Technology, Safety and Costs of Decommissioning a Reference Uranium Hexafluoride Conversion Plant

Prepared by H. K. Elder

Pacific Northwest Laboratory
Operated by
Battelle Memorial Institute

Prepared for
U.S. Nuclear Regulatory
Commission

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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 will be prepared as the studies of the various facilities are completed.

The first report in this series, published in FY 1977, covers a fuel reprocessing plant;⁽²⁾ the second, published in FY 1978, covers a pressurized water reactor;⁽³⁾ and the third, published in FY 1979, deals with a small mixed oxide fuel fabrication plant.⁽⁴⁾ An addendum to the pressurized water reactor report,⁽⁵⁾ issued in FY 1979, 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, published in FY 1980, deals with a low-level waste burial ground;⁽⁶⁾ the sixth, published in FY 1980, covers a large boiling water reactor power station;⁽⁷⁾ and the seventh, published in FY 1981, examines a uranium fuel fabrication plant.⁽⁸⁾ The eighth report, published in FY 1981, covers the decommissioning of non-fuel-cycle nuclear facilities.⁽⁹⁾ This report, ninth in the series, deals with the decommissioning of a reference uranium hexafluoride conversion plant.

Additional topics will be reported on the tentative schedule as follows:

- FY 1982 • Multiple Reactor Facilities
- FY 1982 • Research/Test Reactors
- FY 1982 • LWR Post-Accident
- FY 1982 • Independent Spent Fuel Storage Installations

The second series of studies covers supporting information on the decommissioning of nuclear facilities. Three reports have been issued in the second series. The first, published in FY 1979, consists of an annotated bibliography on the decommissioning of nuclear facilities.⁽¹⁰⁾ The second, published in FY 1979, is a review and analysis of current decommissioning regulations.⁽¹¹⁾ The third, published in FY 1980, covers the facilitation of the decommissioning of light water reactors,⁽¹²⁾ identifying modifications or design changes to facilities, equipment, and procedures that will improve safety and/or reduce costs.

The information provided in this report on uranium hexafluoride conversion facilities, 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:

Chief
Fuel Process Systems Standards Branch
Division of Engineering Standards
Office of Standards Development
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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4. Technology, Safety and Costs of Decommissioning a Reference Small Mixed Oxide Fuel Fabrication Plant. NUREG/CR-0129, Pacific Northwest Laboratory for U.S. Nuclear Regulatory Commission, February 1979.
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ABSTRACT

Safety and cost information is developed for the conceptual decommissioning of a commercial uranium hexafluoride conversion (UF_6) plant. Two basic decommissioning alternatives are studied to obtain comparisons between cost and safety impacts: DECON, and passive SAFSTOR. A third alternative, DECON of the plant and equipment with stabilization and long-term care of lagoon wastes, is also examined.

DECON includes the immediate removal (following plant shutdown) of all radioactivity in excess of unrestricted release levels, with subsequent release of the site for public use. Passive SAFSTOR requires decontamination, preparation, maintenance, and surveillance for a period of time after shutdown, followed by deferred decontamination and unrestricted release. DECON with stabilization and long-term care of lagoon wastes (process wastes generated at the reference plant and stored onsite during plant operation) is also considered as a decommissioning method, although its acceptability has not yet been determined by the NRC.

The decommissioning methods assumed for use in each decommissioning alternative are based on state-of-the-art technology. The elapsed time following plant shutdown required to perform the decommissioning work in each alternative is estimated to be: for DECON, 8 months; for passive SAFSTOR, 3 months to prepare the plant for safe storage and 8 months to accomplish deferred decontamination. Planning and preparation for decommissioning prior to plant shutdown is estimated to require about 6 months for either DECON or passive SAFSTOR. Planning and preparation prior to starting deferred decontamination is estimated to require an additional 6 months. DECON with lagoon waste stabilization is estimated to take 6 months for planning and about 8 months to perform the decommissioning work.

Decommissioning cost, in 1981 dollars, is estimated to be \$5.91 million for DECON. For passive SAFSTOR, preparing the facility for safe storage is estimated to cost \$0.88 million, the annual maintenance and surveillance cost is estimated to be about \$0.095 million, and deferred decontamination is estimated to cost about \$6.50 million. Therefore, passive SAFSTOR for 10 years is estimated to cost \$8.33 million in nondiscounted 1981 dollars. DECON with lagoon waste

stabilization is estimated to cost about \$4.59 million, with an annual cost of \$0.011 million for long-term care. All of these estimates include a 25% contingency. Waste management costs for DECON, including the net cost of disposal of the solvent extraction lagoon wastes by shipping those wastes to a uranium mill for recovery of residual uranium, comprise about 38% of the total decommissioning cost. Disposal of lagoon waste at a commercial low-level waste burial ground is estimated to add \$10.01 million to decommissioning costs.

Safety analyses indicate that radiological and nonradiological safety impacts from decommissioning activities should be small. The 50-year committed dose equivalent to members of the public from airborne releases during normal decommissioning activities is estimated to be about 4.0 man-rem.

Radiation doses to the public from accidents are found to be very low for all phases of decommissioning. Occupational radiation doses from normal decommissioning operations (excluding transport operations) are estimated to be about 79 man-rem for DECON and about 80 man-rem for passive SAFSTOR with 10 years of safe storage. Doses from DECON with lagoon waste stabilization are about the same as for DECON except there is less dose resulting from transportation of radioactive waste. The number of fatalities and serious lost-time injuries not related to radiation is found to be very small for all decommissioning alternatives.

Comparison of the cost estimates shows that DECON with lagoon waste stabilization is the least expensive method. However, this alternative does not allow unrestricted release of the site. The cumulative cost of maintenance and surveillance and the higher cost of deferred decontamination makes passive SAFSTOR more expensive than DECON.

Several methods to assure that the licensee has adequate funds for decommissioning are considered. The methods investigated (all based on expected decommissioning costs) range from a single payment when plant operations begin, to payments into a sinking fund during the normal plant operating period, to a single payment when normal plant operations cease and decommissioning begins.

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1.D INTRODUCTION

The purpose of this study is to provide information on the technology, safety, and costs of decommissioning uranium hexafluoride conversion (UF_6) plants. This information is intended for use as background data for the development of regulations and regulatory guides pertaining to decommissioning of UF_6 plants and for use by plant licensees in planning for decommissioning of their facilities. Decommissioning techniques are reviewed and conceptually applied to a reference facility.

Decommissioning of a nuclear facility can be defined as the measures taken at the end of the facility's operating lifetime to ensure the continued protection of the public from any residual radioactivity and other potential safety concerns associated with the retired facility. A spectrum of decommissioning alternatives, all resulting in unrestricted release, are possible for such a facility. For this study, two specific alternatives are examined: DECON and passive SAFSTOR.

As used in this study, these decommissioning alternatives are defined as follows:

- DECON - Radioactive materials are removed and the facility is decontaminated and disassembled immediately following final shutdown. Upon completion, the property is released for unrestricted use.
- Passive SAFSTOR - Radioactive materials and contaminated areas are secured, and the structures and support systems are deactivated for a period of time ending in deferred decontamination. Until deferred decontamination is finished, the facility remains under a modified nuclear license. Decontamination is deferred for reasons specified by the facility owner, with NRC approval. Upon completion of decontamination, the property is released for unrestricted use.

The reference plant selected for this conceptual analysis is defined using a combination of existing facilities and no attempt is made to use a single plant as a model for all the design features.

For each of the decommissioning alternatives studied, a work plan is developed for the conceptual decommissioning of the reference UF_6 plant. These plans describe decommissioning methods, technology, and scheduling, from the planning phase through disposal of material and completion of decommissioning. From these plans, estimates are developed of manpower, major equipment and material needs, material disposal requirements, and the resultant costs. The primary guidelines for the development of these plans are:

- to ensure adequate public and occupational safety while utilizing cost-effective decommissioning practices
- to use only current, proven decommissioning techniques.

To accomplish the decommissioning options, a number of variations in the work plans and techniques described in this study is possible. However, the methods postulated in this study appear to be representative of activities expected to be used for decommissioning a UF_6 plant, and are believed to reflect an appropriate balance of safety and cost.

The safety aspects of performing the decommissioning activities are assessed, as they affect both the general public and the decommissioning workers. Safety and cost issues are evaluated for the safe storage period of passive SAFSTOR, wherein periodic surveillance and maintenance at the site will be required to assure the continued protection of the public from the radioactive materials remaining in the facility.

Safeguards and accountability for handling fissile materials, quality assurance needs, and methods for assuring decommissioning finances are examined. Relative environmental and societal advantages and disadvantages are compared, where possible, for the decommissioning alternatives studied. Suggested criteria are developed for permissible contamination levels for unrestricted facility and site reuse, based on the potential for radiation exposure to the public.

Many aspects of decommissioning (e.g., plans, methods, safety, and costs) may be sensitive to variations in facility location, specific facility shutdown conditions, and residual contamination levels in the plant. The bases

and assumptions used in this study must be carefully examined before the results can be applied to a different facility and site.

The results of this study are presented in sections and appendices. The sections summarize the key information developed and contain a summary as well as general background information (i.e., past experience in decommissioning selected types of facilities, decommissioning alternative definitions, study approach, applicable regulations and safeguards considerations, plant and site descriptions, and an overview of the suggested methodology used to develop acceptable residual contamination levels). Decommissioning techniques are described, and cost and safety analyses for each of the decommissioning alternatives are presented. Also included is a discussion of basic methods for assuring financial capability for decommissioning, and a glossary of terms used in the report. The appendices contain the supporting data, methodology, and analyses that support the results and are organized corresponding to the report sections. References are presented at the end of each major section and appendix.

2.0 SUMMARY

The results of this study to conceptually decommission a reference uranium hexafluoride conversion (UF_6) plant are summarized in this section. The purpose of the study is to identify the technology available and to evaluate the safety and costs related to decommissioning such a facility. The study is intended to provide background information for the development of regulations and regulatory guides pertaining to decommissioning of UF_6 plants and for use by plant licensees in planning for decommissioning of their facilities.

The reference plant used as the basis for this study is assumed to have processed 10,000 metric tons per year of natural uranium. The basis for the reference plant comes from a combination of existing and retired facilities and no attempt is made to use a single existing plant for the reference design. For this study, the facility is assumed to be located on a reference site having characteristics typical of midwestern or southeastern areas. Decommissioning plans, procedures, and schedules are developed for all plant areas that contain radioactive materials and for the inseparable adjacent areas that contain no radioactivity.

Two basic decommissioning alternatives^(a) are considered in detail for the UF_6 plant and site: DECON, and passive SAFSTOR. Waste generated by the reference plant is stored onsite in waste lagoons during plant operation. Potential decommissioning methods for these waste lagoons are: 1) onsite burial including site stabilization followed by long-term care, and 2) removal of the waste to an authorized disposal site. Since onsite burial of the waste lagoons does not result in unrestricted release of the site (as required by DECON and SAFSTOR), another decommissioning method is also presented, DECON of the plant with stabilization followed by long-term care for the waste lagoons. Costs and safety impacts are estimated for these alternatives, and comparisons of overall costs and potential risks are made. Methodology developed for previous decommissioning studies is modified and applied to determine example acceptable contamination levels for the reference facility and site.

(a) See Section 4 for descriptions of these alternatives.

