaminated following accidents during the early 1960s. Loysen's report is significant because it illustrates the importance of factors such as the age of the facility, materials of construction, residual contamination limits, and the effects of preplanning on the costs of decommissioning.

The sites examined in Loysen's report consisted of a plutonium research facility and a truck terminal that were both accidentally contaminated with plutonium. The truck terminal was decontaminated to permit unrestricted commercial occupancy, and the plutonium research facility was cleaned for reoccupancy.

Data on the decontamination of these two facilities are summarized in Table 3.3-1. (The data are taken from Reference 27.) In some instances, decontamination costs include significant waste disposal costs. Wastes disposed of include ductwork, drains and piping, and some contaminated building materials (floor tiles, sections of wood floor, etc.). Reference 27 does not indicate that any facility components (such as fume hoods, glove boxes, or laboratory work benches) were decontaminated.

Several factors contribute to the wide difference in decontamination costs for the two facilities shown in Table 3.3-1. One important cost factor is the

TABLE 3-3-1. Data on the Decontamination of Non-Fuel Cycle Facilities That were Decommissioned Following Accidents in the Early 1960s

Original Use	Type of Construction	Area (ft ²)	Use Following Decontamination	Decontamination Procedures	Average Alpha Contamination (d/m per 100 cm ²)		Labor Requirements for Decontamination		Total Cost (\$)	Cost per ft. ² (\$)
					Before	After	Man-days	Dollars		
Plutonium research facility	One-level cinderblock; concrete floor covered with epoxy	8 000	Reuse for nuclear work	Wet mopping and wiping	4 000 total 200 removable	<20 removable	8	480	860	0.11
Truck terminal accidentally contaminated with plutonium	Concrete shipping platform and tiled office area	8 300	Unrestricted use	Mopping and steam cleaning	5 500 total	<500 total <0.3 removable	265	11 200	21 070	2.54

(a) Source: Reference 27.

type of construction. The easily cleanable surfaces of the plutonium laboratory lent these surfaces to rapid decontamination. The high cost of cleaning the floor of the truck terminal was due to several factors. First, while the office areas were easy to clean, the shipping platform did not respond to the usual decontamination procedures, and steam cleaning was necessary to remove the heavy layer of soil. Second, decontamination was hampered by extremely cold weather, which necessitated the use of aircraft heaters to prevent the condensed steam from freezing and to aid in drying the floors for radiation monitoring. Finally, the emergency nature of the operations contributed to the high decontamination cost of the truck terminal. Had more time been available to plan the decontamination, the false starts and wasted motions that occurred could have been precluded.

3.4 ORE PROCESSING TAILINGS ACCIDENTS

Uranium mill tailings slurry releases have occurred in the past, and the consequences associated with these events have been documented to varying levels of detail in reports to the NRC and to Agreement States. For this study it is assumed that ore processing tailings accidents would be similar to accidents for mill tailing sites. Table 3.4-1, from Reference 14, contains a summary of uranium mill tailings accidents from 1959 to 1971.

Table 3.4-2, from Reference 28, contains a summary of recorded accidental tailings slurry releases for the period 1959 through 1979. From these historical data, the average releases from tailings embankment failure or flooding were approximately 5.5×10^7 l (1.4 x 10⁷ gallons) of liquids and 1.4×10^7 kg (3.2 x 10⁷ lb) of solids. Six out of ten of the releases from embankment failure or flooding reached the watercourse. Mills having dikes similar in construction to those that failed were required to strengthen the dikes, and the design of the embankment retention systems for new mills is expected to conform to Regulatory Guide 3.11.⁽²⁹⁾

Most failures in the tailing distribution piping would result in release of the slurry to the tailings pond and not to the environment. However, if the failure were to occur in the length of piping between the mill and the tailings area, the slurry could conceivably reach the watercourse. Based on the historical data given in Table 3.4-2, the average releases to the watercourse from the piping failure were approximately 3.5×10^6 l (9.1 x 10⁵ gallons) of liquid and 8.2×10^6 kg (1.8 x 10⁷ lb) of solids. The values, in Table 3.4-2 are probably conservatively high for comparison to ore processing sites.

The AEC received a report of a leak in the retention dam of a liquid waste lagoon, resulting in the loss of an estimated 1.4 million gallons of tailings slurry.⁽¹⁴⁾ All liquids flowed into a lake on the licensee's property, the discharge from which is controlled by the licensee. The uranium concentration

TABLE 3.4-1. Uranium Mill Tailings Releases(a)

Date	Mill and Location	Incident
8/19/59	Union Carbide Green River, Utah (Concentrator) Tailings Dike Failure	Tailings dam washed out; calculated 15,000 T sands lost to Browns Wash and Green River due to flash flood; No increase in dissolved Ra was noted in river.
3/22/60	Kerr-McGee Shiprock, New Mexico Raffinate Pond Dike Failure	240,000 gal. of raffinate released into San Juan River; $\sim 50 \times 10^{-8}$ $\mu\text{Ci}/\text{ml}$ radium-226; river samples collected several days after release showed no increase in radium-226 background; river at Medicine Hat (100 mi downstream of plant) showed 0.36×10^{-9} $\mu\text{Ci}/\text{ml}$ radium-226 on 8/30/60.
12/6/61	Union Carbide Maybell, Colorado Tailings Dike Failure	Calculated 500 T solids released from tailings area; 200 T reached unrestricted area; no liquid reached any flowing stream. "The presence of these tailings (off-site) does not constitute a hazard, as there are no persons living in the area, nor is there any drinking water taken from surface or ground water in the near vicinity."
6/11/62	Mines Development, Inc. Edgemont, South Dakota Tailings Dike Failure	200 T solids washed into Cottonwood Creek and some carried 25 mi into Angostura Reservoir.
8/17/62	Atlas-Zinc Minerals Mexican Hat, Utah Tailings Line Failure	Estimate 280 T solids + 240 T liquids released from broken tailings discharge line into draw 1.5 mi from San Juan River. Calculated concentration of river water would have been below 10 CFR 20 MPC.
6/16/63	Utah Construction Riverton, Wyoming Tailings Dike Procautionary Release	Material released by 2-ft drainage cut made to prevent cresting due to heavy rains; material released below 10 CFR 20 values.
11/17/66	VCA Shiprock, New Mexico Raffinate Line Failure	Estimated 16,000 gal of liquid lost because of break in raffinate line; material spread over 1/4 acre; break occurred 1 mile from San Juan River with some small amount reaching river.

TABLE 3.4-1. (contd)

Date	Mill and Location	Incident
2/6/67	Atlas Corporation Moab, Utah Auxiliary Decant Line Failure	Overflow from main tailings pond overflowed aux. decant system; 440,000 gal lost; average radium-226 concentration was 5.5×10^{-8} $\mu\text{Ci}/\text{ml}$.
7/2/67	Climax Uranium Grand Junction, Colorado Tailings Dike Failure	Dike failure of unapproved retention system released; calculated 1-10 acre-ft of waste liquid into Colorado River; No indication that Ra conc. in river exceeded 10 CFR 20 limits.
11/23/68	Atlas Corporation Moab, Utah Tailings Distribution Pipe Break	33,000 gal. of tailings slurry lost; effluent flowed down drywash and then 1/2 mile to Colorado River; river flow sufficient to give 10,000: 1 dilution; most solids settled out in drywash; measurements of river downstream of plant immediately after release and at 4-hr intervals in 24 hr following release showed uranium, radium-226, thorium-230 below 10 CFR 20 limits.
2/16/71	Petrotomics Shirley Basin, Wyoming Secondary Tailings Dike Failure	2000 gal of liquid lost to unrestricted area; break in dike of effluent sump; spill frozen in place.
3/23/71	Western Nuclear Jeffrey City, Wyoming Tailings Line - Dike Failure	Break in sand tails slurry line caused a dike failure allowing sand tails to flow for 2 hr into natural basin adjacent to tailings site on licensee's property; fence extended to make this area restricted.

(a) Source: Reference 14.

TABLE 3.4-2. Summary of Accidental Tailings Slurry Releases, 1959-1979(a)

Cause	Solids Released, kg	Released, Liters	Reached Watercourse
Flash Flood	14×10^6	$1.2 \times 10^7(b)$	Yes
Dam Failure	$9 \times 10^5(b)$	9.1×10^5	Yes
Dam Failure	5×10^5	$4 \times 10^5(b)$	No
Dam Failure	2×10^5	$2 \times 10^5(b)$	Yes
Pipeline Failure	3×10^5	2×10^5	Yes
Flooding	$1 \times 10^8(b)$	8.7×10^7	Yes
Pipeline Failure	$6.4 \times 10^4(b)$	6.1×10^4	Small Amount
Pipeline Failure	$2 \times 10^6(b)$	1.7×10^6	Yes
Dam Failure	$1-14 \times 10^6(b)$	$1-11 \times 10^6$	Yes
Pipeline Failure	$1 \times 10^5(b)$	1.3×10^5	Yes
Dam Failure	$9 \times 10^3(b)$	8×10^3	No
Pipeline Failure	4.5×10^7	$8-30 \times 10^6$	No
Dam Failure	$8.2 \times 10^6(b)$	7.6×10^6	No
Pipeline Failure	1.1×10^3	1.5×10^4	Yes
Dam Failure	1.0×10^6	3.8×10^8	Yes
Pipeline Failure/Dam Failure	No Quantitative Information		

(a) Source: Reference 28.

(b) This value is based on the assumption that equal weights of solids and liquids are released, and that the density of the liquids is approximately 1.6 g/cm^3 (100 lb/ft^3)

of the water released was less than 10% of the 10 CFR Part 20 limits for release to an unrestricted area. The greatest potential hazard from an uncontrolled release of waste water from a lagoon is the ammonia content, and the hazard is a function of the flow rate of the receiving stream. The normal release of 450,000 gallons per day with 420 mg/l of nitrogen in the form of ammonia requires dilution by approximately $600 \text{ ft}^3/\text{sec}$ of ammonia-free water to meet recommended levels for drinking water supplies. An accidental release would contain approximately the same concentration of ammonia, but would require greater dilution if the volume released exceeded normal operating releases.

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4.0 STUDY APPROACH AND KEY STUDY BASES

This report describes the post-accident decommissioning of fuel cycle and non-fuel cycle facilities including post-accident cleanup. The overall approach used in the study is discussed in this chapter. Accident cleanup and decommissioning activities are outlined in Section 4.1. The study bases (i.e., ground rules) established to guide the study are given in Section 4.2. The technical approach used in the study is described in Section 4.3.

4.1 ACCIDENT CLEANUP AND DECOMMISSIONING ACTIVITIES

Accidents of the types postulated in this report could result in radioactive contamination and potential physical damage to the facility structures. Details of the accidents considered in this study are discussed in Chapter 8.

The first activities following an accident are designed to bring the accident under control and to stabilize the plant. Once the situation is stabilized, accident cleanup can begin. Accident cleanup is followed by either decommissioning or refurbishment of the facility. The sequence of accident cleanup and subsequent decommissioning or refurbishment activities for a nuclear facility that has been involved in a serious accident is shown in Figure 4.1-1. As discussed in Section 4.1.2, the accident cleanup activities are necessary and would be similar whether the facility is ultimately refurbished or decommissioned.

The study does not address the details of the activities that would be needed to stabilize the facility or to refurbish and restart the plant. Accident cleanup and decommissioning activities, which are analyzed in detail in the various chapters of this report and which are described briefly in the following subsections, include: 1) preparations for accident cleanup, 2) accident cleanup of the process building and other buildings as required, and 3) decommissioning of the facility by the DECON, SAFSTOR, or ENTOMB alternative.

4.1.1 Preparations for Accident Cleanup

A period of planning and preparation precedes the actual performance of cleanup operations within the accident-damaged facility. Several tasks must be performed during this period to ensure the proper performance and the success of the operational phase of accident cleanup. These tasks include:

- contaminated area entry and data acquisition
- preparation of documentation for regulatory agencies
- design, fabrication, and installation of special equipment
- development of detailed work plans and procedures
- selection and training of accident cleanup staff.

FIGURE 4.1-1. Postulated Sequence of Accident Cleanup and Decommissioning Activities for a Facility Involved in a Serious Accident

The tasks postulated for the planning and preparations phase are estimated to require from six months to one year for completion, depending on the severity of the accident, the time needed to design, fabricate, install, and test special facilities and equipment, and the time required to secure the necessary regulatory approvals to proceed with the cleanup operations.

4.1.2 Accident Cleanup Activities

The operational phase of the cleanup and decommissioning of an accident-damaged nuclear plant begins with an accident cleanup campaign with the following principal goals:

1. to reduce the initial levels of radioactive contamination present on building surfaces and equipment, thereby reducing the radiation dose received by workers engaged in cleanup and decommissioning operations
2. to collect and package for disposal the readily dispersible radioactivity present in the plant.

To achieve these goals, the accident cleanup campaign is postulated to include the following tasks (not necessarily in this order):

- initial decontamination of building surfaces and decontamination or disposal of some equipment
- cleanup of the plant systems as required
- solidification and packaging of wastes from accident cleanup operations.

Details of accident cleanup tasks are presented in Appendices D, E, F, and G.

Some of the decontamination procedures and the chemical solutions used for cleanup might be different because of the effort to minimize the effect of cleanup operations on a plant that was being returned to operation. However, whether accident cleanup is to be followed by restart of the plant or by decommissioning, the use of highly corrosive chemicals would probably not be desirable because such chemicals are too destructive. Destructive chemical solutions could compromise the integrity of systems that need to remain intact during operations or decommissioning activities. Highly corrosive chemicals can also have an adverse effect on personnel using them.

4.1.3 Decommissioning the Facility

The alternatives for completing the decommissioning of a nuclear facility that has been involved in a serious accident are DECON (immediate decontamination to unrestricted release), SAFSTOR (safe storage with deferred decontamination to unrestricted release), and ENTOMB (entombment of radioactive materials with decay to unrestricted release).^(1,2) DECON permits termination of the facility operating license, while SAFSTOR and ENTOMB require continuance of an amended version of the license for extended periods of time. The amended nuclear license allows the licensee to possess but not to operate the facility. These alternatives, applied to post-accident decommissioning, are discussed in the following paragraphs. Since the SAFSTOR and ENTOMB alternatives are not likely to be used for non-reactor facilities, only the DECON alternative is discussed in detail.

DECON

DECON is a pseudoacronym defined by the NRC as the immediate removal of all radioactive material to permit license termination and release of the property for unrestricted use. To achieve a condition of unrestricted use, the residual radioactivity levels in the facility must be reduced to values that do not exceed limiting values set by regulatory guidance.^(a) If DECON is chosen as the alternative for decommissioning, the following tasks must be performed after accident cleanup:

- remove contaminated materials from plant buildings
- decontaminate buildings to release levels
- package and ship radioactive materials
- survey site and decontaminate as necessary
- demolish buildings and complete site restoration (optional).

DECON meets the requirements for termination of the facility license and renders the facility and site available for unrestricted use within a finite time period. Other advantages of DECON include the availability of personnel who are knowledgeable about the facility to form a decommissioning work force and the elimination of the need for long-term security, surveillance, and maintenance that would be required for the other decommissioning alternatives. Disadvantages of DECON are the larger initial commitments of personnel, radiation exposure, waste-disposal-site space, and money than are required for the other alternatives.

For DECON, nonradioactive equipment and structures need not be torn down or removed as part of the decontamination procedure. In addition, once the radioactive structures are decontaminated to levels permitting unrestricted use, they may be put to some other use or demolished, at the owner's option. When a facility and site have been decontaminated to levels that permit unrestricted use of the property, the nuclear license is terminated and the NRC's responsibilities at the station are also terminated. There are no provisions in any NRC regulations that imply that the decontaminated structures must be demolished and the site restored to pre-facility conditions. Therefore, demolition and site restoration are not required to complete the decommissioning and are done at the owner's option. Costs of demolition and site restoration are not included in this study.

(a) Current regulatory guidance on acceptable surface contamination levels for unrestricted release of a nuclear reactor is contained in Regulatory Guide 1.86. The NRC is considering defining acceptable conditions for the unrestricted release of a decommissioned facility in terms of the potential radiation dose to an individual who uses the facility.⁽²⁾

SAFSTOR

SAFSTOR means to fix and maintain property so that risk to public safety is acceptable for a period of storage followed by decontamination and/or decay of residual radioactivity to an unrestricted level. SAFSTOR consists of: 1) a period of facility and site preparation (preparations for safe storage) that includes concentration and immobilization of dispersible radioactive materials, 2) an interim period of continuing care (safe storage) that encompasses security, surveillance, and maintenance, and 3) the deferred removal of any remaining contamination to permit release of the facility for unrestricted use (deferred decontamination). An amended version of the nuclear license that does not permit operation of the facility would remain in force throughout the safe storage period, since materials having radioactivity levels above unrestricted release levels remain onsite. The duration of the safe storage period is undefined; however, periods of up to 100 years are consistent with recommended EPA policy on institutional control reliance for radioactivity containment.⁽²⁾ The use of SAFSTOR for fuel cycle and non-fuel cycle facilities is not likely because the safe storage period would be very long due to long half-life radionuclides that would not decay to release levels in 100 years.

ENTOMB

ENTOMB means to encase and maintain property in a strong and structurally long-lived material (e.g., concrete) to assure retention and isolation from the environment until the contained radioactivity decays to an unrestricted level. ENTOMB is intended for use where the residual radioactivity will decay to levels permitting unrestricted release of the facility within reasonable time periods (i.e., within the time period of continued structural integrity of the entombing structure). Recommended EPA policy on institutional control reliance for radioactivity containment suggests that the entombing period not exceed approximately 100 years.⁽²⁾ The use of ENTOMB following a serious accident is unlikely because of the presence of significant quantities of long-lived (i.e., 30-year half-life) radionuclides that will not decay to release levels in 100 years.

ENTOMB is similar to SAFSTOR in that it consists of a period of facility and site preparations, followed by a period of safe storage that includes security, surveillance, and maintenance activities. Following entombment, these activities are minimal unless the irradiated fuel is stored onsite.

4.2 KEY STUDY BASES

This study is intended to provide post-accident cleanup and decommissioning information useful to regulators, designers, and operators of nuclear fuel cycle and non-fuel cycle facilities. A number of key study bases (i.e., study

ground rules) are established to guide the emphasis of the study and to ensure that the primary objective of the study is achieved.

The requirements and costs of accident cleanup and decommissioning following an accident depend on the nature of the accident and on the radioactive contamination and physical damage that result, as well as on such factors as facility location, specific facility design, and operating practices during the facility lifetime. In addition, requirements and costs depend upon the specific methods and techniques used in the accident cleanup and decommissioning, and upon the capability to dispose of wastes. There can also be an impact from specific rules and orders imposed on the facility after the accident. The bases and assumptions used in this study have a major impact on the estimates of time and manpower requirements, safety, and costs. These bases and assumptions must therefore be carefully examined before the results can be applied to a different facility.

The key bases are listed below.

1. The intent of the study is to yield realistic and up-to-date results. This primary basis is a requisite to meeting the objective of the study and provides the foundation for most of the other bases.
2. Accident scenarios are chosen that are illustrative of a range of accident cleanup and decommissioning requirements and for which the costs of cleanup and decommissioning are substantially greater than those for decommissioning following normal shutdown. An attempt is made to choose credible accident scenarios related to NRC licensing criteria for design basis accidents. Scenarios are not restricted to accidents that have occurred.
3. The reference MOX plant for this study is the same plant that was used as the reference plant for a previous study (NUREG/CR-0129)⁽³⁾ of MOX plant decommissioning following normal shutdown. The reference U-Fab plant is the same plant that was used as the reference plant for a previous study (NUREG/CR-1266)⁽⁴⁾ of U-Fab decommissioning following normal shutdown. The reference non-fuel cycle plant and ore processing site are taken from NUREG/CR-1754,⁽⁵⁾ which described normal decommissioning of the reference facilities. The use of these plants as reference facilities should not be construed as implying anything about their reliability and/or safety relative to other facilities in operation or under construction. Their use facilitates comparisons with the earlier, non-accident, decommissioning studies.

4. The reference plants are assumed to have operated for 30 years and to be operating at full capacity at the time of a postulated accident.
5. This study is conducted within the framework of existing regulations and regulatory guidance. No assumptions are made regarding what future regulatory requirements or guidance might be.
6. This study focuses on the technical effort and associated costs of accident cleanup and decommissioning effort. It is recognized, however, that these considerations can have significant impacts on cleanup and decommissioning requirements, delaying the time of completion and increasing the cost of these activities.
7. Current technology and techniques are used in descriptions of accident cleanup and decommissioning procedures. Where developmental techniques are specified in the study, they are in an advanced state of development and believed to be ready for application.
8. Decontamination and decommissioning and radiation protection philosophies and techniques applied in the study conform to the principle of keeping public and occupational radiation doses as low as reasonably achievable (ALARA).
9. Sufficient funding is available to carry out the accident cleanup and decommissioning without significant delays.
10. Release of facilities and/or equipment for unrestricted use is assumed to be predicated on the decontamination of surfaces and equipment to residual levels of radioactivity as specified by present regulatory guidance.^(6,7)
11. It is recognized that work in high radiation areas proceeds much more slowly than work in areas with minimal contamination. Allowances are made in preparing work schedules for the unavoidable inefficiencies associated with this work. However, no allowances are made for unforeseen events that might impede the conduct of the work. A 25% contingency is added to cost totals to account for work delays and unanticipated material and equipment costs.
12. Some of the tasks related to cleanup and decommissioning must be performed in high radiation areas where workers can receive their allowed quarterly or annual radiation dose in relatively short periods of time. In this study, manpower requirements for accident cleanup and decommissioning are adjusted upward beyond the numbers

required for efficient performance of the work to provide a sufficient manpower pool so that no individual worker exceeds 5 rem/year.⁽⁸⁾

13. Costs are in 1981 dollars. To make cost comparisons with previous studies^(3,4,5) of decommissioning following normal shutdown, the cost results from these earlier studies, which are in 1978 dollars, are converted to the 1981 cost base.
14. Based on 10 CFR Part 61 criteria, low-level radioactive waste from accident cleanup and decommissioning operations are assumed to be transported by truck to an authorized shallow-land burial ground for disposal. The criteria of 10 CFR Part 61 may result in certain highly radioactive and/or transuranic wastes from accident cleanup and decommissioning being deemed unsuitable for near-surface disposal. Because criteria for disposal of these wastes are not defined, and because a deep geologic facility to dispose of these wastes does not now exist, the estimated requirements and costs of interim storage of highly radioactive wastes at a federal repository are evaluated in this study.

4.3 TECHNICAL APPROACH

The technical approach used to conduct this study is illustrated in Figure 4.3-1.

FIGURE 4.3-1. Technical Approach of this Study

The first step is to describe the reference facility. The reference MOX plant is the Cimarron, Oklahoma, plant of Kerr-McGee Company. The reference U-Fab plant is the Wilmington, North Carolina, plant of General Electric Company. The reference laboratory for the non-fuel cycle part is based on New England Nuclear Company's labs and the reference processing site is taken from Reference 5. These facilities were used as reference facilities for previous studies^(3,4,5) of decommissioning following normal shutdown. Use of these same plants in this study facilitates comparisons between the requirements and costs of post-accident cleanup and decommissioning and of normal-shutdown decommissioning and is not intended to imply anything about their safety and reliability relative to other plants in operation.

Accident scenarios are chosen that are believed to be credible based on reviews of reactor accident experience, safety analysis reports, and current safety NRC safety evaluation reports and licensing criteria. The accidents analyzed require a significant cleanup and decommissioning effort. Each accident scenario is characterized in terms of radioactive contamination of building surfaces and equipment, and radiation dose rates to decommissioning workers.

This study provides an analysis of the technical requirements for decommissioning accident-damaged nuclear fuel cycle and non-fuel cycle facilities, including planning and preparation, accident cleanup activities, and procedures for decommissioning the facility after accident cleanup is completed. The study does not include considerations of immediate stabilizing measures taken following an accident (such as bringing the accident under control, rescuing casualties, or extinguishing fires). However, the costs and occupational exposures for activities needed to prepare for decontamination of the buildings, entries into the contaminated areas and supporting buildings to make measurements, or repairing essential systems and services are included in this analysis. As discussed previously, the accident cleanup activities would be similar whether the plant is refurbished for restart or decommissioned. Hence, the requirements, costs, and safety analyses given in this report are considered to be a good representation independent of the ultimate use of the plant. The DECON alternative for completing the decommissioning after accident cleanup is the only alternative studied in detail. The study does not include consideration of the activities related to the refurbishment and restart of a plant following the accident cleanup period.

Regulatory guidance is reviewed, summarized, and used as an aid and basis in the study. Guidance is provided for determining allowable contamination levels for unrestricted release of a facility, based on realistic dose assessments to an exposed individual.

Techniques for the decontamination of facilities are reviewed. Work and time schedules are developed to conceptually perform accident cleanup activities and to decommission the reference plants for the DECON decommissioning alternative. Postulated work schedules include allowances for time needed to prepare for entry into contaminated areas and for inefficiencies associated with work in high radiation fields.

Safety assessments are performed to estimate radiological hazards to workers and to the public from accident cleanup and decommissioning operations. These analyses include radiological exposures to workers and the public from normal cleanup and decommissioning operations and from potential accidents. Nonradiological industrial accidents to workers are also estimated. The safety analyses use established data and methodology to estimate the release mechanisms, dispersion, and pathways and exposure modes of the released materials.

Direct costs of accident cleanup and decommissioning are estimated, including labor, materials, equipment, and the packaging, transportation, and disposal of radioactive wastes. Costs are estimated for planning and preparation, for accident cleanup, and for completion of the decommissioning. Cost ranges are defined to estimate the sensitivity of the total cost to variations in key cost elements such as levels of radioactive contamination and waste treatment practices. Alternatives for financing the costs of post-accident cleanup and of decommissioning are examined.

The requirements and costs of post-accident cleanup and of decommissioning are compared with those for normal-shutdown decommissioning as defined in previous studies.^(3,4,5) In addition, because accident cleanup is an activity which takes place at a time when the condition of the plant is uncertain and when social, political, financial, and regulatory constraints can affect the progress and costs of cleanup activities, the study also analyzes the sensitivity of accident cleanup costs to various factors which could impact these costs.

The reference plants are described in Chapter 5. Accident scenarios are described in Chapter 6. The technical requirements, costs, and safety impacts of accident cleanup and decommissioning are summarized in Chapters 8 through 12.

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5.0 CHARACTERISTICS OF REFERENCE FUEL CYCLE AND NON-FUEL CYCLE FACILITIES

This chapter contains brief descriptions of the reference small mixed-oxide (MOX) plant, the reference uranium fuel fabrication (U-Fab) plant, the reference non-fuel cycle plant, and the ore processing facility site. Included is a description of the reference site on which the reference MOX and U-Fab plants are assumed to be located, together with estimates of the radioactivity in the facilities as a result of normal operations. Additional information about the reference site and the reference MOX and U-Fab plants is given in Appendix A of Volume 2. More detailed descriptions than those presented in this chapter for the non-fuel cycle facilities and the ore processing facility site are not available. The reference facilities used in this study are the same as those used in previous decommissioning studies in this series.⁽¹⁻³⁾

The reference accident scenarios for this study, described in Chapter 8, contain information about radioactive contamination, physical damage to structures and equipment, and radiation exposure rates that result from the postulated accidents at the reference facilities.

5.1 DESCRIPTION OF THE REFERENCE SITE

A reference site, described briefly in this section, is used to assess the public safety effects of post-accident cleanup and decommissioning of the reference facilities. The meteorological parameters and population distributions used for this reference site are taken from the ALAP Study⁽⁴⁾ for the river site in the year 2000. The ecological data are taken from environmental information provided for an operating nuclear reactor.⁽⁵⁾ The remainder of the information is obtained from a variety of sources, and is thought to be representative of potential sites for nuclear fuel cycle facilities in the midwestern or southeastern United States. This reference site description is developed for use in a series of studies examining decommissioning of nuclear fuel cycle facilities. The detailed supporting information relating to this abbreviated site description is found in Appendix B.

Individual features for specific sites will likely vary from those of the reference site described in this study. It is believed, however, that use of a reference site rather than a specific site will result in a more meaningful overall analysis of the potential safety impacts associated with the decommissioning of nuclear fuel cycle facilities. Site-specific environmental information will be required for the detailed safety analysis and the environmental report submitted with the request for license modification prior to decommissioning a particular facility.

The reference site occupies 4.7 km^2 in a rectangular shape of 2 km by 2.35 km. A river of moderate size runs through one corner. The site is located in a rural area that has a relatively low population density. Higher population densities are located at distances 16 to 64 km away, and gradually reducing population densities are encountered out to 177 km. The closest moderately large city, population 40,000, is about 32 km away. The closest large city, population 1,800,000, is about 48 km away. The total population in a radius of 80 km is 3.52 million.

The plant facilities are located within a fenced portion of the site. The minimum distance from the point of plant atmospheric releases to the outer boundary of the reference site is 1 km. About 80% of the land surrounding the reference site is used for farming.

The relatively clean river flowing through the site has an average flow rate of $1420 \text{ m}^3/\text{sec}$. The river is used for irrigation, fishing, boating, and other aquatic recreational activities, and is a source of drinking water for larger communities. Large supplies of flowing ground water exist at modest depths around the site. This water is widely used for drinking and irrigation.

Atmospheric dispersion factors used in this study are derived as an average from the meteorological data of 16 nuclear sites. The resulting annual average atmospheric dispersion factor at the closest point on the site boundary (i.e., 1 km) is about $5 \times 10^{-8} \text{ sec/m}^3$.⁽⁵⁾

Prior to the postulated accidents, the site is assumed to be slightly contaminated with radioactive material as a result of deposition from normal operating effluents over the 40-year plant operating life. Site contamination that results from the postulated accidents is discussed in Chapter 6.

5.2 DESCRIPTION OF THE REFERENCE MIXED-OXIDE PLANT

The Cimarron, Oklahoma, plant of Kerr-McGee Company^(6,7) was chosen as the reference facility because it is believed to be representative of small MOX plants in the United States. The Cimarron plant uses a head-end process involving coprecipitation of plutonium and uranium solutions to blend uranium and plutonium. To extend the applicability of the present study to other MOX plants, a conceptual addition has been made to the existing plant to provide an additional head-end process based on dry blending of preformed oxides.

Details of the reference MOX plant, the plant process descriptions and site description details are given in Appendix A, and the bases for residual radioactivity estimates are presented in Reference 1.

5.2.1 Process Description

Overall processing characteristics assumed for the reference MOX plant are presented in Table 5.2-1. A simplified block flow diagram of the process is shown in Figure 5.2-1. Details of the process are given in Section A.2 of Appendix A. All processes associated with fuel pellet fabrication are performed inside glove boxes located in three rooms in the process area of the building (the wet processing room, room 128; the dry blending room, room 155; and the pellet processing room, room 124). Transfer of product materials from one process step to the next is sometimes accomplished automatically and sometimes manually through use of a portable glove box.

Production activities fall into six major categories:

1. mixing and blending of plutonium nitrate and uranyl nitrate hexahydrate (UNH), coprecipitation and calcining to a mixed oxide
2. an alternative head-end process involving the dry mixing and blending of uranium and plutonium oxides (UO_2 and PuO_2)
3. mechanical pressing of mixed oxide into pellets
4. fabrication of pellets into fuel pins
5. inspection of fuel pins and packaging for shipment offsite
6. recovery of plutonium and/or uranium from process materials and liquids.

Coprecipitation Head-End

For the coprecipitation head-end process, plutonium nitrate solution arrives at the plant by truck in 10- $\frac{1}{2}$ L-10 containers. Plutonium is carefully weighed and sampled upon receipt and either pumped directly to weigh tanks or stored in the process building vault room until needed. Uranyl nitrate hexahydrate (UNH) arrives at the plant in 208- $\frac{1}{2}$ (55-gal) steel drums with polyethylene liners. The temperature of the UNH must be carefully controlled to maintain UNH as a fluid and to prevent crystals from forming that would unbalance the process assay. Overhead insulated lines with electrical heating traces transfer the solution from the drums to the weigh tanks or to holding tanks in a wall of the process building vault.

For the coprecipitation process, precisely measured quantities of $\text{Pu}(\text{NO}_3)_4$ and $\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$ are fed to a weigh tank located inside a large glove box in the wet processing room. A circulating pump incorporated in the tank is

TABLE 5.2-1. Characteristics of the Reference MOX Plant

ANNUAL OPERATING CAPACITY

2.3 MT MOX, 22.76 wt% PuO_2 , 77.24 wt% UO_2

ANNUAL INPUT

Coprecipitation Mode (Assumes all material processed by this mode)

472.5 kg of contained plutonium in the form of 250 g/l Pu as $\text{Pu}(\text{NO}_3)_4$ solution in 3-6 molar nitric acid; packaged in 189 type L-10 containers.

1618 kg of contained natural uranium in the form of 2M $\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$ solution; packaged in ten 208-l (55-gal) drums.

Dry Oxide Mode (Assumes all material processed by this mode)

540 kg of PuO_2 ; packaged in 120 type 6M containers.

1820 kg of UO_2 ; packaged in seven 208-l (55-gal) drums.

ANNUAL OUTPUT

12,750 mixed oxide fuel pins per year, containing 144 pellets each fuel pin, 1-1/4 grams per pellet.

SCRAP RATES

10% of U+Pu throughput recycled as clean scrap;
1 to 2% of U+Pu throughput recycled as dirty scrap.

SHIFT STRUCTURE

Around-the-clock operation for 365 days per year.

Basic operating cycle is a 7-day cycle with 19 or 20 processing shifts followed by 1 or 2 cleanout shifts. This is done with four rotating shifts plus one permanent day shift.

Total of 120 people working inside the plant area.

operated to thoroughly blend the batch. The mixture is then stored in one of the 107-l wall storage tanks or transferred to a precipitation column in another glove box.

FIGURE 5.2-1. Simplified Process Flow Diagram for the Reference MOX Plant

In the coprecipitation step, the uranium-plutonium feed and ammonia water are precisely metered into a precipitation column where the plutonium-uranium blend is continuously precipitated as plutonium hydroxide and ammonium diuranate. The precipitate slurry is continuously transferred to aging columns equipped with temperature control. Additional ammonia is added to ensure completion of precipitation. The slurry is filtered; the solid is loaded into metal trays, dried, calcined and reduced to $(U, Pu)O_2$ in a furnace using nitrogen-8% hydrogen by volume as a reducing atmosphere. The finished mixed oxide is hammer-milled to a uniformly sized powder, cross-blended into uniform batches, packaged in safe containers and loaded into a portable glove box. The glove box is stored in the concrete vault and then moved to the pelletizing room when the mixed oxide is needed for making pellets.

Dry Process Head-End

For the alternative dry process head-end, plutonium oxide and uranium oxide pellets (or powder) are presumed to be received at the plant with a quality that permits direct blending for fuel manufacture without the need for wet processing. The PuO_2 , received in type 6M containers, is first weighed and then manually transferred to the PuO_2 feed hopper. The UO_2 , received in 208-l drums, is transferred to a similar feed hopper. For each batch, the two oxides are metered from their respective feed hoppers into separate batch-weighing hoppers, and then released to the blender. After blending, the mixed oxide may be hammer-milled, if necessary, to a uniformly sized powder. The finished mixed oxide is then packaged in safe containers and transferred by portable glove box to the concrete vault or to the pellet processing room.

Pellet Formation and Fuel Rod Assembly

Pellet formation is a two-step process. In the first step, the mixed oxide powder is fed, one container at a time, into a hydraulically driven slugging press. This press is used for "slugging" the oxide, a preliminary forming of pellets to increase the oxide density. Pellets from this initial operation are ground up by a screen granulator. The granulated material is again calcined and then pressed into fuel pellets. These green pellets are sintered to ceramic grade pellets, precisely dimensioned in a centerless grinder, outgassed in a heated vacuum furnace, and transferred to the fuel rod fabrication room.

Finished pellets are incorporated into fuel rods in a glove box operation in the fabrication room. Pellets are inspected, cleaned, and assembled into rows. The pellet rows are loaded into horizontally oriented cladding tubes that have the first end cap welded in place. Assembled fuel pins are evacuated, filled with argon or helium to slightly negative pressure, and the final end cap is welded to the cladding tube.

Rod welds are tested at an inert gas leak test station and by x-ray analysis. The rods are then thoroughly cleaned and tested for bowing on a high-precision granite surface plate. Rods that pass all the required quality assurance tests are loaded into containers for shipment to the customer.

Scrap Recovery

There are three product scrap recovery areas (Rooms 127, B01 and B05). One area recovers product from clean scrap; the other two recover product from dirty scrap. Processes are available to recover plutonium from the following materials:

1. non-contaminated (U, Pu) O_2 or UO_2 - PuO_2 pellets or powder that is rejected for physical reasons
2. chemically contaminated (U, Pu) O_2 reject pellets or powder
3. combustible plutonium-contaminated solids (paper, cloth, etc.)
4. organic "non-combustible" solids (gloves, plastics, etc.)
5. highly refractory solids, of assay up to and including pure PuO_2
6. contaminated aqueous and organic plutonium-bearing solutions.

Unit scrap recovery processes include digestion, leaching, calcination, incineration, precipitation, two-cycle solvent extraction for separation and purification of plutonium and uranium, and ion exchange. Arrangements and layout are sufficiently flexible to permit several different sequences of processing.

Substantial fractions of (U, Pu) O_2 reject pellets, powder, etc., are rejected on physical characteristics alone; they are of adequate chemical purity for immediate reuse. Such material is processed through a clean dissolution circuit and is returned to ADU- $Pu(OH)_4$ precipitation without purification processing.

Chemically contaminated reject pellets or powder are dissolved in HNO_3 or HNO_3 -HF and purified by the use of two tributyl phosphate (TBP) solvent extraction cycles and/or a batch anion exchange cycle. The plutonium product is recycled as $Pu(NO_3)_4$ solution or converted to PuO_2 via precipitation as plutonium oxalate and calcination to PuO_2 . Purified uranium is removed as uranyl nitrate solution which is then recycled to the coprecipitation step.

The purification operations generate low-to-intermediate-level waste streams that can be processed for additional product recovery by back-cycling to the first solvent extraction cycle.

Combustible solids are roasted in an oxidizing atmosphere to remove organic material. The ash from this roasting contains essentially all the plutonium and uranium in the original waste, and is processed through dissolution. Certain poorly combustible organic materials such as glove box gloves, plastic bags, etc., are processed through one or more leaching operations with warm HNO_3 or $\text{HNO}_3\text{-HF}$ and then discarded to offsite burial.

Processing of clean scrap ceramic materials is accomplished in processing modules that carry out the following distinct functions:

1. high-temperature oxidizing roast of high assay (uranium, plutonium) ceramic materials
2. dissolution and precipitation of plutonium and uranium as oxalates
3. calcination of plutonium oxalate to plutonium oxide
4. conversion of the oxide to plutonium nitrate for blending back into the coprecipitation head-end process.

Product recovered from processing ceramic materials is returned to the wet head-end process as the nitrate or to the dry head-end process as the oxide.

Waste and Effluent Processing

All low-level plutonium-contaminated aqueous and organic wastes from the plant are filtered to eliminate large particulate or fibrous material. The filter media with its collected material is calcined to dry waste and discarded to burial. The clear filtrate is evaporated to minimize the liquid volume and mixed with cement in 208-l (55-gal) drums. The resulting solid is shipped offsite for disposal in a federal repository. Dry plutonium waste material is similarly processed for burial in 208-l drums.

5.2.2 Plant Description

A MOX plant is a manufacturing facility designed and constructed for the production of $(\text{U-Pu})\text{O}_2$ pellets and incorporation of these pellets into clad fuel rods. The plant also has facilities for the recovery of plutonium from unirradiated scrap materials. An isometric drawing of the reference plant (the Cimarron plant of Kerr-McGee Corporation) is shown in Figure 5.2-2. Details of the plant description are given in Section A.2 of Appendix A. The plant is assumed to be located on a fenced-in restricted area of about $12,000 \text{ m}^2$ (3 acres). It consists of about 2400 m^2 ($26,000 \text{ ft}^2$) of manufacturing, maintenance, office and laboratory floor space located in one building. Auxiliary facilities include a cooling tower, electrical substation, effluent storage and gas supply.

FIGURE 5.2-2. Isometric of Reference MOX Plant

Facility Description

The exterior walls and roof of the plant building are constructed of precast, prestressed concrete strengthened with additional reinforcing steel to withstand a wind load of 190 km/hr (120 mph). A concrete floor was poured in place after the precast concrete building was erected. The roof construction over the concrete roof deck is a poured-in-place insulation with built up roofing and gravel. All exterior and interior joints in the precast building are caulked to make an airtight structure.

The process area includes a poured-in-place reinforced concrete storage vault and adjoining basements. The vault is used for storage of plutonium not currently being processed. It is capable of withstanding wind loads of 480 km/hr (300 mph), and a pressure change of 0.34 atm (5 lb/in.²). Two of the walls of the vault are constructed to contain storage tanks for plutonium and uranium liquid solutions. Mild steel sleeves 15.2 cm (6 in.) in diameter and 9.1 m (30 ft) long are encased in these two reinforced concrete walls. Storage tanks are located inside the steel sleeves. Basement areas provide access to the bottom of the plutonium and uranium solution storage tanks and house the solvent extraction process, the scrap recovery operation and the waste solution storage tanks.

The walls, ceiling and floors throughout the building are coated with three layers of an organic coating to provide a smooth coated surface that may be easily cleaned.

Glove boxes for all processing equipment provide additional containment inside the building. These boxes are built of sand-blasted stainless steel that is 1 cm (3/8 in.) thick. Windows are made of plexiglass. Processing is generally done in process equipment inside these glove boxes.

To provide appropriate separation of functions and improve ventilation control, the building is divided into work areas with each area subdivided into rooms in accordance with their respective functions. The work areas include the following:

- office area
- central services core
- analytical laboratory area
- process area.

The office area is considered a clean area and is separated from all other facilities. It contains a reception area, director's office and staff lunchroom. A central service core contains change rooms, shower facilities, laundry and health physics laboratories. The analytical laboratory area includes rooms for general wet chemistry, metallurgical analysis, radiological analysis, emission spectroscopy and mass spectroscopy. A maintenance shop with a glove box for equipment maintenance is located at a central corridor. The process area

is subdivided into wet processing, dry blending, pellet fabrication, rod fabrication and inspection, and scrap recovery rooms. The vault is located in the center of the processing area. Table 5.2-2 lists the individual rooms in the processing area and briefly describes the functions carried out in each room. Detailed descriptions and equipment lists for process rooms are given in Section A.2 of Appendix A.

The only personnel entrance is at the front of the building. Emergency "exit only" doors are provided throughout the plant. The emergency doors are sealed to prevent inflow or outflow of air and are equipped to sound an alarm when opened. The shipping and receiving area at the rear west side of the plant is a material air lock. Two air locks are provided between the change rooms and the process area. Another air lock is provided for entry to the laboratory rooms. All air locks are interlocked so that only one door to the air lock can be opened at one time.

Ventilation System Description

A simplified flow diagram for the reference small MOX plant ventilation system is shown in Figure 5.2-3. Air is supplied to the building through an intake system that provides for dust filtration, heating or chilling as appropriate to the season, and distribution to the various rooms in the building. The intake system is located in the supply air fan room (room 202) located on the second floor of the building. The flow of air within the building is controlled. Air moves from clean areas to areas with successively higher contamination potential and finally into the glove box enclosures containing process equipment, where the ventilation system becomes effectively a gaseous radioactive effluent treatment system.

Air is supplied to individual rooms through ducts and diffusers located in the ceilings of the rooms. Room air is exhausted through a roughing filter followed by a high-efficiency particulate air (HEPA) filter installed in the floor of each room. The air then enters an underground ductwork system and is filtered by a final HEPA filter system in the exhaust air fan room before being exhausted from the plant.

Room air entering a glove box is filtered by roughing and HEPA filters. Air leaving a glove box is filtered first by roughing and HEPA filters at the glove box and then by a second HEPA filter located in each room for all the glove boxes in the room. The air then enters a central ductwork system and is filtered by a third HEPA filter in the exhaust air fan room before being exhausted from the plant.

The exhaust air fan room (room 201) is located on the second floor of the building adjacent to the stack that extends from the building roof (see Figure 5.2-2).

TABLE 5.2-2. Functional Uses of Process Rooms in the Reference MOX Plant

Room Number	General Operations	Overall Dimensions (m)	Functional Use
155	Dry Blending	6.23 x 4.68 x 4.27 high	Weighing, blending and milling of PuO_2 and UO_2 powder to form a mixed oxide powder suitable for fuel pellet fabrication.
128	Wet Processing	14.53 x 7.16 x 7.32 high	$\text{Pu}(\text{NO}_3)_4$ and $\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$ solutions are combined and chemically processed to precipitate Pu hydroxide and ADU. The precipitate is calcined, milled and blended to form a mixed oxide powder suitable for fuel pellet fabrication.
127	Clean Scrap Recovery	16.0 x 5.19 x 7.32 high	Room contains dissolvers, ion exchange columns and a calciner to process clean scrap and recover plutonium in a form suitable for return to production.
126	Storage Vault	13.72 x 5.56 x 4.27 high	Temporary storage of plutonium and uranium solutions, plutonium oxide and mixed oxide powder.
124	Pellet Processing	19.93 x 4.95 x 7.32 high	Room contains presses, furnaces and grinders to convert mixed oxide powder into precisely dimensioned fuel pellets.
123-122	Fuel Fabrication and X-ray	19.93 x 4.72 x 7.32 high	Pellets are inspected and assembled in cladding tubes. Cladding tube end caps are welded and welds are inspected and x-rayed.
121	Fuel Rod Inspection	18.29 x 5.87 x 7.32 high	Completed fuel rods are washed, inspected and packaged for shipment off-site.
801	Dirty Scrap Recovery	10.82 x 5.18 x 4.88 high	Contains equipment for processing dirty scrap to recover plutonium and convert it to a form suitable for return to production.
802	Waste Treatment	16.46 x 7.16 x 4.88 high	This room contains liquid waste storage tanks and evaporators.
805	Solvent Extraction	5.18 x 3.05 x 12.50 high	This room houses the solvent extraction columns for processing liquid waste to recover plutonium and uranium.
129	General Chemistry Laboratory	10.67 x 10.67 x 4.27 high	Chemical assay of solutions and powders containing uranium and plutonium.
116	Equipment Maintenance	10.67 x 5.95 x 4.27 high	This room contains a glove box for maintenance and repair of contaminated equipment.
156	UNH Receiving	4.68 x 3.66 x 4.27 high	Room contains piping for unloading $\text{UO}_2(\text{NO}_3)_2 \cdot 6\text{H}_2\text{O}$ for feed to wet process; also drum storage space.

FIGURE 5.2-3. Simplified Flow Diagram for the Reference MOX Plant Ventilation System

5.2.3 Radioactive Contamination

Table 5.2-3 summarizes estimated inventories of plutonium in each process room at the reference MOX plant at shutdown. Only plutonium numbers are shown because the plutonium radiological hazard completely dominates the uranium radiological hazard. Key assumptions and rationale on which the inventory estimates are based and detailed inventories for each room are presented in Appendix C of Reference 1.

At shutdown the plutonium inventory in the reference plant is estimated to be between 40 and 50 kg.

5.3 DESCRIPTION OF THE REFERENCE URANIUM FUEL FABRICATION PLANT

The Wilmington, North Carolina, plant of General Electric Company^(8,9) is the reference facility for this study because it is representative of contemporary U-Fab plants in the United States. The Wilmington plant currently uses two head-end processes for converting gaseous UF_6 to UO_2 . The primary method used is a chemical process involving hydrolysis of vaporized UF_6 to ammonium diuranate (ADU) precipitate using ammonia, and reduction and calcining of the ADU to dry UO_2 powder. The secondary method involves direct conversion of UF_6 vapor to U_3O_8 in a flame conversion reactor and reduction of U_3O_8 to UO_2 powder in a reduction-calciner. The associated facility components for both processes are analyzed in this study, and the methods and costs for decommissioning each head-end process are discussed individually.

Details of the reference U-Fab plant, the plant process descriptions and site description details are given in Appendix A, and the bases for residual radioactivity estimates are presented in Reference 2.

5.3.1 Process Description

Overall processing characteristics assumed for the reference U-Fab plant are presented in Table 5.3-1. A simplified block flow diagram of the process is shown in Figure 5.3-1. Details of the process are given in Section A.3 of Appendix A.

All processes associated with fuel pellet fabrication are performed inside enclosures located in seven rooms in the process areas of the building (the UF_6 vaporization area; the first-floor chemical area; the second-floor chemical area; the press area; the sintering area; the grind, load, weld area; and the gadolinia area). Transfer of product materials from one process step to the next is accomplished both automatically and manually, using 19-l cans (one safe batch).

TABLE 5.2-3. Estimated Radioactivity Levels in Reference MOX Plant at Shutdown

Process Area		At Shutdown
Room Number	Operation	kg of Plutonium (a)
155	Dry Blending	0.74
128	Wet Processing	6.7
B02	Wet Processing and Waste Treatment	3.9
128-B02	Piping	1.7
127	Scrap Recovery	4.6
B01	Scrap Recovery	7.2
SX-127-B01	Piping	5.4
SX	Solvent Extraction	11.3
124	Pellet Processing	8.7
123	Fuel Fabrication	0.24
126	Storage Vault	0.37
129	Laboratory	0.19
116	Maintenance	<u>0.07</u>
Rounded Totals		50

(a) Values are calculated; the number of significant figures is greater than the expected accuracy.

TABLE 5.3-1. Overall Processing Characteristics
of the Reference U-Fab Plant

Daily Processing Capacity

3300 kg U per day (i.e., 3800 kg UO_2 per day)

Average Yearly Production

1 million kg U per year

Plant Efficiency

~80%

Annual Input

1 million kg U as UF_6 gas enriched 2 to 4 wt% in ^{235}U ; shipped in 2300-kg, 76-cm-diameter cylinders (Model OR-30B), packaged in a protective shipping container.

Annual Output

99% UO_2 Pellets--79% in fuel rods, 1% UO_2 Powder;
20% as pellets only

Scrap Rate

20% of U throughput recycled as clean scrap

Effluent Losses

0.7 wt% U in solids
0.3 wt% U in liquids

Unrecovered Losses

0.45 wt% U

Production activities fall into three basic categories:

1. chemical conversion of UF_6 to UO_2
2. mechanical processing, including pellet production, fuel rod fabrication, and fuel bundle assembly
3. recovery of uranium from scrap, off-specification material, and chemical wastes.

FIGURE 5.3-1. Simplified Process Flow Diagram for the Reference U-Fab Plant

Chemical Conversion Head-End

Uranium is received by truck at the plant site as UF_6 sealed in pressure cylinders. The UF_6 , enriched from 2 to 4 wt% in ^{235}U , arrives in 2300-kg, 76-cm-diameter cylinders (Model OR-30B) that are protected in transit by an outer shipping container. Uranium slightly enriched above 4 wt% in ^{235}U arrives in 25-kg, 12.7-cm-diameter cylinders (Model OR-5A) that are also packed in protective shipping containers.

Upon receipt, the cylinders are carefully weighed for accountability purposes and then either stored or transferred on a cart from the storage platform to the adjacent UF_6 vaporization room. By an overhead bridge crane, each individual cylinder is lifted into position in one of 11 different vaporization chambers. The cylinder is properly connected to a flexible discharge line, then heated by electrically heated air to volatilize the stored UF_6 . The UF_6 gas is piped to a vessel filled with water, where the gas reacts with water (hydrolyzed) to form UO_2F_2 and HF. The HF in the off gas is scrubbed and recovered as a byproduct. The UO_2F_2 solution is recirculated through one of two identical hydrolysis storage tanks until the proper UO_2F_2 concentration is reached. The process is then switched to the second storage tank and the UO_2F_2 solution in the first tank is transferred to the chemical precipitation tank. In the precipitation tank, ammonium hydroxide is added to the UO_2F_2 solution to precipitate uranium as ADU. The resulting ADU solution is transferred to a digester tank to aid the precipitation process. The resulting ADU slurry is then dewatered and concentrated (by centrifuging) to a paste. The paste is continuously fed to the horizontal chamber of a gas-fired reduction calciner, where the ADU is defluorinated, reduced, and dried to produce ceramic-grade uranium dioxide powder. The UO_2 powder is placed in 19-l cans for transport to the second-floor UO_2 powder preparation area.

Direct Conversion Head-End

Four flame conversion reactors are used to directly convert UF_6 cylinders, in a cycle that permits continuous operation of the reactors. In the reactors, UF_6 , natural gas, and oxygen are mixed at a carefully controlled temperature to optimally convert UF_6 to U_3O_8 and HF. The HF contained in the reactor off gas is scrubbed and recovered as a byproduct, while the U_3O_8 is retained in filter tubes and collected in one of two identical powder collection pots. U_3O_8 is then pneumatically transferred to a hopper, collected in 19-l cans, and weighed before further processing. The U_3O_8 is further treated by crushing and granulating it to a powder suitable for further processing as necessary. The U_3O_8 powder is then fed to a horizontal, gas-fired reduction calciner, where it is defluorinated, reduced, and dried to produce ceramic-grade UO_2 powder. The calciner discharges the UO_2 powder through a double rotary air lock system into 19-l buckets. The UO_2 powder is then transported to the second-floor UO_2 powder preparation area for treatment to enhance its pellet-forming properties.

Pellet Formation and Fuel Rod Assembly

Pellet formation is a two-step operation. In the first step, the uranium dioxide is pulverized, compacted, and granulated to increase its density and size distribution to that desired for pellet formation. To perform these steps, UO_2 from the calciners is processed through a size-reduction hammer mill, a predensifier press, and a granulator that crushes the compacts from the press into material of uniform particle size. The sized UO_2 particles are placed in 19-l buckets, sealed, and transferred to the mezzanine storage area. After samples of the densified material are analyzed for uranium and moisture content and found acceptable, the UO_2 particles are ready for the second step in the process of forming UO_2 pellets.

In the second step, the densified UO_2 powder is pelletized, and the pellets are sintered in a reducing (hydrogen) atmosphere. They are then ground to finished dimensions and loaded into fuel rods. To perform these steps, the densified UO_2 powder stored on the mezzanine is loaded in hoppers that feed the pellet presses located on the main floor. The pellets formed are sintered in a reducing atmosphere in one of five electrically heated furnaces (sintered to about 95% of theoretical UO_2 density) and ground to correct size through one of five centerless grinders equipped with diamond-grit work wheels. The finished pellets are placed in trays and stored until needed in storage cabinets located at one of four pellet-loading stations. Fuel pellets removed from storage are placed on fuel mock-up channels, weighted, and measured to meet required specifications. An acceptable string of pellets is pushed into an empty zircaloy tube previously welded at one end. Loaded tubes (now fuel rods) are placed in trays and transferred to one of three rod-outgas ovens to remove all traces of moisture. From the ovens, individual rods are inserted into a controlled atmosphere weld box. After air is evacuated, the rod is backfilled with helium and the end plug automatically inserted and welded into position. The rods are placed in trays and stored in cabinets for eventual assembly into fuel bundles.

Fuel Bundle Assembly and Final Inspection

Rod trays are removed from the storage cabinets and each fuel rod is scanned for ^{235}U content, using a neutron source (^{252}Cf) and gamma radiation detectors to confirm the enrichment. Based on rod enrichment requirements for a bundle matrix, the required number of rods are removed from the tray and laid out in a specific order on one of five assembly tables. This procedure is repeated for each enrichment, until the total number of rods needed to satisfy the bundle matrix is obtained. The correct number of rods with the right enrichments for the bundle to be assembled is now on the bundle make-up table in proper loading order. The rods are visually inspected for cleanliness and damage and replaced as necessary.

The spacer hardware is positioned with the fixtures on one of five horizontal bundle loading tables. The fuel rods on the assembly tables are inserted into the spacer hardware according to a fixed insertion schedule. When all rods are in place, the end pieces or tie plates are bolted into position and the assembly is raised vertically, unlocked from the fixture, and removed by an overhead crane to the lead-test and inspection station.

The assembled bundles are lead-tested in a vacuum chamber. Helium leaking from fuel rods is detected in the exhaust air, using a helium mass spectograph.

The bundle is next placed in a lighted inspection fixture, where the rods are inspected for straightness, linearity, and other design requirements. If the bundle passes the inspection, it is wrapped in a plastic dust cover and removed with an overhead crane to the bundle storage racks. The bundles are vertically suspended from hooks on the racks by their upper tie plates until they are ready for shipment.

Scrap Recovery

Internally generated "clean" uranium scrap in various physical forms that does not meet quality standards or that is mixed with combustible foreign material (i.e., slugger and press residue, ADU from pad, grinder sludge, rejected pellets, etc.) is reprocessed through scrap recovery equipment known collectively as the uranium purification system (UPS). The simplified process flow-sheet diagram for the UPS is shown in Figure 5.3-1. Material to be reprocessed through the UPS is accumulated in safe batches (normally 19- $\frac{1}{2}$ buckets) and stored until processed.

Scrap uranium to be reprocessed (~21-kg lots) is slowly charged through a discharge chute into one of nine different dissolver tanks (arranged in sets of three) filled with hot nitric acid (88°C). After dissolution, the uranyl nitrate is filtered, cooled, and sent to one of three precipitation tanks. Ammonium hydroxide and hydrogen peroxide are used to precipitate uranium tetroxide ($\text{UO}_4 \cdot \text{H}_2\text{O}$) from the solution. The UO_4 slurry is dewatered in a centrifuge, and the UO_4 cake discharged is pumped from the long feed tank (UO_4 receiver tank) to a gas-fired reduction calciner. In the calciner, the UO_4 is converted to UO_2 powder, collected in 19- $\frac{1}{2}$ buckets, and returned to the powder storage area in the main process area for use in making fuel pellets.

Uranium scrap mixed with foreign materials (gadolinia slugger and press residue, gadolinia grinder sludge, rejected gadolinia pellets, nitrate wastes, combusted wastes, solid wastes, radwastes, etc.) not meeting UPS quality standards for processing is shipped to an offsite contractor for rework and recovery of uranium. The recovered uranium is returned to the facility in the form of UNH and is added back into the main process through the UPS.

Liquid Waste Effluent Processing

Chemically different liquid waste streams containing uranium (generated in fuel manufacturing operations) are kept separate to facilitate uranium recovery operations. They are classified as nitrate wastes, fluoride wastes, and rad-wastes. This separation allows individual treatment of each process waste stream in a manner compatible with the recycle of valuable materials back into the main process and/or the release of effluents into the environs. Figure 5.3-2 illustrates schematically how each of these waste streams is treated before eventual impoundment or release offsite. Each stream has its own quarantine tank system within the main building to permit temporary storage of the waste liquid. Based on uranium content, the liquid wastes are either released for final waste treatment or are recycled to recover the uranium residuals.

Solid Waste Processing

Solid wastes such as paper, rags, mops, plastic, wood, protective clothing, damaged tools, and equipment are constantly generated during plant operations. These waste materials are collected at the point of origin in designated containers designed to prevent the loss of contents. Filled containers are sealed, tagged, stored, and eventually transferred to the waste handling facility in the fuel manufacturing building. In the waste handling facility, uranium-bearing materials in containers are segregated into noncombustible and combustible categories and treated or disposed of as illustrated in Figure 5.3-3.

5.3.2 Plant Description

The reference U-Fab plant is designed and constructed for the production of UO_2 pellets, the incorporation of these pellets into fuel rods, and the assembly of fuel rods into fuel bundles. The plant also has facilities to recover uranium from scrap and waste materials and to recover valuable chemicals from gaseous and liquid wastes. An isometric drawing of the reference plant is shown in Figure 5.3-4. Details of the plant description are given in Section A.3 of Appendix A.

The plant is assumed to be located within a fenced-in restricted area of approximately $30,000 \text{ m}^2$ that is controlled and policed by security personnel. The main plant building occupies $19,300 \text{ m}^2$ of manufacturing, laboratory, maintenance, decontamination, storage, and office floor space. Adjacent and connected to the south side of the main building are two separate but inner-connected single-story structures. The structures house the chemical-metallurgical laboratory, the waste recycle control room, and the UO_2 powder

FIGURE 5.3-2. Process Liquid Waste Treatment

warehouse, and occupy 770 m², 340 m², and 810 m² of floor space, respectively. Contaminated waste incineration operations occupy another 220 m² of floor space in a separate building located 30 m to the west of the main building. Other auxiliary facilities include a fluoride and nitrate waste treatment plant and associated lagoons, liquid chemical waste treatment lagoons, a sanitary waste treatment plant, a propane station, a tank and pump station, equipment storage yards, an electrical substation, and warehouse and CaF_2 storage grounds.

FIGURE 5.3-3. Solid Waste Treatment

Facility Description

The main fuel manufacturing building is a 80-m by 211-m structure that is built to allow for future expansion to double the current production capacity. It is fabricated of 3.8-cm insulated metal siding attached to a steel framework. The interior walls are constructed of 20- by 20- by 41-cm concrete block and 1.3-cm sheet rock. The ground floor and the first meter of the outside walls are reinforced concrete 10 to 15 cm thick. The mezzanine area floors are steel grating or reinforced concrete. The steel beams supporting the building steel framework are anchored in the outside 1-m-high walls. The roof consists of a 3.5-cm insulated corrugated metal deck that is capped with 3.8 cm of asphalt and gravel.

MEZZANINE LEGEND

1 Decon Area	12 Control Room
2	13 Gadolinia Facility
3	14 HVAC Equipment Room
4 Q A Office	15 Womens Change Room
5 Mens Change Room	16 Mens Room
6 Che Me Office	17 Equipment Room
7 Mens Change Room	18 Cafeteria
8	19 Conference Room
9 Laundry Room	20 Office Area
10 Chemical Area	21 Computer Room
11 Powder Storage Area	22 Office Area

Areas open through Mezzanine Floor

GROUND FLOOR LEGEND

1 Storage Area	11. Maint. Storage Rm.	27. Design Services
2	12. Deionizer Area	28. Hot Instrument Lab
3	13. Air Compressor Rm.	29. Fluid Bed Blender Area
4. Trucking Area	14. Rad. Waste Area	30. Pelletizing Area
5	15. Supply Room	31.
6. GECO Equipment	16. Red Cap Area	32. Sintering Development Area
7 Decon. Room	17.	33. Ceramic Lab
8 Cylinder Storage Area	18. Maint. Hot & Cold Shop	34. Sintering Area
9 Vaporization Area	19. GECO Area	35. Gadolinia Facility
10 Boiler Area	20. Storage Area	36. Fuel Mfg. Engineering
	21. Chemical Area	37. GECO Development Lab.
	22. UPS Area	38. Fuel Rod Fab. Development
	23. Powder Storage Warehouse	39. Metals Tech. Lab
	24. Men's & Women's Change Rooms	40. Segmentizing Area
	25. Chem. Labs	41. Box Factory
	26. Rad. Safety Office	42. Neutron Radiography Facility
		43. Elevator
		44. Q. T. Development Lab
		45. N.D.T. Lab
		46. Instrument Lab
		47. Office Area
		48. Rod Storage
		49. Bundle Assembly Area
		50. Fuel Bundle Area
		51. Elevator
		52. Fuel Mgr
		53. Fuel Mgr's office Area
		54. First Weld End Plug Station
		55. Small Parts Storage Area
		56. Shipping Area
		57. Bundle Shipping & Tube Storage

FIGURE 5.3-4. Isometric of the Reference U-Fab Plant

In the main building, the fuel fabrication operations are divided into the following unit operations:^(a)

- UO_2 powder production (ground and upper level)
- pellet production
- rod loading and welding
- gadolinia rod production
- bundle assembly.

The process areas of the main building are divided into a series of enclosed or partially enclosed rooms located on both the ground and upper floors. These interior processing rooms are referred to as follows:

- UO_6 vaporization room
- chemical area
- UO_2 powder production (upper floor)
- powder storage and feeding (upper floor)
- press area
- sintering area
- grind, load, and weld area
- gadolinia room
- quality control, bundle assembly, and packaging area.

Uranium recovery, chemical recovery, waste handling maintenance, laundry, storage, boiler and water treatment, air filtering, and laboratory operations are conducted in support of production operations within the main building. Change rooms, offices, control stations (shown in Figure 5.3-4) are involved in the above support activities.

The processing areas or rooms within the main building are generally partitioned with painted cinderblock walls or, in some instances, with painted wall board. The floors of the production areas are generally covered with 30.5-cm-square tiles that are removed and replaced yearly. The chemical area has a sealed concrete floor. Nonproduction and noncontaminated areas in the facility generally have a concrete floor. Offices and change rooms have either tiled or painted floors.

The chemical-metallurgical (chem-met) laboratory is a single-story structure 21 m wide by 37 m long that is separated from the main building by a 3-m enclosed corridor. It is constructed of concrete slab siding and concrete floors and has the same type of roof as the main building. The laboratory is divided equally into a controlled and a uncontrolled section. The controlled section has hoods, enclosures, and glove boxes for testing, handling, and containment of uranium in various physical and chemical forms.

(a) These unit operations may involve more than one room within the plant.

The stacker building is a separate single-story structure that is attached to both the main building and the chem-met laboratory. It is 27 m wide by 47 m long and houses the UO_2 powder warehouse and the uranium waste and scrap processing (red cap) areas. The building is constructed of insulated metal siding attached to a metal framework that is anchored to a concrete foundation and floor. The roof is sloped and constructed of insulated, corrugated metal decking.

The warehouse portion of the stacker building contains sets of storage racks for storing 19-l buckets of uranium oxide. The red cap area of the stacker building contains a crusher and blender and a drying or heat-treating furnace. Uranium scrap materials are pretreated in the red cap area before they are processed through the UPS, shipped to an offsite contractor for uranium recovery, or directly recycled back into the main process.

The waste incineration building is 12.2 m wide by 18.3 m long and approximately two stories high. It is constructed of insulated corrugated metal siding attached to a metal frame. The frame and metal siding rest on a concrete floor-foundation. The roof is similar to that of the main building. Combustible materials from plant operations are incinerated in a blower furnace and, depending on the uranium content, the ash is either shipped for burial in a government-licensed burial ground or is reworked by an outside contractor to recover the uranium.

The calcium fluoride and nitrate waste treatment plant is a multistoried, roofed structure that is open on four sides. It is constructed of large steel framework members that serve as supports for the numerous tanks, columns, pumps, and piping associated with the facility. A small enclosed area on the ground floor of the building houses a control room for the facility and a uranium recovery (centrifuge) room. A separate building next to the main facility houses a boiler for supplying steam to the NH_3 stripping column. Adjacent to the waste treatment facility are six storage lagoons for the temporary impoundment of waste liquids and solids. These lagoons are lined to prevent seepage of waste liquids into ground water and are periodically cleaned of all solid residues.

Ventilation System Description

A simplified flow diagram for the reference U-Fab plant ventilation systems is shown in Figure 5.3-5. Each operational area in the fuels building has its own separate air handling system that is designed specifically to provide clean, thermally conditioned air to the building and to exhaust filtered or scrubbed and filtered air to the environs. The flow of air within the building is controlled so that the air moves from clean areas to areas with successively higher contamination potential.

FIGURE 5.3.5. Ventilation and Hood Exhaust Systems in Main Production Areas of the Reference U-Fab Plant

Fresh air is supplied to each process area within the building after it has been filtered for dust, chilled to remove moisture, and heated as appropriate for the season. In the chemical (UF_6 - UO_2 conversion), radwaste, and "hot" maintenance shop areas, fresh air is directly drawn to make up for all the air being exhausted. In the vaporization, sintering, UO_2 powder storage, segmenting, and pelletizing areas of the building, fresh make-up air is drawn as needed into the recirculating air ventilation system. The fresh air and recirculated air are filtered through a dust and a HEPA filter, respectively, before being mixed, chilled, and heated in the recirculation ventilation system for distribution in the building. Any additional air needed for a contaminated room is drawn from building areas having little or no contamination. The recirculating ventilation system in the vaporization area is equipped with a scrubber to remove contaminants from the recirculating air. An emergency dump fan is also located in the vaporization room that boosts exhaust air from the room into the UF_6 - UO_2 conversion area exhaust system (water scrubber-HEPA system), in the event of an accidental UF_6 release.

There are 32 stacks associated with exhaust air systems for fuel manufacturing operations. Twenty-eight of these stacks are located on the roof of the main building. The other four are located on the roof of the incineration building. These separate air exhaust systems exhaust air from the building either through enclosures containing highly contaminated process equipment or through room exhaust systems located in potentially contaminated rooms. Exhaust air from the vaporization and chemical processing (UO_2 - UF_6 conversion) areas and associated equipment is scrubbed and dried before being filtered through HEPA's. Exhaust air from other processing areas generally is filtered through two HEPA's. One HEPA is generally located at the work station for quick changeout. Some heavily contaminated enclosures have a roughing filter that precedes the first HEPA to eliminate large-diameter aerosols.

5.3.3 Radioactive Contamination

Table 5.3-2 summarizes estimated residual radionuclide inventories of uranium in major pieces of equipment at the reference U-Fab plant, and Table 5.3-3 shows the estimated distribution of residual radioactivity in the plant ventilation system at shutdown. Key assumptions and rationale on which the inventory estimates are based and detailed inventories for each room are presented in Reference 2. At shutdown and before final inventory cleanout, the uranium inventory in the reference U-Fab plant is estimated to be about 790 kg.

5.4 DESCRIPTION OF THE REFERENCE NON-FUEL CYCLE FACILITIES

About 20,000 specific licenses of the NRC and the agreement states are in effect authorizing the handling of byproduct, source, and special nuclear material. Most of these are not involved in the nuclear fuel cycle. Licensed

TABLE 5.3-2. Estimated Uranium Inventory in Major Pieces of Equipment at the Reference U-Fab Plant at Shutdown

Equipment	kg of Uranium
UF ₆ Vaporization	
Vaporizers and Equipment	2.1
Chemical Process Equipment	
Hydrolyzer Tanks and Equipment	1.4
Precipitation and Digestion Tanks and Equipment	0.2
Centrifuges and Equipment	0.3
Calciners and Equipment	21.8
Milling, Pressing and Granulation	43.7
Bucket Loading Equipment	1.5
GECO Flame Reactors and Equipment	6.4
Uranium Purification System	<u>3.1</u>
Subtotal	78.4
Pellet Production and Assembly	
Blenders and Powder Storage	2.5
Pelletizing	4.6
Sintering	0.5
Grinding and Rod Loading	3.5
Granulators	<u>39.7</u>
Subtotal	50.8
Gadolinia Production	
Blenders	1.2
Sluggers	0.3
Granulators	0.1
Pelletizers	0.1
Sintering Furnace	0.3
Grinders	0.7
Rod Evacuators	<u><0.1</u>
Subtotal	2.7
Waste and Scrap	
Scrap Recovery	1.4
Incinerator	<u>8.4</u>
Subtotal	9.8
Other	<u>7.0</u>
Total	150.8

TABLE 5.3-3. Estimated Distribution of Residual Radionuclide Inventories in the Reference U-Fab Plant Ventilation System at Shutdown

<u>Ventilation Stack Area Served</u>	<u>Total Uranium (kg)</u>	<u>% of Total</u>
Chemical Process, North	24.63	20.7
Chemical Process, South	19.16	16.1
Grinders and Rod Loading	37.37	31.4
Sluggers	15.83	13.3
Powder Storage	0.48	0.4
Pelletizer	0.48	0.4
Pelletizer	2.86	2.4
Scintillation Furnace	1.07	0.9
Gadolinia	2.26	1.9
Pail Filling	1.31	1.1
Chemistry Laboratory	0.48	0.4
Waste Treatment Centrifuge Room	0.04	0.03
Waste Treatment Laboratory	0.02	0.02
Mill	0.83	0.7
New Decontamination Room	0.36	0.3
Hot Maintenance	0.71	0.6
Scrap Treatment Warehouse	1.43	1.2
Women's Change Room	0.12	0.1
Men's Change Room	0.1	0.08
Laundry	0.24	0.2
Mezzanine Warehouse	0.24	0.2
Chemical Process, FMOX, North	0.36	0.3
Incinerator Room, North	0.24	0.2
Incinerator Room, South	0.24	0.2
Incinerator Exhaust	6.66	5.6
Incinerator Exhaust	0.36	0.3
GECO Process	0.48	0.4
Uranium Purification System	--	--
Chemical Process, FMOX South	0.36	0.3
Stacker Warehouse	0.1	0.08
Rod Out-Gas	<u>0.12</u>	<u>0.1</u>
Totals	119	100

operations include a variety of applications in industry, medicine, education, and research. Operations range from the occasional use of short-lived isotopes by physicians for the diagnosis and treatment of their patients to the large-scale processing of radioactive materials by industrial licensees engaged in the manufacture of labeled compounds and sealed sources.

NRC and agreement state licenses in effect as of June 1978 are summarized in Table 5.4-1. The number of licensee facilities is less than the number of

TABLE 5.4-1. Radioactive Material Licenses
as of June 1978

Nuclear Regulatory Commission Licenses

<u>Byproduct</u>		
Medical	2 239	
Academic	384	
Industrial	4 205	
Civil Defense	104	
Other	<u>27</u>	
Total Byproduct	6 959	(6 924) ^(a)
Source	400	(332)
Special Nuclear	<u>720</u>	(583)
Total	8 079	(7 839)

Agreement State Licenses

Medical	4 749
Academic	867
Industrial	5 030
Civil Defense	185
Other	<u>900</u>
Total	11 731

(a) Non-fuel-cycle licenses are in parenthesis. These numbers were obtained by subtracting fuel cycle facilities and also export/import licenses which are, in effect, paper transactions and do not represent separate facilities.

licenses because some facilities are licensed under more than one part of the regulations and therefore possess more than one license. A more complete breakdown of radioactive material licenses, with some indication of possible decommissioning efforts that might be involved, is presented in Appendix A of Reference 3.

The focus of this section is on materials licensee facilities that represent a significant decommissioning effort. Because of their large number, it is not practical to attempt a description of decommissioning requirements for all possible types. The examples selected are believed to be representative of real facilities. The example facilities are nuclear material processing and use laboratories. These facilities can be categorized as:

- Radioactive material processors - These licensees include companies that manufacture the radiopharmaceuticals, radiochemicals, and sealed radioactive sources used by medical, academic, and industrial licensees.
- Institutional users - These licensees include hospitals, medical schools, and other academic institutions that use radioisotopes in both unsealed and sealed source form for medical, research, and educational purposes.

Examples from each of these categories are described in the balance of this section.

5.4.1 Reference Radioactive Material Processor Laboratories

In this study, the term "radioactive material processor" is used to designate companies with specific licenses that manufacture and distribute radiopharmaceuticals, radiochemicals, and sealed radioactive sources. Specific licenses issued to these companies under the provisions of 10 CFR 30, 40, and/or 70, or comparable agreement state regulations authorize the possession, storage, use in research and in manufacturing, and distribution to authorized persons of a wide variety of radioisotopes. Typical radionuclide possession limits for selected isotopes that might be authorized for a large radioactive material processor are shown in Table 5.4-2.

Radiopharmaceuticals are biochemicals such as nucleic acids, amino acids, proteins, carbohydrates, hormones, drugs, carcinogens, and toxins that are labeled with radioactive atoms. The radioisotopes most commonly used for radioactive labeling of biochemicals are ^3H and ^{14}C . ^{32}P , ^{35}S , ^{125}I , and ^{131}I are also used. Details of the manufacture of labeled biochemical and sealed sources are given in Appendix A of Reference 3.

TABLE 5.4-2. Typical Possession Limits for Selected Radioisotopes for a Radioactive Material Processor

Isotope	Possession Limit (curies)
^3H	100 000
^{14}C	1 000
^{32}P	100
^{35}S	100
^{55}Fe	100
^{60}Co	10 000
^{85}Kr	10 000
^{90}Sr	1 000
^{125}I	100
^{131}I	100
^{137}Cs	10 000
^{241}Am	100

The preparation of labeled compounds or radioactive sources is generally handled in such a way that work with different radioisotopes is performed in separate rooms or separate work spaces. To illustrate the diversity in equipment requirements for work with different radioisotopes, five reference laboratories are described. These laboratories include a laboratory for the manufacture of:

- ^3H -labeled compounds
- ^{14}C -labeled compounds
- ^{125}I -labeled compounds
- ^{137}Cs sealed sources
- ^{241}Am sealed sources.

Most labeled compounds are hydrocarbons, and the most common labeling agents for these compounds are ^3H and ^{14}C . These isotopes are low-energy beta emitters that do not require the use of radiation shielding beyond that provided by laboratory glassware and hoods or glove boxes. Radioisotopes of iodine are both radiotoxic and highly volatile. The labeling of compounds with ^{125}I requires the use of special equipment and procedures to ensure the proper containment of the material and adequate filtration of the exhaust from work enclosures. To protect workers from excessive exposure to radiation, processing operations that involve curie quantities of gamma-emitting radionuclides

such as ^{137}Cs are normally carried out in shielded enclosures. The reference laboratory for the manufacture of ^{137}Cs sealed sources includes a small hot cell where operations are performed remotely with master-slave manipulators. The design of the ^{241}Am laboratory illustrates some special requirements for work with transuranic elements.

The five radioactive material processor laboratories are described in Sections 5.4.2 through 5.4.6. The descriptions include estimates of residual contamination levels on building surfaces and facility components at the time of decommissioning. These contamination levels are based on experience at a large commercial laboratory (New England Nuclear Corporation) for the manufacture of labeled compounds and sealed sources.

5.4.2 Laboratory for the Manufacture of ^3H -Labeled Compounds

The reference laboratory for the manufacture of ^3H -labeled compounds is shown schematically in Figure 5.4-1. Work enclosures and equipment are provided in the laboratory for labeling organic compounds by catalytic reduction with tritium gas, for incorporating these labeled compounds in appropriate chemical solutions, and for packaging the labeled compounds for shipment.

Tritium has a radioactive half-life of 12.26 years. It decays by emitting a beta particle with a maximum energy of 0.0186 MeV. Some important properties of tritium are discussed in Appendix A of Reference 3.

Tritium labeling of organic compounds is usually effected by exchange methods. (Several exchange methods for labeling with tritium are described in Appendix A of Reference 3.) Equipment in the reference facility includes the vacuum lines and distillation equipment necessary to label and purify the product. Because of the potential radiation hazard from tritium, the equipment is set up inside fume hoods and glove boxes. Tritium gas is supplied from 1000-Ci steel cylinders that are connected directly to the vacuum manifolds. Valves and gages are provided with the containers to regulate the flow of gas.

The reference laboratory has 120 m² of floor space and is designed for occupancy by six people. The walls are plasterboard painted with a latex enamel paint. The floor is plywood covered with asphalt tile. Overhead utility lines and the ventilation ductwork are concealed by a false ceiling constructed of acoustically treated fiberboard panels. Fluorescent lighting is located in receptacles recessed in the ceiling.

There are five fume hoods in the reference laboratory. (Fume hoods are described in Appendix A of Reference 3.) Each hood is of steel construction with interior surfaces of stainless steel and a hood window of acrylic plastic. Hoods are required whenever an operation may give rise to airborne

B = BENCH
 C = CABINET
 F = FREEZER
 GB = GLOVE BOX
 H = HOOD
 R = REFRIGERATOR

FIGURE 5.4-1. Reference Laboratory for the Manufacture of ^3H -Labeled Compounds

contamination from radioisotopes or other toxic chemicals. Three of the hoods are used for labeling and purifying multicurie batches of reagents that are stored until needed to fill specific orders. The remaining hoods are used for packaging reagents in small quantities.

There are six glove boxes in the reference laboratory. (Glove boxes are described in Appendix A of Reference 3.) They are fabricated from steel with an acrylic plastic viewing panel and gloves made from neoprene or other appropriate material. One glove box houses the apparatus for the transfer of the

tritium gas from the 1000-Ci gas cylinder to the vacuum lines located in a fume hood. Additional glove boxes contain equipment for heating reactants, for purifying the labeled product, and for special operations such as the handling of light powders and toxic or biologically hazardous chemicals. Operations involving substances that are especially hazardous are performed in glove boxes of special design located inside fume hoods.

Each fume hood and each glove box is separately vented through a roughing and a HEPA filter.

There are no sinks in the reference facility. All liquid wastes are disposed of as described in Section 5.4.7.

The reference laboratory includes 20 linear meters of laboratory workbenches where apparatus is assembled and nonradioactive solutions used in the labeling process are prepared. The workbenches are 0.75 m wide and the bench tops are 0.9 m above the floor. The workbenches are of steel construction with plastic-laminate tops.

Refrigerators, freezers, and storage cabinets are provided for the storage of labeled compounds and nonradioactive chemicals.

The laboratory is air conditioned to control the movement of air, to remove dust from the air, and to maintain constant temperature and humidity. Air enters the room through air diffusers in the ceiling. Exhaust air is removed by the fume hoods. Roughing and HEPA filters are located in the exhaust ductwork immediately above each hood. There is no additional filtering of the air before it leaves the room. The exhaust ductwork is fabricated from sheet metal.

Assumed residual surface contamination levels at the time of decommissioning the facility are given in Table 5.4-3. Values listed in the table represent removable contamination as determined by wiping the area with dry filter or soft absorbent paper and counting the wipe sample with an appropriate instrument of known efficiency. The contamination is entirely from tritium, usually in the form of tritiated water vapor.

5.4.3 Laboratory for the Manufacture of ^{14}C -Labeled Compounds

The reference laboratory for the manufacture of ^{14}C -labeled compounds is shown schematically in Figure 5.4-2. Work enclosures and equipment are provided for the chemical synthesis of labeled organic compounds, for incorporating these compounds in appropriate chemical solutions, and for packaging the labeled compounds for shipment.

TABLE 5.4-3. Residual Surface Contamination Levels Prior to Decommissioning the Reference Laboratory for the Manufacture of ^3H -Labeled Compounds^(a)

Facility Surface	Removable Contamination per 100 cm ² of Surface
Walls	Generally less than 1 000 d/m, with some spots in the range of 10 000 to 100 000 d/m
Floors	Generally less than 1 000 d/m, with some spots as high as 50 000 d/m (usually at edges of benches)
Bench Tops	Generally less than 1 000 d/m, with some spots as high as 50 000 d/m
Inside Hoods	1 millicurie
Inside Glove Boxes	10 to 100 millicuries
Inside Ventilation Ductwork	50 000 to 100 000 d/m

(a) Contamination levels are based on experience at a large commercial laboratory for the manufacture of labeled compounds (New England Nuclear Corporation).

Carbon-14 has a radioactive half-life of 5740 years. It decays by emitting a beta particle having a maximum energy of 0.156 MeV. Some important properties of ^{14}C are discussed in Appendix A of Reference 3.

A ^{14}C label may be introduced into a wide variety of chemical compounds by standard synthesis procedures of organic chemistry. (Chemical synthesis of labeled compounds is discussed in Appendix A of Reference 3.) The reference facility contains vacuum manifolds and distillation equipment typical of an organic chemistry laboratory. (An example of a vacuum manifold for the carbonation of a Grignard reagent is shown in Figure A.3-2 of Appendix A of Reference 3.) Because of the potential radiation hazard to persons working with ^{14}C , all processing operations are performed within the confines of fume hoods and glove boxes.

The reference laboratory has 80 m² of floor space and is designed for occupancy by four people. The walls are plasterboard painted with latex enamel paint. The floor is plywood covered with asphalt tile. Overhead utility lines and the ventilation ductwork are hidden by a false ceiling constructed of acoustically treated fiberboard panels. Fluorescent lighting is located in receptacles recessed in the ceiling.

B = BENCH
 C = CABINET
 D = DESK
 F = FREEZER
 GB = GLOVE BOX
 H = HOOD
 R = REFRIGERATOR
 S = SINK

FIGURE 5.4-2. Reference Laboratory for the Manufacture of ^{14}C -Labeled Compounds

There are four fume hoods in the reference laboratory. Each hood is of steel construction with interior surfaces of stainless steel. The hoods contain the vacuum lines and distillation apparatus needed for the synthesis and purification of the labeled compounds. One hood is used as a weighing hood and contains a balance.

There are four glove boxes in the reference laboratory. Each glove box is fabricated from steel with an acrylic plastic viewing panel and neoprene gloves. The following special operations are carried out in the glove boxes:

- #1 - handling light powders
- #2 - handling toxic or biologically hazardous chemicals
- #3 - scraping thin-layer plates
- #4 - operations that require an inert atmosphere.

Each hood or glove box is separately vented by a roughing and a HEPA filter.

There is one stainless steel sink in the reference facility. It is used for hand washing and for rinsing laboratory glassware. Low levels of radioactivity are discharged to the sanitary sewer via the sink. The specific activity of liquid waste discharged via the sink is approximately 1.5×10^{-5} $\mu\text{Ci}/\text{ml}$. About 200 millicuries of ^{14}C is discharged in this manner each year.

The reference laboratory contains 15 linear meters of laboratory workbenches where apparatus is assembled and nonradioactive solutions are prepared for use in the labeling process. The workbenches are of wood construction and are painted with a latex enamel paint. The bench tops are covered with plastic-laminate.

Refrigerators, freezers, and storage cabinets are provided for the storage of labeled compounds and nonradioactive chemicals.

The laboratory is air conditioned to control the movement of air, to remove dust from the air, and to maintain constant temperature and humidity. Air enters the room through air diffusers in the ceiling. Exhaust air is removed by the fume hoods. Roughing and HEPA filters are located in the exhaust ductwork immediately above each hood. There is no additional filtering of air before it leaves the room. The exhaust ductwork is fabricated from sheet metal.

Normally, about 100 mCi of ^{14}C is processed in the reference facility at any given time. However, as much as 10 Ci might be handled on occasion. Labeled compounds are packaged in glass or plastic vials, glass ampoules, and breakseal tubes with activities per package ranging from a fraction of a microcurie to tens of millicuries.

Assumed residual surface contamination levels at the time of decommissioning the facility are given in Table 5.4-4. Values listed in the table represent removable contamination as determined by wiping the area with dry filter or soft absorbent paper and counting the wipe sample with an appropriate instrument of known efficiency. The contamination is entirely from ^{14}C .

5.4.4 Laboratory for the Manufacture of ^{125}I -Labeled Compounds

The reference laboratory for the manufacture of ^{125}I -labeled compounds is shown schematically in Figure 5.4-3. Work enclosures and equipment are provided for the exchange labeling of chemical compounds with ^{125}I , generally in the form of sodium iodide.

TABLE 5.4-4. Residual Surface Contamination Levels Prior to Decommissioning the Reference Laboratory for the Manufacture of ^{14}C -Labeled Compounds^(a)

<u>Facility Surface</u>	<u>Removable Contamination per 100 cm² of Surface</u>
Walls	Generally less than 1 000 d/m, with some spots as high as 10 000 d/m
Floors	Generally less than 1 000 d/m, with some spots as high as 50 000 d/m
Bench Tops	Generally less than 1 000 d/m, with some spots as high as 50 000 d/m
Drains	From 10 000 to 50 000 d/m
Inside Hoods	Generally less than 100 000 d/m, with some spots in excess of 1 000 000 d/m
Inside Glove Boxes	Generally about 1 000 000 d/m, with some spots in excess of this value
Inside Ventilation Ductwork	Generally less than 100 000 d/m, with some spots in excess of 1 000 000 d/m

(a) Contamination levels are based on experience at a large commercial laboratory for the manufacture of labeled compounds (New England Nuclear Corporation).

Iodine-125 has a radioactive half-life of 60 days. It decays by electron capture to the 0.035 MeV level of ^{125}Te . Therefore, the decay of ^{125}I results in the emission of very low energy gamma rays and tellurium x-rays. Some properties of ^{125}I are discussed in Appendix A of Reference 3.

Labeling of chemical compounds with radioactive iodine is usually accomplished by exchange methods. The reference facility contains the equipment needed for chemical operations of labeling and purifying the product. Because of the high radiotoxicity of radioactive iodine, labeling operations are performed in specially constructed glove boxes located inside fume hoods. This arrangement improves ventilation control and minimizes the contamination of the room atmosphere and laboratory surfaces by radioactive iodine.

The reference laboratory has 48 m² of floor space and is designed for occupancy by three people. The walls and the ceiling are concrete sealed with an epoxy paint. The floor is concrete covered with asphalt tile. Overhead utility lines, ventilation ductwork, and fluorescent light fixtures are all suspended from the concrete ceiling.

B = BENCH
 C = CABINET
 H = HOOD
 R = REFRIGERATOR
 S = SINK
 SH = SHELVES

FIGURE 5.4-3. Reference Laboratory for the Manufacture of
¹²⁵I-Labeled Compounds

Major facility components include four fume hoods with a specially designed glove box located inside each hood. The fume hoods are of standard size and are constructed of steel with stainless steel interiors. The glove boxes are constructed entirely of acrylic plastic and are 1.2 m wide, 0.6 m deep, and 0.6 m high. Each glove box is equipped with an activated charcoal filter. The glove box vents to the fume hood in which it is located. Each fume hood is equipped with a charcoal filter.

Room air is also exhausted through the fume hoods. A charcoal filter, a roughing filter, and a HEPA filter are located in the ductwork at the point where the exhaust air leaves the room. The ductwork is fabricated from sheet metal.

The reference facility contains one stainless steel sink that is used for personal cleanliness and for washing nonradioactive glassware. Liquid effluent from this sink is discharged to a tank where it is held for radioactive decay, monitored, and diluted as necessary before discharge to the sanitary sewer. Waste liquids from processing operations are not discharged to the laboratory sink, but are disposed of as described in Section 5.4.7.

The reference laboratory contains 8 linear meters of laboratory workbench surface, and a refrigerator, a storage cabinet, and shelves for the storage of labeled chemical compounds and nonradioactive chemicals.

Approximately 1 curie of ^{125}I is processed in the reference facility at any given time. Labeled compounds are packaged in glass or plastic vials. The radioactivity per package ranges from 1 microcurie to 1 millicurie.

Assumed residual surface contamination levels at the time of decommissioning the facilities are given in Table 5.4-5. Values listed in the table represent removable contamination as determined by wiping the area with dry filter or soft absorbent paper and counting the wipe sample with an appropriate instrument of known efficiency. The contamination is entirely from ^{125}I .

5.4.5 Laboratory for the Manufacture of ^{137}Cs Sealed Sources

The reference laboratory for the manufacture of ^{137}Cs sealed sources is shown schematically in Figure 5.4-4. Work enclosures and equipment are provided in the laboratory for processing ^{137}Cs to fabricate a source and seal it in a metal capsule.

Cesium-137 has a radioactive half-life of 30 years. It decays by beta decay to ^{137}Ba . The decay scheme is shown in Figure 5.4-5. In approximately 94% of the decays, a gamma photon with an energy of 0.662 MeV is emitted. The half-value thickness in lead for a 0.662 MeV gamma ray is about 6 mm (6.7 g/cm^2). To protect workers from excessive exposure to radiation, several centimeters of lead shielding are required for work with millicurie to curie quantities of ^{137}Cs .

TABLE 5.4-5. Residual Surface Contamination Levels Prior to Decommissioning the Reference Laboratory for the Manufacture of ^{125}I -Labeled Compounds^(a)

<u>Facility Surface</u>	<u>Removable Contamination per 100 cm² of Surface</u>
Walls	Generally less than 100 d/m, with some spots as high as 1 000 d/m
Floors	Generally less than 100 d/m, with some spots as high as 1 000 d/m
Bench Tops	Generally less than 100 d/m, with some spots as high as 1 000 d/m
Drains	Less than 100 d/m
Inside Hoods	Generally less than 100 d/m, with some spots as high as 1 000 d/m
Inside Glove Boxes	Microcurie levels
Inside Ventilation Ductwork	200 to 2 000 d/m

(a) Contamination levels are based on experience at a large commercial laboratory for the manufacture of labeled compounds (New England Nuclear Corporation).

The manufacture of sealed sources is described in Appendix A of Reference 3. For a ^{137}Cs source, the process begins with the radioactive material in the form of cesium-oxide powder. The radioactive material is converted to anhydrous chloride and fused with alumina by heating the mixture to a high temperature. The fused material is then doubly encapsulated in stainless steel. Most of the steps in the fabrication process (except for final encapsulation and testing of individual sources) are performed in a small hot cell using remote handling equipment.

Up to 50 curies of ^{137}Cs may be processed in a hot cell at any given time. Individual sources range in strength from 1 millicurie up to 10 curies.

The reference laboratory has 48 m² of floor space and is designed for occupancy by three people. The walls and the ceiling are concrete painted with latex enamel paint. The floor is concrete covered with asphalt tile. Overhead utility lines, ventilation ductwork, and fluorescent light fixtures are all suspended from the concrete ceiling.

B = BENCH
 C = LOCKED CASK
 H = HOOD
 HC = HOT CELL
 S = SINK

FIGURE 5.4-4. Reference Laboratory for the Manufacture of ^{137}Cs Sealed Sources

FIGURE 5.4-5. Radioactive Decay Scheme for ^{137}Cs

Major facility components include two small hot cells constructed from interlocking lead bricks, two fume hoods, 4 linear meters of laboratory work-bench surface, and locked storage casks for the storage of radioactive material awaiting processing and for manufactured sources. The hot cell and the fume hood are described in Appendix A of Reference 3. A sliding lead door allows material and equipment to be moved into and out of a hot cell. Entrance is by way of the fume hood located adjacent to the cell. Operations performed in a hot cell include opening the cesium-oxide capsule, chemical conversion of the radioactive material to the anhydrous chloride, heating the material in a furnace to fuse it with alumina, and encapsulation of the source in an inner capsule. Operations performed in a hood include sealing the inner capsule in an outer capsule and leak testing and wipe testing the completed assembly.

A single stainless steel sink, used for personal cleanliness, is located in the reference facility. The effluent from the sink goes to a centrally located hold-up tank for monitoring and dilution prior to discharge to the sanitary sewer. Liquid waste generated during processing operations is disposed of as described in Section 5.4.7.

The laboratory is air conditioned to control the movement of air, to remove dust from the air, and to maintain constant temperature and humidity. Air enters the laboratory through air diffusers in the ceiling and is exhausted via the fume hoods and hot cells. Each hot cell or fume hood is equipped with a roughing and a HEPA filter for filtration of the air leaving the component. An additional roughing and HEPA filter are located at the point where the exhaust ductwork from the room joins the building exhaust plenum. Ventilation ductwork is fabricated from sheet metal.

Assumed residual surface contamination levels at the time of decommissioning the facility are given in Table 5.4-6. Values listed in the table represent removable contamination as determined by wiping the area with dry filter or soft absorbent paper and counting the wipe sample with an appropriate instrument of known efficiency. The contamination is entirely from ^{137}Cs .

TABLE 5.4-6. Residual Surface Contamination Levels Prior to Decommissioning the Reference Laboratory for the Manufacture of ^{137}Cs Sealed Sources^(a)

<u>Facility Surface</u>	<u>Removable Contamination per 100 cm² of Surface</u>
Walls	Generally less than 1 000 d/m, with some spots as high as 10 000 d/m
Floors	Generally less than 1 000 d/m, with some spots as high as 50 000 d/m
Bench Tops	Generally less than 1 000 d/m, with some spots as high as 50 000 d/m
Drains	Generally less than 1 000 d/m, with some spots as high as 50 000 d/m
Inside Hoods	1 to 10 microcuries
Inside Hot Cells	10 millicuries to 1 curie
Inside Ventilation Ductwork	1 000 to 50 000 d/m

(a) Contamination levels are based on experience at a large commercial laboratory for the manufacture of sealed sources (New England Nuclear Corporation).

5.4.6 Laboratory for the Manufacture of ^{241}Am Sealed Sources

The reference laboratory for the manufacture of ^{241}Am sealed sources is shown schematically in Figure 5.4-6. Work enclosures and equipment are provided for processing ^{241}Am and fabricating doubly encapsulated x-ray excitation sources.

Americium-241 decays by alpha emission with a radioactive half-life of 458 years. The alpha decay is usually accompanied by the emission of a gamma photon with an energy of 0.103 MeV (13% of the decays) or 0.059 MeV (86% of the decays). Thus, ^{241}Am is a useful source of low-energy gamma radiation.

B = BENCH
 GB = GLOVE BOX
 H = HOOD
 T = TRANSFER TUNNEL

FIGURE 5.4-6. Reference Laboratory for the Manufacture of ^{241}Am Sealed Sources

The manufacture of ^{241}Am sources is described in Appendix A of Reference 3. X-ray excitation sources are prepared by incorporating americium-oxide powder in an aluminum-oxide matrix and compressing the mixture into a capsule that is sealed by arc welding. Glove boxes, which are maintained at a slightly negative pressure, are used to prepare and encapsulate the ^{241}Am sources. The transfer of radioactive material between adjacent glove boxes is made through transfer tunnels. Fume hoods are used to perform the second encapsulation and to wipe test the sources. Up to 100 curies of ^{241}Am may be processed in the reference facility at any given time. Individual sources range in strength from 1 to 300 millicuries.

The reference laboratory has 60 m² of floor space and is designed for occupancy by three people. The walls and ceiling are concrete sealed with an acrylic paint. The concrete floor is covered with linoleum. All linoleum seams are heat sealed. The linoleum is turned up at the walls to form a 0.15-m cove base.

Major facility components include two fume hoods, seven glove boxes, a storage cabinet for nonradioactive supplies, and a small laboratory workbench. Radioactive material is stored in lead shields.

Fume hoods are of the standard size and construction described in Appendix A of Reference 3. Glove boxes are 1.2 m wide, 0.6 m deep, and 0.6 m high and fabricated from stainless steel with acrylic plastic viewing panels and neoprene gloves. The following operations are performed in the glove boxes:

- #1 - receive encapsulated americium oxide
- #2 - open capsule; weigh radioactive material
- #3 - prepare americium oxide-aluminum oxide matrix
- #4 - encapsulate source
- #5 - seal source by arc welding
- #6 - remove encapsulated source.

Roughing and HEPA filters are attached to each hood or glove box to filter the exhaust air. A second absolute filter system is in series with the main duct system prior to release to the environment.

There are no sinks or drains in the room. Minimal quantities of liquid effluent are generated in this source-fabrication operation. (Liquids are used only in Glove Box #6 or Hood #2, where capsules are cleaned by wiping with an absorbent material wetted with an organic solvent.)

The laboratory is air conditioned to control the movement of air, to remove dust from the air, and to maintain constant temperature and humidity. Air enters the room through air diffusers in the ceiling and is exhausted through the fume hoods. Additional filtration of the exhaust air is provided by roughing and HEPA filters located at the point where exhaust ductwork from the room joins the building exhaust plenum. Ductwork is constructed from polyvinylchloride (PVC) pipe.

Assumed residual surface contamination levels at the time of decommissioning the facility are given in Table 5.4-7. Values listed in the table represent removable contamination as determined by wiping the area with dry filter or soft absorbent paper and counting the wipe sample with an appropriate instrument of known efficiency. The contamination is entirely from ^{241}Am .

TABLE 5.4-7. Residual Surface Contamination Levels Prior to Decommissioning the Reference Laboratory for the Manufacture of ^{241}Am Sealed Sources^(a)

<u>Facility Surface</u>	<u>Removable Contamination per 100 cm² of Surface</u>
Walls	Less than 20 d/m
Floors	Generally less than 20 d/m, with some spots as high as 1 000 d/m
Bench Tops	Generally less than 20 d/m, with some spots as high as 200 d/m
Inside Hoods	Generally about 1 000 d/m, with some spots as high as 5 000 d/m
Inside Glove Boxes	10 microcuries to 1 millicurie
Inside Ventilation Ductwork	Generally less than 50 d/m, with spots as high as 1 000 d/m

(a) Contamination levels are based on experience at a large commercial laboratory for the manufacture of labeled compounds (New England Nuclear Corporation).

5.4.7 Housekeeping and Waste Disposal Procedures

The NRC has published a regulatory guide⁽¹⁰⁾ on health physics surveys for byproduct material at NRC-licensed processing and manufacturing plants. The guide describes methods and procedures that are acceptable to the NRC staff for establishing survey programs that conform with the as low as is reasonably achievable (ALARA) philosophy. The frequency of routine surveys depends on the nature, quantity, and use of radioactive materials, as well as the specific protective facilities, equipment, and procedures that are designed to protect workers from external and internal exposure. The minimum acceptable survey frequencies that are considered by the NRC staff to meet the requirements of 10 CFR 20.101⁽¹¹⁾ are given in Table 5.4-8. Recommended action levels for removable surface contamination in manufacturing plants, also from

TABLE 5.4-8. Acceptable Frequencies for Health Physics Surveys^(a)

Radionuclide Group	External Radiation Surveys (Nuclides with Asterisks Only)*		Amounts (curies) in Process at Any One Time or Placed into Process in Any 3-Month Period Within Any Room Requiring Surveys					
	Weekly	Monthly	Air Sampling**			Surface Contamination		
			Weekly	Monthly	Quarterly	Weekly	Monthly	Quarterly
I: H-3, C-14, F-18,* K-42,* Cu-64,* Tc-99m,* In-113m*	If point source of activity could exceed 50 mrad/h at 1 meter	If point source of activity could exceed 0.5 mrad/h at 1 meter	≥ 10	≥ 1 < 10	< 1	≥ 100	≥ 10 < 100	< 10
II: Br-82, Cr-51,* Fe-55, I-123,* Hg-197*			≥ 1	≥ 0.1 < 1	< 0.1	≥ 10	≥ 1 < 10	< 1
III: S-35, Au-198, Ca-47, I-132, Ce-141, Mixed fission products,* Sr-85, La-140, Nb-95, Zn-65, Co-58,* Fe-59,* Na-24,* Co-57,* Se-75,* MO-99*			≥ 0.1	≥ 0.01 < 0.1	< 0.01	≥ 1	≥ 0.1 < 1	< 0.1
IV: Hf-181, Pm-147, P-32,* Ba-140,* Th-234, Kr-85, Ir-192,* Cl-36, Y-91, Ta-182, Ca-45, Sr-89, Cs-137, Co-60,* Ce-144,* I-126, Eu-154, I-131,* I-125,* Tm-170, Na-22,* Mn-54,* Ag-110m,* Hg-203,* Rn-222,* Sn-113*			≥ 0.01	≥ 0.001 < 0.01	< 0.001	≥ 0.1	≥ 0.001 < 0.1	< 0.01
V: Tc-99, I-129, Ru-106			≥ 0.001	$\geq 10^{-4}$ < 0.01	< 10^{-4}	≥ 0.01	≥ 0.001 < 0.1	< 0.001
VI: Ra-223, Po-210, Th-227, Si-90, Pb-210, Cm-242, U-233*			$\geq 10^{-4}$	$\geq 10^{-5}$ < 10^{-4}	< 10^{-5}	≥ 0.001	$\geq 10^{-4}$ < 0.001	< 10^{-4}
VII: Sm-147, Nd-144, Ra-226,* Cm-244, Ra-228, Pu-241			$\geq 10^{-5}$	$\geq 10^{-6}$ < 10^{-5}	< 10^{-6}	$\geq 10^{-4}$	$\geq 10^{-5}$ < 10^{-4}	< 10^{-5}
VIII: Am-243, Am-241,* Np-237, Ac-227, Th-230, Pu-242, Pu-238, Pu-240, Pu-239, Th-228, Cf-252			$\geq 10^{-6}$	$\geq 10^{-7}$ < 10^{-6}	< 10^{-8}	$\geq 10^{-5}$	$\geq 10^{-6}$ < 10^{-5}	< 10^{-6}

* Nuclides with asterisks are those more likely to require external radiation surveys.