Some of the key bases for the study are:

- Decommissioning plans are selected on the basis of providing good public and occupational safety in a cost-effective manner.
- Decommissioning operations are evaluated assuming efficient performance of the work.
- Current decommissioning technology and techniques are used.
- Expected contamination levels within the facility/site at the time of plant shutdown are based on known typical housekeeping practices during plant operation. The residual radioactive material is assumed to have accumulated at the rate of 1/40 per year for the assumed 40-year plant life.
- A radiation dose of 50 mrem/year to the maximum-exposed individual is used as the basis for the determination of suggested levels of radioactivity that can remain on the site and the facility when the property is released for unrestricted use.

The results obtained in this study are specific to the above key bases and to the other bases and assumptions used in this study. Use of other conditions, bases, and assumptions (e.g., contamination levels) may change the results significantly.

2.1 REVIEW OF DECOMMISSIONING EXPERIENCE

A review of information on UF_6 facilities in the United States shows that although several UF_6 plants have been shut down, none have been completely decommissioned. A proposed plan for decommissioning a U.S. Department of Energy UF_6 plant is used as an example of potential decommissioning experience. Two low-enriched uranium fuel fabrication facilities have been decommissioned, which provides some related experience. The review shows that: 1) experience exists in government and private organizations regarding methods and equipment for performing decommissioning operations of nuclear fuel cycle facilities, and 2) technology exists to successfully decommission UF_6 plants.

2.2 FEDERAL REGULATIONS AND GUIDELINES

Regulatory and federal guidelines are reviewed relative to their general application to decommissioning of UF_6 plants. The review shows that in many cases the regulations do not speak specifically to decommissioning but can be interpreted as being applicable.

Areas where our review of current regulations indicate that more specific guidance could be helpful are:

- Financial qualifications and responsibilities for decommissioning need to be clarified to better define the commitments of the facility owner for achieving the final decommissioned status of the property. Specific definitions need to be established as to what are acceptable methods for assuring the availability of funds at the time of decommissioning.
- Some centralization, or a central indexing, of regulations pertaining to decommissioning in the Code of Federal Regulations would be very helpful.
- Existing guidance on what levels of residual radioactivity are acceptable on materials, structures, and sites for unrestricted use is somewhat fragmentary and does not have a common identifiable basis. The suggested methodology demonstrated in this study could form that basis, predicated on a decision by regulatory agencies as to what constitutes an acceptable annual radiation dose to the maximum-exposed individual from unrestricted use of decommissioned property.

Both site stabilization and the removal of the contaminated material are considered in this study as alternatives for the decommissioning of the waste lagoons. The NRC has not yet determined whether site stabilization would be acceptable; its acceptability would have to be considered on a case-by-case basis. Legislation and information pertaining to uranium mill tailings piles contained in the Uranium Mill Tailings Control Act of 1978,⁽¹⁾ in the Generic Environmental Impact Statement on Uranium Milling,⁽²⁾ and in new regulations on uranium mill tailings licensing⁽³⁾ can be used for guidance. However, there are differences between uranium mill tailings and the reference plant wastes and other considerations will be involved.

2.3 APPROACHES TO FINANCING DECOMMISSIONING

Three general approaches to financing future decommissioning costs are identified: 1) creation of a sinking fund by annual payments during the operating lifetime of the facility, 2) an initial payment into a trust fund at the time of facility startup, and 3) purchase of a surety bond or an equivalent method of guaranteeing funds for decommissioning. A set of five criteria is identified that may be helpful in evaluating the desirability of each of these financing approaches. These criteria are: 1) the extent to which decommissioning is financially assured, 2) the present value cost of each approach, 3) the extent to which the beneficiaries of the operation of the facility pay for its decommissioning, 4) the extent to which the approach facilitates the consideration of decommissioning costs when making selections between alternative power generation systems, and 5) the ease with which the approach can be administered.

2.4 SUGGESTED METHODOLOGY FOR DETERMINING ACCEPTABLE RESIDUAL RADIOACTIVE AND CHEMICAL CONTAMINATION LEVELS FOR A UF₆ PLANT

Methodology presented in this report is used to develop numerical values for acceptable residual radioactive and chemical contamination levels for decommissioned UF₆ facilities and sites. The suggested methodology for radioactive levels is based on maximum annual doses to any member of the public from all probable radiation exposure pathways resulting from unrestricted use of the reference facility or site.

Numerical dose limits for unrestricted use of decommissioned facilities and sites by members of the public are currently being investigated by the Nuclear Regulatory Commission and the Environmental Protection Agency. For this study, it is assumed that the limit that will apply is an annual dose of 10 mrem. The use of 10 mrem in this study is consistent with the current NRC recommendations for an annual dose range to be used as guidance in decommissioned nuclear facilities. Example calculations for a maximum annual dose of 10 mrem are summarized in Table 2.4-1. These numbers are based on a specific radionuclide mixture expected to be present in the reference UF₆ facility and in the site soil, resulting from normal production operations and associated

TABLE 2.4-1. Summary of Example Acceptable Residual Radioactive Contamination Levels for Unrestricted Release of the Reference UF₆ Conversion Plant and Site

	Translocation Class	Critical Organ	Acceptable Residual Contamination Levels for an Annual Dose of 10 mrem ^(a)		
			Surface Contamination	Soil Contamination Mixed To:	
			($\mu\text{Ci}/\text{m}^2$)	10 mm (pCi/g)	150 mm (pCi/g)
Facility	W	Lungs	0.04	-- ^(b)	--
	Y	Lungs	0.009	--	--
Site	W	Bone	2.6	170	13
	Y	Lungs	2.8	160	6.8

(a) Based on the natural uranium mixtures shown in Table 7.4-1 for the facility and Table 7.1-1 for the site.

(b) A dash indicates that soil contamination levels are not calculated for surface contamination inside the facility.

atmospheric releases. For the site, acceptable residual contamination levels are determined for various times between plant shutdown and final decommissioning. The principal contributors to the calculated annual dose are found to be ²³⁴U, ²³⁵U, and ²³⁸U.

The methodology developed to determine chemical contamination levels is based on the radiological methodology. Acceptable residual chemical contamination levels are determined for inhalation and ingestion pathways. Inhalation exposure level limits are based on 0.01 of the threshold limit value, and ingestion limits are based on exposure not exceeding EPA drinking water standards. These levels provide a safety factor that accounts for exposure to the most susceptible individuals.

Since dose limits for decommissioned facilities and sites are promulgated by federal agencies, corresponding acceptable residual contamination levels can be derived using the suggested methodology developed in this study for conditions specific to a particular facility. The examples of acceptable contamination levels derived in this study are specific to the facility and site conditions assumed to exist at the reference UF₆ plant.

2.5 FACILITY CHARACTERISTICS

The plant is assumed to be operated for 40 years at a production rate of 10,000 metric tons of natural uranium per year. The feed to the plant is natural uranium U_3O_8 (yellowcake). Yellowcake concentrate extracted from uranium ore is converted to the volatile compound uranium hexafluoride (UF_6) for enrichment by the gaseous diffusion process. Two processes are in commercial use at this time which differ primarily in the method of purification of the product. The purification step is added either at the beginning using a solvent extraction process or at the end by a fractional distillation of the UF_6 . The reference plant employs the wet chemical solvent extraction step at the head end of the process to prepare a high-purity uranium feed to the reduction, hydrofluorination, and fluorination steps. The effluents consist of impurities contained as dissolved solids in the solvent extraction waste stream. The second waste stream, fluoride waste, is generated primarily by the hydrofluorination system. Solvent extraction and fluoride wastes are entrapped in waste lagoons onsite.

2.6 ESTIMATED RADIONUCLIDE INVENTORIES

Estimates are made of the amount of residual radioactivity within the plant (after final operational flushing and chemical decontamination) and on the plant site from 40 years of normal operation. Numerous activities could occur during the operational phase of the facility that could significantly affect radionuclide inventories. The inventories used in this study are presented in Table 7.4-2 of Section 7 and are based on engineering judgment that considers the characteristics of the reference facility. After operational inventory cleanout, the total uranium inventory in the main plant building is estimated to be about 12,400 kg. Chemical decontamination activities during decommissioning reduce this inventory to about 5,700 kg of uranium.

2.7 DECOMMISSIONING METHODS

A plan and a set of procedures are developed for each of the two basic alternatives studied for decommissioning the reference UF_6 plant. Decommissioning is

assumed to start after termination of production operations. Termination includes a process inventory cleanout and audit similar to that done periodically between operating campaigns for material segregation and accountability.

The first decommissioning phase for each alternative is termed "planning and preparation." This phase takes place during the last year of normal plant operation. During this phase, the decommissioning staff is assembled; a decommissioning plan and procedures are prepared; safety analysis reports and an environmental impact evaluation are prepared; an application for an amended license is prepared; a quality assurance program is established; health and safety requirements are developed; and bulk quantities of unneeded process chemicals, radioactive materials, and nonessential uncontaminated equipment are removed.

In general, decommissioning work is assumed to be done on the basis of 5 days per week with one shift of workers. Certain operations (such as extraction waste processing and plant security) are carried out on a 3-shift-day, 7-day-week basis.

2.7.1 DECON Procedures

After about 6 months of planning and preparation, DECON activities are initiated. These activities are generally divided into four phases:

- physical and chemical decontamination of equipment and facilities
- removal of equipment and facilities
- radioactive materials handling, packaging, and shipping
- final cleaning and survey.

These phases can proceed simultaneously in different parts of the facility. Approximately 8 months are needed to complete all phases and release the site for unrestricted use.

Chemical decontamination involves flushing of internal surfaces of process piping and equipment. Physical decontamination involves disassembly of equipment and enclosures and removal of the contaminated materials. Physical decontamination also involves removal of contaminated portions of structural and site materials. These contaminated materials are packaged and transported offsite

as waste, or they are processed through the plant radwaste and incinerator facilities for recovery. Upon completion of dismantlement, decontamination, shipping, and final cleaning and survey, the facility can be released for unrestricted use.

2.7.2 Passive SAFSTOR Procedures

After about 6 months of planning and preparation, active decommissioning efforts (preparations for safe storage) are divided into four phases:

- waste treatment facilities stabilization
- equipment deactivation
- isolation of contaminated areas
- final preparations for safe storage.

Many of the decommissioning activities associated with preparations for safe storage can proceed simultaneously. It is estimated that approximately 3 months are required to place the plant and site in safe storage.

Decontamination efforts for passive SAFSTOR are similar to those performed for DECON, but are performed to a lesser extent. Also involved are deactivation and isolation of contaminated areas, sealing of contamination by adding durable seals or covering with paint, and installation of improved alarm and protection systems for fire, intrusion, or malfunctioning equipment.

Activities during the safe storage period include routine inspection, corrective and preventive maintenance on safety systems, environmental surveillance, and prevention of unauthorized intrusion by man.

Safe storage must be terminated eventually by deferred decontamination. Activities are generally similar to those for DECON, with allowances for the prior decontamination efforts and for retraining of new decommissioning staff. An estimated 14 months are needed to decontaminate the facility at the conclusion of the period of safe storage, including 6 months for planning and preparation and 8 months for active decommissioning.

2.7.3 Site Stabilization Procedures

Site stabilization is assumed to include the following procedures. The lagoon waste is covered with a 50-mm-thick layer of asphalt. This asphalt

layer is then covered with 1 m of soil. The soil is mounded slightly at the center to allow water to drain from the soil cover and to prevent the accumulation of runoff from rainfall or snow melt. After compaction and contouring of the soil cover, the area is seeded with grass.