** Assuming continuous sampling is unnecessary.

(a) From Reference 1.

Reference 10, are given in Table 5.4-9. Any contamination in excess of the values shown in Table 5.4-9 requires immediate remedial action to reduce the level of residual radioactivity.

TABLE 5.4-9. Recommended Action Levels for Removable Surface Contamination in Manufacturing Plants^(a)

Type of Surface	Type of Radioactive Material ^(b)			
	Alpha Emitters		Beta or X-Ray Emitters ($\mu\text{Ci}/\text{cm}^2$)	Low-Risk Beta or X-Ray Emitters ($\mu\text{Ci}/\text{cm}^2$)
	High Toxicity ($\mu\text{Ci}/\text{cm}^2$)	Lower Toxicity ($\mu\text{Ci}/\text{cm}^2$)		
1. Unrestricted areas ^(c)	10^{-7}	10^{-7}	10^{-6}	10^{-6}
2. Restricted areas ^(d)	10^{-4}	10^{-3}	10^{-3}	10^{-2}
3. Personal clothing worn outside of restricted areas 10^{-7}	10^{-7}	10^{-6}	10^{-6}	
4. Protective clothing worn only in restricted areas	10^{-5}	10^{-5}	10^{-4}	10^{-4}

(a) From Reference 10.

(b) High toxicity alpha emitters include Am-243, Am-241, Np-237, Ac-227, Th-230, Pu-242, Pu-238, Pu-240, Pu-239, Th-228, and Cf-252. Lower toxicity alpha emitters include those having permissible concentrations in air greater than that for Ra-226(s) in 10 CFR Part 20, Appendix B, Table 1, Column 1. Beta or x-ray emitter values are applicable for all beta or x-ray emitters other than those considered low-risk. Low-risk nuclides include those whose beta energies are less than 0.2 MeV, whose gamma or x-ray emission is less than 0.1 R/h at 1 meter per curie, and whose permissible concentration in air in 10 CFR Part 20, Appendix B, Table 1, is greater than 10^{-6} mCi/mL.

(c) Contamination limits for unrestricted (non-contamination-controlled) areas in this table are considered to be compatible in level of safety with those for release of facilities and equipment for unrestricted use, as given in Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," and in "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material."

(d) Averaging is acceptable over inanimate areas of up to 300 cm^2 or, for floors, walls, and ceiling, 100 cm^2 . These limits are allowed only in those restricted areas where appropriate protective clothing is worn.

Solid and liquid wastes generated during manufacturing operations and routine housekeeping are packaged and shipped to a commercial shallow-land burial site for disposal. Packaging requirements are determined by federal regulations and by the specific requirements of the burial site. For the reference non-fuel cycle facilities examined in this study, the following waste packaging procedures are assumed:

1. Solid wastes are packaged in 208-l steel drums lined with a 0.15-mm-thick polyethylene liner.
2. Aqueous liquids are solidified by the addition of cement and are packaged in 208-l steel drums that have a 0.15-mm-thick polyethylene liner.
3. Organic liquids are absorbed on vermiculite, perlite, diatomaceous earth, or some other absorbent material that meets burial ground specifications. The waste is packaged in 113-l drums lined with polyethylene. The 113-l drums are overpacked in 208-l drums lined with polyethylene. The space between the two drums is filled with vermiculite, perlite, diatomaceous earth, or other approved absorbent material.

5.5 DESCRIPTION OF THE REFERENCE ORE PROCESSING SITE

The reference ore processing site of this study is a tailings pile containing uranium and thorium residues from processing operations for the recovery of rare metals from tin slag. The reference site description details can be found in Appendix A of Reference 3.

The term "rare-metals refinery" refers to an ore processing facility that separates nonradioactive rare metals from slags, raw ores, or ore concentrates that also contain licensable quantities of thorium and uranium. The radioactivity remains with the residue, which is stored in settling ponds or tailings piles. It is this stored residue that is of primary concern from a decommissioning standpoint. Some general information about rare-metals refinery tailings is given in Appendix A of Reference 3.

The reference rare-metals refinery is a plant that refines raw material for the recovery of tantalum and niobium. The raw material is the slag produced by tin smelters located on the Malay Peninsula. The tin slag consists of glassy flakes or pellets that contain, in addition to the tantalum and niobium, approximately 0.1 to 0.5 wt% U_3O_8 and ThO_2 . The ^{238}U and ^{232}Th are assumed to be in secular equilibrium with their daughter products.

The tin slag arrives at the ore processing plant by rail in hopper cars or by truck in 208-l drums. At the plant it is stored in the steel drums in which it arrives or in large concrete storage bins. To begin the rare metals recovery operation, the slag is ground, roasted, and digested with hydrofluoric acid. The hydrofluoric acid is filtered off and passed to a facility for the chemical extraction of niobium and tantalum. The sludge, which contains essentially all of the thorium and uranium radioactivity, is pumped to a settling pond located about 100 m from the refinery. In the settling pond, the water is allowed to evaporate, converting the sludge to a glassy solid.

Figure 5.5-1 shows a schematic of the reference settling pond. The pond is 100 m long by 50 m wide by 5 m deep, with a 2-1/2 to 1 slope on each side. The soil from which the pond is excavated is clayey silt that extends to a depth of about 10 m. Beneath the clay deposit is a 10-m-thick layer comprised of a mixture of coarse sand, coarse to medium gravel, and some rock fragments. A layer of shale is located beneath the sand and gravel layer. The water table is at a depth of about 15 m below the ground surface. The nearest river is assumed to be located at a distance of 1 km.

The settling pond is assumed to be unlined. The low permeability of the clay restricts, but does not entirely prevent, the percolation of moisture into the ground beneath the pond. This is probably representative of construction procedures at older ponds. To reduce percolation, newer settling ponds are lined with a fluorocarbon-type material approximately 1 mm thick. There is no covering over the glassy residue in the settling pond. However, because of the nature of the glassy surface, the airborne dust problem is not serious. The gamma dose rate measured at the edge of the pond is approximately 1 mrem/hr. Additional information about the reference tailings pond and its contents is given in Table 5.5-1. Some properties of the series of radioactive isotopes that exist in secular equilibrium with ^{238}U and ^{232}Th are listed in Tables 5.5-2 and 5.5-3.

For purposes of estimating site survey costs, the settling pond is assumed to be located on a $20,000\text{-m}^2$ site (approximately 5 acres).

FIGURE 5.5-1. Schematic of Reference Settling Pond

TABLE 5.5-1. Reference Rare Metals Refinery Tailings Pond Parameters

Parameter	Value
Volume of Pond	16 400 m ³
Density of Residue	2 500 kg/m ³
Weight of Residue	4.10 x 10 ⁷ kg
U ₃ O ₈ Concentration	0.2 wt%
ThO ₂ Concentration	0.5 wt%
Contained U ₃ O ₈	8.20 x 10 ⁴ kg
Contained ThO ₂	2.05 x 10 ⁵ kg
²⁶⁶ Ra Activity	23.2 Ci
²²⁸ Ra Activity	20.0 Ci

TABLE 5.5-2. Radioisotopes in Secular Equilibrium with ^{238}U

Radionuclide	Half-Life	Type of Disintegration	Decay Energy (MeV)
^{238}U	4.51×10^9 yr	Alpha	4.27
^{234}Th	24.1 day	Beta	0.26
^{234}Pa	6.75 hr	Beta	2.23
^{234}U	2.47×10^5 yr	Alpha	4.86
^{230}Th	8.0×10^4 yr	Alpha	4.77
^{226}Ra	1602 yr	Alpha	4.87
^{222}Rn	3.82 day	Alpha	5.59
^{218}Po	3.05 min	Alpha, Beta	6.11, 0.33
^{214}Pb	26.8 min	Beta	1.04
^{218}At	2 sec	Alpha	6.82
^{214}Bi	19.7 min	Alpha, Beta	5.62, 3.28
^{214}Po	1.64×10^{-4} sec	Alpha	7.84
^{210}Tl	1.3 min	Beta	5.50
^{210}Pb	21 yr	Beta	0.061
^{210}Bi	5.01 day	Beta	1.16
^{210}Po	138.4 day	Alpha	5.41
^{206}Tl	4.19 min	Beta	1.52
^{206}Pb	Stable		

TABLE 5.5-3. Radioisotopes in Secular Equilibrium with ^{232}Th

Radionuclide	Half-Life	Type of Disintegration	Decay Energy (MeV)
^{232}Th	1.41×10^{10} yr	Alpha	4.08
^{228}Ra	6.7 yr	Beta	0.055
^{228}Ac	6.13 hr	Beta	2.15
^{228}Th	1.91 yr	Alpha	5.52
^{224}Ra	3.64 day	Alpha	5.79
^{220}Rn	55 sec	Alpha	6.40
^{216}Po	0.15 sec	Alpha, Beta	6.91
^{212}Pb	10.6 hr	Beta	0.58
^{216}At	3×10^{-4} sec	Alpha	7.95
^{212}Bi	60.6 min	Alpha, Beta	6.21, 2.25
^{212}Po	3.0×10^{-7} sec	Alpha	8.95
^{208}Tl	3.10 min	Beta	4.99
^{208}Pb	Stable		

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6.0 REFERENCE ACCIDENT SCENARIOS AND RESULTING CONTAMINATION LEVELS

Reference accident scenarios described in this chapter provide the bases for determining contamination levels and the condition of the facility following an accident. This information is used in the analysis of decommissioning cost and safety estimates in subsequent chapters. In considering post-accident decommissioning, the consequences of an accident (i.e., the radiological and physical condition of the plant following an accident) are more important than the sequence of events that occur during the accident. Reference accident-consequence scenarios provide information about radioactive contamination, physical damage, and radiation exposure rates that result from the postulated accidents. Scenarios have been selected that could potentially occur but have been provided for in safety systems design.

Environmental reports prepared as part of the licensing process for fuel cycle facilities and selected research reports include descriptions of a spectrum of postulated accidents, including criticality, fires, and explosions. The reference scenarios developed in this chapter are derived from a variety of sources, including fuel cycle facility environmental reports and various referenced research studies.

For this post-accident decommissioning study, accident consequence scenarios are reviewed that are illustrative of a range of decommissioning requirements and for which the occupational radiation doses and costs of decommissioning are substantially greater than those estimated in previous reference studies of fuel cycle facility decommissioning following normal shutdown. Scenarios are not restricted to accidents that have occurred, but an attempt is made to choose credible scenarios.

Accidents are unique occurrences. Their consequences depend for the most part upon the sequence of events leading to and following the initial malfunction and to the amount and character of radioactive material present. For this reason, not every conceivable accident is covered. A range of potential accidents is discussed, leading to the selection of a release scenario having the potential for the greatest consequences for decommissioning. The consequences of these hypothetical accidents are theoretical and should not necessarily be expected to occur during actual plant operation.

In examining various accidents deemed credible in the reference facilities, it becomes obvious that the effects on decommissioning would be inconsequential unless a significant release of radioactivity to the buildings occurred. With the amount of radioactive material limited in each process area, the typical operationally induced accidents result only in inconvenient in-plant conditions with negligible effects. An accident that results in extensive radioactive contamination and major physical damage to the facilities

(such as might occur as the result of a major facility fire and explosion) is the most serious accident considered.

The reference accident scenarios are described in the following sections. Parameters that characterize these scenarios include:

- amounts of radioactivity released
- radioactive contamination levels
- radiation exposure levels
- damage to the plant buildings and equipment.

The contamination condition of a nuclear facility following an accident is related to the radioactivity released during the accident. The radioactivity release is dependent on the source term (i.e., on the amount of radioactivity present at the time of a postulated accident) and on the release fraction postulated.

The reference accident scenarios for each facility are summarized in Table 6.0-1 and are described in more detail in the following sections. For the reference facilities, source terms that provide the bases for estimates of radioactive contamination, estimated inventories of radioactivity released, details of contamination levels, radiation exposure rates, and the extent of damage to buildings are provided for each of the reference accident scenarios.

6.1 ACCIDENTS AT A REFERENCE SMALL MIXED-OXIDE FUEL FABRICATION PLANT

The reference MOX plant produces mixed oxide fuel, so an accident in this plant will involve plutonium and uranium powders and solutions and hazardous chemicals. Information from MOX plant accident scenarios described in environmental reports,⁽¹⁻⁵⁾ the GESMO study⁽⁶⁾ and research reports⁽⁷⁻⁹⁾ is used to derive the reference scenarios presented in this section.

Source terms that provide the bases for estimates of radioactive contamination following the postulated accidents are described in Section 6.1.1. Potential accident scenarios that might occur at a MOX plant are discussed in Section 6.1.2, followed by a discussion of the selected reference scenarios in Section 6.1.3. Discussions on the contamination and exposure rates expected for the reference accident scenarios are provided in Sections 6.1.4 and 6.1.5.

6.1.1 Accident Source Terms

The contamination level in the reference MOX plant following an accident is dependent on the radioactivity present in the fuel mixture at the time of the accident. The number of curies of each radionuclide present in the reference MOX fuel mixture is listed in Table 6.1-1, taken from Reference 10. The column labeled "Time of Fuel Fabrication" contains the calculated radioactivity

TABLE 6.0-1. Summary of Accident Scenarios

Small MOX Plant	
<u>Accident Parameter</u>	<u>Accident Description</u>
• Accident	Explosion followed by major facility fire
• Inventory at Risk	125 kg Pu
• Release Mechanism	1. Blending tank failure 100% liquid release 2. 4% aerosol release due to fire
• Radioactivity Released	
- Inside Building	35 kg Pu
- Outside Building	5 g Pu
• Physical Damage to Building	Fire damage to interior, building exterior not breached

U-Fab Plant		
<u>Accident Parameter</u>	<u>U-Fab Plant Accident Description</u>	
• Accident Description	UF ₆ cylinder rupture	UF ₆ cylinder rupture (2 cylinders damaged)
• Inventory at Risk	2160 kg UF ₆	4400 kg UF ₆
• Release Mechanism	UF ₆ vaporization 1/3 of inventory released	UF ₆ vaporization 1/3 of inventory released
• Radioactivity Released		
- Inside Building	700 kg UF (440 kg U ₆)	None
- Outside Building	2 kg UF ₆ (~1 kg U)	1400 kg of UF ₆
• Physical Damage to Building	None	None

TABLE 6.0-1. (contd)

Accident Parameter	Non-Fuel Cycle Facility Accident Description	
• Accident	Major facility fire	Fire in storage area
• Inventory at Risk	Max. inventory in each laboratory	Max. inventory in plant storage area
• Release Mechanism	100% of ^3H , ^{14}C , ^{125}I 1% of ^{137}Cs and ^{241}Am	100% of ^3H , ^{14}C , ^{125}I 1% of ^{137}Cs and ^{241}Am
• Radioactivity Released		
- Inside Building	2000 Ci ^3H , 10 Ci ^{14}C , 1 Ci ^{125}I , 0.5 Ci ^{137}Cs , 1 Ci ^{241}Am	4000 Ci ^3H , 15 Ci ^{14}C , 4 Ci ^{125}I , 1 Ci ^{241}Am
- Outside Building	None	None
• Physical Damage to Building	Fire Damage to Building	Fire Damage to Building

Accident Parameter	Ore Processing Facility Accident Description
• Accident	Tailings dam failure
• Inventory at risk	8.2×10^4 kg of U_3O_8 2.05×10^5 kg ThO_2
• Release mechanism	Waterborne solids released from tailings pond
• Radioactivity Released	Uranium and daughters

TABLE 6.1-1. Radioactivity of Mixed-Oxide Fuel at the Time of Fabrication, Shutdown, and Various Times After Plant Shutdown

Nuclide	Curies/MTHM (Except when shown otherwise)								
	At Time of Fuel Fabrication ^(a)	At Time of Operational Shutdown ^(b)	Grams/MTHM	Materials Held up in Plant (Various Times After Shutdown in Years)					
				5	10	20	30	50	100
²³¹ Th	1.17E-2 ^(c)	1.18E-2	2.22E-8	1.18E-2	1.19E-2	1.20E-2	1.21E-2	1.24E-2	1.30E-2
²³⁴ Th	2.55E-1	2.57E-1	1.11E-5	2.55E-1	2.55E-1	2.55E-1	2.55E-1	2.55E-1	2.55E-1
²³³ Pa	8.35E-5	3.98E-3	1.94E-7	1.09E-2	2.08E-2	4.71E-2	7.92E-2	1.52E-1	3.43E-1
^{234m} Pa	2.55E-1	2.57E-1	3.74E-10	2.55E-1	2.55E-1	2.55E-1	2.55E-1	2.55E-1	2.55E-1
²³⁴ Pa	2.55E-4	2.57E-4	1.30E-10	2.55E-4	2.55E-4	2.55E-4	2.55E-4	2.55E-4	2.55E-4
²³⁴ U	4.81E-1	5.02E-1	8.11E+1	5.22E-1	5.42E-1	5.79E-1	6.13E-1	6.74E-1	7.91E-1
²³⁵ U	1.17E-2	1.18E-2	5.50E+3	1.18E-2	1.19E-2	1.20E-2	1.21E-2	1.24E-2	1.30E-2
²³⁶ U	1.74E-4	1.04E-3	1.64E+1	1.91E-3	2.77E-3	4.51E-3	6.23E-3	9.69E-3	1.83E-2
²³⁷ U	9.36E+0	7.50E+0	9.18E-5	5.92E+0	4.68E+0	2.93E+0	1.83E+0	7.19E-1	6.92E-2
²³⁸ U	2.55E-1	2.55E-1	7.64E+5	2.55E-1	2.55E-1	2.55E-1	2.55E-1	2.55E-1	2.55E-1
²³⁷ Np	1.03E-4	3.98E-3	5.64E+0	1.09E-2	2.08E-2	4.71E-2	7.92E-2	1.52E-1	3.43E-1
²³⁸ Pu	1.54E+3	1.48E+3	8.78E+1	1.43E+3	1.37E+3	1.27E+3	1.17E+3	1.00E+3	6.81E+3
²³⁹ Pu	1.22E+4	1.22E+4	1.98E+5	1.22E+4	1.22E+4	1.22E+4	1.22E+4	1.22E+4	1.21E+4
²⁴⁰ Pu	5.98E+3	5.98E+3	2.71E+4	5.98E+3	5.97E+3	5.97E+3	5.96E+3	5.95E+3	5.92E+3
²⁴¹ Pu	3.74E+5	2.99E+5	2.98E+3	2.36E+5	1.87E+5	1.17E+5	7.33E+4	2.87E+4	2.76E+3
²⁴² Pu	1.34E+0	1.34E+0	3.45E+2	1.34E+0	1.34E+0	1.34E+0	1.34E+0	1.34E+0	1.34E+0
²⁴¹ Am	6.12E+2	3.17E+3	9.25E+2	5.27E+3	6.90E+3	9.16E+3	1.05E+4	1.17E+4	1.16E+4
TOTAL	3.94E+5	3.22E+5	1.00E+6	2.61E+5	2.13E+5	1.46E+5	1.03E+5	5.96E+4	3.31E+4

(a) Radioactivity based on assumptions listed in Section C.1 of Reference 10 (1 year after reprocessing)

(b) Based on 1 MTHM mixed-oxide fuel held up over 10-year life of plant (1/10 MTHM held up per year).

(c) 1.17E-2 is equivalent to 1.17×10^{-2} .

of the reference MOX fuel mixture at the time it is fabricated into fuel pellets. The columns labeled "Time of Operational Shutdown" give the calculated total quantities (curies and grams) of each radionuclide per MTHM of the reference MOX fuel mixture held up in the plant over its 10-year operating life (1/10 accumulated per year). The remaining columns depict the number of curies of each radionuclide per MTHM at various times after plant operation and shutdown, taking into account radioactive decay.

6.1.2 Potential Accident Scenarios

There are four criteria for selection of potential accidents: amount of plutonium present, difficulty of generating plutonium releases, probability of occurrences, and exposure by other means (e.g., criticality). Based on this set of criteria, the attention is focused on four process areas or steps in the MOX fabrication process. These areas are fuel storage, conversion, powder treatment, and scrap recovery. In other areas, the fuel material is not easily dispersible, it is diluted by UO_2 or contained, or was present only in small quantities. A large quantity of dispersible plutonium is in tanks during blending of the nitrate solution.

Potential accidents discussed below for the MOX plant are criticality, explosion, fire, major facility fire, and a loading dock accident.

Criticality Accident

As an example of a criticality accident in the reference MOX facility, a criticality event is assumed in an "unsafe" tank of plutonium nitrate.⁽⁷⁾ Simplifying assumptions are made to maximize the consequences and are important only in the mechanism(s) and the rate of terminating the criticality. It is assumed that the accident is caused by exceeding the critical volume by 10 liters during the filling of the tank, resulting in 2×10^{18} total fissions. Termination of the event is assumed to be by evaporation and explosion of the 10-liter excess volume from the tank. An 8-inch wall of ordinary concrete is assumed to shield the plant environs from the direct radiation from the accident.

As a result of the excursion, fission products are formed and a large amount of energy is released. About 85% of the approximate 200 MeV released per fission is available for heating the solution. The energy input to the solution provides the power-limiting and ultimate terminating mechanism for the accident. For this accident greater than 6×10^6 calories of heat energy is initially available to the system. The solution would, more than likely, critically pulse at an "equilibrium" fission rate, until the solution is below the critical volume.

Vaporization of some of the solution during the initial power pulse could cause a rapid volume expansion or pressurization of sufficient magnitude to rupture the glove box. This is assumed to occur.

The excursion is assumed to terminate following the evaporation of the 10 liters of excess solution. Airborne release fractions for plutonium in nitrate solution vary widely, depending upon temperature and activity of the solution, the plutonium concentration, and air flow over the solution. The heat energy from the excursion would raise a 1000-liter solution about 15°F if the heat were uniformly distributed. However, it was conservatively assumed that all of the heat was used to create localized boiling, vaporization of the excess volume, and surface film breakup. A release fraction of 0.2% was conservatively used for this accident. Work by Mishima et al.⁽¹¹⁾ indicates that as much as 0.18% of the plutonium in a dilute solution was made airborne during evaporation of approximately 90% of the solution in a deep form beaker of the solution at a rolling boil. For less extreme conditions, the release fractions would be in the range of 10^{-2} to 10^{-4} %.

Assuming a solution containing 150 g of plutonium per liter, and assuming that 1500 g of plutonium are released to the building, a maximum of 3 g of plutonium could become airborne. The hood filter is assumed to have plugged, releasing everything into the room and out through the final filter system.

The fission products generated as a result of the criticality event are also considered. For this analysis, two simplifying assumptions are made to determine the maximum condition possible:

- All of the noble gases generated are released from the building. One fourth of the iodine is released. The material is released over a 10-minute interval.
- The radioactive material is released at ground level and dispersion is during a Pasquill type-F meteorological inversion.

The inventory of fission products generated during the criticality is calculated using the computer code RIBD.⁽¹²⁾ The number of curies for groupings of radionuclides at selected times after termination of the event is tabulated in Table 6.1-2. Cumulative yields for thermal neutrons are used to determine the activity of the fission products. The volatile solids and remaining fission products are not released from the building to any significant extent.

The source terms for this accident are: Quantity - 1500 g of plutonium released to the building with 3 g of plutonium airborne in the building; Chemical Form - soluble; Duration of Release - less than 2 hours.

TABLE 6.1-2. Fission Product Inventory from Criticality Accident^(a)10¹⁸ Total Fissions (32 MW-sec) Quantity in Curies

Nuclide Classification	Duration of Event	Decay Time						
		Shutdown	10 min	60 min	2 hr	5 hr	10 hr	1 day
Noble Gases	0.5 sec	2.5E6 ^(c)	2.4E3	2.1E2	1.1E2	5.9E1	3.2E1	1.4E1
	5 min	3.8E4	1.8E3	2.0E2	1.1E2	5.9E1	3.2E1	1.4E1
	1 hr	4.0E3	5.3E2	1.5E2	9.9E1	5.4E1	3.1E1	1.4E1
Halogens	0.5 sec	6.0E5	4.7E2	2.7E2	2.1E2	7.4E1	3.4E1	1.5E1
	5 min	2.1E4	3.7E2	2.7E2	2.1E2	7.4E1	3.4E1	1.5E1
	1 hr	2.1E3	2.8E2	2.5E2	1.8E2	6.5E1	3.2E1	1.4E1
Volatile Solids	0.5 sec	9.0E5	2.3E3	6.7E2	2.4E2	1.8E1	5.8E0	3.3E0
	5 min	3.4E4	2.0E3	6.4E2	2.3E2	1.8E1	5.8E0	3.3E0
	1 hr	4.1E3	1.1E3	4.2E2	1.5E2	1.4E1	5.6E0	3.3E0
ARFP ^(b)	0.5 sec	1.5E6	1.5E4	2.0E3	7.5E2	3.3E2	1.8E2	6.4E1
	5 min	1.1E5	1.2E4	1.9E3	7.3E2	3.3E2	1.8E2	6.4E1
	1 hr	1.6E4	4.7E3	1.2E2	6.0E2	3.1E2	1.7E2	6.2E1

^(a) From Reference 7.^(b) ARFP - All remaining fission products.^(c) 2.5E6 means 2.5×10^6 .

Explosions

Several types of explosions are conceivable in a MOX facility. The most common are explosions in sintering furnaces, autoclaves, and plutonium solution tanks. These types of events are localized and will result in damage only to a restricted area. No explosion can be conceived that would affect the entire facility and result in major damage to the building.

Hydrogen explosions in pellet sintering furnaces have a limited amount of energy. The damage that could result from this type of event would likewise result in limited consequences. The explosion would probably be directed out the ends of the furnace. The glove box could be breached, and pellets and a small amount of mixed oxide fines could be spread around the room. This type of accident would result in an in-plant contamination spread.

The rupture of an autoclave wall during operation could result in the high-speed projection of fuel pins at the building walls or ceiling. Building design against missiles precludes penetration of the structure from this type of accident. Significant source terms of plutonium are not credible either within or outside the building from this type of event.

An explosion involving a plutonium solution could result in the production of a substantial quantity of plutonium particles. Some liquid would be pushed out of the vessel and some plutonium particles would be generated. If sufficient force is not available to cause extensive film breakup, it seems probable that most of the liquid would impact on and adhere to adjacent surfaces. If an explosion is of sufficient magnitude to rupture a heavy-walled, closed vessel, a considerable number of fine particles could be generated by the liquid passing through the jagged opening. The number of particles that persist in the air is limited. Swain and Haberman reviewed data from non-nuclear sources and calculated that 33 mg of plutonium per cubic meter was the maximum airborne concentration a few minutes after an accident.⁽¹³⁾ Considerable care and effort are required to generate monodispersed particles. Generally, particles generated by a single mechanism are log-normally distributed. Thus, the anticipated weight concentration of 10- μ m AED particles would be much lower than that calculated for monodispersed particles. An upper value of 100 mg of plutonium per cubic meter for particles in the respirable range appears reasonable. Because of this upper-mass limit for airborne plutonium particles, the consequences for explosion accidents appear to be limited by the material that can be maintained in the air and not by the total volume or mass initially involved. An explosion will, of course, splatter the solution on the wall and floor and this solution may later become airborne as it dries.

Fire

The occurrence of a fire near the nitrate blending tanks represents one of the greatest potentials for release of plutonium. Airborne releases from fire around a blending tank can be envisioned in several ways. The fire could heat the liquid and cause it to boil. Boiling a dilute plutonium nitrate solution until a 90% volume reduction is attained could result in an airborne release no greater than 0.2%.⁽¹¹⁾ Fractional releases from concentrated solutions could be higher. After a substantial quantity of liquid is lost, the liquid thickens and vapor trapped under the surface tends to throw solids into the air. The salt content would be high and the tendency would be for the majority of material to form a coarse cake that is sintered to the surface of the vessel. Less than 0.5% of a uranium-nitrate solution (applicable to plutonium) as a thin film on a stainless-steel surface involved in a gasoline fire was made airborne by air at a nominal velocity of 2.5 mph passing over the surface.⁽¹⁴⁾ Approximately 80% of the uranium aerosol was in the respirable size range. A smaller fraction is anticipated under the accident conditions due to the lack of air-flow and depth of the vessel.

If the pressure is high enough, the vessel could rupture. If the rupture occurs above the liquid level, some plutonium would become airborne but not a significant quantity. If the rupture occurs near the bottom of the tank and is large, a large stream of solution would pour onto the floor and extinguish part or all of the fire. If the fire is extinguished, the fractional airborne release is low--less than 0.003%.⁽¹¹⁾ If the fire continues and dries the nitrate solution, the airborne release could be as high as 0.5%.

To estimate the amount of radioactivity released, 150 kg of plutonium as a nitrate in a 1000-liter blending solution was assumed to be present. If the blending tank is vented, a fire that boils the vessel dry could release as much as 0.7% of the plutonium present, or about 1050 g of plutonium into the available air space.⁽⁷⁾ If the vessel is closed and is ruptured by the internal pressure, the nitrate would spill on the floor and be dried by the fire. The fractional airborne release could be as high as 750 g of plutonium into the available air space.⁽⁷⁾ An internal detonation of significant magnitude could rupture the vessel and generate a large quantity of fine droplets into the available air space. For a maximum release, a conservative estimate assumes an instantaneous uniform air concentration. The airborne concentration would be approximately 100 mg of plutonium/m³ of air for a few minutes.

The source terms are Quantity - 1100 g of plutonium released to the building atmosphere; Particle Size - 80% in respirable range; Chemical Form - insoluble; Duration of Release - less than 2 hours.

Major Facility Fire

It would require a major disaster to breach facility confinement and release unfiltered plutonium to the environs. Only a few accidents in this "plant disaster" class could theoretically produce damage of sufficient magnitude to compromise the final confinement barriers of the plant. The plant is, at a minimum, structurally designed and built to satisfy criteria relative to earthquakes and tornadoes. However, finite possibilities exist that the facility could be stressed by forces beyond those used for design. Major facility fires also seem unlikely in buildings where combustibles are limited, but experience indicates that they can occur. Plant disaster accidents that can cause major facility damage are not "incredible," but highly improbable.

The bulk of plutonium in the facility will be in storage in a hardened area, like a vault. For the reference MOX facility, only 125 kg of plutonium will be in-process. About one-half of this plutonium will be in dispersible form.

A plant fire that could cause catastrophic breaching of the final barrier could not be conceived. With the expected concrete construction of a facility of this type, the low fire loading, and the expected airtight nature, total burning is incredible. For analysis, the final barrier is assumed to remain intact after a facility fire involving the glove boxes and other combustibles in process, and the deluge system is assumed to remain operable. It is estimated that less than 1% of the in-process dispersible inventory could become airborne with the facility. It is assumed that 125 kg of plutonium would be available for release.

The source terms would be: Quantity - 5 kg of plutonium; Particle size - all in respirable range; Chemical Form - insoluble; Duration of Release - extended (greater than 2 hours and less than 8).

6.1.3 Reference Accident Scenario

The reference accident scenario inside the reference MOX plant is assumed to be an explosion and major facility fire in the wet processing room involving the nitrate blending tanks. The nitrate blending tanks are 107-liter tanks each containing 15 kg of plutonium solution. The fire results in overheating and two tanks explode releasing 30 kg of plutonium solution to the waste treatment solvent extraction and wet process rooms. The plutonium is assumed to be spread throughout the rooms. The fire is assumed to spread to the scrap recovery and the pellet processing rooms before it is extinguished. It is also assumed that plutonium contamination is spread to the scrap recovery and pellet processing rooms due to the effects of the fire.

Four percent of the maximum involved inventory of 125 kg is assumed to be affected by the fire. The material is airborne and spread to all the affected rooms. In addition, it is picked up by the ventilation system. The pre-filters are assumed to fail during the fire or explosion. The final barrier, consisting of the building and HEPA filters, is assumed to remain intact. The building ventilation system is assumed to be contaminated by the fire up to the final HEPA filters.

It is conservatively assumed that the entire maximum in-process inventory of 125 kg of plutonium is affected by the fire. The release is estimated to be 4% of the inventory. Thus, an additional 5 kg of plutonium are assumed to be dispersed in the area affected by the fire. The building ventilation system is designed to prevent spread of contamination, and it is assumed that other rooms in the MOX plant are not affected by plutonium contamination from the fire and explosion.

For the maximum release outside the plant building, it is assumed that only the final HEPA filter is intact from the accident and the final filter is 99.9% efficient. The release of plutonium outside the plant building is assumed to be about 5 g of plutonium.

A summary of the releases from the reference scenario is shown in Table 6.1-3.

TABLE 6.1-3. Reference MOX Plant Accident Scenario

<u>Accident Parameter</u>	<u>Accident Scenario</u>
● Accident Description	Explosion followed by major facility fire
● Inventory at Risk	125 kg Pu
● Release Mechanism	Blending tank failure 100% liquid release 4% aerosol release due to fire
● Radioactivity Released in Accident	
- Inside Building	35 kg Pu
- Outside Building	5 g Pu
● Physical Damage to Building	Fire damage to interior. Building interior not breached.

6.1.4 Contamination Levels

Estimates of the radioactivity deposited on building and equipment surfaces at a time $t = 1$ year after the postulated accident are shown in Table 6.1-4. Total radioactivity on building and equipment surface is estimated to be about 44 kg of plutonium at the time of shutdown and a total of 35 kg of plutonium are assumed to be released in the accident.

Estimates of radioactivity deposited from the accident are derived from the quantities of plutonium assumed to be released by making the following assumptions.

1. 30 kg of plutonium released from failure of the blending tank is assumed to be evenly distributed throughout the contaminated areas.
2. 5 kg of plutonium released in the fire is assumed to be distributed throughout the contaminated rooms plus the ventilation system.
3. Surface areas of the plant and equipment are taken from the reference decommissioning study.⁽¹¹⁾
4. The radioactivity deposited on building and equipment surfaces is assumed to be uniformly distributed over the total available surface area.

The distribution of the radioactivity in the reference MOX plant and on the equipment is a function of the conditions in the accident, of the propagation and precipitation mechanisms of the radionuclides, and of the effect of safety equipment such as building sprinklers and ventilation system filters. Means are not available for theoretical determination and no test data are available for radioactive contamination levels following an accident. Therefore, estimates of contamination levels and dose rates are based on engineering judgment and related available information.

Radionuclides that are originally present in the release from the accident as particles or aerosols may experience one or more of the following processes:

- They may be released to the environment through the final HEPA filters or through leaks in the building.
- They may be collected in the HEPA filters.
- They may fall out of the building atmosphere by the action of gravity.

TABLE 6.1-4. Estimated Levels of Radioactivity on Structure and Equipment Surfaces

Process Area		Inventory at Normal Shutdown, kg of Plutonium ^(d)	Radioactive Contamination from Accident, kg of Plutonium	Dose Rate (mrem/hr) ^(a) following Accident			
Room Number	Operation			Gamma		Neutron	
				Room ^(b)	Equip. ^(c)	Room ^(b)	Equip. ^(c)
155	Dry Blending	0.7	0.0	0.4	2.0	0.1	0.2
128	Wet Processing	6.2	8.1	0.6	2.0	0.6	1.0
B02	Wet Proc. & Waste Treat.	3.6	6.6	0.6	2.0	0.6	0.4
128-B02	Piping	0.8	0.0	--	2.0	--	0.06
127	Scrap Recovery	4.2	7.4	0.6	2.0	0.6	0.4
B01	Scrap Recovery	6.6	4.2	0.4	2.0	0.5	0.6
SX-127- B01	Piping	2.5	0.0	--	2.0	--	0.2
SX	Solvent Extrac- tion	10.4	4.0	0.8	4.0	1.6	20.0
124	Pellet Proces- sing	8.0	4.7	0.8	2.0	1.2	1.0
123	Fuel Fabrication	0.2	0.0	0.4	2.0	0.01	0.04
126	Storage Vault	0.4	0.0	0.1	--	0.02	--
129	Laboratory	0.2	0.0	0.1	0.1	0.01	0.03
116	Maintenance	0.1	0.0	0.1	0.1	0.01	0.01
Rounded Total		43.9	35.0				

(a) Whole-body radiation exposure rates estimated.

(b) Working in room.

(c) Working in glove box.

(d) Values are calculated.

- They may be precipitated or condensed on building and equipment surfaces.
- The fire is assumed to distribute the airborne radionuclides uniformly throughout the area affected by the fire.

6.1.5 Radiation Exposure Levels

Estimated average radiation exposure rates from surface contamination after the postulated accident are shown in Table 6.1-4. Estimated average exposure rates range from 0.2 mR/hr to 1.2 mR/hr for the reference accidents in different areas of the plant.

The exposure rate in any given area includes contributions from contamination on all building surfaces and all equipment contamination consisting of residual radioactivity from plant operation and from the accident.

Exposure rate calculations use the method developed in the Reactor Shielding Design Manual.⁽¹⁵⁾ Details of the methodology are given in Appendix B of Volume 2.

To calculate the exposure rate from surface contamination, the postulated contamination on floors and walls is approximated by a uniformly contaminated infinite plane. Results of a sensitivity analysis made for a reference decommissioning study⁽¹⁶⁾ indicate that the calculated gamma exposure rate at 1 m from a uniformly contaminated infinite plane closely approximates calculated exposure rates at 1 m from the floor of rooms with floor areas greater than about 30 m², when wall surface contamination levels are much lower than floor contamination levels. This analysis is reproduced in Appendix B.

The estimated exposure rates from radioactive contamination shown in Table 6.1-4 are average values that are used as a basis to estimate occupational radiation doses to workers engaged in cleanup and decommissioning operations at the reference MOX plant. Actual exposure rates would vary depending on the location of the worker in the building or room. Localized hot spots could result in much higher exposure rates than those shown in Table 6.1-4.

6.2 ACCIDENTS AT A REFERENCE URANIUM FUEL FABRICATION PLANT

Accidents of varying degrees of severity may occur at a uranium fuel fabrication plant. Typically, occurrences such as loss of electrical power, valve leaks, inadequate ventilation of enclosures, and spills of UO₂ powder have trivial consequences. These incidents and others (such as filter failures, minor fires, UF₆ releases and inadvertent release of liquid wastes) have occurred or might occur in a U-Fab plant lifetime. Such accidents can result

in increased occupational exposure, equipment damage, and loss of operational time, but not in premature plant closure.

U-Fab plant accident scenarios described in environmental reports^(17,18) and the Environmental Survey of the Uranium Fuel Cycle⁽¹⁹⁾ are used as reference material to develop the accident scenarios presented in this section.

Radioactive source terms that are used to develop contamination levels are described in Section 6.2.1. Potential accident scenarios that could occur are described in Section 6.2.2, and the reference scenarios are summarized in Section 6.2.3. Contamination levels developed from the reference scenarios and radiation exposure levels are given in Sections 6.2.4 and 6.2.5.

6.2.1 Plant Accident Source Terms

Contamination levels in the reference U-Fab plant following an accident are dependent on the radioactivity present in the fuel mixture at the time the accident occurs. The specific activities ($\mu\text{Ci/g}$) of the radionuclides in the reference U-Fab uranium fuel mixture at the time of shutdown and at 5, 10, 20, and 100 years after shutdown are listed in Table 6.2-1, from Reference 20.

TABLE 6.2-1. Specific Radioactivity of Residual Uranium Fuel Mixture at the Time of Shutdown and at Various Times After Shutdown^(a)

Radionuclide	Specific Activity, $\mu\text{Ci/g}$ of Mixture ^(b)				
	Shutdown	5 Years	10 Years	30 Years	100 Years
^{230}Th	3.10×10^{-4}	3.89×10^{-4}	4.58×10^{-4}	7.6×10^{-4}	1.81×10^{-3}
^{231}Th	6.54×10^{-2}	6.54×10^{-2}	6.54×10^{-2}	6.54×10^{-2}	6.54×10^{-2}
^{234}Th	3.36×10^{-1}	3.36×10^{-1}	3.36×10^{-1}	3.36×10^{-1}	3.36×10^{-1}
^{231}Pa	2.93×10^{-5}	3.63×10^{-5}	4.32×10^{-5}	7.17×10^{-5}	1.73×10^{-4}
^{234m}Pa	3.36×10^{-1}	3.36×10^{-1}	3.36×10^{-1}	3.36×10^{-1}	3.36×10^{-1}
^{234}Pa	3.36×10^{-4}	3.28×10^{-4}	3.28×10^{-4}	3.28×10^{-4}	3.28×10^{-6}
^{234}U	1.72	1.72	1.72	1.72	1.72
^{235}U	6.54×10^{-2}	6.54×10^{-2}	6.54×10^{-2}	6.54×10^{-2}	6.54×10^{-2}
^{238}U	3.27×10^{-1}	3.27×10^{-1}	3.27×10^{-1}	3.27×10^{-1}	3.27×10^{-1}
Totals	2.85	2.85	2.85	2.85	2.85

(a) From Reference 20.

(b) Based on average of 3% enriched uranium feed material.

6.2.2 Potential Accident Scenarios

For a U-Fab plant, accidents will involve uranium powders and solutions, UF_6 , and hazardous chemicals. Three accidents with a significant potential impact on decommissioning considerations are postulated for this accident analysis. These are: a criticality accident, which creates direct neutron and gamma radiation and releases fission products; rupture of a hot UF_6 cylinder, which releases hydrogen fluoride and soluble uranium; and an explosion in a sintering furnace, which releases uranium in an insoluble form.

Criticality Accident

Nuclear criticality safety at U-Fab plants is based on a double contingency policy. Equipment design, systems parameters, and administrative procedures are such that two independent errors need to occur to create an accidental criticality accident.

The regulatory guide for evaluating criticality in U-Fab plants⁽²¹⁾ requires simultaneous breakdown of at least two independent controls throughout all elements of the operation. The standard criticality accident is assumed for the purpose of assessing this accident.

An excursion is assumed to occur in a vented vessel of unfavorable geometry containing a solution of 400 g/g of uranium enriched in ^{235}U . The excursion produces an initial burst of 10^{18} fissions in 0.5 seconds, followed successively at 10-minute intervals by 47 bursts of 1.9×10^{17} fissions, for a total of 10^{19} fissions in 8 hours. The excursion is assumed to be terminated by evaporation of 100 liters of the solution.

It is assumed that all of the noble gas fission products and 25% of the iodine radionuclides resulting from the excursion are released directly to a ventilated room atmosphere. It is assumed that an aerosol that is generated from the evaporation of solution during the excursion is released directly to room atmosphere. This aerosol is assumed to comprise 0.05% of the salt content of the solution that is evaporated. This is 20 g of uranium enriched in ^{235}U . The soluble fission product cloud is released from the liquid system and is drawn into the building ventilation system.

The amount released from the plant is mitigated by the HEPA system. A decontamination factor of 2.5×10^7 is considered applicable to two HEPAs, so the release to the environment is not over 5×10^{-6} g enriched uranium.

The radioactive volatile solids and gases evolved in a nominal criticality accident are listed in Table 6.2-2, for a total release of 1.2×10^5 Ci. It is assumed that all of the volatile solids and noble gases evolved would be released from the critical system and escape to the environment.⁽¹⁹⁾

TABLE 6.2-2. Radioactivity of Important Nuclides Released from Criticality Accident at a U-Fab Plant^(a)

Nuclide	Half-life	Radioactivity (Ci)		Total
		0-0.5 hr	0.5-8 hr	
^{83m} Kr	1.8 h	2.2×10^1	1.4×10^2	1.6×10^2
^{85m} Kr	4.5 h	2.1×10^1	1.3×10^2	1.5×10^2
⁸⁵ Kr	10.7	2.2×10^{-4}	1.4×10^{-3}	1.6×10^{-3}
⁸⁷ Kr	76.3 m	1.4×10^2	8.5×10^2	9.9×10^2
⁸⁸ Kr	2.8 h	9.1×10^1	5.6×10^2	6.5×10^2
⁸⁹ Kr	3.2 m	5.9×10^3	3.6×10^4	4.2×10^4
^{131m} Xe	11.9 d	1.1×10^{-2}	7.0×10^{-2}	8.2×10^{-2}
^{133m} Xe	2.0 d	2.5×10^{-1}	1.6	1.8
¹³³ Xe	5.2 d	3.8×10	2.3×10^1	2.7×10^1
^{135m} Xe	15.6 m	3.1×10^2	1.9×10^3	2.2×10^3
¹³⁵ Xe	9.1 h	5.0×10^1	3.1×10^2	3.6×10^2
¹³⁷ Xe	3.8 m	6.9×10^3	4.2×10^4	4.9×10^4
¹³⁸ Xe	14.2 m	1.8×10^3	1.1×10^4	1.3×10^4
¹³¹ I	8.0 d	1.2	7.5	8.7
¹³² I	2.3 h	1.5×10^2	9.5×10^2	1.1×10^3
¹³³ I	20.8 h	2.2×10^1	1.4×10^2	1.6×10^2
¹³⁴ I	52.6 m	6.3×10^2	3.9×10^3	4.5×10^3
¹³⁵ I	6.6 h	6.6×10^1	4.0×10^2	4.7×10^2

(a) From Reference 21.

A Major Release of UF₆ Gas in the Building

The rupture of a UF₆ cylinder is an extremely unlikely event. The cylinders are stored either on a dock or in fire-protected shipping containers in a paved storage area adjacent to the UO₂ building. While a cylinder is in the vaporization chest, it is protected from overheating that might result in hydraulic rupture by three interlocked thermocouples and a pressure switch, any one of which can deenergize the electric heaters if preset limits are exceeded.

It is postulated that a 2.5-ton UF₆ cylinder is received that has been over-filled with UF₆ or contaminated with some foreign gas or vapor, such that overpressurization occurs upon heating in spite of the protective interlock system. It is assumed that the cylinder ruptures and that no immediate corrective action is taken. Approximately 700 kg of UF₆ is released into the vaporization room in about 35 minutes, after which the contents of the cylinder cool to the solid state. The material released into the room would be exhausted through a scrubber system, a HEPA filter, and out the stack. Upon release, it

is assumed that the UF_6 becomes hydrolized from contact with water vapor in the air and with the scrubber system, forming UO_2F_2 and HF. It is further assumed that the HEPA filters become plugged with hydrolized UF_6 , HF destroying the negative pressure differential and allowing UF_6 , UO_2F_2 , and HF gas to seep from the building. Over a 2-hour period, 1% of the uranium and 10% of the HF gas that escaped from the cylinder are assumed released from the building at essentially ground level.

Sintering Furnace Explosion

UO_2 pellets are sintered in furnaces with a reducing atmosphere of hydrogen. The procedure for starting a sintering furnace requires that air in the furnace be displaced by nitrogen before any hydrogen is introduced to minimize the potential for a hydrogen air explosion. It is hypothesized that these controls fail and an explosion occurs in one of the furnaces. The explosion would result in overpressurizing the furnace, but would not produce sufficient force to destroy the furnace. The force of the explosion would blow out the ends of the furnace. It is assumed that the furnace contains 30 kg of uranium, all in the form of UO_2 pellets and that all the uranium is blown out. UO_2 pellets and granules of UO_2 would be ejected into the room. Pellets and granules ejected from the furnace would contaminate the room, but the expected size of the particles would be too large for much of the contamination to spread throughout the plant. It is hypothesized that 0.3 kg of the uranium is drawn into the air exhaust system.

UF_6 Cylinder Rupture Outside the Building

Uranium hexafluoride cylinders could be ruptured accidentally in several situations: puncture, dropping, natural, or operational events. The consequence depends on the variables involved: number of cylinders, heat generated by the event, location of the accident, and mitigation.

An accident is postulated whereby a delivery truck strikes the UF_6 storage area with sufficient force to rupture two cylinders. A hundred gallons of gasoline is postulated to spill to the ground and be ignited by sparks created by contact of the metal and concrete. It is estimated that from 250 to 1260 kg of UF_6 (5 to 30% of two cylinders) could be made airborne during the course of the fire if no remedial action is taken. Some UF_6 would also be continuously vaporized from the ruptured cylinders at the temperature of the cylinder and its contents after the fire.

Undamaged cylinders in the immediate vicinity of the ruptured cylinders could also be overheated. Under the assumption stated above, the radiative heat transfer is sufficient to raise the temperature of the cylinder and its contents from 20° to 101°F, increasing the vapor pressure of the UF_6 inside the cylinder by only 3.5, to 33 psi (well within the operating pressure of the cylinders).

The 100 gallons of gasoline postulated to be involved in the fire has a density of 0.703 g/cm^3 , for a total mass of $2.66 \times 10^5 \text{ g}$. Assuming that the gasoline is 15.5% hydrogen by weight, $3.78 \times 10^5 \text{ g}$ of water is formed by the complete combustion of 100 gal. This amount of water could react with up to $3.7 \times 10^3 \text{ kg}$ of UF_6 to produce particulate UO_2F_2 according to the following equation:

Consequently, the 1,260 kg of UF_6 assumed to be airborne from this accident could be considered as completely converted to UO_2F_2 particulate.

6.2.3 Reference Accident Scenarios

Since a criticality accident is considered highly unlikely to occur in the powder process, the reference accident scenario selected for the U-Fab plant is the failure of a UF_6 cylinder in the vaporization area. The postulated initiating event is the overpressurization of a UF_6 cylinder by overfilling or by contamination with some foreign gas or vapor, such that failure occurs upon heating. The material release is approximately 700 kg of UF_6 into the vaporization room. The material released is assumed to be exhausted through a scrubber system, a HEPA filter, and out the stack. It is assumed that 1% of the uranium and 10% of the HF that escaped from the cylinder are released from the building at essentially ground level.

The reference scenario with the maximum release outside the plant buildings is the rupture of UF_6 cylinders in the loading dock area. The postulated initiating event is a truck delivering supplies impacts the storage area, rupturing two UF_6 cylinders. A fire involving 100 gallons of gasoline from the truck follows and the material released is 1260 kg of UF_6 , which becomes airborne and is completely converted to UO_2F_2 particulate.

A summary of the reference U-Fab plant accident scenarios is given in Table 6.2-3.

6.2.4 Estimated Contamination Levels

Estimates of the radioactivity deposited on building and equipment surfaces inside the building at a time $t = 1$ year after the postulated accident are shown in Table 6.2-4. The table also shows the estimated residual radionuclide inventories in the rooms of the plant at the time of shutdown, from Reference 20. At shutdown, the maximum uranium inventory in the process building is estimated to be about 790 kg.⁽²⁰⁾ A total of 440 kg of uranium is assumed to be released in the accident.

TABLE 6.2-3. Reference U-Fab Plant Accident Scenarios

<u>Accident Parameter</u>	<u>Inside Accident</u>	<u>Outside Accident</u>
● Accident Description	UF ₆ Cylinder Rupture	UF ₆ Cylinder Rupture
● Inventory at Risk	2160 kg UF ₆	4400 kg UF ₆
● Release Mechanism	UF ₆ Vaporization 1/3 of Inventory Released	UF ₆ Vaporization 1/3 of Inventory Released
● Radioactivity Released		
- Inside Building	700 kg of UF ₆ (440 kg of U)	None
- Outside Building	2 kg of UF ₆ [~1 kg of U]	1260 kg of UF ₆
● Physical Damage to Building	None	None

Estimates of the radioactivity deposited on building and equipment surfaces from the accident are derived from the quantities of uranium released by making the following assumptions:

1. Approximately 50% of the uranium released to the building is assumed to be deposited on the building surfaces. The remaining 50% is assumed to be released to the building atmosphere.
2. 10% of the amount airborne remains airborne long enough to be intercepted by the final filters.
3. The radioactivity plated out on building and equipment surfaces is initially distributed uniformly over the building and equipment surface area.
4. Surface areas of the plant and equipment are taken from the reference U-Fab decommissioning study.⁽²⁰⁾

These assumptions result in about 5% of the radioactivity released in the accident assumed to be trapped on the filters, about 1% released to the atmosphere, and the remainder deposited on the building surfaces and equipment.

TABLE 6.2-4. Estimated U-Fab Plant Uranium Inventory Levels After Reference Accident

Room or Area	Uranium Deposition ^(a) at Time of Accident (kg of uranium)	Uranium Deposited from Accident	Total Dose Rate Following Accident (mrem/hr)
UF ₆ Vaporization Area	9.0	140	0.5
Chemical Process Area	204.7	160	0.5
Powder Room	5.5	10	0.1
Pelletizing Room	131.0	105	0.5
Sintering Room	2.1	4	0.1
Grinding Room	7.2	8	0.1
Rodding Room	3.4	5	0.1
Gadolinia Room	8.0	8	0.1
Uranium Scrap Recovery Room	5.2	--	--
Analytical Lab	1.2	--	--
Process Dev. Lab	6.5	--	--
Hot Maint. Shop	2.5	--	--
Hot Inst. Shop	0.6	--	--
Radwaste Room	1.3	--	--
Decon Facility	1.6	--	--
Incinerator	24.3	--	--
HEPA Filter Room	0.6	--	--
Fluoride Waste Treatment Area	3.2	--	--
Nitrate Waste Treatment Area	1.4	--	--
Radwaste Treatment Effluent Area	1.2	--	--
Total (rounded)	421	440	

(a) From Reference 20.

These assumptions are consistent with the amounts of radioactivity assumed to be released from building containment systems in the reference environmental studies.

The distribution of radioactivity on building and equipment surfaces is a function of the conditions of the accident, the propagation and precipitation mechanisms of the radionuclides, and the effect of safety equipment such as building sprinklers and ventilation system filters.

6.2.5 Radiation Exposure Levels

Estimates of the radiation exposure levels in the contaminated areas of the building are made by approximating the walls, ceiling, and floor of each room as a disk and then using this to calculate the estimated dose rates for each room. The calculated dose rate contribution from the accidental release of contamination is shown in Table 6.2-4.

A person working in a room would receive a dose contribution from the radioactive inventory at the time of shutdown and from the accidental release of radioactivity that is assumed to be in the room.

6.3 ACCIDENTS AT A REFERENCE NON-FUEL CYCLE FACILITY

The non-fuel cycle facility covered in this study⁽²²⁾ involves operation of laboratories within the facility using the radionuclides ^3H , ^{14}C , ^{125}I , ^{137}Cs , and ^{241}Am . Physical forms of these radioisotopes vary within the different laboratories in the facility. Tritium is available in 1000-curie gas cylinders; ^{14}C is available as a barium carbonate compound, ^{125}I is in the form of sodium iodide, ^{137}Cs is in the form of a cesium-oxide powder contained in capsules, and ^{241}Am is in the form of americium-oxide powder, also in capsules. These nuclides will all respond differently to accident stresses. For example, gases will become airborne by breaching the container; tritium can be volatilized, forming water vapor; radioiodides are volatile; and cesium and americium are semi-volatile, depending on temperature. Thus, dispersibility of these radionuclides is a function of the form of the radionuclide plus its physical properties.

Quantities of maximum inventories of the different nuclides in the laboratory are shown in Table 6.3-1. They range from 1 curie of ^{125}I to 10,000 curies of tritium.

All of the isotopes have half-lives long enough to be of concern in accident scenarios resulting in premature shutdowns, and the impact could be dependent on the process phase of the system. For example, nuclides with short half-lives would not be considered a problem for decommissioning after a period of time, such as 1 year after the accident.

TABLE 6.3-1. Maximum Radionuclide Inventory in Non-Fuel Cycle Facility^(a)

<u>Nuclide</u>	<u>Possession Limit (Ci)</u>	<u>Maximum Inventory in Building (Ci)</u>	<u>Maximum Inventory in a Laboratory (Ci)</u>
³ H	100 000	10 000	2 000
¹⁴ C	1 000	100	10
¹²⁵ I	100	10	1
¹³⁷ Cs	10 000	5 000	50
²⁴¹ Am	6 000	6 000	6 000

(a) From Reference 22.

Non-fuel cycle facility accident scenarios described in facility radiological contingency plans⁽²³⁾ are used as reference material to develop the reference accident scenarios in this section.

6.3.1 Potential Accident Scenarios

Two areas of the radiopharmaceutical operation may contain relatively large amounts of radionuclide inventory and could be the site of potentially large releases in the event of an accident. These are the shipping/receiving area and the waste storage area. The isotopes are used mostly in small amounts in different locations (hoods, caves, glove boxes, etc.).

Events that are considered for this category include a loading dock accident, explosion (including natural gas), facility fire, and an incident involving waste.

Loading Dock Accident

Radioactive byproduct material is delivered to the radiopharmaceutical manufacturer by truck. The largest inventories of nuclides are usually found in these areas on receiving days.

It is postulated that a truck delivering supplies and equipment strikes the delivery dock. The truck fuel tank is torn open in the crash and the entire 100 gallons of gasoline in the tank spills to the ground and is ignited in the accident. The accident is assumed to occur with curie amounts of isotopes in the bulk shipment that would be at risk in the fire. The largest inventory at risk would be 2000 Ci of ³H in two pressurized cylinders. The integrity of these cylinders could be breached by the impact damaging valves, or heating with subsequent pressurization could cause failure. This would

result in ^3H becoming airborne. However, heat would not be required for volatilization, and the entire 2000 Ci ^3H inventory is assumed to be airborne.⁽³²⁾ The ^3H release to the environment would usually be converted to the oxide form quite rapidly and dispersed like ordinary water.⁽²⁴⁾

Iodides could also be of concern if the shipping container is breached in the accident. Iodides are volatile at ambient temperature.⁽²⁵⁾ NaI solutions achieve a steady-state vaporization rate⁽²⁶⁾ and volatilization can be a function of increasing temperature.⁽²⁷⁾ All available iodine is assumed to be volatilized. Shipments of ^{125}I are approximately 4 Ci each, and the entire inventory is assumed airborne.

Ten Ci of ^{14}C is the other radioisotope potentially involved in the fire and is assumed to be completely airborne, since ^{14}C is easily volatilized. ^{137}Cs and ^{241}Am exhibit volatility on heating, for example 1% per hour for ^{137}Cs .⁽²⁸⁾ Since the fire is assumed to last 12.5 minutes, the ^{137}Cs release would be 0.2% and thus be negligible. Shipping amounts for these isotopes are assumed to be small since they can be sealed sources.

Radioactive waste may be stored in the shipping/receiving area, and therefore could potentially become involved.

It is assumed the fire spreads to the waste area, heating and rupturing drums containing lab trash. A fraction of the waste, perhaps 10%, is expelled from the drums and contributes to the fire. Volatile nuclides associated with the waste become airborne. Nonvolatiles become airborne, with a fractional release of 1.5×10^{-4} . Lab trash contains 2000 Ci of ^3H and 10 Ci of ^{125}I , all assumed to be volatilized. Fifty Ci of ^{14}C are airborne as particulates, with a release of 7.5×10^{-4} Ci.

Fires

A fire could occur in a glove box containing ^{241}Am powders as they are used for manufacturing of neutron sources. They are ball-milled to mix with beryllium in one phase. As the first step, acetone is used to degrease the equipment. Acetone is a potential fire initiator since it is easily ignited: many acetone fires have been started by static electricity.⁽³³⁾ Since ball milling is a contained operation, the times when powder would be in jeopardy are during filling or unloading operations. It is assumed that a beaker of degreasing solvent (acetone) is left in the glove box after degreasing and is inadvertently spilled during filling of the ball mill. Static charge ignites the spilled acetone and a fire follows. For this solvent fire, the combustion source is limited: it could be assumed that there are limited celluloseics present to contribute to the fire. Therefore, the fire would burn out when the acetone is expended. The burning rate of acetone is 2.41 mm/min,⁽³⁴⁾ so a

5-mm-deep spill would burn about 5 minutes. Flash fires can cause pressurization of the glove box and glove ruptures,⁽³⁵⁾ allowing material to be released into the room.

Only a fraction of the powder in the glove box can be assumed "at risk" (i.e., loose so that it is subject to dispersal mechanisms). This is estimated as 10% of the inventory. It appears that the maximum inventory is 6000 Ci of ²⁴¹Am, so 600 Ci are at risk. About 10% of this inventory, or 6 Ci, is estimated ejected into the room. Much of the release would remain in the room, but, for a conservative estimate, it is assumed the release reaches the HEPA filters and about 1 to 2 μ Ci is released to the atmosphere, assuming a filter transmission factor of 2.5×10^{-7} .

Explosion

Use of solvents in the process provides opportunity for explosion. Since the operations are diffuse, a single event of this type would be unlikely to lead to a significant release.

Natural gas used as a building heat source could pose an explosion hazard. The lower flammability limit for natural gas is 4% by volume.⁽²⁹⁾ Gas explosions can occur with a fraction of a room, 0.25 or less, filled with gas. The structure of the process building would seem to ensure safety from natural gas explosions. Fans provide for continual removal of contaminated air, and it is unlikely that an explosive concentration could be reached in a room.

However, the natural gas is carried through lines in the center of the building where airflow could be restricted. Loss or reduction of airflow in conjunction with a natural gas-line rupture with gas flow continuing could result from an explosion. A leak could also occur in a "dead" air space, allowing opportunity for an explosion.

To illustrate this accident scenario, it is hypothesized that a natural gas leak occurs and gas accumulates to the 4% explosive level in a 10 ft x 10 ft x 10 ft (100 ft³) volume, approximately the size of a small room. The gas mixed with air ignites, heated air expands, pressure rises and an explosion follows. It is assumed that a similar event occurs in the radiopharmaceutical manufacturing plant. This event could subsequently lead to the major facility fire that is discussed in the following section.

Facility Fire

A flammable vapor leak (i.e., natural gas) accumulates and subsequently explodes. Oxygen lines (located at each glove box) breached by the initial explosion could accelerate burning of solvents, paper, and wood. In a steel and reinforced concrete structure, the damage would be limited. In a wood-structure building, a full-scale facility fire could ensue. In an older wooden

structure, it would be possible for the entire building to become engulfed in fire and eventually be gutted. The entire inventory of tritium in the building could be released in this accident. The release could be 4000 Ci of tritium.

Fire response of the radionuclides will vary: ^{137}Cs is semi-volatile.⁽²⁸⁾ Americium-241 can be volatile at elevated temperatures, based on americium purification of trace amounts by volatilization giving a yield on the order of 50%.⁽³⁶⁾ Americium is separated from lanthanum, based on volatility⁽³⁷⁾ and the vapor pressure of metallic americium can be calculated. Based on vapor pressure studies, ^{241}Am can be volatile above 1727°C .⁽³⁸⁾ Impacts from the fire will vary therefore with the radionuclide involved. For ^{137}Cs liquid solutions in the hot cells, fire engulfing the building could cause heat generation in the hot cells and melt the HEPA filters. This could lead to partial airborne release of these semi-volatiles. At high temperatures, the ^{137}Cs is 1% per hour.⁽²⁸⁾ From 2000 Ci of ^{137}Cs , the release is 20 Ci.

Waste Accident (Fire)

Waste is gathered together in a facility and either solidified and drummed or held for decay. Drummed waste commonly contains lab trash such as paper, glassware, and clothing. 2000 Ci of ^3H could be contained in 55-gallon steel drums with 6 inches of concrete surrounding each container, resulting in a matrix not conducive to accidents. A large fire involving waste is discussed as part of the loading dock accident and will not be repeated here; a smaller event is covered.

It is postulated that a glass container of acid is included in the waste and broken during compacting. The spilled acid saturates cellulose and consequently fire develops spontaneously. The heat pressurizes the drum and ruptures it.

Lab trash is assumed to contain 50 Ci of ^{14}C , 2000 Ci of ^3H , and 10 Ci of ^{125}I . This material is assumed to be contained in 480 drums, stored in the building in 10 stacks of pallets, each stack four pallets by three pallets high with 4 drums per pallet.

Therefore, one drum of lab trash would contain about 4 Ci of ^3H and 0.02 Ci of ^{125}I , the volatile radionuclides that would become airborne. It is considered highly unlikely that all the waste would be involved in the fire because of the use of fire-suppression equipment.

6.3.2 Reference Accident Scenarios

Major non-fuel cycle facility accidents that might affect large areas of the plant would result in releases of radioactivity that might escape to the

environment. The amounts of radioactivity that could be affected in the typical plant are not expected to result in sufficient contamination levels outside the building to require decontamination outside the plant buildings.

Two accidents are analyzed for their effects on decommissioning of these facilities. One reference accident scenario is assumed to be an accident involving a major facility fire. It is assumed that each of the different types of laboratories is affected by the fire before it is extinguished by the building fire-protection systems. For tritium, ^{14}C , and ^{125}I , the entire inventory of each laboratory is assumed to be affected by the fire and released to the room, resulting in contamination. For the ^{137}Cs and ^{241}Am laboratories, a release of 1% per hour is assumed. Since it is difficult to predict the duration of the fire, it is conservatively assumed that 1% of the cesium and americium will be volatilized and escape to the laboratory. The amounts of contamination assumed to be released for each laboratory are listed in Table 6.3-2.

The second accident scenario involves a fire in the inventory storage and waste storage area of the plant, but does not breach the building exterior. This accident would release radioactive material into the plant primarily from tritium, carbon, and iodine. Amounts of radioactivity released to the building in the accidents are shown in Table 6.3-3.

TABLE 6.3-2. Reference Non-Fuel Cycle Accident Scenarios

Accident Parameter	Accident Description	
• Accident	Major facility fire	Fire in storage area
• Inventory at Risk	Maximum inventory in each laboratory	Maximum inventory in plant storage area
• Release Assumptions	100% of ^3H , ^{14}C , ^{125}I 1% of ^{137}Cs , ^{241}Am	Same
• Radioactivity Released		
- Inside Building	2000 Ci of ^3H , 10 Ci of ^{14}C 1 Ci of ^{125}I , 0.5 Ci of ^{137}Cs , 1 Ci of ^{241}Am	4000 Ci of ^3H , 15 Ci of ^{14}C 4 Ci of ^{125}I , 1 Ci of ^{241}Am
- Outside Building	None	None
• Physical Damage to Building	Fire Damage to Building	Fire Damage to Building

TABLE 6.3-3. Non-Fuel Cycle Facility Accident Releases

Laboratory	Radioactivity Released to each Laboratory (Ci)	Radioactivity Released to Building (Ci)
Tritium- labeled compounds	2000	4000
¹⁴ C- labeled compounds	10	15
¹²⁵ I- labeled compounds	1	4
¹³⁷ Cs- labeled compounds	0.5	1
²⁴¹ Am- labeled compounds	1	2

6.3.3 Contamination Levels

Estimates of the radioactivity deposited on building and equipment surfaces at the time of the accident in the facility process building are shown in Table 6.3-4.

6.3.4 Radiation Exposure Levels

Estimated average radiation exposure levels from surface contamination after the postulated accidents are shown in Table 6.3-4. Estimated average exposure rates range from 0.01 mR/hr to 10 mR/hr for the reference accident in different laboratory areas of the plant.

TABLE 6.3-4. Non-Fuel Cycle Facility Radioactive Contamination Levels Following an Accident

Lab	Lab Floor Area (ft ²)	Radioactive Contamination From Accident (Ci)	Dose Rate At Shutdown (mrem/hr)
³ H	120	4000	1.0
¹⁴ C	80	10	0.1
¹²⁵ I	48	1	0.01
¹³⁷ Cs	48	0.5	0.01
²⁴¹ Am	63	1	0.01

6.4 ACCIDENTS AT A REFERENCE ORE PROCESSING FACILITY

The radioactivity from a rare-metals refinery remains with the residue that is stored in settling ponds or tailings piles. Releases of uranium and uranium daughter products (^{230}Th and ^{226}Ra) can come from the ore piles and the tailing retention ponds. Radon is also released from the ore piles and tailing ponds.

Accident scenarios for tailings retention systems published in Wash 1248⁽¹⁹⁾ and research reports^(9,30) are used as a basis to derive the reference scenarios presented in this section.

6.4.1 Source Terms for Tailings Piles

The primary source of radioactivity from a tailings pond or pile is the radon and particulates of uranium and its daughter products that escape from the tailings pile. Some of the properties of the series of radioactive isotopes that exist in secular equilibrium with ^{238}U and ^{232}Th are listed in Tables 6.4-1 and 6.4-2, respectively.

TABLE 6.4-1. Radioisotopes in Secular Equilibrium with ^{238}U (a)

Radionuclide	Half-Life	Type of Disintegration	Decay Energy (MeV)
^{238}U	4.51×10^9 yr	Alpha	4.27
^{234}Th	24.1 day	Beta	0.26
^{234}Pa	6.75 hr	Beta	2.23
^{234}U	2.47×10^5 yr	Alpha	4.86
^{230}Th	8.0×10^4 yr	Alpha	4.77
^{226}Ra	1602 yr	Alpha	4.87
^{222}Rn	3.82 day	Alpha	5.59
^{218}Po	3.05 min	Alpha, Beta	6.11, 0.33
^{214}Pb	25.8 min	Beta	1.04
^{218}At	2 sec	Alpha	6.82
^{214}Bi	19.7 min	Alpha, Beta	5.62, 3.28
^{214}Po	1.64×10^{-4} sec	Alpha	7.84
^{210}Tl	1.3 min	Beta	5.50
^{210}Pb	21 yr	Beta	0.061
^{210}Bi	5.01 day	Beta	1.16
^{210}Po	138.4 day	Alpha	5.41
^{206}Tl	4.19 min	Beta	1.52
^{206}Pb	Stable		

(a) From Reference 22.

TABLE 6.4-2. Radioisotopes in Secular Equilibrium with ^{232}Th (a)

Radionuclide	Half-Life	Type of Disintegration	Decay Energy (MeV)
^{232}Th	1.41×10^{10} yr	Alpha	4.08
^{228}Ra	6.7 yr	Beta	0.055
^{228}Ac	6.13 hr	Beta	2.15
^{228}Th	1.91 yr	Alpha	5.52
^{224}Ra	3.64 day	Alpha	5.79
^{220}Rn	55 sec	Alpha	6.40
^{216}Po	0.15 sec	Alpha, Beta	6.91
^{212}Pb	10.6 hr	Beta	0.58
^{216}At	3×10^{-4} sec	Alpha	7.95
^{212}Bi	60.6 min	alpha, Beta	6.21, 2.25
^{212}Po	3.0×10^{-7} sec	Alpha	8.95
^{208}Tl	3.10 min	Beta	4.99
^{208}Pb	Stable		

(a) From Reference 22.

6.4.2 Potential Accident Scenarios

The most likely type of accident for the reference ore processing facility tailings pile is the accidental discharge of tailings slurry from the ponds or retention system. Inadvertent discharge of the liquid held in the tailing ponds as a result of dam failure would result in spread of contamination. The exposed pond bed would become subject to wind stresses, resulting in airborne resuspension releases. A fractional airborne release rate of 10^{-8} to $10^{-10}/\text{sec}$ ⁽³⁹⁾ could apply to this release. Two of the failure modes for a tailings system are related to natural phenomena (i.e., failure due to flooding or failure due to an earthquake). Releases could also result from equipment failure (e.g., rupture of a tailings distribution pipeline) or from operating errors (e.g., operator inattention leading to tailings pond overflow).

Rupture of a Pipe in the Tailings Disposal System

Approximately 4×10^7 kg of sand, silt, and clay-sized particles are transported to the tailings pond through the tailings disposal system piping. This material, usually transported as a slurry (~50% water), contains refinery chemicals and radioactive materials. Ruptures in the piping would be expected to occur; however, the majority of the length of the piping would probably parallel the tailings pond, and the flow of the slurry released from the ruptures would be toward the tailings pond, where it would be contained along with the existing tailings material. Should a rupture occur in the length of piping

between the refinery and the tailings area, the slurry could conceivably reach the watercourse. This case is considered in the following section on tailings pond releases.

Releases of Tailings Slurry

Inadvertent release of the tailings slurry to the environment might result from an overflow of the tailings slurry, a rupture in the tailings distribution piping, or a failure of the tailings embankment in addition to washout. Failure of the tailings dam could be caused by a destructive earthquake, flood-water breaching, or structural failure.

Although an incident has been reported involving the release of about 2000 gallons of clear tailings liquid due to a break in a secondary tailings dike caused by unusually high runoff from melting snow, failure of a primary tailings dam by flooding is not considered to be a likely hazard. The risk of flooding is decreased by locating dams below areas of limited drainage, by providing cut-off dams and diversion ditches above the pond area, and by the standard practice of requiring that a 5-foot minimum free-board be maintained during operation of the tailings impoundment system.

For the expected rates of precipitation during the life of the plant, the tailings pond could overflow only if the processing system were allowed to operate unattended for several weeks. An independent analysis of the Bear Creek tailings dam showed that it would not be overtopped by a 100-year return-period flood, or by one-half of the probable maximum precipitation event.⁽⁴⁰⁾ Also, if tailings are deposited above grade, current regulations require that the tailings dam be designed to withstand the probable maximum flood.

Failure of a tailings dam due to an earthquake is highly unlikely since most sites are in the Zone-One Seismic category, a region of "minor damage."^(41,42) An earthquake of intensity MM VI might be expected to have occurred within recent history.

Tailings slurry releases have occurred in the past, and the consequences associated with these events have been documented to varying levels of detail in reports to the NRC (AEC) and to Agreement States and are used to estimate the nominal model mill release. Table 6.4-3 contains a summary of recorded incidents in the period 1959 through 1979.

From these historical data, the average releases from tailings embankment failure or flooding are approximately $5.5 \times 10^4 \text{ m}^3$ ($1.4 \times 10^7 \text{ gal}$) of liquids and $1.4 \times 10^7 \text{ kg}$ ($3.2 \times 10^7 \text{ lb}$) of solids. Six out of ten of the releases from embankment failure or flooding reached the watercourse. One processing plants having dikes similar in construction to those that failed would be required to strengthen the dikes, and for new plants, the design of the embankment retention system is expected to conform to the qualifications in Regulatory Guide 3.11.⁽⁴³⁾

TABLE 6.4-3. Summary of Accidental Tailings Slurry Releases from Uranium Mills, 1959-1979^(a)

	Solids Released, kg	Liquids Released, liters	Reached Watercourse
Flash flood	14×10^6	$1.2 \times 10^7(b)$	Yes
Dam failure	$9 \times 10^5(b)$	9.1×10^5	Yes
Dam failure	5×10^5	$4 \times 10^5(b)$	No
Dam failure	2×10^5	$2 \times 10^5(b)$	Yes
Pipeline failure	3×10^5	2×10^5	Yes
Flooding	$1 \times 10^8(b)$	8.7×10^7	Yes
Pipeline failure	$6.4 \times 10^4(b)$	6.1×10^4	Small amount
Pipeline failure	$2 \times 10^6(b)$	1.7×10^6	Yes
Dam failure	$1-14 \times 10^6(b)$	$1-11 \times 10^6$	Yes
Pipeline failure	$1 \times 10^5(b)$	1.3×10^5	Yes
Dam failure	$9 \times 10^3(b)$	8×10^3	No
Pipeline failure	4.5×10^7	$8-30 \times 10^6$	No
Dam failure	$8.2 \times 10^6(b)$	7.6×10^6	No
Pipeline failure	1.1×10^3	1.5×10^4	Yes
Dam failure	1.0×10^6	3.8×10^8	Yes
Pipeline failure/dam failure	No quantitative information		

(a) From "Environmental Survey of the Uranium Fuel Cycle," WASH-1248, U.S. Atomic Energy Commission, Fuels and Materials, Directorate of Licensing, April 1974, and "Summary of Tailings Slurry Releases 1972-1977," prepared by Teknekron, 28 February 1978.

(b) This value is based on the assumption that equal weights of solids and liquids are released, and that the density of the liquids is approximately 1.6 g/cm^3 (100 lb/ft^3).

Most failures in the tailings distribution piping would result in release of the slurry to the tailings pond and not to the environment. However, if the failure were to occur in the length of piping between the mill and the tailings area, the slurry could conceivably reach the watercourse. Based on the historical data given in Table 6.4-3, the average releases to the watercourse from piping failure were approximately $3.5 \times 10^3 \text{ m}^3$ ($9.1 \times 10^5 \text{ gal}$) of liquid and $3.2 \times 10^6 \text{ kg}$ ($1.8 \times 10^7 \text{ lb}$) of solids.

Since both the historical consequences and likelihood of piping failures are lower than those of embankment failure, only releases from embankment failures or flooding are considered to be likely to occur.

If an operational or equipment failure accident occurred, it would probably not involve a sudden catastrophic failure of the dam. The stored sand and slimes would be somewhat resistant to flow and not readily transported by the water in the pond that is retained behind the low head dam. The flow of any leakage from a tailings line or from the tailings pond would be in the direction of the natural drainage course. The materials carried in the runoff would be rapidly deposited by sedimentation over a relatively short distance. Liquids would flow down the natural drainage system for an indeterminate distance until depleted by seepage and evaporation.

The estimated 4.1×10^7 kg of solid tailings released from the impoundment area in the event of an overtopping or failure of the embankment would be expected to settle out below the embankment. The extent of the area covered would depend upon the type of failure and is difficult to predict. Scaling from previous estimates on the basis of the total mass of tailings released,⁽⁴⁴⁾ the material may be assumed to follow the tributary stream channel for a distance of approximately 2100 m, covering a width of approximately 130 m, and forming a wedge 3 cm (1-1/4 inches) in average thickness.

The main radiological concern associated with the deposition of the tailings material is the small increase in background radiation levels in the affected and adjacent areas and the eventual transport of these low levels of contamination by wind and rain. These long-term effects may be prevented by removing the contaminated material from the environment.

Table 6.4-4 summarizes the effects of tailings pond accidents.

TABLE 6.4-4. Potential Tailings Pond Accidents Resulting in Release of Radioactive Effluents

<u>Site</u>	<u>Accident and Source of Effluent</u>	<u>Transport Mode</u>	<u>Effluent</u>	<u>Estimated Effect of Effluent</u>
Rare Metals Refinery Tailings Pile	Tailings dam failure due to: a) flooding, b) earthquake, or c) equipment failure.	Waterborne solids carried into nearby streams.	Waterborne uranium and daughters	Possible ingestion pathway.
Tailings Pond	Tailings pond spill over or accidental discharge.	Waterborne solids carried into nearby streams.	Waterborne uranium and daughters	Minor ingestion pathway.

6.4.3 Reference Accident Scenario

The total pond area is $5,000 \text{ m}^2$. However, it is not likely that the entire volume would be released. Based on historical data, about $1.8 \times 10^6 \text{ kg}$ of solids would be an average release. The material is assumed to form a 3-cm average thickness cover over an area approximately $2.7 \times 10^5 \text{ m}^2$.⁽³¹⁾ This area would be subject to resuspension. Total sludge volume is $16,400 \text{ m}^3$. It is assumed the sludge will have a density similar to mill tailings at 1.6 g/cm^3 . The total mass subject to resuspension is thus about $1.8 \times 10^6 \text{ kg}$.

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7.0 ACCEPTABLE RESIDUAL RADIOACTIVE CONTAMINATION LEVELS FOR DECOMMISSIONED FUEL CYCLE AND NON-FUEL CYCLE FACILITIES

Unrestricted release of a nuclear facility following decommissioning requires that the radioactivity remaining in the facility and on the site be reduced to levels considered acceptable for unrestricted access and subsequent NRC license termination. This chapter provides a summary of the criteria being developed for determining acceptable residual contamination levels for public use of decommissioned fuel cycle and non-fuel cycle facilities.

Some guidance currently exists for defining the levels of radioactive surface contamination that are acceptable to the NRC for the termination of operating licenses.^(1,2) This guidance by itself is not explicit enough in relating contamination levels to annual dose rates or justifying that the values presented are ALARA, nor does it sufficiently accommodate the various radionuclide mixtures or site-specific features found at each unique nuclear facility. This has resulted in the need for case-by-case licensing decisions to establish acceptable residual levels. These facts suggest that the methodology used to calculate acceptable levels of residual radionuclide contamination at decommissioned nuclear facilities should be based on a more general concept capable of accommodating unique radionuclide mixtures and site-specific features.

The draft generic environmental impact statement (GEIS) on decommissioning of nuclear facilities⁽³⁾ contains a recommendation that the allowable residual radioactivity levels for facility release be based on the dose anticipated to be received by individuals who use the facility or site after the license is terminated. The GEIS includes a discussion of several basic requirements that must be considered in selecting an acceptable level for facility release. These requirements include the following:

1. The level must be low enough to comply with the ALARA concept.
2. The level must be verifiable through actual detailed radiological survey measurements of the facility and site.
3. Dose rates and associated contamination levels should be based on realistic dose assessment methodology.

Guidance on establishing residual radioactivity levels, on the costs and methodology of radiological surveys, and on realistic dose methodology is contained in the GEIS⁽³⁾ and in two reports prepared by Oak Ridge National Laboratory.^(4,5)

It is not within the scope of this study to recommend annual dose limits for the exposure of the public to radioactive materials. Under the provisions of the Energy Reorganization Act of 1974, the Environmental Protection Agency (EPA) has responsibility for establishing radiation dose standards for the protection of the public health and safety and for establishing radiation dose limits for decommissioning a nuclear facility to unrestricted access. The EPA has not yet promulgated these limits and is not scheduled to begin this activity until 1984, with standards being set at some later date. In the absence of standards in this area, the NRC, based on the statutory authority contained in the Atomic Energy Act to protect the health and safety of the public, has issued the guidance in the GEIS and announced its intention to promulgate regulations establishing criteria for acceptable residual radioactivity levels for the release of decommissioned facilities and sites.⁽⁶⁾

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8.0 MIXED-OXIDE PLANT CLEANUP AND DECOMMISSIONING

The first activities following an accident at a small mixed-oxide (MOX) plant are designed to bring the accident under control and to stabilize the facility to prevent further releases of radioactivity. Initial stabilization is followed by accident cleanup, which is followed by either refurbishment or decommissioning of the facility. This chapter provides information about the activities and manpower requirements and costs for accident cleanup and decommissioning following a serious accident at a small MOX plant. It does not analyze the detailed activities related to refurbishment of the facility.

This chapter summarizes the detailed discussion of accident cleanup, costs, and decommissioning of the MOX plant presented in Appendix D of Volume 2. General information on decontamination, liquid waste treatment, and packaging and disposing of wastes is given in Appendix C of Volume 2. The purpose of accident cleanup is discussed in Section 8.1. Activities and manpower requirements for accident cleanup are discussed in Section 8.2. The cost of accident cleanup following the reference accident is discussed in Section 8.3. Activities and manpower requirements for decommissioning are discussed in Section 8.4. The costs of decommissioning following an accident are presented in Section 8.5.

8.1 PURPOSE OF ACCIDENT CLEANUP AT A REFERENCE MIXED-OXIDE PLANT

The reference accident postulated for the MOX plant (see Chapter 8) results in severe contamination of the building and equipment. Either decommissioning or refurbishment of a plant following a postulated accident would begin with an accident cleanup campaign with the following principal goals:

1. to reduce the initial relatively high levels of radioactive contamination present on building surfaces and equipment from the accident, thereby reducing the radiation dose received by workers engaged in cleanup and subsequent decommissioning or refurbishment operations
2. to collect, package for disposal, and dispose of the readily dispersible radioactivity present in the plant.

To achieve these goals, accident cleanup is postulated to include the following tasks:

- processing of contaminated liquids released during the accident (and by decontamination operations) to remove and immobilize radioactive contaminants

- initial decontamination of building surfaces and decontamination or disposal of some equipment
- solidification and packaging of wastes from accident cleanup operations.

Accident cleanup activities must establish conditions within the process building that permit reasonable occupancy times without excessive radiation exposure to workers engaged in subsequent decommissioning operations. In keeping with ALARA principles, decontamination during accident cleanup is restricted to operations that result in the greatest reduction in residual radioactive contamination with the least radiation dose to decommissioning workers.

The tasks required for accident cleanup are assumed to be essentially independent of whether the facility is ultimately restarted or decommissioned, and if decommissioned, of the alternative (DECON, SAFSTOR, or ENTOMB) chosen. (The rationale for this assumption is discussed in Section D.1 of Appendix D.) Because accident cleanup activities are similar, the requirements and tasks presented in this section are considered to be representative of the activities carried out during accident cleanup independent of the ultimate use of the plant. The work required to accomplish each cleanup task is, of course, affected by the severity of the accident. Technical requirements for accident cleanup are discussed in the following sections.

As defined in this study, accident cleanup does not include the extensive hands-on decontamination operations required to reduce surface contamination inside the plant to levels suitable for release of the facility for unrestricted use; that additional decontamination would take place during the decommissioning activities. In addition, as defined in this study, accident cleanup does not include the additional hands-on decontamination to reduce radiation levels to low enough values to permit the extensive work necessary during refurbishment of the facility. Accident cleanup also does not include the decontamination or disposal of permanently installed equipment items, such as the nitrate blending system that would not be removed or refurbished until the decommissioning or refurbishment of the facility after completion of the accident cleanup activities.

8.2 ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP AT A REFERENCE MIXED-OXIDE PLANT

Following an accident, several planning and preparation tasks, needed to ensure the proper performance and the success of accident cleanup, must be completed before cleanup operations begin. Planning and preparation activities and the manpower requirements for their performance are described in this section.

8.2.1 Planning and Preparation Activities

Planning and preparation activities that must be performed prior to the operational phase of building cleanup include (not necessarily or exactly in this order):

- contaminated area entry and data acquisition
- prepare documentation for regulatory agencies
- design, fabricate, and install special equipment
- develop detailed work plans and procedures
- select and train accident cleanup staff.

Contaminated Area Entry and Data Acquisition

Data on the post-accident radiological and physical condition of the plant are obtained and analyzed during the planning and preparations phase. These data include measurements of contamination levels and radiation exposure rates in the building, estimates of physical damage to structures and equipment, and information about the operational status of plant systems and services. The data provide a basis for planning accident cleanup operations and also provide some of the information needed to prepare documentation for regulatory agencies.

Data on post-accident conditions in the process building are obtained by teams of workers (generally two to five persons) who spend short periods (generally a few minutes to a few hours) inside the building. The numbers and durations of entries into the contaminated areas and the information obtainable during these entries depend on radiation exposure levels inside the building.

Prepare Documentation for Regulatory Agencies

A major planning task is the preparation by the licensee of the necessary documentation to amend the facility operating license to maintain the plant in a safe shutdown condition and to obtain regulatory approvals to proceed with cleanup operations. A discussion of existing regulations, guides, and standards that apply to nuclear facility accident cleanup and decommissioning is given in Chapter 5. Documentation that must be provided by the licensee prior to the start of cleanup operations is summarized in Section D.2 of Appendix D (Volume 2). The time needed to furnish information to regulatory agencies, issue environmental statements and assessments, and secure regulatory approvals to proceed with specific cleanup tasks is a critical factor in determining the length of the planning and preparations period and, hence, could delay the start of actual cleanup operations.

Design, Fabricate, and Install Special Equipment

Major facilities and equipment items required for accident cleanup include the following:

- an evaporator/solidification facility to process the decontamination solutions generated
- a volume reduction incinerator to reduce the total quantity of waste requiring disposal
- shielded and unshielded storage facilities for interim storage of radioactive wastes; these may be required if there is difficulty in disposing of wastes because of regulatory or political constraints
- a laundry facility.

Some items such as evaporators, incinerators, and laundry facilities are commercially available. Other items must be designed and fabricated during preparations for cleanup.

This study assumes that for the reference facility, except for the storage facilities indicated, sufficient space is available for the addition of necessary special equipment for accident cleanup. Because of potentially insufficient available space, it might be necessary to construct other new buildings. Section 8.3 includes an analysis of the sensitivity of the costs of accident cleanup to the need to construct a new building to house certain special equipment items.

Designs and specifications are prepared for each special equipment item required. When the item is procured, it is inspected to verify that it meets specifications and complies with applicable quality assurance and safety requirements. It is then tested to ensure that it performs as required. The testing also serves to train personnel in the use of the equipment and to provide pertinent data on its operation. The time requirement for purchasing, installing, and testing this special equipment is a critical factor in determining when actual cleanup activities can begin. Delays in any step in making the equipment ready could cause delays in accident cleanup.

Develop Detailed Work Plans and Procedures

Detailed work plans and procedures are developed based on an evaluation of the condition of the plant following an accident and on the requirements for accident cleanup. Work plans are included in documentation provided to the NRC with the request for license amendment. The detailed plans and procedures contain all the information required to actually carry out the accident cleanup tasks. They address the following items:

- regulatory requirements and constraints
- decontamination methods and procedures
- schedules and sequences of events
- manpower requirements

- equipment requirements
- contamination control
- radiological and industrial safety
- packaging and disposal of radioactive wastes
- quality assurance.

Physical security and environmental constraints are also considered. Plans are updated as the accident cleanup work proceeds and additional data on the physical and radiological status of the facility becomes available.

Select and Train Accident Cleanup Staff

The selection and training of operations staff for accident cleanup begins during preparations for cleanup. Because detailed knowledge of and familiarity with the facility being decontaminated increase the effectiveness of cleanup workers, key staff positions are filled, whenever possible, with personnel familiar with the construction and operation of the plant. Additional training in specific cleanup tasks is provided, with emphasis on the use of new or unique equipment and procedures.

Training of cleanup staff continues throughout the accident cleanup period. Because of the possible high exposure rates encountered and the need to limit individual radiation doses, a number of persons may be involved in accident cleanup operations. Some of these persons may be unfamiliar with the plant, and some will be unfamiliar with the basic principles of radiation protection. These persons require an orientation in the layout of the plant and in basic radiation protection procedures as well as specific instruction in the tasks to be performed.

8.2.2 Time Requirements for Preparations for Accident Cleanup

Time requirements for preparing for accident cleanup depend on several factors, including accident severity, time needed to design, fabricate and test special equipment facilities, and time required to secure regulatory approvals to proceed with specific cleanup tasks. The time required to secure regulatory approvals for specific cleanup operations is a critical factor in determining when these operations can begin.

To provide a basis for computing staff labor costs, preparations for accident cleanup are assumed to require 0.5 years following the reference accident. To enable the user of this study to compute planning and preparations costs for time periods other than those assumed in this study, staff labor costs for preparations for accident cleanup are given on an annual basis in Table D.3-1 of Appendix D (Volume 2).

8.2.3 Staff Requirements for Preparations for Accident Cleanup

The plant operations staff has the responsibility to maintain the plant in a safe shutdown condition. Plant operations staff members also assist the cleanup planning staff in the acquisition of data on the radiological and physical condition of the building and in the installation and testing of equipment and systems required for accident cleanup.

Major activities of the cleanup planning branch include:

- preparing documentation for regulatory agencies
- preparing detailed work plans and work schedules
- acquiring data on the radiological and physical condition of the plant
- testing equipment and procedures to be used in cleanup operations
- installing or repairing systems required for accident cleanup (e.g., rerouting piping connections, installing systems for remote monitoring, etc.).

Site support labor requirements for preparations for accident cleanup are shown in Table 8.2-1. These labor requirements include 5.25 man-years for preparations for accident cleanup following the reference accident scenario.

The staff labor requirements shown in Table 8.2-1 do not include contractor labor to provide engineering support services during preparations for cleanup. An engineering contractor is assumed to provide assistance to the staff in preparing documentation for regulatory agencies, developing detailed work plans and work schedules, and preparing design specifications for the special equipment and facilities required for accident cleanup. Contractor labor for engineering services during preparations for cleanup is estimated to be six man-years per year (three total man-years) following the reference accident.

8.2.4 Accident Cleanup

Accident cleanup procedures, cleanup schedules, and staff labor requirements for accident cleanup at the reference MOX plant are summarized in the following subsections. Details of accident cleanup are given in Section D.2 of Appendix D.

TABLE 8.2-1. Estimated Staff Labor Requirements for Preparations for Accident Cleanup

<u>Position</u>	<u>Staff Labor Requirements (man-years) for Preparations for Accident Cleanup^(a)</u>
Project Manager	0.50
Project Engineer	0.50
Health and Safety Supervisor	0.50
Contracts and Accounting Specialist	0.50
Q. A. Engineer	0.50
Radioactive Shipment Specialist	0.25
Planning Engineer	0.5
Operations Supervisor	0.5
Foreman	0.5
Secretary	<u>1.00</u>
Total Man-Years	5.25

(a) Based on a preparations for cleanup period of 0.5 years.

The reference accident is described in Chapter 6. Accident cleanup operations are assumed to reduce general area radiation exposure rates in the plant. As discussed in Section 8.1, decontamination activities during accident cleanup are not designed to reduce exposure rates to levels permitting unrestricted use of the facility, but only to limit the doses to workers engaged in accident cleanup. Additional decontamination would be required during decommissioning (or refurbishment) to limit the doses to workers engaged in these activities.

Procedures for Accident Cleanup

The tasks that must be completed for accident cleanup at the MOX plant include the following:

1. Decontaminate the building to permit access.
2. Install the evaporator solidification facility.
3. Flush contaminated tanks and pipes and process the contaminated liquid.
4. Continue the decontamination of building surfaces and equipment.

5. Process contaminated liquids from cleanup operations.
6. Replace contaminated filters.
7. Solidify and package wastes from accident cleanup.
8. Treat and dispose of wastes from accident cleanup.
9. Perform radiation survey to determine the extent of residual contamination following accident cleanup and to verify the effectiveness of cleanup operations.

Processing of Contaminated Liquids. Contaminated liquids that must be processed during accident cleanup include water from the fire suppression systems, chemical decontamination solutions generated during decontamination of building surfaces, and plant wet processing systems chemical and flush solutions. Estimated volumes of contaminated liquids from accident cleanup following the postulated accident and the curies of radioactivity removed from these liquids are given in Section D.2 of Appendix D.

Chemical decontamination solutions from initial cleanup operations have radionuclide concentrations in the range from 1 to 10 Ci/m³. Evaporation is a suitable alternative for treatment of these wastes. An evaporator/solidification facility is postulated to be obtained from a commercial supplier and installed in the process building during preparations for cleanup. The evaporator bottom liquids are solidified with vinyl ester styrene and packaged in stainless steel liners for interim onsite storage in the shielded storage facility that is constructed during preparations for accident cleanup.

Initial Decontamination. The objective of initial decontamination at the MOX plant is to reduce surface contamination levels and resultant radiation exposure levels to permit reasonable occupancy times for workers engaged in cleanup operations. In addition to surface decontamination procedures, reduction of general area radiation exposure rates requires the removal and processing of any liquids and the removal or shielding of any contaminated sludge deposits that remain on the walls and floors after the liquids are removed. The reduction of general area radiation exposure rates requires that "hot spots" be shielded by using lead sheet or lead bricks, high-density concrete blocks, or containers filled with water.

A general discussion of procedures for the decontamination of surfaces and equipment is given in Appendix D. Initial decontamination of the plant includes the following steps:

1. Remove and package debris and small items of contaminated equipment that are easily disposed of.

2. Employ high-pressure hose wash techniques for semi-remote decontamination of building surfaces and equipment.
3. Decontaminate and refurbish or replace essential support systems.
4. Perform hands-on decontamination of selected areas where significant reductions in radiation exposure can be achieved with modest effort; decontaminate floors by scrubbing.
5. Provide local shielding of "hot spots."

Water from remote spray and hose-wash decontamination operations is collected in the building sumps. Remote spray and hose-wash operations are carefully coordinated with sump water processing operations.

Waste Treatment and Disposal. Radioactive wastes from accident cleanup operations can be divided into three categories:

1. Solid Materials. Dry radioactive wastes generated from decontamination. These materials consist of trash, contaminated equipment and material, and HEPA and roughing filters.
2. Process Solids. Contaminated sludges and process solid wastes that arise from the treatment of water and decontamination liquids. These solid wastes include filter cartridge assemblies and evaporator bottoms.
3. Chemical Decontamination Solutions. Liquid decontamination wastes that have not been treated to generate process solids. These wastes are immobilized by incorporation in cement or in vinyl ester styrene.

The alternatives assumed in this study for the packaging and disposal of these wastes and the waste volumes generated during accident cleanup in the MOX building are given in Table D.2-1 of Appendix D.

Waste Management Requirements. Because the levels of plutonium contamination are expected to exceed 100 nanocuries per gram of waste, the wastes are all assumed to be packaged in steel boxes and steel drums and shipped to a federal repository for deep geologic disposal.

Table 8.2-2 gives estimated weights and volumes of wastes generated during accident cleanup at the reference MOX plant, together with packaging, shipping and disposal requirements for these wastes. Detailed information about waste quantities is given in Section D.2 of Appendix D.

TABLE 8.2-2. Waste Disposal Requirements of Accident Cleanup^(a,b) at the Reference MOX Plant

Waste Category	Shipping Weight (kg)	Shipping Volume (m ³)	Container Type	Number of Containers	Containers per Shipment	Number of Truck Shipments
HEPA and Roughing Filters	10 200	30.0	1.2 x 1.2 x 1.8-m Steel Box	12	3	4
Solidified Decontamination Liquids	30 500	10.0	208-l (55-gal) Drums	60	24 ^(c)	3
Trash	25 500	25.0	208-l (55-gal) Drums	125	42	3
Totals	66 200	65.0		12 boxes 185 drums		10

(a) All wastes are disposed of by deep geologic disposal.

(b) Number of figures shown is for computational accuracy, and does not imply precision to three significant figures.

(c) Number of containers per shipment is limited by weight rather than by volume.

Schedules for Accident Cleanup

Schedules for accident cleanup at the MOX plant following the reference accident are shown in Figure 8.2-1. Time requirements for accident cleanup depend on accident severity and are based on the cleanup procedures summarized in the previous subsection. Accident cleanup in the MOX plant is estimated to require approximately one year following the reference accident.

(a) SCHEDULE DETAILS ARE GIVEN IN SECTION D.2 OF APPENDIX D.

(b) THE TOTAL TIME REQUIREMENT FOR ACCIDENT CLEANUP IN THE MOX BUILDING FOLLOWING THE REFERENCE ACCIDENT IS 1.0 YEARS

(c) AS REQUIRED DURING THIS TIME PERIOD.

FIGURE 8.2-1. Sequence and Schedule for Accident Cleanup at the Reference MOX Plant Following the Reference Accident^(a)

Details of accident cleanup schedules and the bases and assumptions used to prepare these schedules and to estimate cleanup worker requirements are given in Section D.2 of Appendix D.

Staff Labor Requirements for Accident Cleanup

A postulated staff organization for accident cleanup at the MOX plant is shown in Figure 8.2-2. The staff for accident cleanup includes a plant operations branch and several site support branches (e.g., engineering; health and safety; security; contracts and accounting; and quality assurance) as well as the staff actually involved in cleanup of the process building.

Estimated staff labor requirements for accident cleanup in the MOX plant are shown in Table 8.2-3. These labor requirements are 10 man-years for cleanup following the reference accident.

The accident cleanup staff labor requirements (the man-hours for personnel engaged in cleanup operations in the plant) shown in Table 8.2-3 have been adjusted upward by appropriate factors to ensure that the estimated occupational radiation dose for individual workers does not exceed five rem/year.⁽¹⁾ An explanation of the adjustment factors used to obtain man-years for accident cleanup staff labor is given in Section D.2 of Appendix D. The staff labor requirements shown in Table 8.2-3 are used to compute staff labor costs for accident cleanup at the reference MOX plant. (See Section D.3 of Appendix D.)

The accident cleanup staff labor contribution to the total staff labor requirement for accident cleanup is 33% for cleanup following the reference accident scenario. The management, plant operations, and site support labor contribution to the total utility staff labor requirements is 67% for cleanup following the reference scenario accident. The total labor requirement is a function of the duration of the accident cleanup period.

The staff labor requirements shown in Table 8.2-3 do not include contractor labor to provide engineering support services during accident cleanup. The contractor labor requirement for engineering support services is estimated to be three man-years (two man-years per year) during cleanup following the reference accident.

8.3 COSTS OF ACCIDENT CLEANUP AT A REFERENCE MIXED-OXIDE PLANT

The costs of accident cleanup activities at the reference MOX plant following the postulated accident are summarized in this section. These costs are developed in detail in Section D.3 of Appendix D (Volume 2). Costs are in 1981 dollars.

FIGURE 8.2-2. Postulated Staff Organization for Accident Cleanup at the Reference MOX Plant

TABLE 8.2-3. Estimated Staff Labor Requirements for Accident Cleanup at the Reference MOX Plant

<u>Position</u>	<u>Staff Labor Requirements (man-years) for Cleanup Following the Reference Accident^(a)</u>
Project Manager	1.0
Project Engineer	1.0
Health and Safety Supervisor	1.0
Health Physics Technician	3.0
Security Force Supervisor	1.0
Security Patrolman	8.0
Contracts and Accounting Specialist	1.0
Accountant	0.5
SNM Accounting Specialist	1.0
Q.A. Engineer	1.0
Radioactive Shipment Specialist	1.0
Planning Engineer	1.0
Maintenance and Crafts Supervisor	1.0
Maintenance Man	1.0
Custodian	1.0
Craftsman	4.0
Operations Supervisor	1.0
Foreman	3.0
Technician	12.0
Secretary	<u>2.0</u>
Total Man-Years	45.5

(a) Cleanup staff labor requirements are adjusted to limit individual radiation doses to 5 rem/yr.⁽¹⁾

Accident cleanup costs would be similar whether the plant is refurbished for restart or decommissioned. Therefore, the costs of accident cleanup presented here are believed to be a good representation independent of the ultimate use of the plant. Costs of activities related to refurbishment and restart of a plant beyond the accident cleanup activities are not included in this study. Costs of decommissioning following accident cleanup are summarized in Section 8.5.

8.3.1 Estimated Costs of Accident Cleanup

The estimated costs of accident cleanup at the reference MOX plant are summarized in Table 8.3-1. These activities are estimated to require about 1 year (plus 0.5 years for planning and preparation) at a cost of approximately \$4.8 million.