2.8 COSTS

The estimated costs in 1981 dollars for the decommissioning alternatives in this study are summarized in Table 2.8-1. These cost estimates include a 25% contingency. For DECON, the cost is estimated to be \$5.91 million. For passive SAFSTOR, placing the plant in safe storage is estimated to cost \$0.88 million; the annual cost of maintaining the plant in passive safe storage is estimated to be about \$0.095 million; and deferred decontamination of the plant is estimated to cost \$6.50 million. Therefore, passive SAFSTOR with final decontamination after 10 years is estimated to cost about \$8.33 million. The cost of DECON with stabilization of the lagoon wastes is estimated to be \$4.59 million, with an annual cost of \$11,000 for long-term care. All costs are in non-discounted 1981 dollars. DECON with stabilization of the lagoon wastes is

TABLE 2.8-1. Summary of Estimated Costs for Decommissioning the Reference UF₆ Plant (Millions of 1981 Dollars)^(a)

Item	Millions of 1981 Dollars		
	DECON	Passive SAFSTOR with Deferred Decontamination After 10 Years	DECON With Site Stabilization of Lagoon Wastes
Initial Decommissioning ^(b)	5.91	0.88	4.59 ^(c)
Safe Storage	-- ^(d)	0.95	--
Deferred Decontamination ^(b)	--	6.50	--
Total Costs	5.91	8.33	4.59

(a) Cost estimates include 25% contingencies.

(b) Costs are based on five shifts/week (single shift) for most of the decommissioning. Decommissioning on a two-shift/day basis would reduce time requirements but costs would be about the same.

(c) Plus \$11,000 per year for long-term care. Amended license must be retained.

(d) Dash indicates cost item not applicable to that decommissioning alternative.

the least expensive option. However, the NRC has not yet determined the acceptability of this alternative. This analysis of decommissioning costs indicates an economic disincentive to defer decontamination, primarily because of the cost of safe storage. Deferred decontamination costs more than DECON because of increased labor costs for the following items: 1) removal of seals and barricades erected for safe storage, 2) replacement and testing of ventilation filters, and 3) training of the decommissioning staff.

The breakdown of costs by major cost element for DECON and SAFSTOR is given in Table 2.8-2. Labor costs are about 40% of total cost. Thus, there is considerable incentive to develop plans or techniques that could reduce labor costs. The deferral of decontamination requires additional costs to modify facilities, to reinstitute a trained decommissioning organization, and to provide an analysis of the planned final decontamination to obtain the dismantlement order. Also, passive SAFSTOR costs increase with longer storage time. Other costs of deferred decontamination are about the same as for DECON.

Cost of management of the wastes from DECON amounts to about 38% of the total cost. Of the waste management cost, transportation accounts for about 20% and disposal accounts for about 50%.

TABLE 2.8-2. Decommissioning Cost Distribution of the Reference UF_6 Plant (millions of 1981 dollars)^(a)

Item	Millions of 1981 Dollars	
	DECON	Passive SAFSTOR with Deferred Decontamination After 10 Years ^(b)
Labor	2.56	3.86
Materials	0.19	0.45
Waste Management	2.23	2.37
Subcontracts	--	0.27
Utilities, Taxes	0.93	1.38
Totals	5.91	8.33

(a) Includes 25% contingency.

(b) Includes the costs of safe storage for the years before decontamination.

2.9 SAFETY

Generally conservative estimates are made of the potential safety impacts on the public and on the workers from decommissioning the reference UF_6 plant. Events are analyzed relative to potential consequences and approximate frequency of occurrence. Radiation exposures from normal operations and potential accidents are investigated for immediate and deferred decommissioning activities, storage of partly decommissioned facilities, and transportation of radioactive materials. The results are summarized in Table 2.9-1.

The 50-year committed dose equivalent to the populace located within 80 km of the facility from airborne releases resulting from DECON activities is conservatively estimated to be about 4 man-rem to the lungs. This radiation dose is a very small fraction of the dose received by the affected population from naturally occurring radiation. Radiation doses to members of the public during the period of passive safe storage are essentially negligible. Doses from DECON with stabilization of lagoon wastes are the same as for the DECON alternative. All of the postulated radiation doses are low, primarily because of greatly reduced radionuclide inventories during most of the decommissioning operations and the utilization of efficient process and ventilation filtration systems.

The estimated occupational radiation dose for DECON is 79 man-rem. The doses for passive SAFSTOR are 0.18 man-rem for preparations for safe storage, 0.087 man-rem for each year of safe storage, 62 man-rem for deferred decontamination, and 17 man-rem for transportation, for a total of 80 man-rem. Because of the long-lived radionuclides, deferral of decontamination does not reduce the occupational dose. The occupational dose for DECON with stabilization of lagoon wastes is 66 man-rem.

Potential radiation doses to members of the public from accidents are generally found to be quite low. The major accident postulated to result in the largest atmospheric release of radioactivity is an oxyacetylene explosion during the removal of piping and equipment in the ADU slurry receiving and storage area of the plant. This accident is estimated to release 1.8×10^{-3} Ci of uranium.

TABLE 2.9-1. Summary of Safety Analysis - Decommissioning of Reference UF₆ Plant

Type of Safety Concern	Source of Safety Concern	Units	DECON	Passive SAFSTOR with Deferred Decontamination After 10 Years ^(a)	DECON with Site Stabilization
<u>Public Safety^(b)</u>					
Population Dose Radiation Dose	Decommissioning Operations	man-rem ^(c)	4.0	4.0	4.0
	Transportation	man-rem	1.7	1.7	0.35
	Safe Storage	man-rem ^(c)	N/A ^(d)	0.0003	N/A
	Totals		5.7	5.7	4.35
<u>Occupational Safety</u>					
Serious Lost-Time Injuries	Decommissioning Operations	Injuries	0.25	0.27	0.25
	Transportation	Injuries	1.5	1.5	1.5
	Safe Storage	Injuries	N/A	0.077	N/A
	Totals		1.8	1.8	1.8
Fatalities	Decommissioning Operations	Fatalities	0.0016	0.0017	0.0016
	Transportation	Fatalities	0.09	0.09	0.09
	Safe Storage	Fatalities	N/A	0.0006	N/A
	Totals		0.092	0.092	0.092
Radiation	Decommissioning Operations	man-rem	62	62	62
	Transportation	man-rem	17	17	3.6
	Safe Storage	man-rem	N/A	0.87	N/A
	Totals		79	80	66

(a) Time after reference facility final shutdown; includes 1 year of preparations for safe storage.

(b) Radiation doses from postulated accidents are not included. They are given in Section 11. Both public and occupational doses include the waste exhumation, packaging, and offsite disposal option for the extraction pond, fluoride lagoons, and other onsite wastes.

(c) 50-year committed dose equivalent to the lung.

(d) Not applicable.

Chemical pollutants that could be released during decommissioning activities are found to come from residuals from plant operations and from decontamination chemicals. Chemical releases during decommissioning are examined and the quantities released are found to have no significant effect on the public. Occupational exposure to toxic chemicals is assumed to be limited by conventional industrial contamination control techniques.

2.10 WASTE MANAGEMENT

Radioactive wastes generated during the decommissioning of a UF_6 plant are packaged and shipped to a licensed low-level waste burial ground. There are no high-level or TRU wastes present in the reference plant. The radioactive wastes generated include: process and facility equipment and components, pipe, concrete rubble, filters, trash, decontamination liquids, miscellaneous waste buried onsite, fluorination tower ash and sludge and material from the waste lagoons.

About 1260 m^3 of the radioactive waste from the main process building and the solvent extraction building is assumed to be packaged and shipped to low-level waste burial. About 1870 m^3 of materials and equipment from the buildings and site are decontaminated and excessed or sent to commercial waste burial.

Sludges from the solvent extraction and fluoride lagoons must be disposed of for UF_6 plant decommissioning. Several alternative methods of disposal for the lagoon wastes are available. These are: 1) onsite burial including site stabilization followed by long-term care (similar to SAFSTOR), 2) removal of waste to a uranium mill for recovery of residual uranium, 3) removal of waste to an LLW burial ground, and 4) removal of waste to a chemical or commercial waste dump.

The solvent extraction waste ($21,600\text{ m}^3$) sludge has a uranium concentration of about 0.28 wt% uranium, which puts it at levels slightly higher than the levels in uranium mill tailings. For the DECON alternative, the extraction wastes must be removed from the site to allow unrestricted release.

The NRC has not yet determined whether stabilization of the lagoon wastes would be acceptable; its acceptability would have to be considered on a case-by-case basis. Information about the technology and costs of stabilizing uranium mill tailings and burial grounds that is contained in the uranium milling EIS⁽²⁾ and the burial ground decommissioning study⁽⁴⁾ can be used as guides in developing decommissioning requirements for waste stored onsite during operation.

For the reference plant, the extraction waste can be processed by a milling contractor to recover the residual uranium. The uranium concentration

in the extraction waste for the reference plant is about the level of uranium concentrates in ore being currently mined.

The decision to recover the uranium would be based on an economic study to determine whether the cost of shipping and recovery less the value of the recovered uranium would be less than the cost of burial. Based on current prices for uranium and the inflationary trends in yellowcake cost, it is expected that the economic value of the recovered uranium would make it cost effective to process the extraction waste.

The fluoride waste does not contain sufficient uranium to warrant recovery and that option is not included in this analysis.

If the extraction waste is not processed to remove the residual uranium, it would be disposed of at a low-level waste burial ground at the time of decommissioning. If fluoride waste cannot be stored onsite it would be shipped to LLW burial.

An alternative to low-level waste disposal of the extraction waste would be a reduction of the residual radioactivity levels by incorporation of nonradioactive fill material and storage onsite or shipment to a chemical or commercial waste dump. The concentration of uranium in the resultant waste would need to be less than a value defined by federal regulations as acceptable for unrestricted release. There is presently no clear definition of what constitutes acceptable levels of uranium in bulk waste.

2.11 STUDY CONCLUSIONS

Decommissioning of a UF_6 plant is technically feasible with current technology. Decommissioning can be done with virtually no impact on the safety of the general public. Further development of some techniques (such as decontamination and waste volume reduction) could lead to reductions in costs.

A comparison of the decommissioning alternatives for the various parameters used in this study is given in Table 2.11-1. The main parameters considered are the costs, the potential radiation doses, and the impacts of the DECON and passive SAFSTOR alternatives on staffing requirements and on space requirements at waste disposal facilities.

TABLE 2.11-1. Comparison of Decommissioning Alternatives for the Reference
 UF_6 Plant

Parameter	DECON	Passive SAFSTOR with Deferred Decontamination After 10 Years	DECON with Stabilization of Lagoon Wastes
Decommissioning Cost ^(a) (millions of 1981 dollars)	5.91	8.33	4.59
Occupational Radiation Dose (man-rem)	79	80	66
Staff Required (man-years)	60	72 ^(b)	48
Waste Volume (m^3) (to LLW disposal site)	1280	1280	1280
Final Site Status	Unrestricted	Unrestricted	Restricted with Long- Term Care

(a) Estimates include a 25% contingency.

(b) Includes 17.4 man-years for preparations for safe storage and 9.1 man-years for safe storage.

DECON with stabilization of the lagoon wastes is the least expensive option. However, the NRC has not yet approved this disposal method. It does not allow unrestricted release of the site, and decommissioning methods that convert contaminated sites into waste repositories are not compatible with current national policy and regulatory guidelines for radioactive waste disposal.

Removal of the waste for disposal is more costly than stabilization. However, if the waste is left at the site, the radioactivity would remain for hundreds of thousands of years. Long-term care following stabilization would provide a period for assessing the effectiveness of stabilization procedures.

DECON costs are considerably lower than the cost of passive SAFSTOR, mainly because of the cost of safe storage. Radiation doses to workers are slightly higher for the passive SAFSTOR alternative because of the doses received during safe storage. The total radiation doses (received mostly by the decommissioning workers) do not decrease with time for deferral of decontamination. The waste volume is essentially the same for all decommissioning alternatives.

The decontamination of UF_6 facilities is highly labor intensive. Thus, labor is a major component of the total decommissioning cost. Facility and equipment designs and decontamination systems and techniques that minimize labor could help reduce overall decommissioning costs.

The conditions in effect at a specific facility at the time it is decommissioned, including sociological aspects, may dictate the choice of the decommissioning alternative to be used. Therefore, the results and conclusions in this report should be used only in the context of the reference site and facility studied and the key bases and assumptions used.

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3.0 REVIEW OF DECOMMISSIONING EXPERIENCE

Information specific to the decommissioning of uranium hexafluoride (UF_6) conversion plants is not available in the open literature since no plants have been decommissioned to date. Information regarding decontamination experience at UF_6 facilities that have been shut down is also very limited. A proposed plan for the decommissioning of the U.S. Department of Energy (DOE) Weldon Spring Chemical Plant that has been closed for some time is included in this section as an example of potential decommissioning activities. Information on the decommissioning of other types of uranium processing facilities is also included since some of the decommissioning problems encountered may be similar to those anticipated for decommissioning of commercial UF_6 plants.

3.1 STATUS OF UF_6 PLANTS IN THE UNITED STATES

Table 3.1-1 gives information on the location and operating status of UF_6 facilities in the U.S. Although several UF_6 plants have ceased operations, none have been decommissioned. The DOE has terminated UF_6 conversion at all of its plants listed in Table 3.1-1 except for a small hexafluoride plant for converting returned and reclaimed uranium compounds to feed material for the Cascade Enrichment Plant.

TABLE 3.1-1. Information on UF_6 Conversion Plants in the U.S.

<u>Plant Operator</u>	<u>Plant Location</u>	<u>Present Status</u>
Allied Chemical	Metropolis, IL	Operating
Babcock and Wilcox ^(a)	Apollo, PA	Shut Down
Kerr-McGee	Sequoyah County, OK	Operating
Mallinckrodt Chemical - DOE	Weldon Spring, MI	Shut Down
National Lead of Ohio - DOE	Cincinnati, OH	Shut Down
Portsmouth - DOE ^(b)	Portsmouth, OH	Shut Down
Union Carbide - DOE	Oak Ridge, TN	Shut Down
Union Carbide - DOE	Peducah, KY	Shut Down

(a) Formerly Nuclear Materials and Equipment Corp. (NUMEC.)

(b) The large hexafluoride conversion plant was put into safe storage in the 1961-62 period. It has since been converted to another use. There is currently a small hexafluoride plant for converting returned and reclaimed uranium compounds to feed for the Cascade Enrichment Plant.

3.2 DECOMMISSIONING PLAN FOR WELDON SPRING CHEMICAL PLANT

The Weldon Spring Chemical Plant was operated from 1957 to 1966 by the Mallinckrodt Chemical Works, and in 1966 it was shut down. During operation, the buildings, equipment, and immediate terrain became contaminated with uranium, thorium, and their decay products. In 1967 the Army began a major effort to decontaminate and dispose of three process buildings for other use. The cleanup was not successful and the project was cancelled in 1969. The site entered care and custody status at that time.

A program plan for decontamination of the Weldon Spring Chemical Plant was prepared in 1979. The technical approach was to remove all equipment and buildings from the site and then decontaminate all land so that the approximately 80-hectare (200-acre) site could be released for unrestricted use. Figure 3.2-1 shows the Weldon Spring Program Plan for decontamination and disposition. As shown in the figure, the plan consists of five major tasks performed in sequence by similarity of skills required and by degree of contamination involved.

This approach allows the operation to proceed in stages, staffing as necessary to accommodate funding or resolution of problems such as the potential for recontamination.

FIGURE 3.2-1. Time-Phased Program by Task

The first task involves site preparation for performance of the program and removal of clean facilities. In the second task, all slightly contaminated facilities are removed. In the third task, all highly contaminated (above-ground) facilities are decontaminated and removed. Once this is done, all below-ground and support facilities are removed. In the final task, contaminated soil is removed and a final site survey is performed, and site restoration compatible with adjacent properties is accomplished.

3.3 DECOMMISSIONING EXPERIENCE AT OTHER URANIUM PROCESSING FACILITIES

Other uranium processing facilities, such as low-enriched uranium fuel fabrication (U-Fab) plants, have been decommissioned that might offer experiences valuable to UF_6 plant decommissioning.

Among the low-level enriched U-Fab plants, two facilities that have been shut down are examples of decommissioning experience. A Kerr-McGee plant at Crescent, Oklahoma, has been partly decommissioned. The plant is still intact, and the waste ponds have been cleaned up. Contaminated waste from the plant was loaded into drums and shipped to a low-level waste burial ground.

The most complete experience with decommissioning a low-enriched U-Fab plant has been the General Electric U-Fab plant in San Jose, California. At shutdown, the area was cleaned to administrative control levels not exceeding 1000 dpm/100 cm^2 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 up by removing the berms and wet-grinding any hot spots. The decommissioning effort was more extensive than would have normally been necessary because of an operating accident that had released a relatively large amount of UF_6 inside the plant. This accident contaminated not only all of the building and fixture surfaces in the production areas but also some otherwise clean areas.

3.4 LESSONS LEARNED FROM EXISTING DECOMMISSIONING EXPERIENCE

Technology for decontamination and decommissioning exists and has been applied to a variety of nuclear installations. Because of the uniqueness of each facility, no two have had identical decommissioning problems or conditions. However, the basic approach to any mode of decommissioning remains virtually unchanged (i.e., the gathering of staff manpower and a period of planning and preparation, followed by chemical decontamination and mechanical removal operations). The decommissioning plan and execution varies primarily with plant design and with the inherent refinements potentially available or needed for a given facility. Areas that could use improvements in technology are remote-handling equipment, disassembly techniques, decontamination techniques, and waste volume reduction.

From the standpoint of decontamination, all walls should be seamless and have a smooth durable surface to aid in flushing and cleaning. Separation of process areas into compartments allows for more effective control of radioactive migration. Sealed-off access areas behind processing pipes and highly contaminated equipment provide an effective means of controlling radioactive contamination, while also providing a suitable work area for personnel. Building and hood fluid services, located either beneath the floor level or in some area away from work areas, tend to minimize the hazardous effects of pipe leaks. Location of building service systems (such as vacuum systems, corrosive vapor removal systems, and exhaust systems) in isolated areas allows maintenance work to be performed with little interruption of ongoing processing operations. These are some of the most obvious improvements in design that could enhance decommissioning. A more comprehensive listing of design considerations favorable to decommissioning of the reference facility is given in Section 13. New techniques, as well as improvements in current decommissioning techniques, can be expected to occur. These improvements, in turn, will directly impact future decommissioning considerations.

4.0 DECOMMISSIONING ALTERNATIVES AND STUDY APPROACH

Once a uranium hexafluoride conversion (UF_6) plant reaches the end of its useful operating life, it must be decontaminated or placed in a condition such that future risk to public safety from the facility and its site is within regulatory limits. Several decommissioning alternatives are possible to satisfy these general requirements. These alternatives range from minimal initial cleanup requiring continued surveillance and physical security followed by later more complete cleanup, to immediate complete cleanup and removal of contaminated materials resulting in unrestricted public use of and access to the facility and site. For all of the alternatives categorized, the ultimate goal is unrestricted access to the facility.

In this section, decommissioning alternatives are evaluated for a reference UF_6 conversion plant and the reasons for selecting certain alternatives are discussed. The approach of this decommissioning study is also discussed. Certain assumptions must be made in the absence of specific data, to permit general application of the results. The important overall assumptions for the study and the rationale for their selection are identified.

4.1 DECOMMISSIONING ALTERNATIVES

The general characteristics of the basic decommissioning alternatives are summarized in Table 4.1-1. Each of the alternatives as applied to the reference UF_6 conversion plant is defined and discussed in the following subsections.

4.1.1 Definition of and Rationale for DECON

DECON (immediate decontamination to unrestricted release) provides a way to meet the requirements for termination of a nuclear license in the near term, thus eliminating long-term security, maintenance, and surveillance needs and making the site available for unrestricted use within about 1 year following facility shutdown. To accomplish DECON requires that all potentially contaminated systems be disassembled and removed from the facility and transported to a regulated disposal site.

TABLE 4.1-1. Characteristics of the Various Decommissioning Alternatives

<u>Alternative</u>	<u>Facility Status</u>	<u>Facility/Site Use</u>
<u>DECON</u>	Equipment - removed if radioactive Surveillance Staff - none Security - none Environmental Monitoring - none Radioactivity - removed Surveillance - none Structures - removal optional License - terminated	Facility - Unrestricted Site - Unrestricted
<u>SAFSTOR</u>		
Custodial	Equipment - some operating Surveillance Staff - some required Security - continuous Environmental Monitoring - continuous Radioactivity - confined Surveillance - continuous Structures - intact License - amended version maintained	Facility and site are restricted to nuclear use until deferred decontamination is accomplished.
Passive	Equipment - none operating Surveillance Staff - routine periodic inspections Security - remote alarms Environmental Monitoring - routine periodic Radioactivity - immobilized/sometimes sealed Surveillance - periodic Structures - intact License - amended version maintained	All of the facility and most ^(a) of the site are restricted to nuclear use until deferred decontamination is accomplished.
Hardened	Equipment - none operating Surveillance Staff - none on site Security - temporary hardened barriers; fencing and posting; remote alarms Environmental Monitoring - infrequent Radioactivity - sealed in hardened structures Surveillance - infrequent Structures - partial removal optional License - amended version maintained	Most ^(a) of the facility and most ^(a) of the site are restricted to nuclear use until deferred decontamination is accomplished.
<u>ENTOMB</u>	Equipment - none operating Surveillance Staff - none on site Security - hardened barrier; fencing and posting Environmental Monitoring - infrequent Radioactivity - sealed in monolithic structure Surveillance - infrequent Structures - partial removal optional License - amended version maintained	Most ^(a) of the facility and some ^(a) of the site are restricted to nuclear use until the confined radioactivity has decayed to unrestricted release levels.

(a) Implies a release of part of the site or the facility for unrestricted use, while maintaining control of the licensed portion that contains radioactive materials above releasable levels.

In the DECON alternative, larger initial commitments of money are made in exchange for prompt availability of the plant site for other purposes, reuse of plant components, and elimination of continuing costs for security, maintenance, and surveillance.

4.1.2 Definition of and Rationale for SAFSTOR

SAFSTOR includes all operations needed to prepare for safe storage, surveillance and maintenance during safe storage, and complete decontamination to

unrestricted release following safe storage. The facility is placed in such a condition that risk to the public can be kept within acceptable bounds while the facility is maintained in storage, and the facility can be subsequently decontamination to unrestricted release at the end of the safe storage period.