TABLE 8.3-1. Summary of Estimated Costs of Accident Cleanup
at the Reference MOX Plant

<u>Cost Category</u>	<u>Cost in Millions of 1981 Dollars^(a)</u>	<u>Percent of Total</u>
Manpower -		
Planning and Preparation	0.328	8.9
Accident Cleanup	1.916	51.7
Equipment and Supplies	0.526	14.2
Disposal of Radioactive Material	0.360	9.7
Miscellaneous Owner Expense	0.536	14.5
Specialty Contractors	<u>0.037</u>	<u>1.0</u>
Subtotal	3.703	100.0
25% Contingency	<u>0.926</u>	
Total Costs of Accident Cleanup	4.629	

(a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

Manpower represents about 60% of the total cost of accident cleanup. Manpower costs include both support staff and accident cleanup worker labor. In Table 8.3-1, manpower costs are shown separately for planning and preparation activities and for accident cleanup activities. Manpower costs include onsite labor for packaging radioactive waste materials for shipment. Labor costs related to radioactive waste transportation are included with waste management costs, since transportation is performed by a specialty contractor.

Waste management accounts for about 10% of the total cost of accident cleanup. Waste management costs include shipping container costs, transportation charges, and fees for waste disposal. All radioactive wastes generated during accident cleanup are assumed to go to deep geologic disposal.

Manpower Costs

Table 8.3-2 shows manpower cost estimates for the planning and preparation phase, and Table 8.3-3 shows support staff and worker manpower requirements and costs for the active phase of accident cleanup. About five man-years are estimated to be required for planning and preparation, at a labor cost of about \$0.3 million. About 45 man-years are estimated to be required to chemically and physically decontaminate the facility, at a labor cost of about \$1.9 million. The total labor cost for accident cleanup of the facility is therefore about \$2.2 million without contingencies.

TABLE 8.3-2. Summary of Staff Labor Costs for Planning and Preparation Phase of Accident Cleanup at the Reference MOX Plant

<u>Title or Function</u>	<u>Cost (\$ millions)(a,b)</u>
Project Manager	0.079
Project Engineer	0.068
Health and Safety Supervisor	0.017
Contracts and Accounting Specialist	0.011
Q.A. Engineer	0.024
Radioactive Shipment Specialist	0.006
Planning Engineer	0.031
Operations Supervisor	0.047
Foreman	0.018
Secretary	<u>0.028</u>
Total Cost	0.328

(a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

(b) Contingency of 25% is not included in these costs.

Estimated Material and Equipment Requirements and Costs of Accident Cleanup

Estimates of material and equipment requirements and costs of accident cleanup at the reference MOX plant are shown in Table 8.3-4. Equipment requirements are based on procedures described in detail in Appendix D and summarized in Section 8.2. The total cost of material and equipment for accident cleanup is estimated at about \$0.5 million without contingency.

TABLE 8.3-3. Summary of Staff Costs for Accident Cleanup
at the Reference MOX Plant

<u>Title or Function</u>	<u>Cost (\$ millions)(a,b,c)</u>
Project Manager	0.104
Project Engineer	0.090
Health and Safety Supervisor	0.067
Health Physics Technician	0.107
Security Force Supervisor	0.048
Security Patrolman	0.239
Contracts and Accounting Specialist	0.053
Accountant	0.017
SNM Accounting Specialist	0.053
Q.A. Engineer	0.062
Radioactive Shipment Specialist	0.046
Planning Engineer	0.062
Maintenance and Crafts Supervisor	0.058
Maintenance Man	0.037
Custodian	0.034
Craftsman	0.152
Operations Supervisor	0.062
Foreman	0.133
Technician	0.438
Secretary	<u>0.056</u>
Total Cost	1.916

(a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

(b) Information given is for support staff and decommissioning workers.

(c) Contingency of 25% is not included here.

Estimated Waste Management Costs

The estimated costs for containers, transportation and disposal of the radioactive wastes from accident cleanup are summarized in Table 8.3-5. Cost estimates are based on projected packaging and shipping requirements summarized

TABLE 8.3-4. Estimated Costs of Equipment and Supplies for Accident Cleanup at the Reference MOX Plant

<u>Description</u>	<u>Quantity</u>	<u>Estimated Total Cost (\$ thousands)^(a)</u>
Arc Welder	2 ea	2.8
Paint Sprayer	2 ea	2.8
Ratcheting Pipe Cutter	6 ea	0.4
High-Velocity Liquid Jet	1 ea	6.9
Low-Velocity Liquid Jet	2 ea	5.5
Waste Compactor	1 ea	16.6
HEPA Filter	326 ea	67.8
Roughing Filter	313 ea	21.6
Decontamination Chemicals		13.8
Cleaning Supplies		138.5
Expendable Tools		34.6
Protective Clothing (including laundry)		69.2
Cement	3 m ³	0.3
Office Supplies		48.5
Planning and Preparation		<u>27.7</u>
		526

(a) Contingency of 25% is not included here.

in Table 8.2-2 and on waste management cost data from Section D.3 of Appendix D. The total cost of waste management for accident cleanup at the reference MOX plant is estimated to be \$0.36 million without contingencies.

Miscellaneous Owner Expenses for Accident Cleanup

Estimated miscellaneous owner expenses for accident cleanup at the reference MOX plant are shown in Table 8.3-6.

In estimating the applicable license fee during accident cleanup it is assumed that the full operating fee would be paid, because scrap recovery and waste treatment facilities at the plant would be used during decommissioning to recover plutonium from decommissioning wastes.

TABLE 8.3-5. Estimated Waste Management Costs for Accident Cleanup at the Reference MOX Plant

Waste Category	Costs in Thousands of 1981 Dollars ^(a,b)			
	Containers	Transportation	Burial	Total
Solidified Decontamination Liquids	8.8	41.7	28.2	78.8
HEPA and Roughing Filters	24.3	25.9	102.3	152.4
Trash	<u>8.8</u>	<u>21.8</u>	<u>98.3</u>	<u>128.9</u>
Totals	41.9	89.4	228.8	360.1

(a) Number of figures shown is for computational accuracy and does not imply three place precision in waste management cost figures.

(b) Contingency of 25% is not included here.

TABLE 8.3-6. Estimated Miscellaneous Owner Expenses During Accident Cleanup at the Reference MOX Plant

Cost Category	Cost in Thousands of 1981 Dollars ^(a,b)
Utilities	83.1
Taxes	16.6
License Fee	298.0
Insurance	<u>138.5</u>
Total	536.2

(a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

(b) Contingency of 25% is not included here.

Costs of Specialty Contractors

Major specialty contractor costs include the costs of engineering support, environmental monitoring, and laundry of protective clothing. The bases and assumptions used to estimate these costs are given in Section D.3 of Appendix D. Contractor costs for providing engineering support represent 1% of the costs of accident cleanup following the reference accident.

The sanitary lagoon is decommissioned as described in Section D.2 of Appendix D. Cost for this activity is estimated to be about \$16,900 without contingency.

8.3.2 Summary of Accident Cleanup Costs

Based on the assumptions listed as key study bases in Section 2.2 of Chapter 2, the cleanup activities described in Section 8.2, and the costs of accident cleanup summarized here, the total estimated costs and estimated time requirements for accident cleanup at the reference MOX plant are shown in Table 8.3-7. Accident cleanup is estimated to cost about \$4.6 million and to require 1.5 years for completion. These costs and times include those for planning and preparation as well as the actual costs and times for cleanup operations.

TABLE 8.3-7. Summary of Time and Cost Estimates for
Accident Cleanup at the Reference MOX
Plant Following the Postulated Accident

Activity	Time (years)	Cost (\$ millions)(a)
Preparations for Accident Cleanup	0.5	0.3
Accident Cleanup	<u>1.0</u>	<u>4.3</u>
Totals	1.5	4.6

(a) Costs are in 1981 dollars and include 25% contingency.

8.3.3 Sensitivity of Accident Cleanup Costs to Various Factors

Accident cleanup is an activity that takes place at a time when conditions in a plant are uncertain and when social, political, financial, and regulatory constraints can affect the progress and costs of cleanup activities. In addition, the processing of accident-generated wastes may require the use of specialized procedures and techniques. The sensitivity of accident cleanup costs to various factors is addressed in this section. Several factors can influence accident cleanup costs:

- the potential for delays in accident cleanup due to various causes such as greater damage or contamination than expected, requirements for the design and construction of specialized systems for processing wastes, social or political constraints, regulatory concerns, and financial difficulties

- the need to use more complicated and expensive processing systems, if they are required, for such activities as waste solidification
- the temporary inability to dispose of radioactive wastes offsite due to technical, political, or regulatory constraints
- a requirement to use outside contractors to complete certain cleanup tasks because sufficient staff are not available.

In addition, considerations related to plant design may result in a need for special buildings and equipment for accident cleanup that may be different for different facilities.

Estimates of the sensitivity of accident cleanup costs to the various factors listed above are given in the following subsections.

Sensitivity of Costs to Delays in Completing Accident Cleanup

Delays in accident cleanup can result in increased costs in several ways. One is the added costs that are incurred in paying staff for additional time onsite. Another is the added costs of such items as energy and insurance.

It is beyond the scope of this report to analyze all of the social or political constraints or other regulatory or financial problems that could cause delays. The study assumes the existence of delays and estimates the associated costs. Estimates of the added costs resulting from a one-year delay in either preparations for accident cleanup or the actual accident cleanup activities at the reference MOX plant are given in Table 8.3-8. Since the added costs of delays are expected to vary approximately linearly with length of the delay period, the added costs of delays for shorter or longer time periods can be inferred from the table.

TABLE 8.3-8. Estimated Added Costs of Delays in Completing Accident Cleanup at the Reference MOX Plant

<u>Parameter</u>	<u>Value Following Reference Accident</u>
Reference Time ^(a)	1.0 years
Reference Cost ^(a)	\$4.6 million
Added Cost of 1-yr Delay	\$1.8 million

(a) Reference times and costs are taken from Table 8.3-7.

Other Potential Post-Accident Costs

There are other post-accident costs that are outside the scope of this study. These include the costs of any activities leading to refurbishment of a plant if it is decided to restart the plant. The costs of refurbishment of a facility for restart are not included in this study. Although it is considered that accident cleanup activities (and therefore costs) are relatively independent of whether the plant is ultimately decommissioned or refurbished, the period following system decontamination would result in either decommissioning or additional cleanup leading to refurbishment. The costs of additional cleanup leading to refurbishment could be significant.

Effect of Plant Size on Accident Cleanup Costs

A detailed analysis of the effect of plant size on the cost of accident cleanup following a serious accident is outside the scope of this study. Many individual cost items contribute to the total cost of accident cleanup. To determine the effect of plant size on accident cleanup costs, it is necessary to evaluate the effect of plant size on each cost item. Complex relations may exist between plant size and these individual items, and factors other than plant size can influence accident cleanup costs. Site support and plant operations costs, which are a function of the work force required for these activities and of the time required for preparations for cleanup and for the actual cleanup activities, may be increased. The personnel requirement for site support and plant operations is a function of plant size and of several other factors such as plant design and management philosophy. As indicated previously in this section, the time requirement for accident cleanup is affected by requirements for the design and fabrication of specialized systems and equipment, social and political constraints, regulatory concerns, and financial difficulties, as well as by plant size. It is concluded that no simple relation exists between plant size and the total cost of accident cleanup.

8.4 ACTIVITIES AND MANPOWER REQUIREMENTS FOR DECOMMISSIONING AT A REFERENCE MIXED-OXIDE PLANT

The actual decommissioning of an accident-damaged MOX plant begins following completion of the accident cleanup activities. This section contains information concerning activities and manpower requirements for the post-accident decommissioning of the reference MOX plant via the DECON decommissioning alternative. The information is summarized from the detailed discussion of post-accident decommissioning presented in Section D.4 of Appendix D.

The post-accident decommissioning analyses in this study use the results of previous analyses of MOX plant decommissioning following normal shutdown,⁽²⁾ with appropriate modifications as necessary to account for post-accident conditions. The decommissioning analyses in this study are based on the assumption that the plant has experienced the reference accident described in Chapter 6.

A basic assumption of the decommissioning analyses presented here is that all radioactive waste materials resulting from accident cleanup and from decommissioning are shipped offsite for disposal at the time of decommissioning.

8.4.1 Comparison of Decommissioning Following an Accident and Following Normal Shutdown

Under normal circumstances, decommissioning of a MOX plant follows the orderly shut down of the facility at the end of its planned operating life. However, conditions at an accident-damaged plant are significantly different from normal, with increased levels of radioactive contamination in certain plant areas, and possible physical damage to plant equipment and services. Comparisons of decommissioning activities following normal shutdown with those following an accident are presented in detail in Table D.4-1 of Appendix D. A summary of the differences between normal-shutdown and post-accident decommissioning is presented in this section.

It is assumed in this study that accident cleanup activities are completed prior to the start of the actual decommissioning. The tasks performed during accident cleanup are postulated to be independent of the alternative (DECDN, SAFSTOR, or ENTOMB) chosen to complete the decommissioning. In carrying out accident cleanup, certain tasks that are part of normal-shutdown decommissioning are completed (e.g., comprehensive radiation surveys of the facility). In addition, significant portions of other such tasks are undertaken (e.g., surface decontamination in the contaminated areas).

Accident cleanup also results in certain new tasks that must be completed during decommissioning. These new tasks are the removal of new equipment for processing wastes and the decommissioning of the onsite storage structures specially constructed for the handling of accident-cleanup wastes.

A number of decommissioning tasks are common to both post-accident and normal-shutdown decommissioning. However, the physical and radiological condition of the plant resulting from an accident leads to qualitative changes in these tasks. The major change is that, following an accident, radiation doses to decommissioning workers are higher than those following normal shutdown because of the increased levels of contamination on equipment, piping, and structural surfaces. Because of limitations on the radiation doses that can be accumulated by individual workers,⁽¹⁾ the higher radiation levels result in the need for a greater number of decommissioning workers.

Although the accident cleanup activities remove a large portion of the accident-generated contamination in the plant, the accident does have some impact on decommissioning activities and manpower requirements. The primary impact is on the radiation dose rates to decommissioning workers, which increase because of the accident.

8.4.2 DECON Activities and Manpower Requirements

DECON is the decommissioning alternative that leads to the earliest release of the facility and site for unrestricted use and to the earliest termination of the facility's nuclear license. Compared to the other decommissioning alternatives, DECON results in a greater occupational radiation dose and a greater cost in the first few years after completion of accident cleanup. Planning and preparation activities, decontamination and dismantlement activities, and manpower requirements for post-accident DECON at the reference MOX plant are described in this section.

Planning and Preparation Activities

The success of the project is dependent upon planning and execution of necessary preparatory work prior to completion of the accident cleanup campaign that precedes decommissioning. Planning and preparation for DECON is assumed to be accomplished during the accident cleanup campaign.

Planning and preparation activities for DECON at the reference MOX plant following normal shutdown are discussed in Section 9.1 of Reference 2. Planning and preparation activities applicable to post-accident DECON can be summarized as follows:

- Satisfying regulatory requirements - primarily involves 1) providing the necessary documentation for amending the facility operating license, and 2) obtaining an NRC dismantlement order. In addition, the licensee must submit a radioactive waste handling plan, a quality assurance plan, an environmental report, security and safeguards plans, and possibly updated information concerning the licensee's financial qualifications.
- Gathering and analyzing data - provides input to the documentation and establishes the bases for developing work plans and procedures. The bulk of the required data is assumed to be available as a result of the accident cleanup activities.
- Developing detailed work plans and procedures - provides the decommissioning staff with all the information required to actually carry out the decommissioning tasks. The plans and procedures cover all aspects of the project, and quality assurance, security, and environmental constraints are considered.
- Designing, procuring, and testing special equipment - ensures the availability and proper operation of the required equipment. The testing also serves to train personnel in the use of the equipment and to provide pertinent data on its operation.

- Selecting and training staff - ensures the availability of competent personnel and enables decommissioning to proceed smoothly, safely, and expeditiously.
- Selecting specialty contractors - allows certain specialized decommissioning tasks outside of the expertise or capability of the decommissioning staff to be performed by experts, increasing the overall efficiency and safety of the decommissioning project.

Decontamination and Dismantlement Activities

The decontamination and dismantlement activities during post-accident DECON at the reference MOX plant are very similar to those during DECON following normal shutdown, which are described in detail in Reference 2. The decontamination and dismantlement activities during post-accident DECON are summarized here, with emphasis on activities that differ from those during normal-shutdown DECON.

Following the postulated accident and the subsequent accident cleanup campaign, radioactive contamination levels in the reference MOX plant exceed those that would be present following normal shutdown to an extent that depends on the severity of the accident and on the particular location in the plant. Therefore, to reduce radiation doses to decommissioning workers to practicable levels, the major access routes used by the DECON workers and "hot spots" outside of the access routes that can materially affect worker doses are cleaned up or shielded. This task is undertaken at the start of DECON to obtain the maximum dose-reduction benefits, using the same methods as postulated for similar tasks during accident cleanup or normal-shutdown DECON. The level of effort required for this task is a function of the amount of contamination present.

Decontamination of internal surfaces in the building is initiated during accident cleanup to reduce radiation doses to the cleanup workers. However, the bulk of this work is still carried out during DECON, particularly the removal of contaminated structural material. The methods used during post-accident DECON (i.e., concrete spalling, disassembly or cutting of metal components, etc.) are the same as those employed during DECON following normal shutdown. However, accident-generated contamination results in a somewhat greater level of effort and a greater volume of radioactive waste material produced than following normal shutdown.

Manpower Requirements

The packaging and shipping of radioactive wastes generated during DECON is handled by standing crews that are available over the entire duration of the tasks that generate the waste. The amount and contamination levels of the

wastes handled by these crews and, consequently, the radiation doses to these workers are anticipated to be greater than during normal-shutdown DECON. A major factor affecting manpower requirements for this task is the limitation on radiation doses to individual workers.⁽¹⁾

The accident cleanup wastes stored in the onsite waste storage structures constructed during accident cleanup are removed from the site and shipped to appropriate repositories before the completion of DECON. The waste handling crews described previously perform this task, but the costs for waste transport and disposal are borne by the accident cleanup campaign. After the wastes are removed, the onsite facilities are structurally decontaminated using the same methods employed in the building.

8.4.3 DECON Schedule

The overall schedule and sequence of events for DECON at the reference MOX plant following the reference accident and the subsequent accident cleanup campaign is shown in Figure 8.4-1. Detailed schedules for DECON are presented in Section D.4 of Appendix D. Planning and preparation activities for DECON begin one year prior to the completion of the accident cleanup campaign that precedes DECON, as discussed previously in Section 8.4.1.

DECON begins in the wet-process areas, which comprises the major effort for the decommissioning staff. The succeeding activities are performed on a schedule that depends on the need for the plant areas involved and on the availability of manpower. As shown in Figure 8.4-1, DECON following the reference accident is completed in about 90 weeks.

8.4.4 DECON Staff Requirements

In this subsection, the organization of the decommissioning staff and the types and numbers of decommissioning workers needed for post-accident DECON at the reference MOX plant are discussed.

Organization of the Decommissioning Staff

The staff for post-accident decommissioning of the reference MOX plant is organized as shown in Figure 8.4-2.

The primary decommissioning activities are performed on a one-shift, five-day-week basis. However, selected support activities (i.e., decontamination and radwaste system operation) and security functions are carried out on three shifts, around the clock, seven days per week.

Further discussion pertaining to the staff organization and the functions of key staff members can be found in Chapter 9 of Reference 2.

DECON Manpower

Based on the detailed schedules for DECON activities in the various plant buildings, given in Section D.4 of Appendix D, the types and number of decommissioning workers needed to efficiently complete the radiation-zone work in the allotted time are determined. However, because whole-body radiation doses to the decommissioning workers are limited in accordance with 10 CFR 20.101,⁽¹⁾ the decommissioning worker requirements must be adjusted upward so that average individual radiation doses do not exceed five man-rem/man-year. For DECON at the reference MOX plant following the reference accident, decommissioning worker manpower requirements must be increased by a factor of three as shown in Table D.4-3 of Appendix D.

Manpower requirements for management and support staff are primarily a function of the duration of the DECON project. The assumptions used to calculate management and support staff requirements are presented in Section D.4-4 of Appendix D.

Overall staff labor requirements for DECON at the reference MOX plant following the reference accident are given in Table 8.4-1. These requirements are given in equivalent man-years for the planning and preparation phase as well as for the actual decontamination and dismantlement, and include the management and support staff as well as the decommissioning workers. A total effort of about 128.5 man-years is estimated for completion of DECON following the reference accident. For comparison, the total staff labor requirements for DECON at the reference plant following normal shutdown are about 114 man-years.⁽²⁾

8.5 COSTS OF DECOMMISSIONING AT A REFERENCE MIXED-OXIDE PLANT

The costs of decommissioning the reference MOX plant following a postulated accident and subsequent accident cleanup campaign are developed in detail in Section D.5 of Appendix D. These costs are summarized in this section. Costs of accident cleanup are summarized previously in Section 8.3.

The principal assumptions made in developing the cost estimates for the post-accident decommissioning of the reference MOX plant are as follows:

- The costs of decommissioning are considered separate from the costs of accident cleanup. The costs of decommissioning systems and facilities installed during accident cleanup are included with the decommissioning costs.
- Cost estimates are made for decommissioning via the DECON decommissioning alternative.

FIGURE 8.4-1. Sequence for DECON of Major Areas at the Reference MOX Plant Following an Accident

FIGURE 8.4-2. Postulated Organization Chart for DECON Following an Accident at the Reference MOX Plant

TABLE 8.4-1. Overall Staff Labor Requirements for DECON at the Reference MOX Plant Following the Reference Accident

Position	Staff Labor Requirement (man-years) in Decommissioning Phase	
	Planning and Preparation	DECON
Project Manager	1.0	1.75
Project Engineer	1.0	1.75
Health and Safety Supervisor	1.0	1.75
Health Physics Technician		10.0
Security Force Supervisor		1.75
Security Patrolman		14.0
Contracts and Accounting Specialist	1.0	1.75
Accountant		0.87
SNM Accounting Specialist		1.75
SNM Accounting Clerk		0.87
Radioactive Shipment Specialist	7.0	1.75
Q.A. Engineer	7.0	1.75
Planning Engineer	1.0	1.75
Engineering Technician	3.0	1.75
Maintenance and Crafts Supervisor		3.50
Maintenance Man		3.50
Custodian		1.75
Craftsman		18.0
Operations Supervisor	2.00	3.50
Foreman	3.0	11.5
Technician		40.0
Secretary	3.00	3.50
Total Man-Years	18.0	128.5

- To the maximum extent possible, the decommissioning staff is drawn from the technical and operational staffs of the plant and from the accident cleanup staff. The support services and the part-time assistance of the accident cleanup staff are assumed to be available during planning and preparations for decommissioning with only nominal costs to the decommissioning portion of the overall cleanup and decommissioning project.
- The amended facility license allowing possession but not operation of the plant is in place at the end of the accident cleanup campaign, permitting decommissioning staff to work in direct contact with these systems.
- Piping, equipment, and significant portions of the plant surfaces in the MOX plant are contaminated and require proper management as radioactive wastes.
- The cost estimates presented here are based on the ability to dispose of the radioactive waste materials at offsite facilities at the time of decommissioning.
- Costs are based on 1981 prices and wage rates.

8.5.1 DECON Costs

The estimated cost of DECON at the reference MOX plant following the reference accident, including a 25% contingency, is \$11.4 million, as summarized in Table 8.5-1. Details of the development of these costs are discussed in Section D.5 of Appendix D. Information pertaining to the individual cost categories is summarized in the following subsections.

Costs of Staff Labor

The costs of staff labor during DECON following the reference accident are shown in detail in Table D.5-2 of Appendix D. A major portion of the total DECON cost is associated with staff labor. A total staff labor cost of about \$6.2 million is estimated for DECON following the reference accident. Specialty contractor labor is not included in the labor costs given here, but rather in the costs of specialty contractors presented at the end of Section 8.5.

Costs of Waste Management

The estimated costs for containers, transportation, and disposal of radioactive wastes from DECON at the reference MOX plant are summarized in Table D.5 of Appendix D. Cost estimates are based on projected packaging and shipping

TABLE 8.5-1. Summary of Estimated Costs of Post-Accident DECON at the Reference MOX Plant^(a)

<u>Cost Category</u>	<u>Cost in Millions of 1981 dollars^(b)</u>	<u>Percent of Total</u>
Staff Labor		
Planning and Preparation	0.946	10.3
Decommissioning	5.269	57.5
Equipment and Supplies	1.217	13.3
Disposal of Radioactive Material	0.820	9.0
Miscellaneous Owner Expense	0.670	7.3
Specialty Contractors	<u>0.238</u>	<u>2.6</u>
Subtotal	9.160	100.0
25% Contingency	2.290	
Total Decommissioning Costs	11.450	

(a) Summary does not include the cost of packaging and disposal to shallow-land burial of the sludge from the sewage lagoons that would add an additional \$0.75 million in waste disposal and contingency costs.

(b) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

requirements summarized in Table D.4-1 and on waste management cost data in Section D.4. Waste management cost details are given in Section D.5.

The total waste management cost for DECON is estimated to be about \$0.8 million without contingency. If packaging and disposal of the sludge in the sewage lagoon is required, it would add about \$0.8 million (without contingency) to this cost.

Costs of Equipment and Supplies

The estimated costs of the equipment and supplies that are required for post-accident DECON at the reference MOX plant are presented in Table D.5-4 of Appendix D. The estimated total cost of equipment and supplies is approximately \$1.2 million, which is about 13% of the total DECON cost following the reference accident.

Costs of Miscellaneous Owner Expense

Several miscellaneous owner expenses are incurred during DECON. These include utilities, taxes, license fee, and insurance. The estimated costs of

these items are given in Table D.5-5 of Appendix D. The total estimated cost of miscellaneous supplies during DECON at the reference MOX plant after the reference accident is about \$0.7 million, which represents about 7% of the total DECON cost.

Costs of Specialty Contractors

The estimated requirements for and costs of specialty contractors during DECON are discussed in Section D.5.6 of Appendix D. These specialty contractors perform electropolishing, removal of the wall tanks, and decommissioning of the sanitary lagoons.

The total cost of specialty contractors during DECON at the reference MOX plant is less than \$0.2 million. Specialty contractors account for about 3% of the total DECON costs following a reference accident.

REFERENCES

1. U.S. Code of Federal Regulations, Title 10, Part 20.101, "Radiation Dose Standards for Individuals in Restricted Areas." (10 CFR 20.101), U.S. Government Printing Office, Washington, D.C., 1980.
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9.0 URANIUM FUEL FABRICATION PLANT CLEANUP AND DECOMMISSIONING

The first activities following an accident at a uranium fuel fabrication (U-Fab) plant are designed to bring the accident under control and to stabilize the facility to prevent releases of radioactivity. Initial stabilization is followed by accident cleanup, which is followed by either refurbishment or decommissioning of the facility. This chapter provides information about the activities and manpower requirements and costs for cleanup and decommissioning following a serious accident at a U-Fab plant. It does not analyze the detailed activities related to refurbishment of the facility.

This chapter summarizes the detailed discussion of accident cleanup, costs, and decommissioning of the U-Fab plant presented in Appendix E of Volume 2. General information on decontamination, waste treatment, and packaging and disposing of wastes is given in Appendix C of Volume 2. The purpose of accident cleanup is discussed in Section 9.1. Activities and manpower requirements are discussed in Section 9.2. The cost of cleanup following the reference accident is discussed in Section 9.3. Activities and manpower requirements for decommissioning are discussed in Section 9.4. The costs of decommissioning following an accident are summarized in Section 9.5.

9.1 PURPOSE OF ACCIDENT CLEANUP AT A REFERENCE URANIUM FUEL FABRICATION PLANT

The reference accident postulated for the U-Fab plant (see Chapter 6) results in contamination of the building and equipment. Either decommissioning or refurbishment of a plant following a postulated accident would begin with a cleanup campaign with the following principal goals:

1. to reduce the levels of radioactive contamination present on building surfaces and equipment, thereby reducing the radiation dose received by workers engaged in subsequent decommissioning or refurbishment operations
2. to collect, package for disposal, and dispose of the readily dispersible radioactivity present in the plant.

To achieve these goals, cleanup is postulated to include 1) initial decontamination of building surfaces and decontamination or disposal of some equipment, and 2) solidification and packaging of wastes from cleanup operations.

Cleanup activities must establish conditions within the U-Fab plant that permit reasonable occupancy times without excessive radiation exposure to workers engaged in subsequent decommissioning operations. In keeping with ALARA principles, decontamination during cleanup is restricted to operations

that result in the greatest reduction in residual radioactive contamination with the least radiation dose to decommissioning workers.

The tasks included in cleanup and the procedures used to accomplish these tasks are assumed to be essentially independent of whether the facility is ultimately restarted or decommissioned, and if decommissioned, which alternative (DECON or SAFSTOR) is chosen. Because cleanup activities are similar, the requirements and tasks presented in this section are considered to be representative of the activities carried out during cleanup independent of the ultimate use of the plant. The work required to accomplish each task is, of course, affected by the severity of the accident. Technical requirements for accident cleanup are discussed in the following sections.

9.2 ACTIVITIES AND MANPOWER REQUIREMENTS FOR ACCIDENT CLEANUP AT A REFERENCE URANIUM FUEL FABRICATION PLANT

Following an accident, the plant is shut down and preparations begin for cleanup. Planning and preparation activities and estimates of the manpower required for their performance are described in this section.

9.2.1 Planning and Preparation Activities

Planning and preparation activities that must be performed prior to the operational phase of building cleanup include:

- contaminated area entry and data acquisition
- prepare documentation for regulatory agencies
- design, fabricate, and install special equipment
- develop detailed work plans and procedures
- select and train accident cleanup staff.

Contaminated Area Entry and Data Acquisition

Data on the post-accident radiological and physical condition of the reference U-Fab plant are obtained and analyzed during the planning and preparations phase. These data include measurements of contamination levels and radiation exposure rates in the building, estimates of physical damage to structures and equipment, and information about the operational status of plant systems and services. The data provide a basis for planning accident cleanup operations and also provide some of the information needed to prepare documentation for regulatory agencies.

Data on post-accident conditions in the reference plant are obtained by teams of workers (generally two to four persons) who enter the contaminated areas and obtain information during these entries, depending on radiation exposure levels encountered.

Prepare Documentation for Regulatory Agencies

A major planning task is the preparation by the licensee of the necessary documentation to amend the facility operating license to maintain the plant in a safe shutdown condition and to obtain regulatory approvals to proceed with cleanup operations. Documentation that must be provided by the licensee prior to the start of cleanup operations is summarized in Section E.2 of Appendix E (Volume 2). The time needed to furnish information to regulatory agencies, issue environmental statements and assessments, and secure regulatory approvals to proceed with specific cleanup tasks is a critical factor in determining the length of the planning and preparations period and, hence, could delay the start of actual cleanup operations.

Design, Fabricate, and Install Special Equipment

An evaporator/solidification facility to process the decontamination solutions generated, also a laundry, may be required for cleanup. Items such as evaporators and laundry facilities are commercially available. This study assumes that for the reference facility, except for the storage facilities indicated, sufficient space is available for the addition of necessary special equipment for cleanup.

Develop Detailed Work Plans and Procedures

Work plans and procedures are based on an evaluation of the condition of the plant following an accident and on the requirements for cleanup. Work plans are included in documentation provided to the NRC with the request for license amendment. The detailed plans and procedures contain all the information required to actually carry out the cleanup tasks. They address the following items:

- regulatory requirements and constraints
- decontamination methods and procedures
- schedules and sequences of events
- manpower requirements
- equipment requirements
- contamination control
- radiological and industrial safety
- packaging and disposal of radioactive wastes
- quality assurance.

Physical security and environmental constraints are also considered. Plans are updated as the cleanup work proceeds and additional data on the physical and radiological status of the facility become available.

Select and Train Accident Cleanup Staff

Operations staff are selected and trained for cleanup while preparations for cleanup are being made. Because detailed knowledge of and familiarity with the facility being decontaminated increase the effectiveness of cleanup workers, key staff positions are filled, whenever possible, with personnel familiar with the construction and operation of the plant. Additional training in specific cleanup tasks is provided, with emphasis on the use of new or unique equipment and procedures.

Training of cleanup staff continues throughout the accident cleanup period. Because of the possible exposure rates encountered and the need to limit individual radiation doses, a number of persons may be involved in accident cleanup operations. Some may be unfamiliar with the plant, and some will be unfamiliar with the basic principles of radiation protection. These persons require an orientation in the layout of the plant and in basic radiation protection procedures as well as specific instruction in the tasks to be performed.

9.2.2 Time Requirements for Preparations for Accident Cleanup

Time requirements for planning and preparation depend on several factors, including time needed to design, fabricate and test special equipment and facilities, and time required to secure regulatory approvals to proceed with specific cleanup tasks. The time required to secure regulatory approvals for cleanup operations is a critical factor in determining when these operations can begin.

To provide a basis for computing staff labor costs, preparations for accident cleanup are assumed to require 0.5 years following the reference accident. To enable the user of this study to compute planning and preparation costs for a time period other than those assumed in this study, annual staff labor costs for preparations for accident cleanup are given in Table E.3-1 of Appendix E.

9.2.3 Staff Requirements for Preparations for Accident Cleanup

The plant operations staff has responsibility for maintaining the plant in a safe shutdown condition. Plant operations staff also assist the cleanup planning staff in acquiring of data on the radiological and physical condition of the building and in installing and testing equipment and systems required for accident cleanup.

Major activities of cleanup planning include:

- preparation of documentation for regulatory agencies

- preparation of design specifications for special facilities and equipment
- preparation of detailed work plans and work schedules
- acquisition of data on the radiological and physical condition of the plant
- testing of equipment and procedures to be used in cleanup operations
- installation or repair of systems required for accident cleanup (e.g., reroute piping connections, install systems for remote monitoring, etc.).

Site support includes radiological health, industrial safety, plant security, procurement and accounting, and quality assurance services.

Estimated staff labor requirements for preparations for accident cleanup are shown in Table 9.2-1. The staff labor requirements shown in this table do not include contractor labor to provide engineering support services during preparations for cleanup. An engineering contractor is assumed to provide assistance to the staff in preparing documentation for regulatory agencies, and in developing detailed work plans and schedules.

Contractor labor for engineering services during preparations for cleanup is estimated to be four man-years per year (two total man-years) following the reference accident.

9.2.4 Accident Cleanup

Procedures, schedules, and staff labor requirements for accident cleanup at the reference U-Fab plant following the reference accident are summarized in the following subsections. Details of the accident cleanup in the U-Fab plant are given in Section E.2 of Appendix E.

The reference accident is described in Chapter 8. Accident cleanup operations are assumed to reduce general area radiation exposure rates in the building. As discussed in Section E.1, decontamination activities are not designed to reduce exposure rates to levels permitting unrestricted use of the facility, but only to limit the doses to workers engaged in accident cleanup. Additional decontamination would be required during decommissioning (or refurbishment) to limit the doses to workers engaged in these activities.

TABLE 9.2-1. Estimated Staff Labor Requirements for Preparations for Accident Cleanup at the Reference U-Fab Plant

<u>Position</u>	<u>Staff Labor Requirements (man-years) for Preparations for Accident Cleanup^(a)</u>
Project Manager	0.5
Project Engineer	0.5
Health and Safety Supervisor	0.25
Contracts and Accounting Specialist	0.25
Q. A. Engineer	0.5
Radioactive Shipment Specialist	0.25
Planning Engineer	0.5
Operations Supervisor	0.5
Maintenance and Crafts Supervisor	0.5
Secretary	<u>0.5</u>
Total Man-Years	4.25

(a) Based on a preparations for cleanup period of 0.5 years.

Procedures for Accident Cleanup

Tasks that must be completed for accident cleanup at the reference U-Fab plant include the following:

1. Decontaminate the building to permit access.
2. Install the evaporator solidification facility.
3. Continue the decontamination of building surfaces and equipment.
4. Process any contaminated liquids from cleanup operations.
5. Replace contaminated filters.
6. Solidify and package wastes from accident cleanup.
7. Treat and dispose of wastes from cleanup operations.
8. Perform radiation survey to determine the extent of residual contamination following accident cleanup and to verify the effectiveness of cleanup operations.

Initial Decontamination. The objective of initial decontamination at the reference U-Fab plant is to reduce surface contamination levels and resultant radiation exposure levels to permit reasonable occupancy times for workers engaged in cleanup operations. The reduction of general area radiation exposure rates requires that "hot spots" be shielded by using lead sheet or lead bricks, high-density concrete blocks, or containers filled with water.

A general discussion of procedures for the decontamination of surfaces and equipment is given in Appendix C of Volume 2. Initial decontamination of the plant includes the following steps:

1. Remove and package easily disposable debris and small items of contaminated equipment.
2. Employ high-pressure hose wash techniques for semi-remote decontamination of building surfaces and equipment.
3. Decontaminate and refurbish or replace any essential support systems that have been damaged.
4. Perform hands-on decontamination of selected areas where significant reductions in radiation exposure can be achieved with modest effort. Decontaminate floors by scrubbing
5. Provide local shielding of any "hot spots."

Processing of Contaminated Liquids. Contaminated liquids that must be processed during accident cleanup include chemical decontamination solutions generated during decontamination of building surfaces. Estimated volumes of contaminated liquids from accident cleanup following the postulated accident and the curies of radioactivity removed from these liquids are given in Appendix E.

Chemical decontamination solutions from initial cleanup operations have radionuclide concentrations in the range from 0.5 to 5 Ci/m³. Evaporation is a suitable alternative for treatment of these wastes. An evaporator/solidification facility is postulated to be obtained from a commercial supplier and installed in the U-Fab plant during preparations for cleanup. The evaporator bottom liquids are solidified with vinyl ester styrene and packaged in stainless steel liners for interim onsite storage in the shielded storage facility that is constructed during preparations for accident cleanup.

Waste Treatment and Disposal. Radioactive wastes from accident cleanup operations can be divided into three categories, as follows:

1. Solid Materials. Dry radioactive wastes generated from decontamination. These materials consist of trash, contaminated equipment and material, and HEPA and roughing filters.
2. Process Solids. Contaminated sludges and process solid wastes that arise from the treatment of decontamination liquids. These solid wastes include filter cartridge assemblies and evaporator bottoms.
3. Chemical Decontamination Solutions. Liquid decontamination wastes that have not been treated to generate process solids. These wastes are immobilized by incorporation in cement or in vinyl ester styrene.

The alternatives assumed in this study for the packaging and disposal of these wastes and the waste volumes generated during accident cleanup in the building are given in Table E.2-1 of Appendix E.

Based on the criteria of 10 CFR, Part 61,⁽¹⁾ all of the wastes from U-Fab plant cleanup, except the process solids, are assumed to be transported to a shallow-land burial ground for disposal.

Table 9.2-2 gives estimated weights and volumes of wastes generated during cleanup of the reference U-Fab plant, together with packaging, shipping and disposal requirements for these wastes. Detailed information about waste quantities is given in Section E.2 of Appendix E.

Schedules for Accident Cleanup

The schedule for cleanup at the reference U-Fab plant following the reference accident is shown in Figure 9.2-1. Time requirements for cleanup depend on accident severity and are based on the cleanup procedures summarized in Section 8.2.4. Cleanup in the U-Fab plant is estimated to require approximately 6 months following the reference accident.

Details of cleanup schedules and the bases and assumptions used to prepare these schedules and to estimate cleanup worker requirements are given in Section E.2 of Appendix E.

Staff Labor Requirements for Accident Cleanup

A staff organization postulated for cleanup at the reference U-Fab plant is shown in Figure 9.2-2. The staff includes a plant operations branch and several site support branches (e.g., engineering, health and safety, security, contracts and accounting, and quality assurance) as well as the staff actually involved in cleanup of the plant.

TABLE 9.2-2. Waste Disposal Requirements of Accident Cleanup at the Reference U-Fab Plant^(a,b)

Waste Category	Shipping Weight (kg)	Shipping Volume (m ³)	Container Type	Number of Shipments
HEPA and Roughing Filters	16 000	50	Plywood Boxes	6
Solidified Decontamination	20 000	8	Plywood Boxes	2
Trash	<u>17 000</u>	<u>12</u>	Plywood Boxes	<u>4</u>
Totals	53 000	70		12

- (a) Number of figures shown is for computational accuracy, and does not imply precision to three significant figures.
- (b) Number of containers per shipment is limited by weight rather than by volume.

(a) SCHEDULE DETAILS ARE GIVEN IN SECTION E.2 OF APPENDIX E.

(b) THE TOTAL TIME REQUIREMENT FOR ACCIDENT CLEANUP IN THE U-FAB BUILDING FOLLOWING THE REFERENCE ACCIDENT IS 0.5 YEARS.

FIGURE 9.2-1. Sequence and Schedule for Accident Cleanup at the Reference U-Fab Plant Following the Reference Accident^(a)

FIGURE 9.2-2. Postulated Organization Chart for Accident Cleanup at the Reference U-Fab Plant

Estimated staff labor requirements for cleanup are shown in Table 9.2-3. These labor requirements are 10.25 man-years for cleanup following the reference accident. The staff labor requirements (the man-hours for personnel engaged in cleanup operations in the building) shown in Table 9.2-3 have been adjusted upward by appropriate factors to ensure that the estimated occupational radiation dose for individual workers does not exceed 5 rem/year.⁽²⁾ An explanation of the adjustment factors used to obtain man-years for accident cleanup staff labor is given in Section E.2 of Appendix E. The staff labor requirements shown in Table 9.2-3 are used to compute staff labor costs for accident cleanup.

9.3 COSTS OF ACCIDENT CLEANUP AT A REFERENCE URANIUM FUEL FABRICATION PLANT

The costs of cleanup activities at the reference U-Fab plant following the postulated accident are summarized in this section. These costs are developed in detail in Appendix E. Costs are in 1981 dollars.

TABLE 9.2-3. Estimated Staff Labor Requirements for Accident Cleanup at the Reference U-Fab Plant

<u>Position</u>	<u>Staff Labor Requirements (man-years) for Cleanup Following the Reference Accident (a)</u>
Project Manager	0.5
Project Engineer	0.5
Health and Safety Supervisor	0.5
Health Physics Technician	0.75
Security Force Supervisor	0.5
Security Patrolman	1.0
Contracts and Accounting Specialist	0.5
SNM Accounting Specialist	0.25
Q. A. Engineer	0.25
Radioactive Shipment Specialist	0.25
Maintenance and Crafts Supervisor	0.5
Custodian	0.25
Craftsman	2.0
Operations Supervisor	0.5
Technician	6.0
Secretary	<u>1.5</u>
Total Man-Years	15.75

(a) Cleanup staff labor requirements are adjusted to limit individual radiation doses to 5 rem/yr.⁽²⁾

Cleanup costs would be similar whether the plant is refurbished for restart or decommissioned. The costs presented here are believed to be a good representation independent of the ultimate use of the plant. Costs of activities related to refurbishment and restart of a plant beyond accident cleanup activities are not included in this study. Costs of decommissioning following accident cleanup are summarized in Section 9.5.

9.3.1 Estimated Costs of Accident Cleanup

The estimated costs of accident cleanup at the reference U-Fab plant are summarized in Table 9.3-1. These activities are estimated to require about 0.5 year (plus 0.5 year for planning and preparations), at a cost of approximately \$2.2 million.

Manpower represents about 53% of the total cost of accident cleanup. Manpower costs include both support staff and accident cleanup worker labor.

TABLE 9.3-1. Summary of Estimated Costs of Accident Cleanup
at the Reference U-Fab Plant

<u>Cost Category</u>	<u>Cost in Millions of 1981 Dollars^(a)</u>	<u>Percent of Total</u>
Manpower		
Planning and Preparation	0.266	15.3
Accident Cleanup	0.657	37.9
Equipment and Supplies	0.230	13.3
Disposal of Radioactive Material	0.240	13.8
Miscellaneous Owner Expense	0.302	17.4
Specialty Contractors	<u>0.040</u>	<u>2.3</u>
Subtotal	1.735	100.0
25% Contingency	<u>0.434</u>	
Total Costs of Accident Cleanup	2.169	

(a) Number of figures shown in for computational accuracy and does not imply precision to the nearest thousand dollars.

In Table 9.3-1, manpower costs for planning and preparation activities and for cleanup activities are shown separately. Manpower costs include onsite labor for packaging radioactive waste materials for shipment. Labor costs related to radioactive waste transportation are included with waste management costs.

Waste management accounts for about 14% of the total cost of cleanup. Waste management costs include shipping container costs, transportation charges, and fees for waste disposal. All radioactive wastes generated during cleanup are assumed to go to a low-level waste burial ground.

Manpower Costs

Table 9.3-2 shows manpower cost estimates for the planning and preparation phase, and support staff and worker manpower requirements and costs for the active phase of accident cleanup. Planning and preparation will require an estimated 4.25 man-years, at a labor cost of about \$0.27 million. Chemical and physical decontamination of the facility will require an estimated 15.75 man-years, at a labor cost of about \$0.66 million. The total labor cost for accident cleanup of the facility is therefore about \$0.92 million without contingencies.

TABLE 9.3-2. Summary of Staff Labor Costs for Accident Cleanup at the Reference U-Fab Plant

Title or Function	Cost (\$ millions)(a,b)	
	Planning and Preparation	Cleanup
Project Manager	0.051	0.052
Project Engineer	0.044	0.044
Health and Safety Supervisor	0.017	0.033
Contracts and Accounting Specialist	0.011	0.022
Q. A. Engineer	0.030	0.011
Radioactive Shipment Specialist	0.011	0.017
Planning Engineer	0.030	--
Operations Supervisor	0.030	0.030
Maintenance and Crafts Supervisor	0.028	0.028
Secretary	0.014	0.042
Health Physics Technician	--	0.025
Security Force Supervisor	--	0.025
Security Patrolman	--	0.028
SNM Accounting Specialist	--	0.019
Custodian	--	0.011
Craftsman	--	0.050
Technician	--	<u>0.220</u>
Total Costs	0.266	0.657

- (a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.
 (b) Contingency of 25% is not included in these costs.

Estimated Material and Equipment Requirements and Costs of Accident Cleanup

Estimates of material and equipment required for cleanup at the reference U-Fab plant are shown in Table 9.3-3. Equipment requirements are based on procedures described in detail in Appendix F and summarized in Section 9.2. The total cost of material and equipment for cleanup of the reference U-Fab plant is estimated at about \$0.23 million without contingency.

Estimated Waste Management Costs

The estimated costs for containers, transportation and disposal of the radioactive wastes from accident cleanup are summarized in Table 9.3-4. Cost estimates are based on projected packaging and shipping requirements summarized

TABLE 9.3-3. Estimated Costs of Equipment and Supplies for
Accident Cleanup at the Reference U-Fab Plant

<u>Description</u>	<u>Quantity</u>	<u>Estimated Total Cost (\$ thousands)</u>
Ratcheting Pipe Cutter	6 ea.	0.4
Reciprocating Saw	4 ea.	2.8
Nibbler	2 ea.	2.8
High-Velocity Liquid Jet	1 ea.	6.9
Low-Velocity Liquid Jet	2 ea.	5.5
Hydraulic Concrete Surface Spalling Device	1 ea.	6.9
Concrete Drill	3 ea.	0.8
Electric Pneumatic Hammer	2 ea.	1.4
Portable A-Frames	2 ea.	8.3
Portable Wash Sinks	2 ea.	5.5
Portable Spray Clean Booth	1 ea.	5.5
Portable Greenhouse Erection Kit	1 ea.	2.8
Portable Powered Brushes	20 ea.	4.2
HEPA Filter	10 ea.	2.1
Roughing Filter	100 ea.	6.9
Decontamination Chemicals		78.5
Cleaning Supplies		27.7
Expendable Tools		13.9
Protective Clothing (including laundry)		24.9
Office Supplies: Planning and Preparation		13.9
		<u>8.3</u>
Total		230.0

in Table 9.2-2 and on waste management cost data in Section E.3 of Appendix E. The total cost of waste management for cleanup at the reference U-Fab plant is estimated to be about \$0.24 million without contingencies.

Miscellaneous Owner Expenses for Accident Cleanup

Estimated miscellaneous owner expenses for accident cleanup at the reference U-Fab plant are shown in Table 9.3-5.

In estimating the applicable license fee during accident cleanup, it is assumed that the full operating fee would be paid, because scrap recovery and waste treatment facilities at the plant would be used during decommissioning to recover uranium from decommissioning wastes.

TABLE 9.3-4. Estimated Waste Management Costs for Accident Cleanup at the Reference U-Fab Plant

<u>Waste Category</u>	<u>Cost in Thousands of 1981 Dollars^(a,b)</u>			
	<u>Containers</u>	<u>Transportation</u>	<u>Burial</u>	<u>Total</u>
<u>Low-Level Waste Burial</u>				
Solidified Decontamination Liquids	5.8	27.9	13.2	46.9
HEPA and Roughing Filters	38.1	40.6	47.9	126.6
Trash	<u>5.8</u>	<u>14.6</u>	<u>46.1</u>	<u>66.5</u>
Total	49.7	83.1	107.2	240.0

(a) Number of figures shown is for computational accuracy and does not imply three place precision in waste management cost figures.

(b) Contingency of 25% is not included here.

TABLE 9.3-5. Estimated Miscellaneous Owner Expenses During Accident Cleanup at the Reference U-Fab Plant

<u>Cost Category</u>	<u>Cost in Thousands of 1981 Dollars^(a,b)</u>
Utilities	78
Taxes	50
License Fee	97
Insurance	<u>77</u>
Total	302

(a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

(b) Contingency of 25% is not included here.

Costs of Specialty Contractors

Major specialty contractor costs include the costs of engineering support, environmental monitoring, and laundry of protective clothing. The bases and assumptions used to estimate these costs are given in Section E.3 of Appendix E.

Contractor costs for providing engineering support represent 2% of the costs of accident cleanup following the reference accident.

9.3.2 Summary of Cleanup Costs

Total estimates for both costs and time for cleanup at the reference U-Fab plant are based on the assumptions listed as key study bases in Section 2.2 of Chapter 2, the cleanup activities described in Section 9.2, and the costs of cleanup summarized in Section 9.3. These estimates are shown in Table 9.3-6. Accident cleanup is estimated to cost about \$2.2 million and to require one year for completion. These costs and times include those for planning and preparation as well as the actual costs and times for cleanup operations.

TABLE 9.3-6. Summary of Time and Cost Estimates for Accident Cleanup at the Reference U-Fab Plant Following the Postulated Accident

<u>Activity</u>	<u>Time (years)</u>	<u>Cost (\$ millions)(a)</u>
Preparations for Accident Cleanup	0.5	0.3
Accident Cleanup	<u>0.5</u>	<u>1.9</u>
Totals	1.0	2.2

(a) Costs are in 1981 dollars and include a 25% contingency.

9.3.3 Sensitivity of Accident Cleanup Costs to Various Factors

Accident cleanup takes place at a time when conditions in a plant are uncertain and when social, political, financial, and regulatory constraints can affect the progress and costs of cleanup activities. The following factors can influence accident cleanup costs:

- the potential for delays in accident cleanup due to various causes such as greater damage or contamination than expected
- the need to use more complicated and expensive processing systems, if required, for such activities as waste solidification
- the temporary inability to dispose of radioactive wastes offsite due to technical, political, or regulatory constraints

- a requirement to use outside contractors to complete certain cleanup tasks because sufficient staff are not available
- plant design which may result in a need for special buildings and equipment for cleanup that may be different for different facilities.

Delays in cleanup can result in increased costs in several ways. One added cost is that incurred in paying staff for additional time onsite. Another is the added costs of such items as energy and insurance.