Several subcategories of safe storage for the SAFSTOR alternative are possible. These are:

- Hardened SAFSTOR [temporary entombment^(a,b)] - A comprehensive cleanup effort is coupled with the construction of barriers around areas containing sufficient quantities of radioactivity. These barriers are of sufficient strength to make accidental intrusion impossible and deliberate intrusion extremely difficult. Surveillance requirements during safe storage are limited to detection of intrusion through the barriers and maintenance of the integrity of the structures. The primary restriction to facility and site use is that of prohibiting activities such as excavating, drilling, or any other means of breaking the barriers that confine the radioactivity, until deferred decontamination is accomplished.
- Passive SAFSTOR [mothball,^(a) protective storage^(b)] - A significant cleanup effort is performed initially, sufficient to permit deactivation of the active protective (ventilation and utility) systems during the period of safe storage. The structures are strongly secured and electronic surveillance is provided to detect accidental or deliberate intrusion. During the safe storage period, maintenance of the integrity of the structures is required. Plant use is limited to nuclear only while site use may be non-nuclear, with certain restrictions, until deferred decontamination is accomplished.
- Custodial SAFSTOR [layaway^(b)] - A minimum cleanup effort is made initially, followed by a period of safe storage with the active protection systems (principally the ventilation, utility, and fire protection systems) kept

(a) This nomenclature is used in Regulatory Guide 1.86.⁽¹⁾

(b) This nomenclature is used in NUREG-0278.⁽²⁾

in service throughout the storage period. Onsite personnel will be required for surveillance and maintenance of the operating systems. An electronic surveillance system is required to detect accidental or deliberate intrusion into the facility and the subsequent exposure to radiation or dispersal of radioactivity beyond the confines of the facility. Use of the facility and site is generally limited to nuclear activities until deferred decontamination is accomplished.

All categories of safe storage are open-ended and some positive action is required at the conclusion of the period of safe storage to release the property for unrestricted use and terminate the license for radioactive materials. Depending on the nature of the nuclear facility and its operating history, the necessary action can range from a radiation survey that shows the property to be releasable, to dismantlement and removal of residual radioactive materials. These latter actions, whatever their scale, are generically identified as deferred decontamination.

SAFSTOR is used as a means to satisfy the requirements for protection of the public while minimizing the initial commitments of time, money, occupational radiation exposure, and waste repository space. Modifications to the facility during the preparation stage are limited to those that assure the security of the buildings against intruders and to those required to assure containment of radioactive or toxic material. It is generally not intended that the facility would ever be reactivated, although reactivation is a possible option. For a UF_6 plant, there is no significant decay of residual radioactivity and personnel exposure to radiation is not reduced during reasonable periods of safe storage (a few tens of years). Thus, placing an inactive UF_6 facility in safe storage for a reasonable period of time produces no benefit in terms of reducing radiation exposure to decommissioning personnel.

The reduced initial effort (and cost) of the SAFSTOR alternative is tempered somewhat by the need for continuing surveillance and physical security to assure the protection of the public. For all SAFSTOR alternatives, electronic surveillance devices are in service full-time, with off-shift readouts monitored

in a local law enforcement office or a private security agency. These devices, which monitor for intruders, radiation-level increases, and fire detection, require periodic checks and maintenance. For custodial SAFSTOR, a small operating and security staff is required at the retired facility to provide for equipment operation, general maintenance, and plant security.

Maintenance of the facility's outer-confinement barriers and surfaces and an on-going program of environmental surveillance are also necessary for all SAFSTOR alternatives.

The duration of the period of safe storage before final decommissioning may vary, depending on the needs of the plant owner, based primarily on economic and safety trade-offs. For example, if the value of the site property for unrestricted use is large and the cost of storage is also large, there would be incentive to complete decontamination reasonably soon. On the other hand, a potential alternative use of the facility may suggest maintaining the facility in safe storage for an extended time period. Regulatory requirements and public concerns may also influence the duration of the safe storage period.

At the end of the safe storage period, several things remain to be done before the facility can be made available for unrestricted use and before the amended nuclear license can be terminated. The remaining quantities of long-lived radioactivity that exceed unrestricted release limits must be removed and contaminated equipment must be packaged and removed to a regulated disposal site. Once the remaining levels of radioactivity are less than the unrestricted release limits, the facility nuclear license can be terminated.

For a UF_6 plant there is not much difference in the amount of cleanup required for either the passive or custodial SAFSTOR alternatives. Passive SAFSTOR involves shutdown of the ventilation and utility systems, and less surveillance is required during the safe storage period than for the custodial alternative. Thus, the passive SAFSTOR alternative appears to be most applicable to short-term inactivation (e.g., 5 to 10 years) of a UF_6 plant; to a multifacility site where surveillance, security, maintenance and operating capabilities exist; or to the situation where the likelihood exists for later use of the retired facility.

Deferred decontamination, as would occur at the end of an extended period of safe storage, perhaps a few tens of years, continues decommissioning activities beyond those done during the preparations for safe storage. Decontamination activities will still be controlled by residual long-lived radioactivity in the plant. The benefits to be gained by deferred decontamination of a UF_6 plant (i.e., the possibility of re-use of the facility and the deferral of decontamination costs) will depend on the characteristics of each facility at the time of final production shutdown. These benefits must be weighed against disadvantages of deferring decontamination (i.e., costs of safe storage), value of and need for the reclaimed site, and the need to familiarize the new decommissioning staff with the facility.

4.1.3 Definition of and Rationale for ENTOMB

ENTOMB for a nuclear reactor facility requires the encasement of the radioactive materials in concrete or other structural materials sufficiently strong and structurally long-lived to assure retention of the radioactivity until it has decayed to levels that permit unrestricted use of the site. The amount and half-life of the residual radioactivity in the facility to be entombed determines the time period that the integrity of the structure must be assured and whether or not re-entry for additional decommissioning is required. ENTOMB refers to the entire process of first entombing and then continuing some surveillance to assure the integrity of the structure until the entire site is confirmed to have decayed enough to allow unrestricted release.

The Environmental Protection Agency (EPA) is developing generally applicable environmental protection criteria for management of all radioactive wastes that will impact NRC decommissioning standards and guidelines. In a background report entitled Considerations of Environmental Protection Criteria for Radioactive Waste,⁽³⁾ the EPA proposes a criterion limiting reliance on institutional controls to a finite period of time. The EPA suggests that the use of institutional control to protect the public from hazards in retired nuclear facilities should be limited to a period of 100 years at most and preferably to less than 50 years. After the allowable institutional care period is over, the site would have to meet radioactive protection levels established for release for unrestricted use.

Based on the definition of ENTOMB and the EPA-proposed criteria,⁽³⁾ it is concluded that any "permanently" entombed structure must be designed to outlast any contained radiological or chemical hazard to man, or to be designed perhaps to dilute these hazards to innocuous levels as the structure disintegrates. Unless the structure is to be re-entered later and decommissioned further, these potential chemical and radiological hazards should vanish in no more than about 100 years, in order to fulfill the bases for ENTOMB. Taking no credit for the dilution effects of entombment, these criteria and guidance virtually prohibit entombing any nuclear facility containing long-lived radionuclides or toxic chemical elements.

In addition, while it is reasonable to assume that man can design and construct high-integrity, long-lived surface structures, it is also reasonable to assume that any long-term human controls on or responsibility for that facility will ultimately disappear and that the long-lived radionuclides, chemicals or toxic elements contained therein will ultimately be dispersed into the environment. The ENTOMB alternative also results in the proliferation of decommissioned plant sites containing residual radioactivity. Therefore, ENTOMB is considered not viable for a UF_6 conversion plant.

4.1.4 Decommissioning Methods for Lagoon Waste

Decommissioning methods for lagoon waste are assumed to be the same as those for uranium mill tailings or low-level waste burial grounds, namely, stabilization of the waste lagoon plus long-term care (similar to SAFSTOR), and removal of the waste to an authorized disposal site (DECON). The NRC has not yet determined whether stabilization would be acceptable; its acceptability would have to be determined on a case-by-case basis.

Use of a site as a temporary waste repository might also be necessary because of a shortage of disposal space at authorized waste burial sites. However, decommissioning alternatives that convert contaminated sites into waste repositories are not compatible with current national policy and regulatory guidelines for radioactive waste disposal. The government is presently engaged in a program (the Formerly Utilized Sites Remedial Action Program⁽⁵⁾) to survey sites with suspected radioactive contamination and to take appropriate action to prepare these sites for unrestricted public use.

Information about the technology and costs of uranium mill tailings and burial ground decommissioning that is contained in the uranium milling EIS⁽⁶⁾ and the burial ground decommissioning study⁽⁷⁾ can be used as guides in developing decommissioning requirements for the reference plant lagoon waste. Removal of the waste to an authorized disposal site is more costly than stabilization. However, if the waste is left at the site, the radioactivity would remain for hundreds of thousands of years. Therefore, the purpose of long-term care following stabilization is to provide a period for assessing the effectiveness of stabilization procedures.

To arrive at a decision about which decommissioning alternative to use for lagoon waste management, the following factors need to be considered:

- levels of radioactivity remaining in the waste
- the physical and chemical forms of the material
- the chemical toxicity of the material
- hydrologic and geologic properties of the site
- potential uses of the site
- political and social concerns about site proliferation
- a balance between costs and safety.

4.1.5 Alternatives Selected for Study

The principal alternative selected for study is DECON (immediate decontamination), since the half-lives of the uranium isotopes are so long (^{235}U : 7×10^8 years; ^{238}U : 4.5×10^9 years) deferring decontamination for any reasonable time period (<100 years) would have no effect on the levels of contamination or the radiation dose rates.

The only likely reason to select a safe storage alternative would be if the owner wished to retain the capability to restart the plant sometime in the near future (probably within 5 years). Since passive SAFSTOR would permit restart of the facility, this alternative is also examined.

Hardened SAFSTOR or ENTOMB are not viable alternatives for decommissioning a UF_6 plant because the plant is relatively easy to decontaminate and decay of the uranium isotopes over any reasonable storage period would be negligible.

Since the initial cost of either hardened SAFSTOR or ENTOMB would be greater than that of passive SAFSTOR, and the total cost of these alternatives would exceed that of DECON, there are no incentives to use these alternatives; thus, they are not considered further.

Since stabilization of the lagoon waste on the UF_6 plant site does not result in unrestricted release of the site as do the other decommissioning methods, another method is presented, DECON of the plant with stabilization followed by long-term care for the waste lagoons.

4.2 TECHNICAL APPROACH

The initial effort in this study is to develop a plan to accomplish the study objectives. The plan is developed by a team of key personnel with expertise in the primary areas of interest. The areas of expertise include fuel fabrication plants and their operation, decommissioning techniques, chemical decontamination, chemical and radiological toxicant regulations, safety analyses (including pathways of toxic materials in the environment), operational health physics, and cost and benefit estimating and analyses. The resultant approach is shown in simplified form in Figure 4.2-1. The study is then carried out by the same staff or by staff with similar backgrounds.

FIGURE 4.2-1. Approach for Decommissioning Study

The first step in conducting the analysis is to select and characterize the reference facility in sufficient depth to perform an engineering and safety analysis of decommissioning the facility. An existing plant is selected on the basis of having characteristics typical of UF_6 plants that will be subjected to decommissioning in the foreseeable future.

The total facility is assumed to be on a conceptual generic site that is also being used in similar and related studies of other fuel cycle facilities. A detailed description of the facility is compiled that includes information such as plant equipment and material sizes, volumes, surface areas, and weights. Pre-decommissioning conditions for the plant and site are defined, including residual radioactivity levels.

Viable decommissioning alternatives (i.e., DECON and passive SAFSTOR) and site use limitations for facilities being decommissioned (i.e., restricted to nuclear use only, and unrestricted use) are selected. Related regulatory guidance is reviewed, summarized, and used as an aid and basis in the study.

Methodology is developed for defining suggested residual radioactivity levels in facilities and sites that would permit unrestricted use of decommissioned facilities, in terms of allowable radiation dose to the maximum-exposed member of the public. The variety of potential pathways through which radionuclides could reach man are considered in determining these acceptable levels. This methodology is applied to develop example acceptable levels of residual radionuclides, based on the assumed radionuclide mixtures at the plant/site.

Past decommissioning experience of facilities with characteristics related to the reference facility is reviewed. From this review, a summary of insights from these decommissioning experiences is derived and applied where applicable to this study.

Techniques for decontamination of facilities are reviewed. For both decommissioning alternatives, a work schedule and a time schedule are developed to conceptually decommission the reference facility. The techniques used are selected on the basis of engineering judgment, while maintaining a balance of safety and cost.

Safety analyses are performed for the decommissioning alternatives studied. These analyses include radiological and chemical exposures to the public and workers from normal decommissioning operations and from potential accidents. Nonradiological industrial accidents to workers are also estimated. The safety analyses utilize established data and methodology to estimate the various factors required, such as release mechanisms, dispersion pathways, and exposure modes for the released materials.

Direct costs of decommissioning are estimated, including labor, materials, equipment, packaging, transportation, waste disposal, and surveillance costs where applicable. Alternatives for financing decommissioning are examined. For both decommissioning alternatives, all of these factors are combined into an overall comparison of their safety-costs-benefits and advantages and disadvantages.

The study is summarized in this report, documenting the main study information and appendices containing supporting details.

4.3 KEY STUDY BASES

From the outset, a number of important ground rules are established to guide the emphasis of the study. These bases are derived from the primary objective of the study, which is to provide an analysis of safety, costs, and other factors involved in decommissioning a UF_6 conversion plant. The study is intended to provide background information useful to regulators, plant designers, and operators of such facilities. From these objectives, the key bases are established for all aspects of the study to assure that the overall study objectives (see Section 1) are achieved. These key bases can have major impact on the issues of safety, cost, and time for decommissioning. Many aspects of decommissioning will change with facility location, specific facility shutdown conditions, and residual contamination levels in the plant. The bases and assumptions used in this study must, therefore, be carefully examined before the results can be applied to a different facility and site.

The key study bases are:

1. The study is to yield realistic results based on 1981 cost data. This primary basis is a requisite to meeting the objectives of the study, and provides the foundation for most of the other study bases.

2. The objective of decommissioning is to assure the continued protection of public from the residual radioactivity and any other potential safety hazards in the retired facility.
3. The study is to evaluate a contemporary facility. This basis is an obvious necessity to meet the study objectives and the primary basis above. The reference facility for the study is based on current technology.
4. The study is to include an analysis of the viable decommissioning alternatives: DECON and passive SAFSTOR.
5. Only facilities expected to contain radioactive material and contiguous areas are included in this study. Decommissioning of separate nonradioactive subfacilities is to be accomplished by conventional demolition/salvage techniques and is outside the scope of this study.
6. The UF_6 facility is assumed to have operated for 40 years prior to plant shutdown and the onset of decommissioning operations. During the operation, the plant is assumed to have had a processing capacity of 10,000 metric tons per year of natural uranium.
7. Current and proven decommissioning technology and techniques are used in the study. Where developmental techniques are conceptualized, they are in an advanced state of development and believed to be ready for application in this study.
8. A single decommissioning plan is evaluated for each of the two decommissioning alternatives analyzed. Where different techniques or assumptions have significant impact on the study results, the effects of alternatives are discussed at least qualitatively.
9. Decommissioning techniques conform to the principle of keeping occupational radiation doses As Low As is Reasonably Achievable (ALARA).
10. Decommissioning plans are selected to provide public/occupational safety in a cost-effective manner.
11. All personnel assigned to decommissioning operations are assumed to be experienced radiation workers with previous experience in the operation of a UF_6 plant or other similar nuclear facilities.

12. The performance of decommissioning is assumed to be relatively trouble-free; that is, no scheduling or cost allowances are made for unforeseen events that might impede the conduct of the work. This assumption may lead to somewhat optimistic results, but is believed to be achievable with good planning and preparations.
13. It is assumed that plant process areas have been kept relatively clean during the operating period to allow for easier operational maintenance. As a result, expected contamination levels are generally modest and should be reasonably consistent with the quality of operation expected in modern commercial facilities. Any major contamination episodes are assumed to have been reasonably well cleaned up immediately following the event.
14. A final operational cleanup of the more important inventories of radionuclides is done as part of normal operations, and is not charged to decommissioning. This cleanup is assumed to be routine and similar to those done periodically between normal processing campaigns to improve equipment performance, segregate materials, and to recover materials unaccounted for. Subsequent decontamination efforts are charged to decommissioning.
15. The quantity and mixture of radioactive contamination present at plant shutdown is assumed to represent an accumulation of contamination that is fairly difficult to clean during operations. Specifically, contamination inventories are assumed to accumulate at the rate of 1/40th per year of the total accumulation, for the assumed 40 years of plant operation.
16. The isotopic inventory of input to the plant is typical of that used in the manufacture of uranium hexafluoride.
17. All materials shipped to the plant and waste shipments from the plant are assumed to be transported by truck.
18. Estimates of external radiation exposure to the public and to decommissioning workers from normal decommissioning activities are based on assumptions

believed to be realistic. Estimates of internal radiation exposure (i.e., those from internally deposited radioactive material) from normal decommissioning activities and from potential accidents are based on assumptions believed to be conservative.

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5.0 REGULATORY CONSIDERATIONS

A viable plan for decommissioning a uranium hexafluoride conversion (UF_6) plant must include consideration of applicable regulations that exist to ensure public and occupational safety. The following sections discuss these issues in general and as they apply to the reference UF_6 conversion plant. Also, areas are identified where guidance is incomplete or where application of existing guidance to decommissioning is ambiguous.

5.1 EXISTING REGULATIONS, STANDARDS, AND GUIDES

A general discussion of the background of present regulatory responsibility and the division of the responsibility among the regulatory agencies is provided in this section. Although an extensive framework of government requirements apply to decommissioning of nuclear facilities, many of the requirements that affect decommissioning do so only indirectly. Examples of requirements that apply to decommissioning but were originally directed towards other primary purposes include extensive government requirements regarding nuclear facility construction and operation, the possession of nuclear materials, and limitations of occupational radiation dose.

Regulations and guidelines in this area are dynamic and new regulations are forthcoming. The U.S. Nuclear Regulatory Commission (NRC) is considering the development of a more explicit overall plan with regard to decommissioning.⁽¹⁾ With increasing public and regulatory attention being paid to decommissioning issues, requirements can be expected that will have the purpose of establishment of rules for governing the decommissioning of nuclear facilities.

5.1.1 Federal Jurisdiction

Several federal agencies have jurisdictions that can affect the decommissioning of nuclear facilities. The principal agencies with jurisdiction are the NRC, the Environmental Protection Agency (EPA), the Department of Transportation (DOT), and the Department of Energy (DOE). This subsection briefly identifies these agencies and summarizes their regulatory jurisdiction, as identified in Table 5.1-1.

TABLE 5.1-1. Principal Federal Agencies and Statutory Authority that May Affect Decommissioning

<u>Agency</u>	<u>Statutory Authority</u>
Nuclear Regulatory Commission	<ul style="list-style-type: none"> • Atomic Energy Act of 1954 • Energy Reorganization Act of 1974
Environmental Protection Agency	<ul style="list-style-type: none"> • Reorganization Plan No. 3 of 1970 • Clean Air Act Amendments of 1977 • Safe Drinking Water Act
Department of Transportation	<ul style="list-style-type: none"> • Hazardous Materials Transportation Act
Department of Energy	<ul style="list-style-type: none"> • Energy Reorganization Act of 1974 • Department of Energy Organization Act

Pursuant to the Energy Reorganization Act of 1974, the AEC was abolished.⁽²⁾ The DOE assumed the disbanded AEC's research and development functions and its promotion of new technology activities and the NRC inherited its regulatory authority under the Atomic Energy Act of 1954.⁽³⁾ Among other things, the NRC is responsible for assurance of safety to life and property from the civilian use of nuclear material. NRC authority extends to all persons who possess, use, or transfer byproduct, source, or special nuclear materials.

The EPA is the federal government's chief environmental regulator. The EPA assumed the duties of the Federal Radiation Council under the President's Reorganization Plan No. 3 of 1970.⁽⁴⁾ The EPA has authority to regulate radioactive emissions into the air under the Clean Air Act Amendments of 1977.⁽⁵⁾ The EPA also has authority to regulate doses from radioactive discharges under the Safe Drinking Water Act.⁽⁶⁾ Under these authorities, the EPA has established maximum contaminant levels in public drinking water systems. Regulations by the EPA that establish standards for radioactive contaminant levels in drinking water are contained in 40 CFR 141.

The principal federal agencies concerned with the transportation of radioactive materials are the DOT under the Hazardous Materials Transportation Act of 1974,⁽⁷⁾ and the NRC under the Atomic Energy Act of 1954. Federal safety regulations concerning nuclear materials transportation are outlined in Reference 8.

The following federal regulations are applicable to the transport of the radioactive materials associated with a UF_6 conversion plant that processes only natural uranium:

- Title 49 Code of Federal Regulations Parts 170-199 (49 CFR 170-199) - DOT regulations governing the transport of hazardous materials.
- 10 CFR 71 - NRC regulations governing the packaging and shipment of radioactive materials.
- 14 CFR 103 - FAA regulations for shipment of radioactive materials by air.
- 47 CFR 146 and 149 - U.S. Coast Guard regulations governing the shipment of radioactive materials by water.

The DOT and the NRC regulations are the most important for shipments made during the decommissioning of nuclear facilities.

Occupational safety is also of major importance during decommissioning. Radiation protection to workers is regulated by 10 CFR Part 20. Sections 20.101 through 20.105 define the external exposure limits. The operating philosophy of ALARA (As Low As is Reasonably Achievable) applies to these exposure limits. The NRC describes this operating philosophy in Regulatory Guide 8.8, "Information Relevant to Maintaining Occupational Radiation Exposure as Low as Practicable (Nuclear Reactors)," and Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposure as Low as Reasonably Achievable." Although not specifically cited for application to decommissioning activities, the guides are intended to apply. Additional information can be found on how to comply with the ALARA concept in the NRC Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Section 12.1, "Assuring that Occupational Radiation Exposures are As Low As is Reasonably Achievable."

One of the goals of decommissioning a nuclear facility is to make the land available for other uses if desired. To release the facility and/or site for unrestricted use, the residual radioactive contamination must be at a level sufficiently low to assure public protection. Several attempts have been made to define the permissible levels of residual radioactivity. Guidance is found

in Regulatory Guide 1.86⁽⁹⁾ and 40 CFR 190⁽¹⁰⁾ and the proposed ANSI Standard N328, Control of Radioactive Surface Contamination on Materials, Equipment and Facilities to be Released for Uncontrolled Use. Other guidance⁽¹¹⁾ from the NRC for terminations of byproduct, source, and SNM licenses (similar to Regulatory Guide 1.86) contains a table of "Acceptable Surface Contamination Levels" identical to that in Regulatory Guide 1.86. Thus, the NRC, the EPA, the DOT, and the DOE are the federal agencies with the principal responsibilities affecting decommissioning. To the extent that regulations of more than one agency apply, a nuclear facility operator needs to comply with all such regulations.

5.1.1.1 NRC Regulations

Throughout the decommissioning process there is potential for various levels of radiation to be present. For this reason, there are no changes in regulatory procedures relating to occupational exposure, or lowering of existing safety precautions for decommissioning. 10 CFR 20, "Standards for Protection Against Radiation," specifies standards for allowable radiation doses to workers (10 CFR 20.101 - 20.105), diagnostic requirements (10 CFR 2.107 - 20.108), precautionary procedures (10 CFR 20.201 - 20.207), radiation exposure in waste disposal (10 CFR 20.301 - 20.305), and requirements for records, reports, and notification (10 CFR 20.401 - 20.409).