It is beyond the scope of this report to analyze all of the social or political constraints or other regulatory or financial problems that could cause delays. The study assumes the existence of delays and estimates the associated costs. Estimates of the added costs resulting from a one-year delay in either preparations for cleanup or the actual cleanup activities at the reference U-Fab plant are given in Table 9.3-7. Since the added costs of delays are expected to vary approximately linearly with length of the delay period, the added costs of delays for shorter or longer time periods can be inferred from the table.

TABLE 9.3-7. Estimated Added Costs of Delays in Completing Accident Cleanup at the Reference U-Fab Plant

<u>Parameter</u>	<u>Value Following Reference Accident</u>
<u>Accident Cleanup</u>	
Reference Time ^(a)	0.5 years
Reference Cost ^(a)	\$2.2 million
Added Cost of 1-yr Delay	\$0.8 million

(a) Reference times and costs are taken from Table 9.3-6.

9.3.4 Other Potential Post-Accident Costs

There are other post-accident costs that are outside the scope of this study. These include the costs of any activities leading to refurbishment of a plant if it is decided to restart the plant.

Although it is considered that cleanup activities (and therefore costs) are relatively independent of whether the plant is ultimately decommissioned or

refurbished, the period following system decontamination would result in either decommissioning or additional cleanup leading to refurbishment. The costs of additional cleanup leading to refurbishment could be significant. However, as noted, they are beyond the scope of this study and are not included.

9.4 ACTIVITIES AND MANPOWER REQUIREMENTS FOR DECOMMISSIONING AT A REFERENCE URANIUM FUEL FABRICATION PLANT

The decommissioning of a U-Fab plant following an accident begins when cleanup activities are completed. This section contains information concerning activities and manpower requirements for the post-accident decommissioning of the reference U-Fab plant via the DECON decommissioning alternative. This information is summarized from the detailed discussion of post-accident decommissioning presented in Section E.4 of Appendix E.

The post-accident decommissioning analyses in this study are based on previous analyses of U-Fab plant decommissioning following normal shutdown,⁽³⁾ with modifications as necessary to account for post-accident conditions. The U-Fab plant is assumed to have experienced the reference accident described in Chapter 6.

9.4.1 Comparison of Decommissioning Following an Accident and Following Normal Shutdown

Under normal circumstances, decommissioning of a U-Fab plant follows the orderly shutdown of the facility at the end of its planned operating life. However, conditions at an accident-damaged plant are significantly different from normal, with increased levels of radioactive contamination in certain plant areas, and possible physical damage to plant equipment and services. A summary of the differences between normal-shutdown and post-accident decommissioning is presented in this section.

It is assumed in this study that cleanup activities are completed prior to the start of the actual decommissioning. The tasks performed during cleanup are postulated to be independent of the alternative (DECON or SAFSTOR) chosen to complete the decommissioning. During cleanup, certain tasks that are part of normal-shutdown decommissioning are completed (e.g., comprehensive radiation surveys of the facility).

In addition, significant portions of other such tasks are undertaken (e.g., surface decontamination in the contaminated areas). Cleanup also results in certain new tasks that must be completed during decommissioning (e.g., the removal of new equipment for processing wastes).

A number of decommissioning tasks are common to both post-accident and normal-shutdown decommissioning. However, the physical and radiological condition of the plant resulting from an accident leads to qualitative changes in these tasks. The major change is that, following an accident, radiation doses to decommissioning workers are higher than those following normal shutdown because of the increased levels of contamination on equipment, piping, and structural surfaces. Because of limitations on the radiation doses that can be accumulated by individual workers,⁽²⁾ the higher radiation levels result in the need for a greater number of decommissioning workers.

9.4.2 DECON Activities and Manpower Requirements

DECON is the decommissioning alternative that leads to the earliest release of the facility and site for unrestricted use and to the earliest termination of the facility's nuclear license. Compared to the other decommissioning alternatives, DECON results in a greater occupational radiation dose and a greater cost in the first few years after completion of accident cleanup. Planning and preparation activities, decontamination and dismantlement activities, and the schedule and manpower requirements for post-accident DECON at the reference U-Fab plant are described in this section.

Planning and Preparation Activities

The success of the project is highly dependent upon planning and execution of necessary preparatory work. Planning and preparation for DECON is assumed to be accomplished during the accident cleanup campaign.

Planning and preparation activities for DECON at the reference U-Fab plant following normal shutdown are discussed in Section 9.1 of Reference 3. Planning and preparation activities applicable to post-accident DECON can be summarized as follows:

- Satisfying regulatory requirements - provides the necessary documentation for amending the facility operating license and for obtaining an NRC dismantlement order. In addition, the licensee must submit a radioactive waste handling plan, a quality assurance plan, an environmental report, security and safeguards plans, and possibly updated information concerning the licensee's financial qualifications.
- Gathering and analyzing data - provides input to the documentation and establishes the bases for developing work plans and procedures. The bulk of the required data is assumed to be available as a result of the cleanup activities.
- Developing detailed work plans and procedures - provides the decommissioning staff with all the information required to actually carry

out the decommissioning tasks. The plans and procedures cover all aspects of the project, and quality assurance, security, and environmental constraints are considered.

- Designing, procuring, and testing special equipment - ensures the availability and proper operation of the required equipment. The testing also serves to train personnel in the use of the equipment and to provide pertinent data on its operation.
- Selecting and training staff - ensures the availability of competent personnel and enables decommissioning to proceed smoothly, safely, and expeditiously.
- Selecting specialty contractors - allows certain specialized decommissioning tasks outside the expertise or capability of the decommissioning staff to be performed by experts, increasing the overall efficiency and safety of the decommissioning project.

Decontamination and Dismantlement Activities

The decontamination and dismantlement activities during post-accident DECON at the reference U-Fab plant are very similar to those during DECON following normal shutdown. (See Reference 3 for a detailed description of DECON activities following normal shutdown.) The decontamination and dismantlement activities during post-accident DECON are summarized here, with emphasis on those activities that differ from the ones during normal-shutdown DECON.

Following the postulated accident and subsequent cleanup, radioactive contamination levels in the U-Fab plant exceed those that would be present following normal shutdown to an extent that depends on the severity of the accident and on the particular location in the plant. Therefore, to reduce radiation doses to decommissioning workers to practicable levels, the major access routes used by the DECON workers and "hot spots" outside of the access routes that can materially affect worker doses are cleaned up or shielded. This task is undertaken at the start of DECON to obtain the maximum dose-reduction benefits, using the same methods as postulated for similar tasks during cleanup or during normal-shutdown DECON. The level of effort required for this task is a function of the amount of contamination present.

Although decontamination of internal surfaces in the building is initiated during cleanup to reduce radiation doses to the workers, the bulk of this work is still carried out during DECON, particularly the removal of contaminated structural material. The methods used during post-accident DECON (i.e., concrete spalling, disassembly or cutting of metal components, etc.) are the same as those employed during DECON following normal shutdown. However, accident-generated contamination results in a somewhat greater level of effort and a greater volume of radioactive waste material.

The packaging and shipping of radioactive wastes generated during DECON is handled by crews that are available over the entire DECON period. The amount and contamination levels of the wastes handled by these crews and, consequently, the radiation doses to these workers, are anticipated to be greater than during normal-shutdown DECON.

9.4.3 DECON Schedule

The overall schedule and sequence of events for DECON is shown in Figure 9.4-1. Detailed schedules for DECON are presented in Section E.4 of Appendix E. Planning and preparation for DECON begin 6 months before the completion of the cleanup campaign.

DECON begins in the plant process areas and will comprise the major effort for the decommissioning staff. The schedule for succeeding activities depends on the need for the plant areas involved and on the availability of manpower. As shown in Figure 9.4-1, DECON following the reference accident is completed in about one year.

FIGURE 9.4-1. Sequence and Schedule of Major Activities for U-Fab Plant Decommissioning by DECON Following an Accident

9.4.4 DECON Staff Requirements

In this subsection, the organization of the decommissioning staff and the types and numbers of decommissioning workers needed for post-accident DECON at the reference U-Fab plant are discussed.

Organization of the Decommissioning Staff

The staff for post-accident decommissioning at the reference U-Fab plant is organized as shown in Figure 9.4-2. The primary decommissioning activities are performed on a one-shift, five-day-week basis. However, selected support activities (i.e. decontamination and radwaste system operation) and security functions are carried out on three shifts, around-the-clock, seven days per week. Further discussion pertaining to the staff organization and the functions of key staff members can be found in Chapter 9 of Reference 3.

DECON Manpower

Manpower requirements for management and support staff are primarily a function of the duration of the DECON project. The assumptions used to calculate management and support staff requirements are presented in Section E.4.4 of Appendix E.

Based on the detailed schedules for DECON activities in the various plant buildings given in Section E.4 of Appendix E, the types and number of decommissioning workers needed to efficiently complete the radiation-zone work in the allotted time are determined. However, because whole-body radiation doses to the decommissioning workers are limited in accordance with 10 CFR 20.101,⁽²⁾ the decommissioning worker requirements may be adjusted upward so that average individual radiation doses do not exceed 5 man-rem/man-year.

Overall staff labor requirements for DECON at the reference U-Fab plant following the reference accident are given in Table 9.4-1. These requirements, given in equivalent man-years for the planning and preparation phase as well as for the actual decontamination and dismantlement, include the management and support staff as well as the decommissioning workers. A total effort of about 56.5 man-years is estimated for completion of DECON following the reference accident. For comparison, the total staff labor requirement for DECON at the reference plant following normal shutdown is about 46 man-years.⁽³⁾

9.4.5 Waste Management Requirements for DECON

Waste management requirements for DECON are described in this section. Estimates are made of quantities of radioactive wastes generated during DECON and of packaging, transportation, and disposal requirements for managing these wastes. Radioactive wastes generated during DECON must be properly packaged and shipped to a low-level waste burial site. Radioactive wastes generated during DECON include:

^(a) ASSIGNED TO WORK CREWS AS WORK SITUATION DEMANDS - COUNTED ONLY ONCE

FIGURE 9.4-2. Postulated Organization Chart for DECON at the Reference U-Fab Plant Following an Accident

- contaminated process equipment, tanks, hoods, and piping
- concrete rubble from the mechanical decontamination of contaminated floors and walls
- HEPA and roughing filters
- sections of ventilation ductwork

TABLE 9.4-1. Overall Staff Labor Requirements for DECON at the Reference U-Fab Plant Following the Reference Accident

Position	Staff Labor Requirement (man-years) in Decommissioning Phase	
	Planning and Preparation	DECDN
Project Manager	0.5	1.0
Project Engineer	0.5	1.0
Health and Safety Supervisor	0.25	1.0
Health Physics Technician		1.5
Security Force Supervisor		1.0
Security Patrolman		3.5
Contracts and Accounting Specialist	0.5	1.0
SNM Accounting Specialist		1.0
Radioactive Shipment Specialist	0.25	0.75
Q. A. Engineer	0.5	1.0
Planning Engineer	1.5	1.0
Engineer Technician	0.5	1.0
Maintenance and Crafts Supervisor		1.0
Maintenance Man		0.75
Custodian		1.0
Craftsman		4.5
Operations Supervisor	0.5	1.0
Foreman	1.0	4.0
Technician		28.0
Secretary	<u>1.0</u>	<u>1.5</u>
Total Man-Years	7.0	56.5

- combustible and noncombustible trash (protective clothing, contaminated tools, rags, paper, plastic, metal scrap, etc.)
- sludge, liners, and soil from the waste treatment lagoons.

The bulk of the material that must be packaged for disposal will be contaminated with uranium. The equipment and material wastes are assumed to be disposed of at licensed low-level waste burial sites. Details of assumed waste shipping volumes for contaminated process equipment, tanks, hoods, piping, ducts, etc., postulated to require disposal at a low-level waste burial site are given in Table E.4-2 of Appendix E.

Table 9.4-2 gives estimated weights and volumes of decommissioning wastes from DECON at the reference plant, together with the type of packaging and the number of shipments required for these wastes. Detailed information about

TABLE 9.4-2. Waste Disposal Packaging and Shipping Data for DECON at the Reference U-Fab Plant

Waste Category	Shipping Weight (kg) (a)		Shipping Volume (m ³) (a)	Type of Container	Number of Shipments
<u>To Low-Level Waste Burial:</u>					
Hoods, Equipment and Components	314	800	764.40	Plywood Boxes	18
Pipe, Conduit, Duct, Trays, Fixtures, etc.	133	200	118.52	Plywood Boxes	7
HEPA and Roughing Filters	16	500	51.66	Plywood Boxes	1
Concrete Rubble	23	800	39.66	Steel Drums	2
Contaminated Liner and Soil Materials	63	600	91.00	Plywood Boxes	5
Miscellaneous	17	400	12.5	Steel Drums	1
Miscellaneous	9	100	12.5	Plywood Boxes	1
Totals	578	400	1 090.24		35

(a) Number of significant figures shown is for computational accuracy and does not imply accuracy to three or more significant figures.

waste quantities is given in Section E.4 of Appendix E. It is assumed that the sludge from the two chemical and aeration lagoons would not require removal.

9.5 COSTS OF DECOMMISSIONING AT A REFERENCE URANIUM FUEL FABRICATION PLANT

The costs of decommissioning at the reference U-Fab plant following a postulated accident and subsequent accident cleanup campaign are developed in detail in Section E.5 of Appendix E. These costs are summarized in this section. Costs of accident cleanup are summarized previously in Section 9.3.

The principal assumptions made in developing the cost estimates for the post-accident decommissioning of the reference U-Fab plant are as follows:

- The costs of decommissioning are considered separate from the costs of accident cleanup. The costs of decommissioning systems and facilities installed during accident cleanup are included with the decommissioning costs.
- Cost estimates are made for decommissioning using the DECON alternative.

- To the maximum extent possible, the decommissioning staff is drawn from the technical and operational staffs of the plant and from the accident cleanup staff. The support services and the part-time assistance of the accident cleanup staff are assumed to be available during planning and preparations for decommissioning with only nominal costs to the decommissioning portion of the overall cleanup and decommissioning project.
- The amended facility license is in place at the end of the accident cleanup campaign, permitting decommissioning activities to commence without unnecessary delays.
- Piping, equipment, and significant portions of the building surfaces in the U-Fab plant are contaminated and require proper management as radioactive wastes.
- The cost estimates presented here are based on the ability to dispose of the radioactive waste materials at offsite facilities at the time of decommissioning.
- Costs are based on 1981 prices and wage rates.

The estimated cost of DECON at the reference U-Fab plant following the reference accident, including a 25% contingency, is \$5.4 million, as summarized in Table 9.5-1. Details of the development of these costs are discussed in

TABLE 9.5-1. Summary of Estimated Costs of Post-Accident DECON at the Reference U-Fab Plant^(a)

<u>Cost Category</u>	<u>Cost in Millions of 1981 Dollars^(a)</u>	<u>Percent of Total</u>
Manpower		
Planning and Preparation	0.398	9.3
Decommissioning	2.334	54.6
Equipment and Supplies	0.174	4.1
Disposal of Radioactive Material	0.273	6.4
Miscellaneous Owner Expense	1.011	23.6
Specialty Contractors	<u>0.084</u>	<u>2.0</u>
Subtotal	4.274	100.0
25% Contingency	<u>1.073</u>	
Total Decommissioning Costs	5.347	

(a) Number of figures shown is for computational accuracy and does not imply precision to the nearest thousand dollars.

Section E.5 of Appendix E. Information pertaining to the individual cost categories is summarized in the following subsections.

9.5.1 Costs of Staff Labor

The costs of staff labor during DECON following the reference accident are shown in detail in Table E.5-3 of Appendix E. A major portion of the total DECON cost is associated with staff labor. A total staff labor cost of about \$2.79 million is estimated for DECON following the reference accident.

9.5.2 Waste Management Costs

The estimated costs of containers, transportation and disposal of the radioactive wastes are summarized in Table E.5-5 of Appendix E. Cost estimates are based on projected packaging and shipping requirements summarized in Table E.4-2 and on waste management cost data in Section E.4 of Appendix E. Waste management cost details are given in Appendix E, Section E.5.

The total waste management cost for DECON is estimated to be about \$0.27 million without contingency.

9.5.3 Costs of Equipment and Supplies

The estimated costs of the equipment and supplies needed for post-accident DECON are represented in Table E.5-4 of Appendix E. The estimated total cost of equipment and supplies is approximately \$0.17 million, which is about 4% of the total DECON cost following the reference accident.

9.5.4 Costs of Miscellaneous Owner Expense

Several miscellaneous owner expenses are incurred during DECON. These include utilities, taxes, license fee, and insurance. The estimated costs of these items are given in Table E.5-6 of Appendix E. The total estimated cost of miscellaneous supplies during DECON is about \$1.0 million, which represents about 23% of the total DECON cost.

9.5.5 Costs of Specialty Contractors

The estimated requirements for specialty contractors during DECON are discussed in Section E.5.6 of Appendix E. The total cost of specialty contractors during DECON at the reference U-Fab plant is less than \$0.1 million. Specialty contractors account for about 2% of the total DECON costs following a reference accident.

REFERENCES

1. "Licensing Requirements for Land Disposal of Radioactive Waste," Federal Register, Vol. 46, No. 142, pp. 38081-38100, July 24, 1981.
2. U.S. Code of Federal Regulations, Title 10, Part 20.101, "Radiation Dose Standards for Individuals in Restricted Areas." (10 CFR 20.101), U.S. Government Printing Office, Washington, D.C., 1980.
3. H. K. Elder and D. E. Blahnik, Technology, Safety and Costs of Decommissioning a Reference Uranium Fuel Fabrication Plant. NUREG/CR-1266, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, October 1980.

10.0 NON-FUEL CYCLE FACILITY CLEANUP AND DECOMMISSIONING

Estimated time and manpower requirements and total costs for decommissioning the reference laboratory for processing of radioisotopes are summarized in this section. The reference laboratory for manufacture of ^{241}Am sealed sources is described in Section 5.

The technical approach used for this analysis is described in Section 10.1. The results of the decommissioning analyses are presented in Section 10.2. Details of manpower and of waste management requirements and costs for decommissioning the reference laboratory are given in Appendix F.

10.1 TECHNICAL APPROACH

The technical approach and some key bases used to define requirements and to estimate costs and safety of decommissioning the reference radioactive materials laboratory are discussed in this section.

10.1.1 Reference Laboratory

The decommissioning of a radioactive materials laboratory following an accident includes a period of planning and preparation, with activities performed to ensure that the decommissioning effort is done in a safe and cost-effective manner in accordance with all applicable federal, state, and local regulations. These planning and preparation activities include the preparation of documentation for regulatory agencies, an initial radiation survey of the laboratory, and the development of detailed work plans. Planning and preparation activities are described in Section F.2 of Appendix F.

The cost estimates in this section include the costs of staff labor and supplies for planning and preparation. For the laboratory facilities analyzed in this study, planning and preparation activities are estimated to require about eight weeks prior to the start of actual decommissioning operations.

Decommissioning Alternatives

Two DECON methods may be used to decommission contaminated components and building surfaces of the reference laboratory. The DECON options are 1) decontamination of a component or building surface to unrestricted release levels, or 2) packaging and shipment of the contaminated material to a shallow-land burial ground for disposal. To provide a basis for estimating the requirements and costs of decommissioning the reference laboratory, choices are made of DECON options for facility components.

The DECON method chosen for the components of the reference laboratories is believed to represent reasonable approaches to the decommissioning of particular components. Some of the fume hoods and glove boxes are assumed to be decontaminated to unrestricted release levels, while other hoods and glove boxes are packaged for disposal, depending on the magnitude and type of surface contamination. Laboratory benches and other components such as refrigerators, storage cabinets, etc., are assumed to be decontaminated to unrestricted release levels. Sinks are decontaminated, and drain lines are sectioned and packaged for disposal. Filters are packaged for disposal. Ventilation ductwork is sectioned and packaged for disposal. In general, building surfaces are assumed to be cleaned to unrestricted release levels. (Fiberboard ceiling panels and floor tiles that are difficult to decontaminate are packaged for disposal.) Details of the DECON options assumed for the components of each of the reference laboratories are given in Appendix F.

Costs

Costs for decommissioning the reference laboratory include the costs of staff labor, equipment, supplies, and waste management (the packaging, transportation, and disposal of radioactive waste). Some key bases and assumptions for estimating decommissioning costs are given in Appendix F. Unit costs are listed in Appendix H. To maintain consistency with the cost estimates of other decommissioning reports in this series (see the Foreword by the NRC staff for a list of these reports), all costs are expressed in 1981 dollars. Total costs include a 25% contingency.

Decommissioning of the reference laboratory is assumed to be performed by employees of the owner/operator of the laboratory. The basic cleanup and decommissioning work crew is assumed to include a foreman and three technicians, assisted by a health physicist. Craftsmen (electricians, pipefitters, etc.) are added to this crew on a part-time basis to perform specific tasks. Manpower costs are postulated to include the salary of a supervisor on a half-time basis.

Staff labor costs are determined by multiplying the man-days required to decommission the laboratory by the costs per man-day shown in Appendix H. To determine the total time requirement for decommissioning, an estimate is made of the time required for efficient performance of the work by the postulated work crew. This time estimate is then increased by 50% to provide for preparation and set-up time, rest periods, etc. (ancillary time).

In estimating the requirements and costs of decommissioning the reference laboratory, it is assumed that components intended for shallow-land burial (fume hoods, glove boxes, ventilation ductwork, etc.) are packaged with a minimum of sectioning (i.e., cutting) and no compaction. (Fume hoods and glove boxes are packaged without sectioning, while other components such as drain

lines and ventilation ductwork are sectioned for ease of handling and packaging in boxes that are approximately one meter long.) This approach to waste management conforms with current practice in the decommissioning of materials licensee facilities. It minimizes the time and manpower costs of packaging operations, but maximizes the volume of radioactive waste shipped to the shallow-land burial ground. Transportation and waste disposal charges that are determined on a volume basis are thus maximized. Volume reduction may be required for future decommissioning operations because of limits on available shallow-land disposal space.

10.1.2 Facility Components

Decommissioning activities following an accident begin with a radiation survey to determine the extent and magnitude of surface contamination on facility components. Radioactive contamination survey techniques are described in Appendix J. The results of this initial survey provide guidance in formulating procedures to be followed in decommissioning a particular component. After decontamination of the component, and before it can be released for unrestricted use, a second survey is performed to verify compliance with NRC guidelines on acceptable surface contamination levels for facilities and equipment released for unrestricted use. These guidelines are also discussed in Appendix J. Radiation surveys to verify compliance with release limits include measurements with survey instruments to determine room and surface background levels and wipe tests to determine removable contamination levels. The manpower and cost of radiation surveys represent a significant fraction of the total manpower and cost of decommissioning.

Decontamination Procedures

The first step in the decontamination of a facility component is the removal of contaminated equipment located on or within the component. Equipment to be disposed of is wrapped in plastic or placed in small containers (e.g., paint cans) for transfer to 208-L drums. Some equipment items are decontaminated by soaking, washing, ultrasonic cleaning, vibratory finishing, or electropolishing.

The next step is usually the removal of loose or lightly held contamination. Loose or lightly held contamination can often be removed by relatively simple janitorial techniques such as vacuuming, sweeping, brushing, wiping with a damp cloth or sponge, mopping with a damp mop, or scrubbing. Portable vacuum cleaners for both wet and dry pickup of radioactive contamination are commercially available. These vacuum cleaners usually incorporate a primary filter, a high-efficiency particulate air (HEPA) filter, and a vapor/mist eliminator for absolute filtration of the radioactive mist or fog from wet pickup and the radioactive dust from dry pickup.

Some surfaces, such as walls or the interior surfaces of fume hoods and hot cells, can be decontaminated by using steam or a high-pressure water spray mixed with detergent. The liquid waste from this operation is collected by mopping or by using a wet vacuum cleaner.

Surfaces of hoods, glove boxes, hot cells, and laboratory benches are cleaned by scrubbing or wiping, using scrub brushes, sponges, scrapers, paper products, or steel wool. Cleaning solutions are chosen that are best suited for the type of surface being decontaminated and for the particular radioactive contaminant. These include detergents, solvents, commercial cleaning solutions, and a variety of chemicals.

Painted surfaces can often be decontaminated by scrubbing with an abrasive cleanser. If this fails to remove the contamination, the surface is cleaned by scraping or by removing the paint using a paint-stripping compound.

Floors can be decontaminated by scrubbing with a foam-type decontamination agent or with a household cleanser containing grit or pumice. Scrubbing is accomplished manually or by using powered floor scrubbers equipped with rotating brushes. The liquid is collected by mopping or by using a wet vacuum cleaner. Floor tiles that cannot be decontaminated are removed after first chilling the tile by contact with "dry ice" (solid CO_2).

Disassembly and Disposal Procedures

Procedures for disassembly and disposal of contaminated facility components begin with a radiation survey to determine the extent and magnitude of radioactive contamination. The survey is followed by the removal of loose or lightly held contamination using simple janitorial techniques. Contaminated surfaces are spray painted with a clear acrylic plastic paint or with a water base latex paint to reduce the potential for the spread of contamination during packaging and shipment.

The contaminated component is next disconnected from other components, from ventilation ductwork, and from services such as compressed air, water, electricity, etc. Service line and ductwork separations are made inside plastic bags using the following procedure. A loose-fitting vinyl bag is affixed to each point of separation of the service line or ductwork to be dismantled. The size of the vinyl bag is sufficient to permit unbolting or cutting of the line or duct inside the bag. When the necessary separations are made, the section to be removed is rotated inside the bag, thereby twisting a small section of bag at each end. The twisted sections of bag are wrapped with vinyl tape. The bag is then cut at the taped areas and the section of service line or ductwork is removed. This technique assures containment of the radioactivity both during the separation operation and afterwards.

Process pipe openings are sealed with rubber plugs or with blank flanges and gaskets. All large openings such as duct stub and glove port openings and viewing windows are covered with 19-mm-thick plywood strapped to the cabinet with metal bands. The interiors of glove boxes and hoods can be filled with contaminated equipment, ductwork, filters, and piping. All components are wrapped in plastic before being placed in shipping containers.

It has been a practice at some laboratories to fill the interiors of fume hoods, glove boxes, and ductwork with polyurethane foam to prevent the spread of radioactive contamination during shipment. However, this foam is extremely difficult to remove in the event that a decision is later made to decontaminate and salvage the component. For this reason, spray painting is the method assumed to be used in this study for fixing residual contamination.

Contaminated roughing and HEPA filters are sealed in plastic before being placed in shipping containers. Ductwork and piping are spray painted to fix residual contamination, cut into appropriate lengths, and sealed in plastic.

Most of the contaminated components and equipment and the miscellaneous cleaning supplies (rags, mops, paper products, etc.) from the decommissioning of non-fuel cycle facilities are of sufficiently low activity that they can be packaged for shipment in Type A containers such as 208-L steel drums or plywood boxes. The boxes are made in various shapes and sizes to fit the component being shipped, are internally reinforced, and are closed with additional supports and metal bands. Both the drums and boxes are lined with 0.15-mm-thick polyethylene liners. Some radioactive components, particularly those resulting from decontamination of the ²⁴¹Am laboratory, may have contamination levels that require their transport in Type B containers.

Contaminated liquids must be solidified prior to shipment to a disposal site. Aqueous wastes are mixed with cement or other solidifying agents and poured into plastic-lined 208-L steel drums. Organic wastes are poured into plastic-lined, 113-L steel drums filled with an appropriate absorbent material (e.g., vermiculite, perlite, diatomaceous earth, etc.). Each 113-L steel drum is placed inside a plastic-lined 208-L steel drum, and the space between the drums is filled with absorbent material.

In this study, the base-case scenario for determining the requirements and costs of disassembly and disposal of contaminated facility components assumes that current decommissioning practice is followed and that components are packaged and shipped intact with a minimum of sectioning (i.e., cutting) and compaction. This approach reduces the time and costs of packaging, but maximizes disposal site costs that are determined on a per-unit-volume basis. It is anticipated that future disposal site restrictions will require that waste segregation and volume reduction be practiced where the waste originates. This

will require that bulky items such as fume hoods, glove boxes, and ductwork be cut up and compacted and that combustible wastes be segregated and incinerated prior to being packaged for shipment.

The authorized disposal site is assumed to be a shallow-land burial ground located 800 km from the laboratory being decommissioned. Transportation of radioactive waste to the disposal site is assumed to be by exclusive-use truck. Transport of the waste is made in accordance with applicable federal, state, and local regulations.

Cost Estimates

Costs include manpower, equipment and supplies, and waste management costs. Some key bases and assumptions for estimating costs are given in Appendix F. Unit costs are listed in Appendix H. To maintain consistency with the cost estimates of other decommissioning reports in this series (see the Foreword by the NRC staff for a list of these reports), all costs are expressed in 1981 dollars. Total costs include a 25% contingency.

Decontamination of facility components is assumed to be performed by employees of the owner/operator of the facility. Manpower costs are determined by multiplying the man-days required to decommission a component by the costs per man-day shown in Appendix H. In order to determine the total time required to decommission a component, an estimate is made of the time required for efficient performance of the work by a postulated work crew. This time estimate is then increased by 50% to provide for preparation and set-up time, rest periods, etc. (ancillary time).

The time required to complete a particular decommissioning task is usually estimated on the basis of a work crew consisting of a foreman and two technicians. The technicians are assumed to have had some experience working with radiochemicals, to be trained in radiological safety procedures, and to be capable of operating radiation survey equipment as well as the tools and equipment used to decontaminate the facility. Craftsmen such as electricians, pipefitters, sheet metal workers, etc., are assumed to be added to a work crew as the situation requires.

Several small equipment items such as wet-dry vacuums, power scrubbers, and steam cleaning equipment are used for decontaminating facility components. Because an equipment item is only used for a few days, it is not considered reasonable to charge its entire cost to the decommissioning of one component. However, some fraction of the cost of the equipment must be charged to each operation. To estimate equipment costs, a one-year equipment lifetime is assumed and a charge of $x/250$ of the cost of the item is made, where x is the number of days required to decontaminate the component. Equipment for

radiation survey and for the analysis of wipe samples is assumed to be readily available and not chargeable to decommissioning because such equipment is also used during the operation of the facility.

Waste management costs include container costs, transportation costs, and waste disposal charges. Transportation charges are based on the fraction of a truckload required to transport the decommissioning wastes from an individual facility component. It is assumed that one truckload consists of 120 steel drums (208-l) or 30 m³ of plywood boxes. Because transportation and waste disposal operations are contracted activities, manpower costs for the transportation and disposal of radioactive wastes are included in the total costs of these items.

10.2 DECOMMISSIONING ANALYSES

Results of analyses of time and manpower requirements and total costs for decommissioning facility components and the reference laboratory are presented in this section. Requirements and costs are presented for the planning and preparation phase, for the actual decommissioning phase, and for the final radiation survey to demonstrate compliance with unrestricted release guidelines.

Details of manpower and waste management requirements and costs are given in Appendix F. Appendix F also contains descriptions of the DECON options postulated for decommissioning the various components and building surfaces of the reference laboratory.

10.2.1 Facility Components

Results of analyses of time and manpower requirements, total costs, and occupational radiation doses for decommissioning facility components are presented in this section. Total costs include the costs of manpower, equipment and supplies, and waste management (e.g., the packaging, transportation, and disposal of radioactive waste).

Details of time and manpower requirements and of estimated total costs for decommissioning facility components are presented in Appendix F. Appendix F also summarizes the key bases and assumptions used in estimating the requirements and costs of decommissioning. Unit costs of manpower, equipment, and supplies, and waste management activities are given in Appendix H.

Fume Hoods

Estimated time and manpower requirements, total costs, and occupational radiation doses for decommissioning a fume hood by the DECON options of

1) decontamination to unrestricted release levels or 2) packaging and disposal of the contaminated hood at an authorized shallow-land burial site are shown in Table 10.2-1.

TABLE 10.2-1. Summary of Estimated Manpower Requirements and Total Costs for DECON of a Fume Hood Contaminated by ^{241}Am

Decontamination	
Time (days)	7.1
Manpower (man-days)	21.2
Costs (\$ thousands)	7.1
Packaging and Disposal w/o Volume Reduction	
Time (days)	4.5
Manpower (man-days)	15.0
Costs (\$ thousands)(a)	7.1

(a) Costs are in 1981 dollars.

Requirements and costs for the decontamination option are based on the cleaning of hood interiors to reduce residual surface contamination to unrestricted release levels. These contamination levels are shown in Table 10.2-2. A work crew that includes a foreman and two technicians is assumed to perform the work.

The estimated total costs of decontamination of fume hoods having the initial surface contamination levels shown in Table 10.2-2 are about \$7000. Manpower costs represent between 40 and 50% of the total costs of decontamination. About one-third of the manpower costs are for the radiation surveys needed to establish residual contamination levels prior to starting decontamination operations and to verify compliance with unrestricted-release guidelines when decontamination is completed.

Requirements and costs for the packaging and disposal option are shown (Table 10.2-1) with the hood packaged without sectioning.

It is assumed that hoods contaminated with ^{241}Am can be disposed of by shallow-land burial. This may not be the case if the residual contamination

TABLE 10.2-2. Comparison of Residual Contamination Levels on Fume Hoods with Allowable Contamination Limits for Unrestricted Release^(a)

Residual Contamination ^(b) (d/m/100 cm ²)	10 ³ to 10 ⁴
Release Level ^(c) (d/m/100 cm ²)	2 x 10 ¹

(a) The numbers refer to removable contamination.

(b) From Section 7.4.

(c) Based on NRC Guidelines for decontamination of facilities and equipment prior to release for unrestricted use (Reference 1).

level is greater than 10 nCi/gram of waste, equivalent to an average surface contamination on the interior surfaces of a steel hood of about 4×10^6 d/m/100 cm². If the average surface contamination exceeds this value, it may be necessary to partially decontaminate the hood or to provide for interim storage of the contaminated hood, since facilities for the permanent disposal of transuranic wastes are not yet available.

Glove Boxes

Table 10.2-3 presents the estimated time and manpower requirements, and total costs for decommissioning a glove box by the DECON options of 1) decontamination to unrestricted release levels or 2) packaging and disposal of the contaminated glove box at an authorized shallow-land burial site. A work crew consisting of a foreman and one technician is assumed to perform the work.

The estimated total costs of decontamination of the glove box with the initial surface contamination levels shown in Table 10.2-4 are about \$5500. Manpower costs represent about one-third of the total cost of decontamination. An increase (or decrease) of one day in the total time required to decontaminate a glove box to unrestricted release levels would increase (or decrease) the total cost of decontamination by about \$300.

Requirements and costs for the packaging and disposal option are shown in Table 10.2-3 for the case in which the glove box is packaged without sectioning.

TABLE 10.2-3. Summary of Estimated Manpower Requirements and Total Costs for DECON of a Glove Box Contaminated by ^{241}Am

Decontamination	
Time (days)	7.5
Manpower (man-days)	15.0
Costs (\$ thousands)	5.5
Packaging and Disposal w/o Volume Reduction	
Time (days)	3.4
Manpower (man-days)	9.0
Costs (\$ thousands) ^(a)	4.2

(a) Costs are in 1981 dollars.

TABLE 10.2-4. Comparison of Residual Contamination Levels on Glove Boxes with Allowable Contamination Limits for Unrestricted Release^(a)

Residual Contamination ^(b) (d/m/100 cm ²)	10^3 to 10^4
Release Level ^(c) (d/m/100 cm ²)	2×10^1

(a) The numbers refer to removable contamination.

(b) From Section 7.4

(c) Based on NRC Guidelines for decontamination of facilities and equipment prior to release for unrestricted use (Reference 1).

Laboratory Workbenches

Estimated time and manpower requirements, total costs, and occupational radiation doses for decommissioning a laboratory workbench by the DECON options of 1) decontamination to unrestricted release levels or 2) packaging and disposal of the contaminated workbench at an authorized shallow-land burial site are shown in Table 10.2-5. Laboratory workbenches come in a range of sizes.

TABLE 10.2-5. Summary of Estimated Manpower Requirements and Total Costs for DECON of a Laboratory Workbench Contaminated by ^{241}Am

Decontamination	
Time (days)	1.7
Manpower (man-days)	3.4
Costs (\$ thousands)	2.0
Packaging and Disposal w/o Volume Reduction	
Time (days)	1.9
Manpower (man-days)	5.3
Costs (\$ thousands)(a)	4.6

(a) Costs are in 1981 dollars.

The workbench for which decommissioning requirements and costs are estimated is assumed to be 0.9 m high, with a bench top that is 4.6 m long and 0.75 m wide.

Requirements and costs for the decontamination option are based on cleaning the bench top and other surfaces to reduce residual surface contamination to unrestricted levels. These contamination levels are shown in Table 10.2-6. Decontamination is performed by a work crew consisting of a foreman and a technician. The total cost for the decontamination option is estimated to be about \$2000.

TABLE 10.2-6. Comparison of Residual Contamination Levels on Laboratory Workbenches with Allowable Contamination Limits for Unrestricted Release^(a)

Residual Contamination ^(b) (d/m/100 cm ²)	10^1 to 2×10^2
Release Level ^(c) (d/m/100 cm ²)	2×10^1

(a) The numbers refer to removable contamination.

(b) From Section 7.4

(c) Based on NRC Guidelines for decontamination of facilities and equipment prior to release for unrestricted use (Reference 1).

For the packaging and disposal option, an electrician and a pipefitter are temporarily added to the work crew to disconnect services. A second technician is added to the crew to assist in packaging the workbench. The bench is cut into two sections, each 2.3 m long, for ease of packaging. The total cost of the option is estimated to be about \$4600.

Ventilation Ductwork

Dirt that accumulates on inside surfaces of ventilation ductwork makes decontamination very difficult. Therefore, the usual practice when decommissioning a laboratory where radioactive materials have been processed is to package the ductwork for disposal at a shallow-land burial ground. Estimated time and manpower requirements, total costs, and occupational radiation doses for this DECON option are shown in Table 10.2-7. The estimates are based on the packaging and disposal of 20 m of 0.20 m-diameter sheet metal ductwork plus 20 m of rectangular sheet metal ductwork, 0.25 m by 0.60 m. Evaluations are made of both the case in which the ductwork is packaged without compaction and the case in which the ductwork is compacted before being packaged for shipment.

TABLE 10.2-7. Summary of Estimated Manpower Requirements and Total Costs for DECON of Ventilation Ductwork Contaminated by ^{241}Am

Packaging and Disposal w/o Volume Reduction	
Time (days)	4.5
Manpower (man-days)	12.0
Costs (\$ thousands)(a)	6.4

(a) Costs are in 1981 dollars.

Levels of radioactive contamination existing on inside surfaces of the ductwork are shown in Table 10-2.8.

The total costs of packaging and disposal are estimated to be \$6400 without compaction of the ductwork.

Building Surfaces

Building surfaces include walls and floors. Decontamination to unrestricted release levels is the DECON option evaluated for these surfaces. Some contaminated material such as floor tiles or concrete chipped from walls might be packaged and shipped to a shallow-land burial ground.

TABLE 10.2-8. Comparison of Residual Contamination Levels on Ventilation Ductwork with Allowable Contamination Limits for Unrestricted Release^(a)

Residual Contamination ^(b) (d/m/100 cm ²)	10 ¹ to 10 ³
Release Level ^(c) (d/m/100 cm ²)	2 x 10 ¹

(a) The numbers refer to removable contamination.

(b) From Section 7.4.

(c) Based on NRC Guidelines for decontamination of facilities and equipment prior to release for unrestricted use (Reference 1).

The reference laboratory rooms for these evaluations are assumed to measure 6 m by 10 m with walls 3 m high.

Walls. Estimated time and manpower requirements, total costs, and occupational radiation doses for decontamination of the walls of the reference laboratories to unrestricted release levels are shown in Table 10.2-9.

TABLE 10.2-9. Summary of Estimated Manpower Requirements and Total Costs for DECON of Walls Contaminated by ²⁴¹Am

Decontamination	
Time (days)	14.2
Manpower (man-days)	42.7
Costs (\$ thousands)	17.8

(a) Costs are in 1981 dollars.

Requirements and costs for decontamination are based on cleaning wall surfaces to reduce residual surface contamination to unrestricted release levels. These contamination levels are shown in Table 10.2-10. A work crew that includes a foreman and two technicians is assumed to perform the work. The walls are scrubbed with a decontaminating solution. Walls in the ²⁴¹Am laboratory are sealed with epoxy paint and acrylic paint, respectively. These

TABLE 10.2-10. Comparison of Residual Contamination Levels on Walls with Allowable Contamination Limits for Unrestricted Release^(a)

Residual Contamination ^(b) (d/m/100 cm ²)	10 ¹ to 10 ²
Release Level ^(c) (d/m/100 cm ²)	2 x 10 ¹

(a) The numbers refer to removable contamination.

(b) From Section 7.4.

(c) Based on NRC Guidelines for decontamination of facilities and equipment prior to release for unrestricted use (Reference 1).

walls are easier to decontaminate and require less re-cleaning of hot spots than the walls that are covered with latex enamel paint.

The total costs of decontamination are estimated to be about \$17,800. Manpower costs represent almost 50% of those total costs. Decommissioning wastes (cleaning supplies and solidified decontamination liquids) are packaged for disposal in twenty-four 208-l drums.

Floors. Estimated time and manpower requirements, total costs, and occupational radiation doses for decontamination of the floors of the reference laboratories to unrestricted release levels are shown in Table 10.2-11.

TABLE 10.2-11. Summary of Estimated Manpower Requirements and Total Costs for DECON of Floors Contaminated by ²⁴¹Am

Decontamination	
Time (days)	4.9
Manpower (man-days)	14.6
Costs (\$ thousands)	8.8

(a) Costs are in 1981 dollars.

Requirements and costs for decontamination are based on reducing residual surface contamination to unrestricted release levels. These contamination levels are shown in Table 10.2-12. A work crew that includes a foreman and two technicians is assumed to perform the work. The floor in the ^{241}Am laboratory is covered with linoleum with heat-treated seams. Because the linoleum is free from cracks, it is easier to decontaminate and requires less re-cleaning than asphalt tile floors.

TABLE 10.2-12. Comparison of Residual Contamination Levels on Floors with Allowable Contamination Limits for Unrestricted Release^(a)

Residual Contamination ^(b) (d/m/100 cm ²)	10^1 to 10^3
Release Level ^(c) (d/m/100 cm ²)	2×10^1

-
- (a) The numbers refer to removable contamination.
 (b) From Section 7.4.
 (c) Based on NRC Guidelines for decontamination of facilities and equipment prior to release for unrestricted use (Reference 1).

The total costs of decontamination are estimated to be \$8800 for the linoleum floor. Manpower costs represent about one-third of these total costs. Wastes from decontamination operations include four 208-l drums of cleaning supplies and eight 208-l drums of solidified liquids.

10.2.2 Laboratory for the Manufacture of Sealed Sources

The reference laboratory for the manufacture of ^{241}Am sealed sources is described in Section 7.5. Some important parameters that characterize major laboratory components and that influence the requirements and costs of decommissioning this laboratory are shown in Table 10.2-13.

Estimated time and manpower requirements, occupational radiation doses, and costs for decommissioning the reference ^{241}Am laboratory are shown in Table 10.2-14.

Planning and preparation is estimated to require about six weeks and 68 man-days of effort prior to the start of decommissioning operations. Decommissioning operations are estimated to require about 13 weeks and 332 man-days of effort.

TABLE 10.2-13. Major Components in the Reference Laboratory for the Manufacture of ^{241}Am Sealed Sources

Component	Quantity	Description	Contamination Level (d/m) ^(a)
Floor	60 m ²	Concrete covered with inlaid linoleum	2×10^1 to 1×10^3
Walls	168 m ²	Concrete sealed with acrylic paint	2×10^1 to 2×10^2
Ceiling	60 m ²	Concrete sealed with acrylic paint	2×10^1 to 2×10^2
Fume Hood	2	Steel with interior surfaces of stainless steel	1×10^3 to 5×10^3
Glove Box	7	Stainless steel with acrylic plastic viewing panels	2×10^7 to 2×10^9
Hot Cell	--	--	--
Lab Benches	1.5 m ²	Steel with stainless steel top	2×10^1 to 2×10^2
Sink and Drain	--	--	--
Filters			
HEPA	9	0.2 m diameter by 0.2 m high	--
	1	0.25 m x 0.60 m x 0.30 m	
Roughing	9	0.2 m diameter by 0.1 m high	--
	1	0.25 m x 0.60 m x 0.15 m	
Ventilation Ductwork			
0.2 m diameter	18 m	Polyvinyl-chloride	5×10^1 to 1×10^3
0.25 m x 0.60 m	20 m	Polyvinyl-chloride	5×10^1 to 1×10^3
Other Components			
Transfer Tunnels	4 m	0.45 m x 0.45 m stainless steel	1×10^6 to 1×10^8

(a) Removable contamination per 100 cm² of surface.

TABLE 10.2-14. Summary of Estimated Values of Manpower Requirements, Occupational Radiation Doses, and Costs for Decommissioning the Reference Laboratory for the Manufacture of ^{241}Am Sealed Sources

Parameter	Value			
	Planning and Preparation	Decommissioning	Final Radiation Survey	Total
Time (days)	30	63	5	98
Manpower (man-days)	68	332	23	423
Costs (\$ thousands) ^(a)				
Staff Labor	11.76	60.15	3.48	75.39
Equipment	--	2.76	--	2.76
Supplies	1.38	6.23	0.16	7.77
Waste Management	--	34.8	--	34.8
Subtotals	13.14	103.94	3.64	120.72
25% Contingency	3.29	25.99	0.91	30.18
Totals	16.43	129.93	4.55	150.90

(a) Costs are in 1981 dollars. Number of figures shown is for computational accuracy only and does not imply that level of precision.

The total cost of decommissioning the reference laboratory is estimated to be about \$151,000. Planning and preparation activities account for about 16% of the total cost. Approximately 66% of the total cost is for staff labor (including planning and preparation activities) and approximately 21% is for waste management.

It is assumed that components contaminated with ^{241}Am can be disposed of by shallow-land burial. Existing practice at commercial burial sites is to restrict the burial of transuranic waste to concentrations less than 10 nCi/gram of waste. It is assumed that some decontamination of the interiors of glove boxes and transfer tunnels is performed prior to packaging to ensure that the transuranic waste limitation currently in effect at shallow-land burial grounds is not exceeded.

If contaminated components from a ^{241}Am laboratory cannot be disposed of by shallow-land burial, provision would need to be made for interim storage, since facilities for the permanent disposal of transuranic wastes (i.e., greater than 10 nCi of transuranic elements per gram of waste) are not yet available.

REFERENCE

1. Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material, U. S. Nuclear Regulatory Commission, Washington, D.C., November 1976.

11.0 ORE PROCESSING SITE CLEANUP AND DECOMMISSIONING

Information on the technology and costs for decommissioning the reference ore processing site following an accident is presented in this section. The ore processing site is described in Chapter 5. The technical approach used to estimate requirements and costs is described in Section 11.1. The results of decommissioning analyses for individual sites are presented in Section 11.2. Details of decommissioning the reference site is presented in Appendix G.

11.1 TECHNICAL APPROACH

The technical approach and some key bases used to define requirements and estimate costs and safety of decommissioning the reference site are discussed in this section.

Decommissioning activities following an accident begin with a radiological survey to determine current radiological conditions at the site. This survey is particularly important in providing a data base for decisions about the remedial measures or decommissioning procedures needed to protect public safety. A final site survey is also performed prior to release of the site for unrestricted use to verify that any radioactivity remaining on the site is below limits specified for unrestricted release. The manpower and costs of radiological surveys represent significant fractions of the total manpower and costs of decommissioning.

Decommissioning of a contaminated site following an accident involves measures taken to preclude any future concern about inadvertent exposure of people to radiation. Two general approaches possible for the decommissioning of contaminated sites following an accident are taken from Reference 1: 1) site stabilization followed by a period of long-term care, and 2) removal of contaminated soil or buried radioactive material followed by disposal at an approved shallow-land burial site. Site stabilization procedures are summarized in Section 11.1.2. Waste removal procedures are summarized in Section 11.1.3. The background to these approaches is discussed in detail in another report that describes the technology, costs, and safety of decommissioning a low-level waste burial ground.⁽¹⁾

Some key bases and assumptions used for estimating the costs of decommissioning the reference site are discussed in Section 11.1.4. Bases and assumptions for occupational radiation dose estimates are discussed in Section 11.1.5.

11.1.1 Site Surveys

A common method for determining where radiological survey measurements are to be made is to divide the site into survey blocks by a rectangular grid system and to make measurements at the centers of the blocks or at the intersections of the grid lines. The size of the survey blocks can vary from 10 m x 10 m to 100 m x 100 m, depending on the size and nature of the site and the extent and magnitude of suspected radioactive contamination. In addition, measurements are made at specified points where unusual concentrations of radioactivity are suspected, such as along the perimeter of a tailings pile.

The kinds of measurements made for the radiological survey depend on the type of contamination suspected on the site. Some, or all, of the following measurements may be made:

1. measurement of external gamma radiation at the surface and one meter above the surface at all points on the grid system
2. measurement of beta-gamma contamination levels at the ground surface on the same grid
3. collection of surface soil samples on the same grid for measurements of the concentrations of specific isotopes
4. measurement of gamma radiation at various depths in core holes dug throughout the region of known or suspected contamination
- 5) collection of soil samples from some of the core holes for measurements of the concentrations of specific isotopes
- 6) collection of water and mud samples along drainage paths on or adjacent to the property to determine the concentrations of specific radionuclides.

External gamma radiation levels are measured with a portable survey meter that utilizes a NaI(Tl) scintillation probe. The scintillation probe is a 32 mm x 38 mm NaI crystal mounted on a photomultiplier tube and coupled with a sensitive ratemeter.

A portable Geiger-Muller (G-M) survey meter is the primary instrument for measuring beta-gamma contamination. Since the G-M tube is sensitive to both beta and gamma radiation, measurements are taken in both an open-window and a closed-window configuration. Beta radiation cannot penetrate the closed window; thus, the beta reading can be determined by taking the difference between the open- and closed-window readings.

Surface soil samples are obtained using a right circular cylinder cutting tool 0.10 m diameter by 0.15 m long, normally used as a golf-green hole cutter. A typical procedure⁽²⁾ for surface soil sampling is to obtain a core that is 0.30 m long and divided into four segments. The top three segments are each 0.05 m long, and the final segment is 0.15 m long. The reason for the segmented coring is to determine what contamination migration has occurred, if any, and to reduce the dilution of lower level soil with upper level soil segments.

Holes for subsurface soil sampling are drilled with a motorized drilling rig. An auger with a diameter of 0.10 m to 0.20 m is commonly used for drilling. A plastic pipe is placed in the hole after it is drilled. For measurements of gamma radiation as a function of depth below the ground surface, a NaI(Tl) scintillation probe is lowered inside the plastic pipe. The probe is encased in a lead shield with a horizontal row of narrow collimating slits on the side. This arrangement allows measurements of gamma radiation intensities resulting from contamination within small fractions of the hole depth. Measurements are typically made at intervals of 0.15 m to 0.30 m. This "logging" serves to define the profile of radioactivity in the vicinity of the hole, but does not result in absolute measurements of specific radioactivity. The gamma intensity profile can be normalized by making laboratory measurements on soil samples taken from the vicinity of the hole.