Section 40 CFR 190.10(a) provides standards for allowed radiological dose to any member of the public outside the plant, and Section 190.10(b) provides standards for the total quantity of radioactive materials allowed to enter the general environment from the entire uranium fuel cycle. As used in 40 CFR 190, the uranium fuel cycle includes chemical conversion of uranium, milling of uranium ore, isotopic enrichment of uranium, uranium fuel fabrication, reprocessing of spent uranium fuel, and generation of electricity by a light water cooled nuclear power plant using uranium fuel.

UF₆ conversion plants are currently licensed under 10 CFR 40. The only reference made to modification and termination of licenses under Part 40 jurisdiction is included in 10 CFR 40.71(d), and states:

"The Commission may terminate a specific license upon request submitted by the licensee to the Commission in writing."

10 CFR Part 40 does not stipulate the specific decommissioning actions that must be taken prior to termination of a facility license. When a licensee decides to permanently discontinue all activities involving licensed material, Section 40.41(f) of the Code of Federal Regulations requires written notification of such action to the NRC.⁽¹²⁾

Sections 10 CFR 70, "Domestic Licensing of Special Nuclear Material," and 10 CFR 73, "Physical Protection of Plants and Materials," do not apply to UF₆ conversion plants. Section 10 CFR 75, "Safeguards on Nuclear Material - Implementation of the US-IAEA Agreement," applies to UF₆ conversion plants, and deals with establishment of a material control and accounting system for the application of safeguards. Again, decommissioning is not specifically addressed.

There is little in existing regulations that elaborates on the subject of decommissioning financing; some financial information requirements for operating and shutdown reactors are given in 10 CFR 50.33(f). The NRC is considering the need for additional assurance that adequate funds will be available for decommissioning when required.^(13,14)

The goals in the area of financing the decommissioning of nuclear facilities should be to provide reasonable assurance that the licensee will pay the costs and to allow a broad range of approaches to implement that assurance.⁽¹⁵⁾

Another subject that must be considered when dealing with decommissioning activities is effluent monitoring reporting, which is covered in 10 CFR 40.65. However, in 10 CFR 40.65 and the rest of the regulations mentioned, the requirements and criteria for decommissioning are covered in only a very limited fashion.

Radioactive waste will be accumulated, treated, packaged, stored, and transported to a disposal site in accordance with existing regulations. These regulations include:

- 10 CFR 20 Standards for Protection Against Radiation
- 10 CFR 71 Packaging of Radioactive Materials for Transport.

Means for complying with these regulations must be defined in the license amendment request.

Further regulations and guidelines relevant to the shallow-land burial of waste may be forthcoming based on the technology now being developed to characterize the performance of low-level waste burial sites.

Because of the low hazards associated with uranium contamination of components and systems of the front end of the fuel cycle, the standard procedure for decommissioning is decontamination of the facility and disposal of equipment. This is done in accordance with the NRC's guidelines for release of the facility for unrestricted use.⁽¹¹⁾

With the increased attention now being given to decommissioning, more specific guidance is needed to identify public environmental radiation standards that apply explicitly to the decommissioning process, as opposed to facility operation.

Thus, while the NRC is responsible for assuring the public health and safety of all licensed facilities, NRC oversight of decommissioning is not explicitly required by statute and is currently limited to a case-by-case review. In addition, NRC's exercise of decommissioning procedures seems to address only those operational steps involved in the physical process of decontamination and dismantling. Other aspects of decommissioning such as advance planning and financing are not yet formally addressed.

5.1.1.2 License Termination

Under 10 CFR 40.41(f) the licensee must notify the NRC when the decision is made to permanently shut down the facility. Because of the nature of some of the decommissioning activities anticipated at the site, the NRC may elect to issue an amended license with administrative controls and facility requirements appropriate for the decommissioning option selected. The rationale behind this logic is that, although the plant operating functions have changed significantly during decommissioning, many unit operations may be similar (i.e., chemical decontamination and waste treatment). There will be active operations conducted in the plant involving radioactive materials and utilizing existing systems and components that can result in release of effluents to the environment.

The NRC requirements for terminating a license for nuclear reactors and fuel reprocessing plants are contained in 10 CFR 50.82. They require an application that specifies certain information on planned decommissioning procedures. The

regulation authorizes termination procedures, specifies additional conditions, provides for notice to interested persons, and states that if such procedures and conditions are followed, then a termination of license will be granted. In lieu of formal regulatory guidance in 10 CFR 40, the NRC has currently adopted the decommissioning guidelines provided in Reference 11. The clear implication of the 10 CFR 50.82 regulation is that dismantling and disposal are the exclusive objectives of the decommissioning process. However, regulatory guides discussed above also provide for other decommissioning alternatives that may not return a site to unrestricted use.

For facilities utilizing source or by-product materials, 10 CFR 40.71, "Modification, Revocation, and Termination of Licenses" and Section 40.41, "Terms and Conditions of Licenses," provide very general guidance by which a licensee may amend his license. It is likely that rules will be developed in 10 CFR 40 similar to 10 CFR 50.59 and Section 50.90 for reactors, which provide specific regulations and guidance for amending licenses. This amended state for facility licenses results from NRC approval to change requirements in the technical specifications that are applicable to normal facility operations. It seems that such approval can be extended to include the necessary requirements to ensure public safety during decommissioning.

Another potential question arises from the fact that the long-term care outlined in the Regulatory Guides could extend for a period that considerably exceeds the normal licensing period for nuclear facilities (typically 1 to 5 years between licensing renewals for 10 CFR 70 facilities). There is no indication in the regulations what the term of an amended license might be. This raises questions as to whether renewal of an amended license would be permitted and what the standards for such renewal would be.

5.1.2 State and Local Jurisdiction

A nuclear facility operator is also subject to state statutes, regulations, orders, and court decisions. Where conflicts exist between state and local requirements, state requirements generally prevail. Similarly, where there is a conflict between a federal requirement and a state requirement, the federal requirement controls, with some exceptions, as noted in the Clean Air

Act.⁽⁵⁾ Where no conflict exists, or where Congress has elected not to fully occupy a given legislative area, an operator must generally comply with all applicable federal, state, and local requirements in the conduct of his affairs.

Under the terms of 10 CFR 150, "Exemptions and Continued Regulatory Authority in Agreement States Under Section 274," agreement states of the NRC are authorized to issue byproduct, source material, and SNM licenses (limited quantities of SNM) to NRC licensees. Since Section 274(b) of the Atomic Energy Act of 1954 requires agreement state programs to be compatible with NRC regulations, the NRC will require that agreement state programs reflect the NRC's lead in the area of decommissioning.

Section 274k of the Atomic Energy Act of 1954 provides that "nothing in this section shall be construed to affect the authority of any state or local agency to regulate activities for purposes other than protection against radiation hazards."⁽¹⁶⁾ Thus, state requirements relating to such matters as land use, zoning, building construction standards, drainage regulations, and similar requirements are generally not preempted under the Atomic Energy Act, even though they can have an important impact upon the location, construction, and operation of licensed nuclear facilities.

State governments also exercise some control over shipments of radioactive materials. State highway departments regulate gross vehicle weights, vehicular dimensions and other parameters for radioactive shipments just as they do for other kinds of shipments. Currently, about half of the states have adopted the DOT Hazardous Materials Regulations to cover intrastate shipments.⁽¹⁷⁾ Several states have adopted or proposed additional regulations concerning radioactive materials.^(18,19) The variation of regulations between adjacent states can often require special considerations for interstate shipments.

There is potential conflict between some of the proposed state laws and the provisions of the Hazardous Materials Transportation Act of 1974 (Public Law 93-633 signed in 1975).⁽⁷⁾ This law prohibits the states from adopting laws or regulations more stringent than federal regulations unless the state regulations improve transportation safety. Even in this case, such rules can be adopted only if they do not unreasonably burden commerce.

The 1977 Clean Air Act Amendments clarify that states are no longer precluded from establishing and enforcing standards regulating radioactive emissions into the air. Thus, any state or locality may potentially establish standards more stringent than federal standards, or where a federal standard has not been established, may establish any standards it deems appropriate.⁽⁵⁾

5.1.3 Regulation of Effluents

During decommissioning operations, it will be necessary for radioactive air and water emissions to be kept as low as reasonably achievable. Specific regulations pertaining to radioactive emissions during decommissioning have not been issued. In the license termination application, however, an operator must ensure that decommissioning will not be inimical to public health and safety (see 10 CFR 50.82 as an example).

The radioactive effluents from waste processing operations or other activities during decommissioning must comply with EPA regulations as well as with 10 CFR Part 20. Currently, no specific EPA regulations exist for decommissioning. The EPA's 25 mrem/yr limit of exposure to the whole-body for any member of the general public from operating facilities of the nuclear fuel cycle, defined in 40 CFR Part 19D, Environmental Radiation Protection Standards for Nuclear Power Operations, excludes waste management activities, but such limits are now being developed. It is anticipated that a radiation dose limit from waste management operations similar to the 25 mrem/yr fuel cycle limit will be developed by the EPA. This new limit may well include the impact of decommissioning.

The Clean Air Amendments of 1977⁽⁵⁾ include radioactive emissions within the regulatory framework of the Clean Air Act. Section 122 of the Clean Air Act Amendments of 1977 creates an important statutory exception to the NRC's primary jurisdiction over radioactive emissions from licensed nuclear facilities. The 1977 Amendments make it clear that radioactive emissions into the air (which is presumed to be applicable to effluents from decommissioning) are subject to the regulatory framework of the Clean Air Act, in addition to determining whether emissions of radioactive pollutants will endanger public health.

EPA also has authority to regulate radioactive effluent and waste discharge through the Safe Drinking Water Act⁽¹⁹⁾ that establishes maximum contaminant

levels in public drinking water systems. Since contaminant is defined to include "any...radiological substance or matter in water,"⁽²⁰⁾ any activity relating to a nuclear facility that could potentially contaminate public water systems is subject to EPA jurisdiction.

Additional public and environmental radiation standards are found in 10 CFR 40.65 (effluent monitoring reporting requirements) and 10 CFR 20, Appendix B (concentrations in air and water above natural background). Other sources include proposed or existing EPA limits on radioactivity levels in water,⁽⁶⁾ soil, and effluents, and all EPA drinking water standards (40 CFR 141).

5.1.4 Licensing Changes and Decommissioning Plan

The principal regulatory issues during decommissioning of a UF₆ conversion plant will be public and occupational safety, environmental protection, and nuclear material safeguards. The regulatory jurisdictions, cognizant agencies, codes and guides for each of these have been reviewed above. A summary of the regulatory guidance provided in the Code of Federal Regulations is given in Table 5.1-2. When the decommissioning plan is prepared and the license amendment application is made for decommissioning, those regulatory issues will need to be addressed.

Since the major requirement of the planning phase is to provide the NRC with the necessary documentation for amending the facility operating license or negotiating a "decommissioning license," the licensee must provide:

- a description of the current facility status
- an inventory of the onsite radioactive materials
- a description of the proposed decommissioning activities
- a description of the proposed measures to minimize radioactive releases
- safety analyses of the proposed activities.

In addition, the licensee must submit a radioactive waste handling plan, a quality assurance plan, an environmental report, and a safeguards plan.