Subsurface soil samples are collected using a split-spoon sampler,^(a) or they may also be taken from auger turnings as a hole is being drilled. The samples are packaged in plastic bags or bottles for transport to the laboratory, where they are dried for 24 hours at 110°C, weighed, and then pulverized to a particle size of 35 mesh (500 μ m). Aliquots from each sample are transferred to plastic bottles or petri dishes, weighed, and counted using a Ge(Li) detector and a multichannel pulse-height analyzer. Concentrations of ^{226}Ra and ^{238}U in the soil can be determined by this technique. The technique used to count the soil samples is described in some detail in Appendix II of Reference 3. Mud or sediment samples collected along drainage areas can be dried and analyzed in the same way as normal soil samples.

Since activated charcoal readily absorbs ^{222}Rn , an estimate of radon flux from ground surfaces is obtained by placing canisters containing charcoal in direct contact with the ground.⁽⁴⁾ Modified U.S. Army M-11 gas mask canisters have been used for this purpose.⁽³⁾ The canisters are twisted into the soil to a depth of about 10 mm and sealed with additional soil. After a period of exposure ranging from one to two days, the canisters are removed and the radon

(a) A split-spoon, or split-tube, sampler is a drive-type soil sampler with a core barrel that is split lengthwise so that the barrel can be taken apart and the sample removed without breaking up the soil.

daughters are allowed to achieve equilibrium. The amount of radon absorbed on the activated charcoal canisters is determined by counting the gamma emissions from ^{214}Pb and ^{214}Bi using NaI(Tl) or Ge(Li) scintillation detectors coupled to a multichannel pulse-height analyzer.

Measurements of the outdoor concentration of ^{222}Rn in air can be made using a Wrenn chamber.^(5,6) This detector operates on the principle that most of the ^{218}Po (RaA) daughter ions from the decay of ^{222}Rn are positively charged. Radon is allowed to diffuse into the cavity through a hemispherically shaped metal screen covered with foam rubber to filter radon daughters. As radon in the chamber decays, RaA ions are attracted to a thin aluminized mylar film stretched over a zinc sulfide scintillation detector. The potential between this aluminized mylar film and the hemispherically shaped wire screen creates a strong electric field which attracts the charged ions. The ions thus attracted remain on the surface and continue their radioactive decay to other radon daughters. The principal radiation detected by a radon monitor of this type is the alpha particles from ^{218}Po (RaA) and ^{214}Po (RaC'). Alpha pulses are counted and integrated for a fixed time, usually 30 minutes. At the end of each timed counting period, the total count for each channel is printed automatically and the system resets and initiates counting for the next period.

11.1.2 Site Stabilization

Site stabilization is defined as those preventive or remedial measures taken to ensure that the radioactivity is retained within the confines of the site, or that any radionuclide migration that does occur is restricted to acceptable levels. Site stabilization involves the use of engineered procedures to reduce the mobility of the radioactivity and to protect the tailings pile from the effects of potential transport mechanisms. (A transport mechanism is any mechanism such as water or wind erosion, percolation of water through the soil, the burrowing of animals, or human activity, that results in the movement of radioactivity away from where it was intended to be confined.)

A detailed description of site stabilization procedures is given in Reference 1, which describes the decommissioning of low-level waste burial grounds. Potential stabilization activities that might be used for the sites considered in this study include:

- engineered procedures to control the flow of ground or surface water
- stabilization of the land surface and erosion control
- control of plants and animals that might disturb surface stabilization measures or transport radioactivity away from the site
- erection of physical barriers to control human activities at the site.

Some stabilization procedures might have beneficial effects in several areas. For example, the addition of a layer of soil (backfill) over a contaminated site would be used to repair areas damaged by subsidence, erosion, mass wasting, etc. The added material could be compacted and graded to control the flow of surface water over the site. The land surface could then be stabilized with a vegetative cover to reduce erosion. Details of site stabilization procedures that have potential application to the sites considered in this study are given in Section G.1 of Appendix G.

Several factors are used to evaluate stabilization techniques and procedures:

- effectiveness against potential transport mechanisms
- initial cost
- annual maintenance cost
- anticipated useful life
- ease of application.

The most important evaluation criterion is effectiveness in controlling potential transport mechanisms. Cost considerations and potential useful life are of secondary importance. Ease of application is the least important.

Site stabilization implies that the radioactivity remains on the site and that public activities are therefore restricted to land uses that do not compromise stabilization procedures or radionuclide confinement. Stabilization of a site is followed by a period of long-term care, during which some or all of the following activities are carried out at the site:

- environmental monitoring and records maintenance
- inspection and repair of physical features such as fences, soil overburden, etc.
- site drainage control
- erosion control
- vegetation management
- administrative control of the site to ensure that public uses of the site are appropriate to site conditions.

Long-term care continues until measurements and calculations indicate that the radioactivity has decayed to levels permitting unrestricted release of the site or until additional actions are taken to reduce the potential consequences of unrestricted site usage.

11.1.3 Removal of Contaminated Soil and Buried Waste

Waste or soil removal involves exhumation of the contaminated material, packaging, shipment, and burial of the material at an approved shallow-land disposal site. The technology, safety, and costs of removal of the buried waste from a commercial shallow-land burial ground are described in Reference 1. The procedures and packaging requirements for the removal of a radioactively contaminated industrial waste line are described in Reference 2.

Removal of contaminated material is usually much more expensive than stabilization of the site. Therefore, removal would likely be considered only in situations where site stabilization would not adequately ensure that the site did not present an unreasonable risk to public safety. To arrive at a decision about which decommissioning alternative to choose for a particular site, the following factors should be considered:

- levels of radioactivity remaining on the site
- the half-lives and radiotoxicity of the material
- the physical and chemical forms of the material
- hydrologic and geologic properties of the site
- potential uses of the decommissioned site
- the costs of different decommissioning alternatives, including both the dollar costs and the public and occupational radiation exposures resulting from the different alternatives
- political and social concerns about site proliferation.

Formal criteria for the decommissioning of sites have not yet been established by government regulatory agencies.⁽⁹⁾ (An exception to this general statement is the case of uranium mill tailings. The Environmental Protection Agency has recently issued interim standards for the cleanup of open lands and buildings contaminated with residual radioactive material from inactive uranium processing sites.⁽¹⁰⁾ The standards impose a limit of 5 pCi of ^{226}Ra per gram of soil for tailings on open land.) Residual radioactivity levels permitted by formal decommissioning criteria will determine the magnitude of the decommissioning task in terms of the quantity of waste to be removed, as well as the amount of associated contaminated soil requiring packaging and removal.

Exhumation of contaminated soil and waste can be accomplished using conventional earthmoving equipment and techniques. If exhumation requires the removal of overburden, a bulldozer is satisfactory for clearing small area. A

bottom-loading scraper earthmover is a good choice for removing overburden from large areas. A backhoe is used to dig the trench to uncover a contaminated drain line.

After the overburden, if any, has been removed, the choice of equipment for excavating the contaminated waste or soil is dependent on the material being recovered. Bulldozers or front-end loaders can be used for the removal of tailings piles or contaminated top soil. A backhoe is used to remove contaminated soil from a trench or other excavation. Some manual excavation work may be required, particularly in two situations:

- uncovering and dislodging large contaminated items such as drain lines or hold-up tanks
- final trench area cleanup for removal of spots of low-level contamination in trench bottoms and sidewalls.

Several options are available for waste materials packaging and transportation. If contaminated soil or tailings are to be hauled short distances, they may be transported in dump trucks. The dump truck beds are lined with plastic that is folded over the top of the load. A custom-made tarpaulin also covers the load during transit to the waste disposal area.

For shipping contaminated soil or tailings large distances over public highways, it may be necessary to package the material in plywood boxes lined with plastic and sealed by banding. Sections of contaminated pipe and other relatively small objects are also shipped in plywood boxes lined with plastics. Some large objects such as steel hold-up tanks or concrete manholes are sealed and shipped intact to the burial ground on low-boy trailers. Prior to transport, the entire object is wrapped in plastic. It may also be necessary to enclose the tank or manhole in a large, custom-built plywood box.

On the highway, periodic inspections of trucks, trailers, and packages are made to ensure that there is no spillage.

Workers wear anti-contamination clothing during excavation and packaging operations. Operations in a dusty environment or operations involving the excavation of toxic materials may also require the use of protective respiration equipment.

Waste or soil removal procedures are assumed to be performed by a private contractor. This contractor is hired by the site owner/operator who has the responsibility to decommission the site in preparation for the termination of his nuclear materials license. The actual decommissioning of the site is preceded by a period of planning and preparation that includes activities to

ensure that the decommissioning effort is performed in a safe and cost-effective manner in accordance with all applicable federal, state, and local regulations. In the performance of these activities the decommissioning contractor works closely with the site owner/operator.

11.1.4 Cost Estimates

For the ore processing tailings pile, costs are estimated for both the site stabilization and the removal options. Costs are expressed in 1981 dollars and include a 25% contingency. Some key bases and assumptions for estimating costs are given in Appendix G. Unit costs are listed in Appendix H.

Total costs include the costs of manpower, equipment, materials, and waste management (the packaging, transportation, and disposal of radioactive material removed from the site). Because transportation to, and disposal at, a shallow-land burial ground are contracted activities, manpower costs for transportation and disposal are included in the total costs of these items.

Manpower costs are determined by multiplying the man-days required to decommission a site by the costs per man-day shown in Table H.1-1 of Appendix H. For ease in evaluating time and manpower requirements, site decommissioning is divided into a sequence of tasks or steps. The steps for the site stabilization option are listed below.

1. planning and preparation (including initial site survey)
2. mobilization/demobilization
3. site stabilization
4. revegetation.

For the removal option, these steps are as follows:

1. planning and preparation (including initial site survey)
2. mobilization/demobilization
3. removal of overburden
4. exhumation and packaging of contaminated material
5. transport of contaminated material and disposal at a shallow-land burial ground
6. backfill and restoration of site
7. final site survey.

To determine the total time required to decommission a site, an estimate is made of the time required for efficient performance of the work by the postulated work crew. This time estimate is then increased by 50% to provide for preparation and set-up time, rest periods, etc. (ancillary time).

The owner/operator of a site is assumed to perform his own site survey. Site stabilization or waste and soil removal activities are assumed to be performed by a contractor hired by the owner/operator of the site. The contractor is anticipated to receive payment consisting of reimbursement for expenses (i.e., manpower, equipment, and material costs), plus a fee to provide a reasonable profit for his efforts. For this study, the contractor's fee is calculated on the basis of 8% of the sum of his manpower, equipment, and materials costs. This rate is judged to be reasonable for the size and complexity of the decommissioning projects. The costs of shipping containers for radioactive materials are assumed to be subject to the contractor's fee. Transportation and disposal tasks are performed by separate contractors hired by the site owner/operator.

Overhead rates applied to staff labor are expected to be significantly higher for the decommissioning contractor than they are for the site owner/operator. These higher overhead rates apply because of the larger ratio of supervisory and support personnel to direct labor that usually exists in contractor organizations and because of travel and living expenses associated with having personnel in the field rather than in an office. An overhead rate of 110% is applied to direct staff labor for all contractor personnel. The work crew for site decommissioning operations consists of a supervisor (assigned to the project on a half-time basis), a foreman, equipment operators, truck drivers, and technicians who are part of the contractor's staff, and a health physicist from the owner/operator's staff.

Monthly charges for equipment owned by the decommissioning contractor are calculated on the basis of 6% of the capital cost of the equipment and include allowances for equipment depreciation, maintenance and operating expenses (e.g., fuel, lubrication, etc.), the cost of decontamination following use, and return on investment. The equipment costs do not include the operator's wage. Weekly charges are estimated to be approximately one-third of the monthly charges.

Mobilization and demobilization costs are determined by estimating the time required for these activities. Costs of manpower and equipment are adjusted to include these time periods as well as the actual time spent decommissioning the site.

11.2 DECOMMISSIONING ANALYSES

Results of analyses of time and manpower requirements, total costs, and occupational radiation doses for decommissioning the reference ore processing site are presented in this section. Total costs of decommissioning include the costs of manpower, equipment, materials, waste management (e.g., the packaging, transportation, and disposal of radioactive waste), and contractor's fees, where applicable. Details of time and manpower requirements and of total costs for decommissioning the reference sites are presented in Appendix G.

11.2.1 Ore Processing Tailings Pile

Estimated time and manpower requirements, total costs, and occupational radiation doses for decommissioning a tailings pile by the option of stabilization of the pile are presented in Table 11.2-1, summarized from Section G.4 of Appendix G. The annual requirements and costs of long-term care following stabilization of the pile are also shown in Table 11.2-1. The cost of stabilization of the pile is estimated to be about \$157,000, and the occupational

TABLE 11.2-1. Summary of Estimated Manpower Requirements and Costs for Stabilization of a Reference Tailings Pile

Parameter	Site Stabilization Value		Totals	Long-Term Care (Annual Values)
	Planning & Preparation	Decommissioning		
Time (days)	20	12	32	10
Manpower (man-days)	70	104	174	27
Costs (\$ thousands) ^(a)				
Staff Labor	9.81	16.59	26.40	3.17
Equipment	2.00	17.15	19.15	1.00
Materials	1.00	65.60	66.60	0.50
Soil Analyses	5.00	--	5.00	1.00
Contractor's Fee	--	8.28	8.28	--
Waste Management	--	--	--	--
Subtotals	17.8	107.6	125.4	5.7
25% Contingency	4.4	26.9	31.3	1.4
Totals	22.2	134.5	157.0	7.0

(a) Costs are in 1981 dollars. Number of figures shown is for computational accuracy only and does not imply that level of precision.

radiation dose for this option is estimated to be 0.08 man-rem. The annual cost of long-term care is estimated to be about \$7,000, and the annual occupational radiation dose is estimated to be about 0.01 man-rem.

Requirements and costs for removal of the pile are shown in Table 11.2-2. The cost of removal of the pile and its disposal at a shallow-land burial ground is estimated to be about \$8.4 million, and the occupational radiation dose for this option is estimated to be 1.0 man-rem.

TABLE 11.2-2. Summary of Estimated Manpower Requirements and Costs for Removal of a Reference Tailings Pile

Parameter	Value			
	Planning & Preparation	Decommissioning	Final Radiation Survey	Totals
Time (days)	20	114	5	139
Manpower (man-days)	70	1569	18	1657
Costs (\$ thousands)(a)				
Staff Labor	9.81	255.77	2.45	268.03
Equipment	2.00	101.35	1.00	104.35
Materials	1.00	44.70	0.50	46.20
Soil Analysis	5.00	--	2.00	7.00
Contractor's Fee	--	157.42	--	157.42
Waste Management	--	<u>6122.60</u>	--	<u>6122.60</u>
Subtotals	17.8	6681.8	6.0	6705.6
25% Contingency	<u>4.4</u>	<u>1607.5</u>	<u>1.5</u>	<u>1676.4</u>
Totals	22.2	8352.3	7.5	8380.0

(a) Costs are in 1981 dollars. Number of figures shown is for computational accuracy only and does not imply that level of precision.

The ore processing tailings pile is described in Chapter 5. The pile contains the residue from ore refinery operations in which tin slag is processed for the recovery of niobium and tantalum. The tin slag is estimated to contain 0.2 wt% U_3O_8 and 0.5 wt% ThO_2 . The sludge from processing operations, which contains essentially all of the thorium and uranium, is pumped to a settling pond, where the water is allowed to evaporate, converting the sludge to a glassy solid. The reference pond is 100 m long by 50 m wide by 5 m deep, with

a 2.5 to 1 slope on each side. Additional information about the reference tailings pile (or pond) and its contents is shown in Table 11.2-3.

TABLE 11.2-3. Some Characteristics of the Reference Ore Processing Tailings Pile

<u>Parameter</u>	<u>Value</u>
Volume of Pond	16 400 m ³
Weight of Residue	4.1 x 10 ⁷ kg
U ₃ O ₈ Concentration	0.2 wt%
Contained U ₃ O ₈	8.2 x 10 ⁴ kg
ThO ₂ Concentration	0.5 wt%
Contained ThO ₂	2.0 x 10 ⁵ kg

Decommissioning begins with planning and preparation activities that include a radiological survey to determine the radiological condition of the pile and the site where the pile is located. The site survey includes measurements of gamma radiation levels, measurements of the rate of radon emanation from the pile, and the analysis of soil samples.

For the pile stabilization option, the following procedures are assumed. The pile is covered with a layer of asphalt 500 mm thick. This asphalt layer is then covered with one meter of soil. The soil is mounded slightly at the center of the pile to allow water to drain from the soil cover and to prevent the accumulation of runoff from rainfall or snowmelt. After compaction and contouring of the soil cover, the area is seeded with grass.

About one-half of the total cost of the site stabilization option results from the asphalt and the soil used to establish the cover over the tailings pile. Manpower costs represent about 21% of the total cost of this option.

Long-term care activities include administrative control, site maintenance, environmental surveillance, and vegetation management. Manpower costs represent almost 60% of the estimated annual cost of long-term care.

Conventional earthmoving equipment is used to exhume the pile for the removal option. Approximately 16,400 m³ of residue and 3,000 m³ of potentially contaminated soil are packaged in 1.2 m by 1.2 m by 2.4 m (3.4 m³) plastic-lined plywood boxes and shipped to a shallow-land burial ground for disposal. After the tailings pile is removed, the site is backfilled and graded, and grass is planted. The site is then surveyed to verify its suitability for unrestricted release.

Approximately 91% of the total cost of the disposal option is for waste management activities (i.e., the packaging, transportation, and disposal of the exhumed material). Waste management costs could be reduced by about \$1.5 million if the contaminated material was transported to the shallow-land burial ground in plastic-lined 10 m³ capacity dump trucks instead of being packaged in plywood boxes.

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12.0 SAFETY IMPACTS OF DECOMMISSIONING FOLLOWING AN ACCIDENT

Occupational, public, and transportation safety impacts from post-accident decommissioning, including post-accident cleanup, at the reference fuel-cycle and non-fuel cycle facilities are summarized in this chapter. Safety impacts from accident cleanup and decommissioning include: 1) radiation doses to the public from atmospheric releases of radioactivity resulting from routine tasks and industrial accidents during accident cleanup and decommissioning, 2) radiation doses to, and industrial accidents involving, workers performing the cleanup and decommissioning tasks, and 3) radiation doses to and accidents involving the transportation workers and the public during the shipment of radioactive materials from the site. A conservative approach, using parameters that tend to realistically maximize the consequences, is used to evaluate the safety impacts of accident cleanup and decommissioning. The evaluation uses current analysis data and methodology.

This chapter is divided into three sections: technical approach, accident cleanup safety, and decommissioning safety for each of the facilities. Each of the facility sections is divided into subsections that encompass public safety, occupational safety, and transportation safety for both accident cleanup and decommissioning. The information presented in this chapter is summarized from Appendix I. A basic assumption of the safety analyses in this chapter is that the radioactive waste materials from accident cleanup and decommissioning are disposed of offsite at the time of decommissioning.

12.1 TECHNICAL APPROACH

The safety evaluation is divided into two areas of interest: radiological safety and nonradiological safety. Radiological safety impacts are evaluated using a three-part approach. First a description of the reference facility is developed (see Chapter 5). Second, the radionuclide inventories and external dose rates within the facility resulting from normal operations and from the postulated accidents are characterized and quantified (see Chapters 5 and 6). Finally, reference tasks are used to calculate radiation exposures to workers and to the public from accident cleanup and from decommissioning. (The tasks for accident cleanup and decommissioning are described in Chapters 8, 9, 10, and 11.)

The nonradiological safety evaluation is based on the potential for industrial and transportation accidents that result in injuries or fatalities. The technical approach is divided into two parts. First, the total labor requirements for accident cleanup and decommissioning are analyzed and divided into categories of effort, and transportation requirements are quantified; second,

injuries and fatalities are calculated based on statistical information from the literature concerning accident frequencies for the selected categories of effort.

To provide a basis for the estimation of the safety impacts of accident cleanup and decommissioning and to ensure consistency between the various parts of the safety analysis, the following basic assumptions are made:

1. Appropriate radiation protection and contamination control techniques are applied to conform to the principle of keeping occupational radiation doses and radioactivity levels in effluents as low as reasonably achievable (ALARA).
2. The assessments of the safety impacts of post-accident decommissioning are based primarily on information pertaining to the decommissioning of the reference plants following normal shutdown. Appropriate adjustments are made to account for differences between post-accident and normal-shutdown radionuclide inventories and decommissioning requirements.
3. All offsite shipments of radioactive wastes are in accordance with applicable Department of Transportation regulations.
4. The largest potential radiological consequences of a given cleanup or decommissioning task are associated with performing that task in the area of the plant with the largest inventory of radionuclides.
5. The maximum radioactive release from a specific task is assigned to that task whenever it is performed in the facility. In performing the dose calculations for releases from routine tasks, the estimated total releases for the entire cleanup or decommissioning period are assumed to be released at a uniform rate over a one-year period.
6. In calculating atmospheric releases of radioactivity during decommissioning, no credit is taken for the radioactive decay of the radionuclide inventories present at the completion of the accident cleanup campaign.
7. Public radiation doses are calculated using the environmental data and assumptions discussed in Appendix I of Reference 1, consistent with the methods outlined in Regulatory Guide 1.109. Other specific assumptions used in calculating the occupational doses are found in Appendix I. A discussion of the assumptions used for the public and transportation radiation dose calculations is also included in Appendix I.

12.2 MOX PLANT SAFETY

Accident cleanup activities at the reference MOX plant precede the actual refurbishment or decommissioning of the plant and are essentially independent of whether the facility is to be refurbished or decommissioned. The accident cleanup activities are also independent of the alternative chosen for completing the decommissioning. As a practical matter, accident cleanup efforts contribute to the refurbishment or decommissioning effort. In this analysis, accident cleanup is addressed separately from decommissioning for the MOX and U-Fab plants. Accident cleanup and decommissioning are discussed together for the non-fuel cycle plant and the ore processing facility.

12.2.1 MOX Plant Accident Cleanup Safety

This section summarizes the accident cleanup safety of the MOX plant. Three areas of impact are of concern: public safety, occupational (worker) safety, and transportation safety.

Public Safety Impacts of Accident Cleanup

The public radiological safety impacts of onsite activities during accident cleanup are discussed in this section. Safety impacts are evaluated for routine tasks carried out during accident cleanup as well as for postulated industrial accidents that may occur. Nonradiological safety impacts to the public from onsite activities are judged to be negligible and are not considered further.

The consequences of atmospheric releases of radioactivity from routine tasks during accident cleanup are determined by calculating radiation doses to the maximum-exposed individual and to the population residing within 80 km of the site. Radiation exposure pathways considered for these releases are direct external exposure, inhalation, and ingestion of food products. The consequences of postulated industrial accidents that could occur during accident cleanup and result in airborne releases of radioactivity are determined by calculating inhalation radiation doses to the maximum-exposed individual.

Public Radiation Doses from Routine Tasks During Accident Cleanup. Loss of confinement of radioactive materials resulting in public radiation exposure is a primary safety concern during accident cleanup at the reference MOX plant. The radiation doses from these releases are calculated using the dose models discussed in Appendix I of Reference 1.

Tables 12.2-1 and 12.2-2 contain summaries of the calculated radiation doses to the maximum-exposed individual and to the population residing within 80 km of the site as a result of releases from routine tasks during accident cleanup. The radiation doses listed in the tables are the first-year radiation dose and 50-year committed radiation dose equivalent to total body, bone, and

TABLE 12.2-1 Summary of Calculated Doses to the Maximum-Exposed Individual from Releases of Radioactivity from Routine Tasks During Accident Cleanup (MOX)

Accident Cleanup Activity	First-Year Dose (mrem)		Fifty-Year Committed Dose Equivalent (mrem)	
	Bone	Lung	Bone	Lung
Preparations for Accident Cleanup	3.7×10^{-8}	5.8×10^{-8}	2.8×10^{-6}	7.4×10^{-8}
Initial Decontamination	9.2×10^{-6}	6.8×10^{-4}	4.4×10^{-3}	3.2×10^{-3}
Waste Treatment and Packaging	1.8×10^{-4}	1.2×10^{-2}	3.8×10^{-7}	2.8×10^{-7}

TABLE 12.2-2. Summary of Calculated Doses to the Population from Releases of Radioactivity from Routine Tasks During Accident Cleanup (MOX)

Accident Cleanup Activity	First-Year Dose (man-rem)		Fifty-Year Committed Dose Equivalent (man-rem)	
	Bone	Lung	Bone	Lung
Preparations for Accident Cleanup	5.9×10^{-8}	4.4×10^{-6}	2.7×10^{-5}	2.0×10^{-5}
Initial Decontamination	1.0×10^{-5}	8.0×10^{-4}	4.8×10^{-3}	3.6×10^{-3}
Waste Treatment and Packaging	2.0×10^{-4}	1.4×10^{-2}	9.2×10^{-2}	7.0×10^{-2}

lung. Doses to the maximum-exposed individual in any given year are estimated to be below the appropriate dose design objectives set forth in 10 CFR 50, Appendix I.⁽³⁾

The radiation doses shown in the tables include only activities in the reference MOX building.

Public Radiation Doses From Releases Due to Postulated Industrial Accidents During Accident Cleanup. The consequences of postulated industrial accidents that result in releases of radioactivity from the plant are determined by calculating the dose to the maximum-exposed individual.

A summary of estimated doses to the maximum-exposed individual from postulated releases due to industrial accidents during accident cleanup is given in Table 12.2-3. First-year radiation doses and fifty-year committed radiation dose equivalents are listed for the lung of the maximum-exposed individual. The industrial accidents are listed in order of decreasing magnitude of release.

TABLE 12.2-3. Summary of Maximum-Exposed-Individual Radiation Doses from Releases Due to Industrial Accidents During Post-Accident Cleanup (MOX)

Incident	Release to Atmosphere (μCi)	First-Year Dose (mrem)		Fifty-Year Dose Commitment (mrem)	
		Bone	Lung	Bone	Lung
Loss of Intermediate-Stage HEPA Filter after Exhaust Duct Decontamination	1.0×10^4	5.0	30	1.1×10^2	70
Inadvertent Cutting of Undecontaminated Metal	1.6×10^2				
Explosion and/or Fire of Ion Exchange Resin	83	7.0×10^{-2}	6.6×10^{-2}	2.5	6.6×10^{-2}
Inadvertent Dumping of Contaminated Solid Wastes:					
Abraded Firebrick	14				
Concrete Dust	1.4				
Condensed Metal Vapor	7.4×10^{-2}				
Loss of Local Airborne Contamination Control/Loss of Vacuum Filter	3.5	1.0×10^{-4}	1.1×10^{-2}	3.9×10^{-2}	2.8×10^{-2}
Temporary Loss of Service: Electricity (Normal and Emergency)	1.4	7.4×10^{-5}	4.4×10^{-3}	1.5×10^{-2}	1.1×10^{-2}
Liquid Leak:					
Chemical Decontamination	16	1.4×10^{-2}	1.3×10^{-2}	4.8	1.3×10^{-2}
Electropolishing	2.8×10^{-2}				
Fire Involving Contaminated Clothing or Combustible Waste	0.11	9.2×10^{-5}	8.8×10^{-5}	3.3×10^{-3}	8.8×10^{-5}

Occupational Safety Impacts of Accident Cleanup

Occupational safety during accident cleanup at the reference MOX plant is evaluated both for radiation exposure to cleanup workers and for nonradiological industrial accidents.

Estimates of occupational radiation doses are based on the postulated radiation dose rates in various areas of the facility and on the estimated staff labor required to complete the accident cleanup activities.

Occupational Radiation Doses from Accident Cleanup Activities. A summary of the estimated occupational radiation doses for accident cleanup following the reference accident is given in Table 12.2-4. The table contains a listing of the major accident cleanup activities and the associated estimated total external radiation doses to the workers.

TABLE 12.2-4. Summary of Estimated Occupational Radiation Doses from Accident Cleanup of the Reference MOX Plant

<u>Cleanup Activity</u>	<u>Estimated Total Occupational Doses (man-rem)</u>
Planning and Preparations	4.0
Cleanup	22.8
Support Operations	<u>6.2</u>
Total	33.0

The radiation doses to accident cleanup workers are calculated as the product of the estimated radiation zone manpower requirements and the external radiation dose rates postulated for each specific area. The occupational dose estimates are based on the following basic assumptions: 1) personnel exposure to radiation is minimized by using temporary shielding, remote handling techniques, and respiration equipment as appropriate, and by not keeping workers actively engaged in a task out of the radiation fields; 2) decontamination efforts are reasonably successful in reducing radiation dose rates; and 3) careful, prompt accounting of radiation doses is maintained to rapidly identify jobs that are causing excessive dose accumulations so that corrective action can be taken.

The total estimated occupational radiation doses during accident cleanup at the reference MOX plant is 27 man-rem following the reference accident. Accident cleanup workers requirements are adjusted in this study as necessary to ensure compliance with limitations on individual radiation dose accumulations of five man-rem/man-year.⁽⁴⁾

Industrial Safety Impacts of Accident Cleanup. The industrial safety impacts of accident cleanup include potential injuries and fatalities resulting from industrial accidents to cleanup workers. Estimated casualties are calculated by finding the products of 1) the frequencies of injuries and fatalities during various categories of work, and 2) the estimated worker time applied to

each work category. The estimated injuries and fatalities to workers during accident cleanup at the reference MOX plant following the postulated accident considered in this study are summarized in Table 12.2-5. As shown in the table, less than one injury is estimated for accident cleanup following a reference accident. Fatalities resulting from industrial accidents appear unlikely during accident cleanup.

TABLE 12.2-5. Summary of Estimated Occupational Lost-Time Injuries and Fatalities During Accident Cleanup (MOX)

Activity	Frequency Accidents/ 10^6 Man-Hour		Accident Cleanup		
	Lost-Time Injuries ^(a)	Fatalities	Man-Hours ^(b)	Lost-Time Injuries	Fatalities
Heavy Construction ^(c)	10.0	4.2×10^{-2}	3.2×10^3	3.2×10^{-2}	1.3×10^{-4}
Light Construction	5.4	3.0×10^{-2}	2.9×10^4	0.16	8.7×10^{-4}
Operational Support	2.1	2.3×10^{-2}	6.1×10^4	0.13	1.4×10^{-3}
Total				0.32	2.4×10^{-3}

(a) Lost-time injuries are defined in Reference 2.

(b) Labor estimates are given in Tables 10.1-3, 10.2-3, and 10.5-3 from Reference 1.

(c) Primarily facility demolition and equipment disassembly work.

Transportation Safety Impacts of Accident Cleanup

Radioactive waste materials are assumed to be shipped by exclusive-use truck to either a shallow-land burial ground or a federal repository, both of which are assumed to be 1600 km from the site. The safety impacts of transportation activities considered in this study include radiation doses to transport workers and to the public along the transport routes, radiation doses to the maximum-exposed individual from atmospheric releases during transportation accidents, and injuries and fatalities resulting from potential transportation accidents. Radiation doses received by workers unloading the radioactive materials at a repository or disposal site are not estimated in this study, since they are assumed to occur at a separate licensed facility.

Radiation Doses from Routine Transportation Activities During Accident Cleanup. To calculate radiation doses to transport workers and to the public from transportation activities, a number of assumptions are made concerning radiation exposure rates from the shipments and exposure times for the various individuals involved. The primary assumption is that each shipment contains enough radioactive material to result in the maximum exposure rates allowed by Department of Transportation regulations.

The estimated radiation doses from transportation activities during accident cleanup are listed in Table 12.2-6. The values in the table include both truck and rail shipments, and the numbers of each type of shipment required. The total estimated doses from transportation activities following the reference accident are 1.7 man-rem to transport workers and 0.4 man-rem to members of the public along the transportation routes.

TABLE 12.2-6. Summary of Estimated Radiation Doses from Transportation Activities During Accident Cleanup (MOX)

<u>Activity/Group</u>	<u>Number of Shipments</u>	<u>Radiation Dose (man-rem)</u>
<u>Truck Shipments</u>		
Truck Drivers	8	1.6
Garagemen	8	<u>0.1</u>
Total Trucking Worker Dose		1.7
Onlookers	8	0.16
General Public	8	<u>0.24</u>
Total Public Dose from Truck Shipments		0.40

Radiation Doses from Postulated Transportation Accidents During Accident Cleanup. Transportation accidents during the offsite shipment of radioactive materials from accident cleanup can potentially result in inadvertent releases of radioactivity and corresponding radiation doses to individuals near the accident location. For this study, a realistic "worst-case" accident during truck transport is analyzed. It is assumed that a Type B container is broken, releasing 10^{-2} curies of the contained radioactivity. The fifty-year committed dose equivalent to the lung of the maximum-exposed individual for this transportation accident during accident cleanup is 6.5 rem.

Nonradiological Safety Impacts of Transportation Activities During Accident Cleanup. As with any transportation task, a certain potential for accidental injury or death exists from transportation accidents during accident

cleanup activities at the reference MOX plant. A summary of the casualties estimated to result during transportation activities for accident cleanup is shown in Table 12.2-7. As shown, about 0.002 injuries and 0.001 fatalities are estimated for accident cleanup transportation activities following an accident. Casualties from truck transport are estimated to be much greater than those from rail transport because of the greater number of truck shipments required and because of the greater frequency of casualties during truck transport.

TABLE 12.2-7. Summary of Estimated Casualties from Transportation Accidents During Accident Cleanup (MOX)

<u>Transportation Category</u>	<u>No. of Shipments</u>	<u>Injuries</u>	<u>Fatalities</u>
Truck Transport	8	0.02	0.0012

12.2.2 Decommissioning Safety

Decommissioning activities at the reference MOX plant follow completion of the accident cleanup activities. The detailed analysis of the safety impacts resulting from post-accident decommissioning activities at the reference MOX plant presented in Appendix D is summarized in this section.

Public Safety Impacts of Post-Accident Decommissioning

The public radiological safety impacts of onsite activities during post-accident decommissioning are discussed in this section. Nonradiological safety impacts to the public from onsite activities are judged to be negligible and are not considered further.

The consequences of atmospheric releases of radioactivity from routine tasks during post-accident decommissioning are determined by calculating radiation doses to the maximum-exposed individual and to the population residing within 80 km of the site. Radiation exposure pathways considered for these releases are direct external exposure, inhalation, and ingestion of food products. The consequences of postulated industrial accidents that release radioactivity from the plant are determined by calculating inhalation radiation doses to the maximum-exposed individual.

Public Radiation Doses from Routine Tasks During Post-Accident Decommissioning. As during accident cleanup, loss of confinement of radioactive materials resulting in public radiation exposure is a primary safety concern during decommissioning. The radiation doses from these releases are calculated using the dose models discussed in Reference 1.

Tables 12.2-8 and 12.2-9 contain summaries of the calculated radiation doses to the maximum-exposed individual and to the population residing within 80 km of the site as a result of releases from routine tasks performed during decommissioning following the reference accident and the subsequent accident cleanup campaign. Listed in the tables are the first-year radiation dose and fifty-year committed radiation dose equivalent to total body, bone, and lung.

The public radiation doses from routine tasks during post-accident decommissioning are estimated to be small by comparison to the range of annual radiation doses to an individual from natural background in the United States

TABLE 12.2-8. Summary of Radiation Doses to the Maximum-Exposed Individual from Airborne Radionuclides Released During Normal Decommissioning Activities (MOX)

Activity or Location	Release to Atmosphere (μCi)	DECON			
		First-Year Dose (mrem)		Fifty-Year Dose Committed (mrem)	
		Bone	Lung	Bone	Lung
Chemical Decontamination:					
Electropolishing (station)	210	6.6×10^{-4}	1.0×10^{-3}	4.9×10^{-2}	1.3×10^{-3}
Electropolishing (in situ)	2.1×10^{-2}	6.6×10^{-8}	1.0×10^{-7}	4.9×10^{-6}	1.3×10^{-7}
Surface Cleaning:					
Spray Decontamination (Dry Processing Glove Boxes)	4.6×10^{-3}	8.1×10^{-10}	6.2×10^{-8}	3.8×10^{-7}	2.8×10^{-7}
Spray Decontamination (Wet Processing Glove Boxes)	1.2×10^{-2}	3.7×10^{-8}	5.8×10^{-8}	2.8×10^{-6}	7.4×10^{-8}
Physical Decontamination:					
Scraping Metal Surfaces	2.9	5.4×10^{-8}	3.8×10^{-6}	2.4×10^{-5}	1.8×10^{-5}
Scraping Firebrick	5.5	9.7×10^{-8}	7.2×10^{-6}	4.7×10^{-5}	3.4×10^{-5}
Concrete Removal	340	6.4×10^{-5}	4.4×10^{-4}	2.8×10^{-2}	2.1×10^{-2}
Exhaust Duct Decontamination	52	9.2×10^{-6}	6.8×10^{-4}	4.4×10^{-3}	3.2×10^{-3}
Concrete Drilling:					
Floors (Pour holes)					
Segmenting Equipment:					
Arc Saw	97	1.7×10^{-5}	1.3×10^{-3}	8.0×10^{-3}	5.9×10^{-3}
Plasma Torch	2.1×10^4	3.8×10^{-3}	2.8	1.7	1.3
Filter Replacement	990	1.8×10^{-4}	1.3×10^{-2}	7.9×10^{-2}	6.2×10^{-2}
	2.25×10^4	5.1×10^{-3}	2.9	1.9	1.4

TABLE 12.2-9. Summary of Radiation Doses to the Population from Airborne Radionuclides Released During Normal Decommissioning Activities (MOX)

Activity or Location	Release to Atmosphere (μCi)	DECON			
		First-Year Dose (man-rem)		Fifty-Year Dose Commitment (man-rem)	
		Bone	Lung	Bone	Lung
Chemical Decontamination:					
Electropolishing (station)	210	7.2×10^{-4}	1.2×10^{-4}	5.3×10^{-2}	1.4×10^{-3}
Electropolishing (in situ)	2.1×10^{-2}	7.2×10^{-8}	1.2×10^{-7}	5.3×10^{-6}	1.4×10^{-7}
Surface Cleaning:					
Spray Decontamination (Dry Processing Glove Boxes)	4.6×10^{-3}	9.0×10^{-10}	7.1×10^{-8}	4.3×10^{-7}	3.2×10^{-7}
Spray Decontamination (Wet Processing Glove Boxes)	1.2×10^{-2}	4.1×10^{-8}	6.6×10^{-8}	3.0×10^{-6}	8.3×10^{-8}
Physical Decontamination:					
Scraping Metal Surfaces	2.9	5.9×10^{-8}	4.4×10^{-6}	2.7×10^{-5}	2.0×10^{-5}
Scraping Firebrick	5.5	1.1×10^{-7}	8.5×10^{-6}	5.1×10^{-5}	3.9×10^{-5}
Concrete Removal	340	6.9×10^{-5}	5.1×10^{-3}	3.2×10^{-2}	2.4×10^{-2}
Exhaust Duct Decontamination	52	1.0×10^{-5}	8.0×10^{-3}	4.8×10^{-3}	3.6×10^{-3}
Concrete Drilling:					
Floors (Pour holes)					
Segmenting Equipment:					
Arc Saw	97	1.9×10^{-5}	1.5×10^{-3}	8.9×10^{-3}	6.7×10^{-3}
Plasma Torch	2.1×10^4	4.2×10^{-3}	0.32	2.0	1.5
Filter Replacement	990	2.0×10^{-7}	1.5×10^{-2}	9.2×10^{-2}	7.0×10^{-2}
	2.25×10^4	5.5×10^{-3}	3.4×10^1	2.2	1.6

(from 80 to 170 mrem per year).⁽⁵⁾ Furthermore, these estimated radiation doses are several orders of magnitude below the permissible levels of radiation in unrestricted areas as set forth in 10 CFR Part 50, Appendix I.⁽³⁾

Public Radiation Doses from Releases Due to Postulated Industrial Accidents During Post-Accident Decommissioning. The consequences of postulated industrial accidents that result in releases of radioactivity from the facility during decommissioning are determined by calculating the dose to the maximum-exposed individual. The industrial accident situations considered in this study are the same as those analyzed in Reference 1 for decommissioning of the reference MOX plant following normal reactor shutdown. While it is beyond the scope of this study to evaluate every potential industrial accident situation

during decommissioning, the postulated situations presented here are judged to represent the range of credible events and to reflect realistic maximum impacts to the public.

A summary of the estimated radiation doses to the maximum-exposed individual from releases due to postulated industrial accidents during post-accident decommissioning is given in Table 12.2-10. The accidents are listed in order of decreasing magnitude of release. First-year radiation doses and fifty-year committed radiation dose equivalents are listed for the lung of the maximum-exposed individual for DECON following the reference accident. The postulated accident that results in the largest calculated release and the largest doses to the maximum-exposed individual is the loss of HEPA filtration after exhaust duct decontamination. It is calculated that 1.0×10^4 μCi of accident-generated radioactive contamination could be released, resulting in a first-year dose of about 32 mrem and a fifty-year committed dose equivalent of about 78 mrem to the lung of the maximum-exposed individual. The calculated doses from releases due to postulated industrial accidents during post-accident decommissioning are more than three orders of magnitude below those during the accident cleanup campaign that precedes the decommissioning. This is due primarily to the marked reduction in radioactive contamination levels in the facility accomplished during the accident cleanup campaign.

Occupational Safety Impacts of Post-Accident Decommissioning

Occupational safety during post-accident decommissioning at the reference MOX plant is evaluated both for radiation exposure and for nonradiological industrial accidents.

Estimates of occupational radiation doses are based on the postulated radiation dose rates in various areas of the facility following the completion of accident cleanup and on the estimated staff labor required to complete the decommissioning. A summary of the detailed occupational dose information developed in Appendix I is given in this section. Estimates of worker injuries and fatalities resulting from industrial accidents during the decommissioning effort are also presented in this section. These casualty estimates are based on nuclear industry experience.

Occupational Radiation Doses from Post-Accident Decommissioning Activities. The occupational radiation dose from external exposure to gamma radiation during DECON is estimated to be 70.2 mrem at the reference MOX plant, following the reference accident and the subsequent accident cleanup. As discussed previously, the radiation doses to decommissioning workers are calculated in the same manner as the doses to accident cleanup workers.

Industrial Safety Impacts of Post-Accident Decommissioning. The industrial safety impacts of post-accident decommissioning include potential

TABLE 12.2-10. Summary of Radiation Doses to the Maximum-Exposed Individual from Accidental Airborne Radionuclide Releases During Decommissioning Activities (MOX)

Activity or Location	Release to Atmosphere (μCi)	DECON			
		First-Year Dose (mrem)		Fifty-Year Dose Commitment (mrem)	
		Bone	Lung	Bone	Lung
Loss of Intermediate-Stage HEPA Filter after Exhaust Duct Decontamination	1.0×10^4	5.2×10^{-1}	32	1.1×10^2	78
Inadvertent Cutting of Undecontaminated Metal	1.6×10^2	8.5×10^{-3}	5.0	1.8	1.3
Explosion and/or Fire of Ion Exchange Resin	83	7.0×10^{-2}	6.6×10^{-2}	2.5	6.6×10^{-2}
Inadvertent Dumping of Contaminated Solid Wastes:					
Abraded Firebrick	14	7.4×10^{-4}	4.4×10^{-2}	1.5	1.1
Concrete Dust	1.4	7.4×10^{-5}	4.4×10^{-3}	1.5×10^{-2}	1.1×10^{-2}
Condensed Metal Vapor	7.0×10^{-2}	3.8×10^{-6}	2.2×10^{-4}	7.9×10^{-4}	5.7×10^{-4}
Loss of Local Airborne Contamination Control/Loss of Vacuum Filter	3.5	1.9×10^{-4}	1.1×10^{-2}	3.8×10^{-2}	2.8×10^{-2}
Temporary Loss of Services:					
Electricity (Normal and Emergency)	1.4	7.4×10^{-5}	4.4×10^{-3}	1.5×10^{-2}	1.1×10^{-2}
Liquid Leak:					
Chemical Decontamination	16	1.4×10^{-2}	1.3×10^{-2}	4.8	1.3×10^{-2}
Electropolishing	2.8×10^{-2}	5.4×10^{-6}	5.1×10^{-6}	1.9×10^{-4}	5.1×10^{-6}
Fire Involving Contaminated Clothing or Combustible Waste	0.11	9.6×10^{-5}	9.2×10^{-5}	3.4×10^{-3}	9.2×10^{-5}
Explosion of Hydrogen During Electropolishing	7.1×10^{-3}	5.9×10^{-6}	5.5×10^{-6}	2.1×10^{-4}	5.9×10^{-6}
Man Intrusion	(b)				

(a) Frequency of Occurrences: High $>1.0 \times 10^{-2}$; Medium 1.0×10^{-2} ; Low 1.0×10^{-5} per year.

(b) Assumes a 40-hour exposure to an average air concentration of $290 \mu\text{Ci}/\text{m}^3$.

injuries and fatalities from industrial accidents to the decommissioning workers. Estimated casualties during decommissioning are calculated in the same manner as for accident cleanup. Table 12.2-11 contains a summary of the estimated worker injuries and fatalities from industrial accidents during DECON at the reference MOX plant following the reference accident. As shown in the table, lost-time injury is estimated during DECON. [Fatalities from industrial accidents appear to be unlikely during post-accident decommissioning.]

TABLE 12.2-11. Estimated Occupational Lost-Time Injuries and Fatalities from Decommissioning Activities (MOX)^(a)

Activity	Frequency Accidents/10 ⁶ Man-Hour		Man-Hours ^(c)	DECON	
	Lost-Time Injuries ^(b)	Fatalities		Lost-Time Injuries	Fatalities
Heavy Construction ^(b)	10.0	4.2×10^{-2}	3.3×10^4	0.33	1.4×10^{-3}
Light Construction	5.4	3.0×10^{-2}	8.5×10^4	0.46	2.6×10^{-3}
Operational Support	2.1	2.3×10^{-2}	1.2×10^5	0.25	2.8×10^{-3}
Total				1.0	6.8×10^{-3}

(a) Estimates of injuries and fatalities have been rounded to two significant figures.

(b) Lost-time injuries are defined in Reference 3.

(c) Labor estimates are given in Appendix I.

(d) Primarily facility demolition and equipment disassembly work.

Transportation Safety Impacts of Post-Accident Decommissioning

Radioactive wastes generated during post-accident decommissioning are assumed to be shipped offsite to a shallow-land burial site 1600 km from the facility. Exclusive-use trucks are used for these waste shipments. The safety impacts of transportation activities include radiation doses to transport workers and to the public along the transport routes, radiation doses to the maximum-exposed individual from atmospheric releases during transportation accidents, and injuries and fatalities resulting from potential transportation accidents. Radiation doses received by workers unloading the radioactive materials at the site are not estimated, since they are assumed to occur at a separate licensed facility.

Radiation Doses from Routine Transportation Activities During Decommissioning. The assumptions made to estimate radiation doses to transport workers and to the public from exposure to radioactive shipments during decommissioning are the same as those for accident cleanup. The primary assumption is that

each shipment contains enough radioactive material to result in the maximum exposure rates allowed by Department of Transportation regulations.

The estimated radiation doses from transportation activities during DECON at the reference MOX plant following an accident are listed in Table 12.2-12. Data presented in the table include the number of truck shipments required and the resulting doses to the exposed groups. The estimated total doses from transportation activities during DECON are about 6.4 man-rem to transport workers and about 1.5 man-rem to members of the public along the transportation route.

TABLE 12.2-12. Summary of Estimated Radiation Doses from Truck Transportation Activities During Post-Accident Decommissioning (MOX)

Mode	Group	Radiation Dose per Shipment (man-rem)		Total Radiation Dose Per Mode (man-rem)
		To Shallow Land Burial ^(a)	To Deep Geologic Disposal ^(b)	
DECON ^(c)	Truck Drivers	0.07	0.2	6.1
	Garagemen	0.003	0.01	<u>0.29</u>
				6.4 Total Worker Dose
	Onlookers	0.005	0.02	0.57
	General Public	0.01	0.03	<u>0.9</u>
				1.5 Total Public Dose

(a) Number based on 800 km (500 mi) to shallow land burial site.

(b) Number based on 2,400 km (1,500 mi) to deep geologic disposal site.

(c) Total shipments: for Immediate Dismantlement, 18 to shallow land burial and 24 to deep geologic disposal; for Preparations for Safe Storage, 8 to deep geologic disposal; for Entombment, 3 to deep geologic disposal.

Radiological Safety Evaluation of Postulated Transportation Accidents.

Estimated accident frequencies, release amounts and radiation doses to the maximum-exposed individual for selected accidents involving truck transport of radioactive waste materials are shown in Table 12.2-13. These frequencies are calculated by multiplying the total kilometers of radioactive material transport for each decommissioning mode times the total probability of accident per unit distance traveled for each accident severity class.

TABLE 12.2-13. Estimated Frequencies and Radioactivity Releases for Selected Transportation Accidents (MOX)

Accident Description	Release ^(b,c) (Curies)	Radiation Dose for Maximum-Exposed Individual, (rem) ^(a,e)			
		First-Year Dose		Fifty-Year Dose Commitment	
		Bone	Lung	Bone	Lung
Minor Accident with Closed Van	No Release	-- ^(d)	--	--	--
Moderate Accident with Closed Van	1×10^{-4}	6.8×10^{-3}	6.4×10^{-3}	2.4×10^{-1}	6.5×10^{-3}
	4.4×10^{-4}	2.6×10^{-2}	8.9×10^{-2}	6.5×10^{-2}	
Severe Accident with Closed Van	1×10^{-2}	6.8×10^{-1}	6.4×10^{-1}	24	6.5×10^{-1}
	4.4×10^{-2}	2.6	8.9	6.5	

(a) Maximum-exposed individual is assumed at 100 meters from the site of the accident.

(b) Based on an inventory of 100 Ci per truck shipment.

(c) Release fraction of respirable material for moderate and severe accidents are assumed to be 10^{-6} and 10^{-4} , respectively.

(d) A dash indicates that no calculation was made.

(e) The first value for each accident description is assuming the released radioactivity is all soluble; the second value is assuming all released radioactivity is insoluble.

The maximum-exposed individual is assumed to be located approximately 100 m from the point of a transportation accident. The calculated dose values shown in Table 12.2-13 are for the first-year dose and 50-year dose commitment to the bone and lung.

Nonradiological Safety Impacts of Decommissioning Transportation Activities. A certain potential for accidental injury or death exists from transportation accidents during the decommissioning of the reference MOX plants. A summary of the casualties estimated to result during transportation activities for post-accident decommissioning is presented in Table 12.2-14. As shown in the table, about 0.51 injuries and 0.03 fatalities are calculated for transportation activities during DECON following the reference accident.

12.3 U-FAB PLANT SAFETY

Accident cleanup activities at the reference U-Fab plant precede the actual refurbishment or decommissioning of the plant and are essentially independent of whether the facility is to be refurbished or decommissioned and,

TABLE 12.2-14. Estimated Injuries and Fatalities from Decommissioning Transportation Accidents (MOX)^(a)

<u>Transportation Operation</u>	<u>DECON</u>
Probability (Accidents per Vehicle, km)	1.0×10^{-6}
Injuries Per Accident	0.51
Fatalities Per Accident	0.03
km/Operation(b) (Round-Trip)	1.1×10^6
Estimated Nonradiological Impacts of Transportation Accidents ^(c)	Injuries--0.57 Fatalities--0.034

(a) Accident frequencies are from Appendix C, Table 1 of Reference 6.

(b) Assuming truck transport of 1600 km (1000 mi) round trip to shallow-land burial site, and 4800 km (3000 mi) to deep geologic disposal site. Immediate Dismantlement requires 630 trips to shallow-land burial site and 24 trips to deep geologic disposal site.

(c) Estimates of injuries and fatalities are rounded to two significant figures.

in the latter case, of the alternative chosen for completing the decommissioning. As a practical matter, accident cleanup efforts contribute to the refurbishment or decommissioning effort. In this analysis, accident cleanup is addressed separately from decommissioning.

12.3.1 U-Fab Plant Accident Cleanup Safety

This section summarizes the accident cleanup safety of the U-Fab plant. Areas of concern are public safety, occupational safety, and transportation safety.

Public Safety Impacts of Accident Cleanup

The public radiological safety impacts of onsite activities during accident cleanup are discussed in this section. Safety impacts are evaluated for routine tasks carried out during accident cleanup as well as for postulated industrial accidents that may occur. Nonradiological safety impacts to the public from onsite activities are judged to be negligible and are not considered further.

Public Radiation Doses from Routine Tasks During Accident Cleanup. Loss of confinement of radioactive materials resulting in public radiation exposure is a primary safety concern during accident cleanup at the reference U-Fab plant. The radiation doses from these releases are calculated using the dose models discussed in Appendix I of Reference 7.

Tables 12.3-1 and 12.3-2 contain summaries of the calculated radiation doses to the maximum-exposed individual and to the population residing within 80 km of the site as a result of releases from routine tasks during accident cleanup. The radiation doses listed in the tables are the first-year radiation dose and fifty-year committed radiation dose equivalent to total body, bone, and lung. The radiation doses shown in the tables include only activities in the reference U-Fab building.

Public Radiation Doses from Releases Due to Postulated Industrial Accidents During Accident Cleanup. The consequences of postulated industrial accidents that result in releases of radioactivity from the plant are determined by calculating the dose to the maximum-exposed individual.

TABLE 12.3-1. Summary of Calculated Doses to the Maximum-Exposed Individual from Releases of Radioactivity Resulting from Routine Tasks During Accident Cleanup (U-Fab)

Accident Cleanup Activity	First-Year Dose (mrem)		Fifty-Year Committed Dose Equivalent (mrem)	
	Bone	Lung	Bone	Lung
Preparations for Accident Cleanup	1.7×10^{-10}	4.1×10^9	5.9×10^{-10}	2.0×10^{-8}
Initial Decontamination	4.8×10^{-4}	1.2×10^{-2}	1.7×10^{-3}	5.7×10^{-2}
Waste Treatment and Packaging	7.0×10^{-8}	1.7×10^{-6}	2.5×10^{-7}	8.3×10^{-6}

TABLE 12.3-2. Summary of Calculated Doses to the Population from Releases of Radioactivity from Routine Tasks During Accident Cleanup (U-Fab)

Accident Cleanup Activity	First-Year Dose (man-rem)		Fifty-Year Committed Dose Equivalent (man-rem)	
	Bone	Lung	Bone	Lung
Preparations for Accident Cleanup	7.0×10^{-8}	2.0×10^{-6}	2.5×10^{-7}	9.1×10^{-6}
Initial Decontamination	4.8×10^{-4}	1.4×10^{-2}	1.7×10^{-3}	6.3×10^{-2}
Waste Treatment and Packaging	4.7×10^{-7}	1.4×10^{-5}	1.6×10^{-6}	6.2×10^{-5}

A summary of estimated doses to the maximum-exposed individual from postulated releases due to industrial accidents during accident cleanup is given in Table 12.3-3. First-year radiation doses and fifty-year committed radiation dose equivalents are listed for the lung of the maximum-exposed individual. The industrial accidents are listed in order of decreasing magnitude of release.

TABLE 12.3-3. Summary of Maximum-Exposed-Individual Radiation Doses from Releases Due to Industrial Accidents During Post-Accident Cleanup (U-Fab)

Incident	Release to Atmosphere (μCi)	First-Year Dose (mrem)		Fifty-Year Dose Commitment (mrem)	
		Bone	Lung	Bone	Lung
Loss of Intermediate HEPA Filter after Duct Decontamination	2.7	2.3×10^{-3}	7.6×10^{-2}	4.5×10^{-3}	1.9×10^{-1}
Loss of Local Airborne Contamination Control/ Loss of Vacuum Filter	0.70	6.0×10^{-4}	2.0×10^{-2}	1.1×10^{-3}	4.9×10^{-2}
Liquid Leak During Chemical Decontamination	4.5×10^{-3}	3.7×10^{-6}	1.3×10^{-4}	7.3×10^{-6}	3.1×10^{-4}

Occupational Safety Impacts of Accident Cleanup

Occupational safety during accident cleanup at the reference U-Fab plant evaluated both for radiation exposure to cleanup workers and for nonradiological industrial accidents.

Estimates of occupational radiation doses are based on the postulated radiation dose rates in various areas of the facility and on the estimated staff labor required to complete the accident cleanup activities.

Occupational Radiation Doses from Accident Cleanup Activities. A summary of the estimated occupational radiation doses for accident cleanup following the reference accident is given in Table 12.3-4. The table contains a listing of the major accident cleanup activities and the associated estimated total external radiation doses to the workers.

The total estimated occupational radiation dose during accident cleanup at the reference U-Fab plant is 0.81 man-rem following the reference accident.

TABLE 12.3-4. Summary of Estimated Occupational Radiation Doses from Accident Cleanup of the Reference U-Fab Plant

<u>Cleanup Activity</u>	<u>Estimated Total Occupational Doses^(b) (man-rem)</u>
Planning and Preparations	0.20
Cleanup	0.36
Support Operations	<u>0.25</u>
Totals	0.81

Accident cleanup worker requirements are adjusted in this study as necessary to ensure compliance with limitations on individual radiation dose accumulations of five man-rem/year.⁽⁴⁾

Industrial Safety Impacts of Accident Cleanup. The industrial safety impacts of accident cleanup include potential injuries and fatalities resulting from industrial accidents to cleanup workers. Estimated casualties are calculated by finding the products of 1) the frequencies of injuries and fatalities during various categories of work, and 2) the estimated worker time applied to each work category. The estimated worker injuries and fatalities during cleanup at the reference U-Fab plant following the postulated accident are summarized in Table 12.3-5. As shown in the table, less than one injury is estimated for accident cleanup following a reference accident. Fatalities resulting from industrial accidents appear unlikely during accident cleanup.

TABLE 12.3-5. Estimated Occupational Lost-Time Injuries and Fatalities from Decommissioning Activities (U-Fab)^(a)

<u>Activity</u>	<u>Frequency Accidents/10⁶ Man-Hour</u>		<u>Man-Hours^(c)</u>	<u>Lost-Time</u>	
	<u>Lost-Time Injuries^(b)</u>	<u>Fatalities</u>		<u>Injuries</u>	<u>Fatalities</u>
Heavy Construction ^(b)	10.0	4.2×10^{-2}	3.2×10^2	3.2×10^{-3}	1.3×10^{-5}
Light Construction	5.4	3.0×10^{-2}	3.4×10^3	1.8×10^{-2}	1.0×10^{-4}
Operational Support	2.1	2.3×10^{-2}	7.1×10^3	<u>1.5×10^{-2}</u>	<u>1.6×10^{-4}</u>
Total				3.6×10^{-2}	2.8×10^{-4}

(a) Estimates of injuries and fatalities have been rounded to two significant figures.

(b) Lost-time injuries are defined in Reference 8.

(c) Labor estimates are given in Appendix I.

(d) Primarily facility demolition and equipment disassembly work.

Transportation Safety Impacts of Accident Cleanup

Radioactive waste materials are assumed to be shipped by exclusive-use truck to a shallow-land burial ground, which is assumed to be 1600 km from the site.

Radiation Doses from Routine Transportation Activities During Accident Cleanup. To calculate radiation doses to transport workers and to the public from transportation activities, a number of assumptions are made concerning radiation exposure rates from the shipments and exposure times for the various individuals involved. The primary assumption is that each shipment contains enough radioactive material to result in the maximum exposure rates allowed by Department of Transportation regulations.

The estimated radiation doses from transportation activities during accident cleanup are listed in Table 12.3-6. The values in the table include both truck and rail shipments, and the numbers of each type of shipment required. The total estimated doses from transportation activities following the reference accident are 1.25 man-rem to transport workers and 0.24 man-rem to members of the public along the transportation routes.

TABLE 12.3-6. Summary of Estimated Radiation Doses from Transportation Activities During Accident Cleanup (U-Fab)

<u>Activity/Group</u>	<u>Number of Shipments</u>	<u>Radiation Dose (man-rem)</u>
<u>Truck Shipments</u>		
Truck Drivers	35	1.2
Garagemen	35	<u>0.05</u>
Total Trucking Worker Dose		1.25
Onlookers	35	0.08
General Public	35	<u>0.16</u>
Total Public Dose from Truck Shipments		0.24

Radiation Doses from Postulated Transportation Accidents During Accident Cleanup. Transportation accidents during the offsite shipment of radioactive materials from accident cleanup can potentially result in inadvertent releases of radioactivity and corresponding radiation doses to individuals near the accident location. For this study, a realistic "worst-case" accident during truck transport is analyzed. A Type B container is assumed to be broken in a severe accident, releasing about 10^{-4} curies of the contained radioactivity.

The fifty-year committed dose equivalent to the lung of the maximum-exposed individual is 0.032 rem for this transportation accident during accident cleanup.

Nonradiological Safety Impacts of Transportation Activities During Accident Cleanup. As with any transportation task, a certain potential for accidental injury or death from transportation accidents exists during accident cleanup activities at the reference U-Fab plant. A summary of the casualties estimated to result during transportation activities for accident cleanup is shown in Table 12.3-7. As shown in the table, about 0.02 injuries and 0.001 fatalities are estimated for accident cleanup transportation activities following an accident. Casualties from truck transport are estimated to be much greater than those from rail transport because of the greater number of truck shipments required and because of the greater frequency of casualties during truck transport.

TABLE 12.3-7. Summary of Estimated Casualties from Transportation Accidents During Accident Cleanup (U-Fab)

<u>Transportation Category</u>	<u>No. of Shipments</u>	<u>Injuries</u>	<u>Fatalities</u>
Truck Transport	35	0.02	0.001

12.3.2 Decommissioning Safety

Decommissioning activities at the reference U-Fab plant follow completion of the accident cleanup activities. The detailed analysis of the safety impacts of post-accident decommissioning activities at the reference U-Fab plant presented in Appendix D is summarized in this section.

Public Safety Impacts of Post-Accident Decommissioning

The public radiological safety impacts of onsite activities during post-accident decommissioning are discussed in this section. Nonradiological safety impacts to the public from onsite activities are judged to be negligible and are not considered further.

Public Radiation Doses from Routine Tasks During Post-Accident Decommissioning. As during accident cleanup, loss of confinement of radioactive materials resulting in public radiation exposure is a primary safety concern during decommissioning. The radiation doses from these releases are calculated using the dose models discussed in Reference 7.

Tables 12.3-8 and 12.3-9 contain summaries of the calculated radiation doses, to the maximum-exposed individual and to the population residing within 80 km of the site, resulting from releases from routine post-accident decommissioning tasks following the reference accident and the subsequent accident cleanup campaign. Listed in the tables are the first-year radiation dose and fifty-year committed radiation dose equivalent to total body, bone, and lung.

TABLE 12.3-8. Summary of Radiation Doses to the Maximum-Exposed Individual from Airborne Radionuclides Released During Normal Decommissioning Activities (U-Fab)

Activity or Location	Release to Atmosphere (μCi)	DECON		Fifty-Year Committed Dose Equivalent (mrem)	
		First-Year Dose (mrem)		Bone	Lung
		Bone	Lung		
Surface Cleaning					
Spray Decontamination of Hoods and Glove Boxes	2.9×10^{-5}	1.7×10^{-10}	4.1×10^{-9}	5.9×10^{-10}	2.0×10^{-8}
Physical Decontamination of Surfaces					
Scraping Metal Surfaces	1.2×10^{-2}	7.0×10^{-8}	1.7×10^{-6}	2.5×10^{-7}	8.3×10^{-6}
Scraping Firebrick	7.2×10^{-4}	4.0×10^{-9}	1.0×10^{-7}	1.4×10^{-8}	4.8×10^{-7}
Concrete Removal	2.7×10^{-3}	1.5×10^{-8}	3.7×10^{-7}	5.5×10^{-8}	1.8×10^{-6}
Exhaust Duct Decontamination	84	4.8×10^{-4}	1.2×10^{-2}	1.7×10^{-3}	5.7×10^{-2}
Segmenting and Transfer of Equipment					
Reciprocating Saw	2.3×10^{-4}	1.3×10^{-9}	3.2×10^{-8}	4.6×10^{-9}	1.5×10^{-7}
Oxyacetylene Torch	1.8×10^{-2}	1.0×10^{-7}	2.5×10^{-6}	3.6×10^{-7}	1.2×10^{-5}
Exhaust Filter Removal	8.6×10^{-2}	4.7×10^{-7}	1.2×10^{-5}	1.6×10^{-6}	5.5×10^{-5}
Totals	84.1	4.8×10^{-4}	1.2×10^{-2}	1.7×10^{-3}	5.7×10^{-2}

Public Radiation Doses from Releases Due to Postulated Industrial Accidents During Post-Accident Decommissioning. The consequences of postulated industrial accidents that result in releases of radioactivity from the facility during decommissioning are determined by calculating the dose to the maximum-exposed individual. The industrial accident situations considered in this study are the same as those analyzed in Reference 7 for decommissioning of the reference U-Fab plant following normal reactor shutdown. While it is beyond

TABLE 12.3-9. Summary of Radiation Doses to the Population from Airborne Radionuclides Released During Normal Decommissioning Activities (U-Fab)

Activity or Location	Release to Atmosphere (μCi)	DECON			
		First-Year Dose (man-rem)		Fifty-Year Committed Dose Equivalent (man-rem)	
		Bone	Lung	Bone	Lung
Surface Cleaning					
Spray Decontamination of Glove Boxes	2.9×10^{-5}	1.7×10^{-10}	4.8×10^{-9}	5.1×10^{-8}	2.2×10^{-5}
Physical Decontamination of Surfaces					
Spray Metal Surfaces	1.2×10^{-2}	7.0×10^{-3}	2.0×10^{-6}	2.5×10^{-7}	9.1×10^{-6}
Scraping Firebrick	7.2×10^{-4}	4.0×10^{-9}	1.2×10^{-7}	4.4×10^{-8}	5.2×10^{-7}
Concrete Removal	1.1×10^{-2}	1.5×10^{-8}	4.5×10^{-7}	5.5×10^{-8}	2.0×10^{-6}
Exhaust Duct Decontamination	84	4.8×10^{-4}	1.4×10^{-2}	1.7×10^{-3}	6.3×10^{-2}
Segmenting and Transfer of Equipment					
Reciprocating Saw	2.3×10^{-4}	1.3×10^{-9}	3.7×10^{-8}	4.6×10^{-9}	1.7×10^{-7}
Oxyacetylene Torch	1.8×10^{-2}	1.0×10^{-7}	2.9×10^{-6}	3.6×10^{-7}	1.3×10^{-5}
Exhaust Filter Removal	8.6×10^{-2}	4.7×10^{-7}	1.4×10^{-5}	1.6×10^{-6}	6.2×10^{-5}
Total	84	4.8×10^{-4}	1.4×10^{-2}	1.7×10^{-3}	6.3×10^{-2}

the scope of this study to evaluate every potential industrial accident situation during decommissioning, the postulated situations presented here are judged to represent the range of credible events and to reflect realistic maximum impacts from such situations to the public.

A summary of the estimated radiation doses to the maximum-exposed individual from releases due to postulated industrial accidents during post-accident decommissioning is given in Table 12.3-10. The accidents are listed in order of decreasing magnitude of release. First-year radiation doses and fifty-year committed radiation dose equivalents are listed for the lung of the maximum-exposed individual, for DECON following the reference accident. The postulated accident that results in the largest calculated release and the largest doses to the maximum-exposed individual is the loss of HEPA filtration after exhaust duct decontamination. It is calculated that $2.7 \mu\text{Ci}$ of accident-generated radioactive contamination could be released, resulting in a first-year dose of about 0.076 mrem and a fifty-year committed dose equivalent of about 0.19 mrem to the lung of the maximum-exposed individual. The calculated doses from releases due to postulated industrial accidents during post-accident

TABLE 12.3-10. Summary of Radiation Doses to the Maximum-Exposed Individual from Accidental Airborne Radionuclide Releases During Decommissioning Activities (U-Fab)

Incident	Release to Atmosphere (μCi)	DECON			
		First-Year Dose (mrem)		Fifty-Year Committed Dose Equivalent (mrem)	
		Bone	Lung	Bone	Lung
Loss of Intermediate HEPA Filter after Duct Decontamination	2.7	2.3×10^{-3}	7.6×10^{-2}	4.5×10^{-3}	1.9×10^{-1}
Loss of Local Airborne Contamination Control, Loss of Vacuum Filter	0.70	6.0×10^{-4}	2.0×10^{-2}	1.1×10^{-3}	4.9×10^{-2}
Liquid Leak During Chemical Decontamination	4.5×10^{-3}	3.7×10^{-6}	1.3×10^{-4}	7.3×10^{-6}	3.1×10^{-4}

decommissioning are more than 3 orders of magnitude below those during the accident cleanup campaign that precedes the decommissioning. This is due primarily to the marked reduction in radioactive contamination levels in the facility accomplished during the accident cleanup campaign.

Occupational Safety Impacts of Post-Accident Decommissioning.

Occupational safety during post-accident decommissioning at the reference U-Fab plant is evaluated both for radiation exposure and for nonradiological industrial accidents.

Estimates of occupational radiation doses are based on the postulated radiation dose rates in various areas of the facility following the completing of accident cleanup and on the estimated staff labor required to complete the decommissioning. A summary of the detailed occupational dose information developed in Appendix I is given in this section. Estimates of worker injuries and fatalities resulting from industrial accidents during the decommissioning effort are also presented in this section. These casualty estimates are based on nuclear industry experience.

Occupational Radiation Doses from Post-Accident Decommissioning Activities. The occupational radiation dose from external exposure to gamma radiation during DECON at the reference U-Fab plant, following the reference accident and the subsequent accident cleanup, is estimated to be 15.7 man-rem. The radiation doses to decommissioning workers are calculated in the same manner as the doses to accident cleanup workers, as discussed previously.

Industrial Safety Impacts of Post-Accident Decommissioning. The industrial safety impacts of post-accident decommissioning include potential injuries and fatalities from industrial accidents to the decommissioning workers. Estimated casualties during decommissioning are calculated in the same manner as accident cleanup. Table 12.3-11 contains a summary of the estimated worker injuries and fatalities from industrial accidents during DECON at the reference U-Fab plant following the reference accident. As shown in the table, ten lost-time injuries are estimated during DECON. Fatalities from industrial accidents appear unlikely during post-accident decommissioning.

TABLE 12.3-11. Estimated Occupational Lost-Time Injuries and Fatalities from Decommissioning Activities (U-Fab)^(a)

Activity	Frequency (events/10 ⁶ man-hour)		DECON or Deferred Decontamination		
	Lost-Time Injuries ^(b)	Fatalities	Man-Hours ^(c)	Lost-Time Injuries	Fatalities
Heavy Construction ^(b)	10.0	4.2×10^{-2}	1.3×10^4	0.13	5.5×10^{-4}
Light Construction	5.4	3.0×10^{-2}	3.5×10^4	0.19	1.1×10^{-3}
Operational Support	2.1	2.3×10^{-2}	4.8×10^4	0.10	1.1×10^{-3}
Total				0.42	2.8×10^{-3}

(a) Estimates of injuries and fatalities have been rounded to two significant figures.

(b) Lost-time injuries are defined in Reference 8.

(c) Labor estimates are given in Tables 10.1-3, 10.2-3, and 10.5-3. of Reference 7.

(d) Primarily facility demolition and equipment disassembly work.

Transportation Safety Impacts of Post-Accident Decommissioning

Radioactive wastes generated during post-accident decommissioning are assumed to be shipped offsite to a shallow-land burial site 1600 km from the facility. Exclusive-use trucks are used for these waste shipments. The safety impacts of transportation activities include radiation doses to transport workers and to the public along the transport routes, radiation doses to the maximum-exposed individual from atmospheric releases during transportation accidents, and injuries and fatalities resulting from potential transportation accidents. Radiation doses received by workers unloading the radioactive materials at the disposal site are not estimated, since they are assumed to occur at a separate licensed facility.

Radiation Doses from Routine Transportation Activities During Decommissioning. The assumptions made to estimate radiation doses to transport workers

and to the public from exposure to radioactive shipments during decommissioning are the same as those for accident cleanup. The primary assumption is that each shipment contains enough radioactive material to result in the maximum exposure rates allowed by Department of Transportation regulations.

The estimated radiation doses from transportation activities during DECON at the reference U-Fab plant following an accident are listed in Table 12.3-12. Data presented in the table include the number of truck shipments required and the resulting doses to the exposed groups. The estimated total doses from transportation activities during DECON are about 2.6 man-rem to transport workers and about 0.53 man-rem to members of the public along the transportation route.

TABLE 12.3-12. Summary of Estimated Radiation Doses from Truck Transportation Activities During Post-Accident Decommissioning (U-Fab)

<u>Group</u>	<u>Radiation Dose per Shipment (man-rem)^(a)</u>	<u>Total Radiation Dose (man-rem)^(b)</u>
Truck Drivers	0.07	2.5
Garagemen	0.003	<u>0.11</u>
Total Worker Dose		2.6
Onlookers	0.005	0.18
General Public	0.01	<u>0.35</u>
Total Public Dose		0.53

(a) Based on 800 km (500 mi) to burial site.

(b) Based on 35 shipments to burial site.

Radiological Safety Evaluation of Postulated Transportation Accidents.

Estimated accident frequencies, release amounts, and radiation doses to the maximum-exposed individual for selected accidents involving truck transport of radioactive waste materials are shown in Table 12.3-13. These frequencies are calculated by multiplying the total kilometers of radioactive material transport for each decommissioning mode by the total probability of accident per unit distance traveled for each accident severity class.

The maximum-exposed individual is assumed to be located approximately 100 m from the point of a transportation accident. The calculated dose values shown in Table 12.3-13 are for the first-year dose and 50-year dose commitment to the bone and lung.

TABLE 12.3-13. Estimated Frequencies and Radioactivity Releases for Selected Transportation Accidents (U-Fab)

Accident Description	Frequency of Accidents During DECON	Release (Ci) ^(c,d)	Radiation Dose for Maximum-Exposed Individual (rem) ^(a,b)			
			First-Year Dose		Fifty-Year Committed Dose Equivalent	
			Bone	Lungs	Bone	Lungs
Minor Accident	2.3×10^{-2}	No Release	-- ^(e)	--	--	--
Moderate Accident	5.4×10^{-3}	1×10^{-7}	3.9×10^{-6}	1.3×10^{-4}	7.7×10^{-6}	3.2×10^{-4}
Severe Accident	1.4×10^{-4}	1×10^{-5}	3.9×10^{-4}	1.3×10^{-2}	7.6×10^{-4}	3.2×10^{-2}

(a) Maximum-exposed individual is assumed at 100 m from the site of the accident.

(b) The first bone doses for each accident description are calculated assuming the released radioactivity is all soluble; the lung doses are calculated assuming all released radioactivity is insoluble.

(c) Based on an inventory of 100 mCi, the expected maximum per truck shipment.

(d) Release fractions for respirable material for moderate and severe accidents are assumed to be 10^{-6} and 10^{-4} , respectively.

(e) A dash indicates that no calculation is made.

The average radionuclide inventory per truck shipment is conservatively assumed to be 100 Ci of dispersible radioactive material, with shipping distances of about 800 km (500 mi) to the shallow land burial site, and 2400 km (1500 mi) to the deep geologic disposal site.

Nonradiological Safety Impacts of Decommissioning Transportation Activities. A certain potential for accidental injury or death exists from transportation accidents during the decommissioning of the reference U-Fab plant. A summary of the casualties estimated to result during transportation activities for post-accident decommissioning is presented in Table 12.3-14. As shown in the table, about 0.03 injuries and 0.002 fatalities are calculated for transportation activities during DECON following the reference accident.

12.4 SAFETY OF DECOMMISSIONING A NON-FUEL CYCLE FACILITY FOLLOWING AN ACCIDENT

Estimated occupational radiation doses for decommissioning example laboratories that process or use radioisotopes are summarized in this section. The analysis uses safety data for decommissioning laboratory components summarized in Chapter 2. The reference laboratories are described in Chapter 5. The technical approach used for this analysis is described in Section 12.4.1.

TABLE 12.3-14. Estimated Injuries and Fatalities from Decommissioning Transportation Accidents (U-Fab)^(a)

<u>DECON Transportation Operation</u>	
Probability (Accidents per Vehicle km)	1.0×10^{-6}
Injuries Per Accident	0.51
Fatalities Per Accident	0.03
Total Round Trip Travel (km) ^(b)	5.6×10^4
Estimated Nonradiological Impacts Transportation Accidents ^(c)	
Injuries	2.9×10^{-2}
Fatalities	1.7×10^{-3}

-
- (a) Accident frequencies are from Reference 9, Appendix C, Table 1.
- (b) Assuming truck transport of 1600 km roundtrip to low-level waste burial site. DECON requires 35 trips to burial site.
- (c) Estimates of injuries and fatalities are rounded to two significant figures.

12.4.1 Technical Approach

The technical approach and some key bases used to define requirements and to estimate the safety of decommissioning the example radioactive materials laboratory are discussed in this section.

Occupational Radiation Dose Estimates For Lab Components

Estimates of occupational radiation doses for both the decontamination option and the disassembly and disposal option are made for each facility component listed in Table 12.4-1. The estimated worker dose rates that form the bases for occupational dose calculations are also shown in Table 12.4-1. These dose rates are in reasonable agreement with experience at typical radioactive material processing and use laboratories.

Models and assumptions used to calculate the worker dose rates are described in Appendix I. Dose rate calculations are based on residual contamination levels on individual facility components given in Chapter 5. Representative values are used for resuspension rates, worker breathing rates, and dose conversion factors. For work with components contaminated with ^{241}Am , inhalation is the dominant exposure pathway.

TABLE 12.4-1. Estimated Worker Dose Rates for Decommissioning Facility Components^(a)

<u>Component</u>	<u>Contaminating Isotope</u>	<u>Assumed Surface Contamination Level^(b) (d/m/100 cm²)</u>	<u>Dominant Exposure Pathway</u>	<u>Dose Rate^(c) (man-rem/hr)</u>
Hood	²⁴¹ Am	2×10^8	Inhalation	2×10^0
Glove Box	²⁴¹ Am	2×10^8	Inhalation	9×10^1
Workbench	²⁴¹ Am	5×10^1	Inhalation	1×10^{-1}
Ductwork	²⁴¹ Am	1×10^2	Inhalation	2×10^{-1}
Floor	²⁴¹ Am	1×10^2	Inhalation	1×10^0
Wall	²⁴¹ Am	2×10^1	Inhalation	4×10^{-1}

(a) Details to support dose rate estimates are given in Appendix I.

(b) Surface contamination levels are taken from facility descriptions in Chapter 5.

(c) Total dose rate, including contributions from both direct exposure and inhalation.

The dose rates listed in Table 12.4-1 are the sum of the direct exposure dose rate and an inhalation "dose rate." The direct exposure dose rate is a true dose rate. Because inhalation results in the deposition of radioactivity in the body, the inhalation "dose rate" used in the table is not a true dose rate but the dose commitment from one hour's work. For this study, the inhalation "dose rate" is taken to be the 50-year committed dose equivalent from one hour of exposure.

Dose calculations are based on the assumption that the contaminated component is the only item in a room that measures 6 m x 10 m with walls 3 m high. Contamination levels are taken from the facility descriptions of Chapter 5. In calculating occupational radiation doses for decommissioning workers, contributions from the walls and floor, as well as from the contaminated component, are included. (Contributions to dose from all of the contaminated components in a room are included in estimates of occupational radiation doses for the decommissioning of laboratories presented in Section 12.5.2.)

Because of the potentially significant inhalation dose rates associated with the decommissioning of components contaminated with ²⁴¹Am, it may be necessary for persons involved in decommissioning these components to be equipped with protective respiratory equipment. The use of such equipment would reduce inhalation exposure by one or two orders of magnitude.

Occupational Radiation Dose Estimates For the ²⁴¹Am Laboratory

Estimates of occupational radiation dose are made for decommissioning the reference laboratory. The estimated worker dose rates that form the bases for

occupational dose calculations are shown in Table 12.4-1. These dose rates are in reasonable agreement with experience at typical radioactive-materials laboratories. Models and assumptions used to calculate the worker dose rates given in Table 12.4-1 are described in this section.

The dominant exposure pathway for workers engaged in decommissioning the ^{241}Am laboratory is inhalation of airborne radioactivity. Occupational radiation doses are estimated on the basis that protective respiratory equipment is not used by decommissioning workers. The potentially significant inhalation dose estimated for decommissioning the ^{241}Am laboratory illustrates the importance of the use of protective respiratory equipment by workers engaged in decommissioning this laboratory. The use of respiratory equipment could reduce the inhalation dose by one or two orders of magnitude.

Public Safety For the ^{241}Am Laboratory

This study is one in a series of studies of the decommissioning of nuclear facilities. For previous studies, all of which have involved fuel cycle facilities, estimates have been made of radiation doses to the public from airborne releases resulting from routine decommissioning operations and postulated decommissioning accidents. In each case, the 50-year committed dose equivalent to the population within 80 kilometers from airborne releases from normal decommissioning operations has been estimated to be a small fraction of the dose from naturally occurring radiation for the affected population. Potential radiation doses to members of the public from postulated decommissioning accidents have also been estimated to be small as well.

Estimated radionuclide inventories prior to the decommissioning of the reference radioactive materials laboratory are shown in Table 12.4-2. Because radionuclide inventories are small and generally confined to inside surfaces of glove boxes and hot cells, public safety impacts of the decommissioning of non-fuel-cycle nuclear facilities are expected to be less than the negligible impacts estimated previously for the decommissioning of fuel cycle facilities. Many radioactive materials laboratories have been decommissioned with no significant public exposure to radioactivity. Therefore, numerical calculations of public safety impacts are not made for the reference facilities of this study. Public exposure to radioactivity resulting from routine decommissioning operations is expected to be no greater than that experienced during normal operation of the facility.

12.4.2 Safety Analyses of Non-Fuel Cycle Plants

Results of analyses of occupational doses, for decommissioning the reference laboratory are presented in this section.

TABLE 12.4-2. Estimated Residual Radionuclide Inventories
Prior to the Decommissioning of the Reference
Radionuclide Materials Laboratory

Laboratory Component	Radionuclide Inventory for Reference Laboratory (curies) ^(a) ²⁴¹ Am
	Laboratory
Inside Fume Hoods	1×10^{-6}
Inside Glove Boxes	3×10^{-1}
Laboratory Benches	4×10^{-9}
Inside Ventilation Ductwork	2×10^{-7}
Filters	3×10^{-1}
Ceiling	6×10^{-8}
Walls	2×10^{-7}
Floor	3×10^{-7}
Total	6×10^{-1}

(a) Based on facility descriptions given in Chapter 5.

Fume Hoods

Occupational radiation doses are estimated by multiplying the dose rates shown in Table 12.4-1 by the man-days required to decommission the fume hood. Because the exposure rate will decrease as the component is cleaned, the average dose rates used to estimate occupational doses for decontamination are assumed to be one-half the values shown in Table 12.4-1. Occupational radiation doses for both the decontamination option and the packaging and disposal option are all estimated to be less than 0.5 man-rem.

Glove Boxes

Occupational radiation doses are estimated by multiplying the dose rates shown in Table 12.4-1 by the man-days required to decommission the glove box. Because the exposure rate will decrease as the component is cleaned, the average dose rates used to estimate occupational doses for decontamination are assumed to be one-half the values shown in Table 12.4-1. Except for glove boxes contaminated with ²⁴¹Am, occupational radiation doses for both the decontamination option and the packaging and disposal option are all estimated to be less than 0.03 man-rem. The estimated occupational radiation dose for decommissioning a glove box contaminated with ²⁴¹Am is in the range of 1 to 10 man-rem. This estimated worker dose is due primarily to inhalation and would be reduced by one or two orders of magnitude through the use of protective respiration equipment.

Laboratory Workbenches

Occupational radiation doses are estimated to range from less than 1×10^{-6} man-rem to 2×10^{-3} man-rem, depending on the type of contamination.

Ventilation Ductwork

Occupational radiation doses are estimated to be less than 0.1 man-rem. The highest worker exposures are associated with the packaging of ^{241}Am -contaminated ductwork. These radiation exposures can be reduced one or two orders of magnitude if workers use protective respiratory equipment.

Walls

Occupational radiation doses are estimated to range from 2×10^{-5} man-rem to 1×10^{-1} man-rem. The occupational dose from cleaning the walls of the ^{241}Am laboratory can be reduced one or two orders of magnitude if workers use protective respiratory equipment.

Floors

Occupational radiation doses are estimated to range from 2×10^{-6} man-rem to 7×10^{-2} man-rem. The occupational dose from cleaning the floor of the ^{241}Am laboratory can be reduced by worker use of protective respiratory equipment.

Laboratory for the Manufacture of ^{241}Am Sealed Sources

The reference laboratory for the manufacture of ^{241}Am sealed sources is described in Chapter 5. Some important parameters that characterize major laboratory components and influence the requirements and costs of decommissioning this laboratory are shown in Table 12.13. The estimated occupational radiation doses for decommissioning the reference ^{241}Am laboratory are 40 man-rem.

The estimated occupational radiation exposure from decommissioning the reference ^{241}Am laboratory is based on the assumption that decommissioning workers do not use protective respiratory equipment. The estimated radiation dose results almost entirely from inhalation of airborne radioactivity and represents the 50-year committed dose equivalent from inhalation during decommissioning of the glove box line. The inhalation dose could be reduced by one or two orders of magnitude if protective respiratory equipment is used. The magnitude of the potential inhalation dose from this decommissioning operation demonstrates the importance of proper filtration of laboratory air during decommissioning to limit the release of airborne radioactivity from the laboratory and to maintain the population dose at a negligible level.

12.5 SAFETY OF DECOMMISSIONING THE ORE PROCESSING SITE

Information on the occupational radiation doses for decommissioning the ore processing site is presented in this section. The site is described in Chapter 5.

The technical approach used to estimate the safety of the ore processing plant is described in Section 12.5.1. The results of the safety analyses for the reference site are presented in Section 12.5.2.

12.5.1 Technical Approach

The technical approach and some key bases used to define requirements and estimate the safety of decommissioning the reference site are discussed in this section.

Occupational Radiation Dose Estimates

Occupational radiation dose estimates are made by multiplying the man-days required to decommission a site by an average radiation dose rate. The actual worker dose rate experienced during site decommissioning depends on several factors, including the type and amount of radioactive contamination on the site, the location and concentration of the contamination, the site parameters, and the work procedures and schedules. Therefore, worker dose rates are expected to be site- and worker-specific.

Some information exists on dose rates at typical contaminated sites. For example, data from an operating low-level waste burial ground indicate an average dose rate at the site of about 1 mrem per 24-hour day (0.042 mrem/hr).⁽¹⁰⁾ Exposure records of 23 workers engaged in the removal of a contaminated industrial waste line⁽¹¹⁾ showed that over a 3-month period, only four dosimeters recorded monthly doses in excess of 10 millirem. The maximum total exposure for one worker for 1 month was 30 millirem (equivalent to an average dose rate of about 0.2 mrem/hr). Background values of gamma radiation at the inactive uranium mill tailings site at Tuba City, Arizona, have been measured at 0.01 mR/hr.⁽¹²⁾ All of these dose rates refer to direct exposures and do not include contributions from inhalation of radioactive particles.

The inhalation of airborne radioactivity may make a significant contribution to occupational exposure for some site decommissioning operations, depending on the nature of the site and the decommissioning option. The inhalation dose would not normally be significant for site stabilization or for waste removal operations at sites where soil contamination is minimal (e.g., the removal of an underground drain line). This dose could be significant during the removal of a tailings pile because of the dust problem. An example calculation made for the low-level waste burial ground decommissioning study⁽¹⁰⁾ showed that the occupational dose from inhalation might be a factor of one to

ten times as large as the occupational dose from external exposure. For some site decommissioning operations, worker use of face masks or other respiratory equipment or the use of water sprays to reduce dust concentrations would limit the occupational dose from inhalation of airborne radioactivity.

For this study, an average worker dose rate of 0.1 mrem/hr is assumed for site decommissioning operations. This value is believed to be reasonably conservative, based on available information about real sites. Decommissioning workers at potentially dusty sites such as tailings sites are assumed to wear protective respiratory equipment to maintain occupational dose rates at or below this level.

Public Safety

The reference sites of this study are believed to be typical of contaminated sites associated with non-fuel-cycle licensee facilities. Radionuclide inventories for these reference sites are several orders of magnitude smaller than the radionuclide inventory postulated for the low-level waste burial ground described in Reference 1. For example, the rare metals refinery tailings pile is estimated to contain less than 500 curies of uranium and thorium and their daughter nuclides. The other contaminated sites contain smaller amounts of radioactivity. By comparison, a single trench of the reference low-level waste burial ground is estimated to contain about 70,000 curies of radioactivity.⁽¹⁰⁾

Public radiation doses from decommissioning a low-level waste burial ground by the waste removal option are estimated to be small (Reference 10). Estimated 50-year committed dose equivalents to the maximum-exposed individual and to the total exposed population for removal of all the waste from one burial trench are a small fraction of the dose that would be received from natural background radiation over the same time period.

Because of the relatively small radionuclide inventories on most non-fuel-cycle licensee sites, the public safety impacts of removing this radioactivity are expected to be negligible. Calculations of public-radiation doses from decommissioning operations are therefore not made for the reference sites of this study.

12.5.2 Occupational Safety of Ore Tailings Pile

Estimated occupational radiation doses for decommissioning a tailings pile by the option of stabilization of the pile is presented in Table 12.5-1. The annual occupational doses resulting from long-term care following stabilization of the pile are also shown in Table 12.5-1. The occupational radiation dose for this option is estimated to be 0.08 man-rem. The annual occupational radiation dose of long-term care is estimated to be about 0.01 man-rem.

TABLE 12.5-1. Summary of Estimated Occupational Radiation Doses for Stabilization of a Reference Tailings Pile

Parameter	Value			Long-Term Care (Annual Values)
	Site Stabilization			
	Planning and Preparation	Decommis- sioning	Totals	
Occupational Dose (man-rem)	--	0.08	0.08	0.01

The removal of the pile and its disposal at a shallow-land burial ground are estimated to result in an occupational radiation dose estimated to be 1.0 man-rem.

The tailings pile is described in Chapter 5. The pile contains the residue from ore refinery operations in which tin slag is processed for the recovery of niobium and tantalum. The tin slag is estimated to contain 0.2 wt% U_3O_8 and 0.5 wt% ThO_2 . The sludge from processing operations, which contains essentially all of the thorium and uranium, is pumped to a settling pond where the water is allowed to evaporate, converting the sludge to a glassy solid. The reference pond is 100 m long by 50 m wide by 5 m deep, with a slope of 2.5 to 1 on each side.

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13.0 COMPARISONS OF POST-ACCIDENT DECOMMISSIONING WITH NORMAL-SHUTDOWN DECOMMISSIONING

In this chapter, manpower requirements, occupational radiation doses, waste volumes, and costs for post-accident decommissioning at the reference MOX, U-Fab, and non-fuel cycle facilities are compared with the same parameters for decommissioning following normal shutdown. The previous decommissioning studies provided information about the safety and costs of decommissioning following normal shutdown. For ease of comparison, decommissioning costs shown in References 1, 2 and 3 are adjusted to the 1981 cost base used in this study. The factors used for adjusting costs from the original 1978 data base to the 1981 data base are described in Appendix H.

Decommissioning is preceded by accident cleanup in considering the post-accident case for the MOX and U-Fab plants. The accident cleanup activities remove most of the accident-generated waste; however, there still may be greater levels and more widespread contamination present after accident cleanup than after normal shutdown. For the non-fuel cycle facility and the ore processing facility, accident cleanup is considered to be part of decommissioning because it is inconvenient to separate an initial cleanup task from the decommissioning process, due to the small size of those facilities.

For the post-accident case, certain tasks are completed during accident cleanup that would normally be part of the decommissioning activities (e.g. comprehensive radiation surveys). Significant portions of other tasks that are part of normal decommissioning may also be performed during accident cleanup. The planning and preparation period that precedes the decommissioning can be shortened for the post-accident case because of preparatory activities that take place prior to and during accident cleanup. Thus, accident cleanup can have the effect of reducing some of the normal decommissioning requirements and costs.

Accident cleanup also results in some new decommissioning tasks that are not included in normal-shutdown decommissioning. Systems and structures that are specially installed for accident cleanup must be decommissioned. These new tasks increase the requirements and costs of post-accident decommissioning relative to normal-shutdown decommissioning.

Comparisons of manpower requirements, occupational radiation doses, waste management requirements, and costs between post-accident decommissioning following the reference accident and normal-shutdown decommissioning via the DECON alternative are presented in Table 13.0-1. Estimated staff labor requirements

TABLE 13.0-1. Comparisons of Post-Accident and Normal-Shutdown Decommissioning (DECON Alternative) for the MOX and U-Fab Plants

Parameter	Reference MOX		Reference U-Fab	
	Post-Accident DECON ^(a)	Normal-Shutdown DECON ^(b)	Post-Accident DECON ^(a)	Normal-Shutdown DECON ^(c)
Staff Labor (man-years)	158	131	63.8	53.2
Occupational Dose (man-rem)	91	76	15.9	15.7
Waste Volume (m ³) ^(d)	451	430	1155	1100
Cost (\$ millions) ^(d)	12.8	10.7	5.9	4.9

(a) Values are for DECON following cleanup after the reference accident.

(b) Values are from Reference 1 (NUREG/CR-1266)

(c) Values are from Reference 2 (NUREG/CR-1266)

(d) Costs are in 1981 dollars and include a 25% contingency.

and occupational radiation doses for post-accident DECON are approximately 1.5 times larger than estimated staff labor requirements and occupational radiation doses for DECON following normal shutdown. The total costs of post-accident DECON are estimated to be about 20% larger than the costs of DECON following normal shutdown.

Decontamination and dismantlement operations for post-accident DECON are very similar to those for DECON following normal shutdown. However, average radiation dose rates experienced by workers engaged in decommissioning operations are estimated to be 1.2 times higher for post-accident DECON following the reference accident than for normal-shutdown DECON. Staff labor requirements shown in Table 13.0-1 include both decommissioning workers and support staff. Decommissioning worker requirements are adjusted to ensure that occupational radiation doses to individual workers do not exceed five rem per year.⁽⁴⁾

Waste management requirements for post-accident DECON following the reference accident are only slightly larger than those for DECON following normal shutdown. The volumes of contaminated material requiring disposal are comparable for the two cases. (Slightly more contaminated concrete is packaged for disposal during post-accident decommissioning than during decommissioning following normal shutdown.) The major difference is the greater volume of

radioactive waste (e.g., disposable clothing, plastic covers, tools, construction materials, immobilized decontamination solutions, and miscellaneous trash) requiring disposal as a result of the additional decontamination required during post-accident decommissioning.

The major factor affecting the increase in costs for post-accident DECON is the larger labor requirement for post-accident decommissioning.

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14.0 FACILITATION OF POST-ACCIDENT CLEANUP AND DECOMMISSIONING

Methods and procedures are described in this study for the conceptual cleanup and decommissioning of reference fuel cycle and non-fuel cycle facilities that have been involved in serious accidents. The purpose of this chapter is to identify some design features, construction methods, and regulatory procedures that could expedite and simplify the cleanup and decommissioning operations at an accident-damaged nuclear facility.

The primary purpose of the facilitation of accident cleanup and decommissioning is to reduce occupational and public radiation doses during these operations; a secondary purpose is to reduce the costs. If introduced during the design or construction stage, a facilitation procedure ideally will also reduce radiation doses to workers and to the public during normal operation and routine maintenance periods. In reality, however, reductions in cost will not always occur when a design change is made to facilitate cleanup and decommissioning activities, and costs will often be increased rather than decreased.

Radiation doses, quantities of radioactive waste, and even the costs of accident cleanup and decommissioning can also be reduced by the careful planning of decontamination procedures and by the use of special tools and techniques. In the case of cleanup and decommissioning operations that must be performed in high-radiation areas, reductions in the performance time for these operations and consequent reductions in worker exposure can often be achieved by rehearsing an operation in a nonradiation area, sometimes with a mockup of the equipment or facility to be decontaminated, prior to entry into the radiation area for actual performance of the work.

Design considerations and innovative ideas that have been identified in previous studies^(1,2,3) of decommissioning fuel cycle and non-fuel cycle facilities following normal shutdown are also applicable to the facilitation of post-accident cleanup and decommissioning. Table 14.0-1 lists some of these considerations and possible solutions summarized from References 1, 2, and 3. In addition, some facilitation considerations from LWR decommissioning studies are included that are applicable to non-reactor facilities (Reference 4).

A facilitation report,⁽⁵⁾ prepared as part of the series of NRC-sponsored studies on the decommissioning of nuclear facilities, deals specifically with design features, special equipment, and construction methods useful in the facilitation of decommissioning LWRs. While the facilitation issues discussed in Reference 5 are intended to be directly applicable to the decommissioning of

TABLE 14.0-1. Design Considerations and Innovative Ideas
Related to Improvements in Decommissioning
Technology^(a)

<u>Facilitation Consideration</u>	<u>Possible Solution Idea^(b)</u>
1. Elimination of difficulties in decontaminating the internals of pipes and tanks.	1. Build in decontamination spray systems or access ports for their eventual use.
2. Elimination of difficulties in decontaminating and eventual demolition of concrete surfaces.	2. A "double layer" construction concept for concrete surfaces (see Figure 13-1 of Reference 4 for conceptual design).
3. Reclamation, reuse, and/or recycling of valuable materials (both during operation and decommissioning.	3. A/E provide space for inclusion of a commercial-size electropolishing unit in most advantageous facility location.
4. Reduce personnel radiation exposure while changing radioactive filters.	4. Use of gatling gun device (see Figure 13-2 of Reference 4 for conceptual design).
5. Development of remote operations.	5. Improvements in high pressure remote quick disconnect fittings for easier, quicker operation (R&D required).
6. Improvement of waste packaging containers.	6. Acceptable industry-wide standardization of size, design and material of construction of LSA containers would enable cost reductions.
7. Accessibility of equipment for ease of dismantlement.	7. Take down walls and removable roofs in locations where eventual equipment removal needs might dictate; in many cases it is easier to build it into the original structure than to knock it out later.
8. HVAC Systems.	8. Incorporation of devices (unspecified) for purposes of either temporary or permanent isolation including capabilities for various disassembly modes and including design analysis of varying air flow conditions expected during disassembly and dismantlement.
9. Equipment for reduction of volume of radioactive wastes.	9. Digestion, incineration, and/or compaction systems for combustible wastes.

TABLE 14.1. (contd)

Facilitation Consideration	Possible Solution Idea) ^(b)
10. Replacement equipment in operating nuclear plants.	10. Incorporation of design considerations to facilitate decommissioning should become an integral part of all specifications for replacement equipment in aging nuclear power plants. The dual objective of replacement equipment should reflect modifications based on the experience gained from using the original equipment plus the design objectives regarding future decommissioning of that equipment.
11. Radiation exposure reduction.	11. Radical building design alternative in which a shielded working platform on railroad tracks has access from above to all compartments in buildings containing radioactive equipment. Substantial radiation sources could thus be decontaminated, dismantled, removed to the shielded platform, and transported to a disassembly/electro-polishing station at a greatly reduced cost in terms of dollars and personnel exposure. Such a design would pay dividends during the operational lifetime as well to servicing and maintenance personnel. (see Reference 4)
12. Porous surfaces which increase contamination problems.	12. Prepolish metal process equipment surfaces to enhance decontamination. Chemical-resistant paints can also be used on porous surfaces.
13. Crevices in process areas.	13. Minimize crevices in process equipment. Glove boxes and hoods would ease decontamination.
14. Hood and equipment that could be disassembled.	14. Provide equipment that can be disassembled for decontamination rather than cutting apart.

(a) Summarized from References 1, 2, 3, and 4.

(b) No cost benefit analyses have been made. Such analyses could be expected to require participation by and good engineering judgment of the designer, the architect-engineer (A/E), the contractor and the operator of the nuclear facility on a case-by-case basis.

a nuclear reactor following normal shutdown, some also apply to the facilitation of post-accident cleanup and decommissioning of nuclear facilities other than reactors. Applicable facilitation methods discussed in Reference 5 include the following:

- improved documentation
- improved access to contaminated equipment
- techniques for improved protection of concrete and improved removal of contaminated concrete
- improved shielding of decommissioning and maintenance personnel
- techniques for reductions in radioactive waste volumes
- remote maintenance and decommissioning equipment
- special decommissioning tools and techniques.

Accident cleanup usually involves work in areas of relatively high radioactive contamination. To reduce worker exposure to radiation during the early stages of accident cleanup, design modifications should, if possible, facilitate the use of remote decontamination techniques, such as spray systems for the washdown of contaminated surfaces, and of remotely controlled manipulators. Protective coatings should be used wherever possible (e.g., on concrete surfaces) to facilitate surface decontamination. Controls for building systems needed during accident cleanup should be located externally to potentially contaminated areas whenever possible. Attention should be given to locating personnel entries and walkways and critical equipment away from areas within the building where contamination could accumulate as the result of an accident.

Experience gained in the accident cleanup of TMI-2 will undoubtedly provide valuable insights into design modifications and construction practices that could facilitate the cleanup and decommissioning of accident-damaged non-reactor nuclear facilities. Programs have been established in conjunction with TMI-2 cleanup activities to gather information about electrical and mechanical components and systems that were subjected to unusual stresses during the accident.⁽⁵⁾ A goal of these programs is to recommend changes that would improve the reliability of the systems and services needed for accident cleanup.

An instrumentation and electrical program was established at TMI-2 in May, 1980. The purpose of this program is to examine the status, determine failure

modes and changes in operating characteristics, and analyze the impact on system safety of failures of electrical components and electrical equipment such as:

- area radiation monitor instrumentation
- crane electrical components
- cables and cable terminations
- thermocouples.

The program will recommend standards and qualifications of electrical equipment to improve the survivability and reliability of this equipment in case of an accident.

Several recommendations have already come from this program. These include:

- a recommendation for the use of conformal coating of assembled printed wiring boards to improve their performance in humid environments
- a recommendation for improvements in the design, testing, and installation of electrical equipment.

Several simple circuit design changes have also been identified.

Management of the wastes from cleanup and decommissioning of an accident-damaged nuclear facility represents a major cost item and a significant source of worker exposure to radiation. Some of these wastes (such as evaporator bottoms from the processing of contaminated liquids) from the MOX plant in particular may not be suitable for disposal at a shallow-land burial ground. Temporary shielded storage for these wastes may have to be provided onsite. Ultimate disposal will require that the wastes be handled a second time with resulting additional occupational radiation doses and additional costs. No regulatory framework has yet been developed to specifically address the treatment and disposal of wastes from accident cleanup and decommissioning of nuclear facilities. Regulatory attention should be given to defining waste disposal criteria that will minimize the impacts of waste management on costs and occupational exposures for accident cleanup and decommissioning.

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