This information is then incorporated into the detailed work plans and procedures which contain all the information required to actually carry out the

TABLE 5.1-2. Checklist of Principal Federal Regulatory Requirements that Apply to Decommissioning a UF₆ Conversion Facility

<u>Regulatory Documents</u>	<u>Functions Regulated</u>
<u>Material License Requirements</u>	
10 CFR 40	Materials licenses relating to source material
<u>Environmental Protection</u>	
10 CFR 51	Licensing and regulatory policy and procedures for environmental protection
10 CFR 20, Appendix B	Offsite releases
10 CFR 20.300	Waste disposal
40 CFR 190.10	Environmental radiation protection requirements for normal operations in the uranium fuel cycle (EPA)
40 CFR 1500	CEQ guidelines for preparation of environmental impact statements
NRC Staff Report (no number given)	Guidelines for decontamination of facilities and equipment to release unrestricted use or termination of licenses for byproduct, source or special nuclear material (NRC, Nov. 1976)
<u>Employee Protection</u>	
10 CFR 20	Standards for protection against radiation
<u>Transportation</u>	
10 CFR 71	Packaging of radioactive material for transport and transportation of radioactive material under certain conditions
49 CFR 170-199	Department of Transportation hazardous material regulations

decommissioning tasks. The plans and procedures should cover all aspects of the decommissioning project, including:

- decommissioning methods
- radioactive waste management
- radiological and industrial safety
- equipment requirements
- environmental impact

- security and safeguards measures
- quality assurance.

5.2 SAFEGUARDS CONSIDERATIONS

Prior to initiating decommissioning of a UF_6 conversion plant, consideration must be given by the licensee to the necessity of safeguarding the quantities of residual uranium that can be expected to be found in the facility during the dismantlement and decontamination of the equipment, buildings, and site. The regulations for operating facilities licensed to possess and use source material require special accounting and reporting measures, as specified in 10 CFR 40, for all source material receipts, inventories, transfers, and discards.^(a) Similar procedures will probably be required by the NRC during decommissioning for application whenever the licensee possesses, transfers, or disposes of source material found in the facility, or in the wastes generated.⁽²¹⁾ The actual requirements will be specified in the license agreement and may differ from facility to facility because no specific regulations for safeguards during decommissioning are presently in force.

The requirements stemming from the implementation of the US/IAEA agreement on July 31, 1980 (described in 10 CFR 40.31, 10 CFR 70.21, and 10 CFR 75) will probably not be applied to UF_6 conversion plant decommissioning. However, if the licensee expects to possess more than 10,000 kilograms of uranium (one effective kilogram) at any time during decommissioning, the possibility of application of the procedures required by 10 CFR 75 must be considered in the licensing negotiations. It seems unlikely that this will be necessary.

Considerations relating to safeguarding source materials, as well as those pertaining to protection of the environment and the general health and safety, were discussed in the preceding sections. The regulations and guidelines specifically applicable to safeguarding natural uranium in a licensed conversion plant are summarized as follows:

(a) Source material is defined as uranium or thorium or any combination of them, but not including enriched uranium.

The licensee who possesses and uses source material is required to:

- transfer the material only to authorized persons or institutions as defined in 10 CFR 40, Section 40.51
- verify before the transfer that the intended recipient is authorized to possess such material
- keep records showing the receipt, transfer, and disposal of source material
- retain all source material records for the minimum time periods specified in 10 CFR 40, Section 40.61
- afford to the NRC opportunities at reasonable times to inspect the material, premises, facilities and material accounting records, and to perform such tests as the NRC considers appropriate or necessary
- complete and distribute the Nuclear Material Transaction Report, Form NRC-741, whenever a transfer or a receipt of 1000 kg or more of source material is made
- if licensed to possess 1000 kg or more of source material, submit a statement of this inventory once a year
- to report promptly to the NRC any known attempt to commit a theft or unlawful diversion of more than 15 pounds of source material
- if licensed to produce uranium hexafluoride, report every 6 months the quantity and principal radionuclides released to unrestricted areas in liquid or gaseous effluents.

There are no physical protection requirements for a source material licensee except as deemed necessary for the health and safety of the public.

At the end of normal operations, the types of nuclear material remaining in the UF_6 conversion plant will be similar to the types of nuclear materials stored and processed in the plant prior to shutdown. After decommissioning cleanout of the plant and the process equipment, the physical form and concentration of the material may be different from its original form due to the methods used to remove

the material from the areas in which they have been entrapped during normal operation. An estimate of the quantities and locations of uranium in the facility when decommissioning begins is given in Section 7, Table 7.4-2. A summary of those data is given in Table 5.2-1.

TABLE 5.2-1. Estimated Uranium Inventory in Major Pieces of Equipment in the Reference Plant After Final Inventory Cleanout

<u>Equipment</u>	<u>kg of Uranium</u>
Sampling Station Equipment	200
Solvent Extraction	1100
Concentrator Units	2000
Denitration	800
Reduction	400
Hydrofluorination	2100
Fluorination	<u>2600</u>
Total	9200

The problem of safeguarding uranium in a facility during decommissioning is unlike that in an operating facility. During decommissioning, the uranium will be on the surfaces of process enclosures and equipment, entrapped within equipment, in sumps, and distributed in waste and rubble. In general, it will be in inaccessible places. The easily removed SNM would be recovered during the final operational cleanup and any chemical decontamination activity during the initial phases of decommissioning. Measurable quantities of uranium will be obtained in cleanup and decontamination solutions and sludges. In addition, kilogram amounts of uranium compounds may be obtained in some cases during dismantling of certain processing equipment. The decommissioning plan will need to describe the intended procedures for measurement and accounting, record keeping, and reporting of those quantities of uranium. It is likely that NRC will require measurement, accounting, record keeping and reporting of transfers and disposals of uranium in a manner similar to that specified in 10 CFR 40 for operators. These were summarized above.

Most of the uranium will probably be in disposable decommissioning waste. Regulations do not provide restrictions on the quantity or concentration of uranium in disposable waste, but only provide rules for packaging and shipping and the manner of disposal (see 10 CFR 40, 10 CFR 20, 10 CFR 71, 40 CFR 190, and 49 CFR 199). The licensee's safeguards responsibilities are simply to measure and record the quantities of uranium present in an annual inventory accounting, and report all transfers and disposals of uranium-bearing material.

5.3 LICENSE TERMINATION AND FACILITY RELEASE

A primary goal of decommissioning is to terminate the nuclear license and release the facility for unrestricted use. To do this, the residual radioactive contamination must be at acceptable levels for public protection. There are no federal regulations that stipulate the actions to be taken to terminate a license for the facilities of this study. However, regulations, guidelines, and standards do exist that provide guidance on acceptable residual contamination levels for conditional or unrestricted release of uranium mill tailings piles. This guidance is reviewed in this section.

5.3.1 License Termination by Stabilization

Stabilization may be a method of decommissioning considered for the lagoon waste at the reference UF_6 plant. At this time, the NRC has not determined whether this will be acceptable; its acceptability would have to be considered on a case-by-case basis.

In decommissioning a UF_6 waste lagoon by stabilization, attention must be given to two items: first, stabilizing the lagoon to minimize radionuclide release, and second, monitoring the site to detect any such release or human intrusion. Experience in dealing with uranium mill tailings may be relevant to this study.

Legislation and information pertaining to post-operational activities at uranium mill tailings piles are contained in the Uranium Mill Tailings Control Act of 1978⁽²²⁾ and the Generic Environmental Impact Statement (GEIS) on Uranium Milling,⁽²³⁾ respectively. These documents may also be considered for guidance in establishing regulations for the decommissioning and long-term care of the reference plant lagoon wastes.

Four major provisions of the Uranium Mill Tailings Control Act are:

1) government ownership of tailings and tailings disposal sites, 2) financial surety that the mill operator is responsible for the costs of site decommissioning and long-term care, 3) elimination to the extent practicable of long-term maintenance, and 4) reclamation and management of tailings to national standards both before and after termination of a license. The Uranium Mill Tailings Control Act amends the Atomic Energy Act as follows:

- Section 83, covering ownership and custody of tailings and tailings disposal areas, is added. Before either a new license or a license renewal issued after October 1981 can be terminated, the licensee must clean up the site and transfer ownership of the tailings to the federal government or the government of the state in which they are located. In addition, for new licenses three land ownership options are stipulated: 1) ownership by the United States, 2) ownership by the state at the state's option, or 3) private ownership if the NRC determines that government ownership is not necessary.
- Section 161x, covering financial arrangements for decommissioning and long-term care, is added. The new section provides authority to require financial surety arrangements by licensees to permit completion of site cleanup and reclamation prior to termination of the license. If long-term care is required, financial arrangements must also be made; however, the need for long-term care should be minimized to the extent possible.
- Section 274 is expanded to require state standards that are equivalent to, or more stringent than, NRC standards for tailings disposal sites. The amended section describes procedures for state rulemaking paralleling basic federal procedures, and mandates the preparation of a written environmental analysis for each license.
- Section 275 is added, granting authority to the EPA to promulgate general standards for protection of the environment from both radiological and nonradiological releases from mill tailings. The NRC is responsible for enforcement of the EPA-promulgated standards.

The GEIS on uranium milling provides an informational and decisional base for the preparation of regulations covering decommissioning, decontamination, site reclamation, transfer of title on termination of operations, future site use, and long-term care of uranium mill tailings piles. Major institutional questions addressed in the document include:

- the need for land use controls and site monitoring
- the methods of providing financial surety that decommissioning is accomplished by the mill operator
- the need for and funding of long-term care that may be necessary.

The recommendations contained in the uranium milling GEIS are based on government ownership and surveillance of mill tailings. It recommends that decommissioning of other portions of a mill site provide for unrestricted use, but that the mill tailings pile be treated as a low-level waste burial site containing very long-lived radionuclides and requiring continued control to ensure that tailings are not disturbed by either human activities or natural weathering processes. In addition, it recommends that the continued control should be provided through government ownership and custody after the licensee has satisfied decommissioning requirements and the license is terminated, but no transfer of ownership is contemplated for UF_6 plants.

The NRC recently issued a Branch Position stipulating interim performance objectives for low-level waste burial site closure and stabilization.⁽²⁴⁾ These performance objectives could also provide guidance for a rare-metals refinery operator in developing tailings pile closure and stabilization plans in anticipation of transferring the pile to a custodial government agency. Major provisions of this Branch Position deal with: 1) measures to stabilize the site in a condition that requires only passive surveillance and monitoring during the long-term care period that follows license termination, and 2) funding assurance to permit site closure and stabilization.

5.3.2 License Termination by Removal

Removal of UF_6 plant lagoon waste involves the removal, packaging, and offsite shipment of the waste to a disposal facility. Presumably, after

removing the tailings and demonstrating that the contamination in the surrounding soil does not exceed a level to be prescribed by federal regulations, the operating license can be terminated and the site can be released for unrestricted use.

The EPA has issued interim standards for cleanup of open lands and buildings contaminated with residual radioactive materials (mainly tailings) from inactive uranium processing sites.⁽²⁵⁾ The standards apply to open lands and buildings that are part of any processing site designated by the Secretary of Energy under Public Law 95-604, Section 102.⁽²²⁾ The EPA standards might be considered for guidance on a case-by-case basis in defining standards for the cleanup of the reference plant.

The interim EPA standards limit radon decay product concentrations in contaminated buildings to 0.015 Working Level (WL).^(a) Gamma radiation is limited to 0.02 mR/hr above normal background levels. The standards impose a limit of 5 pCi of ²²⁶Ra per gram of soil for tailings on open land. The maximum radon emissions and gamma radiation that will result under the standards are within the variations that would normally occur from undisturbed land areas.

5.4 CONCLUSIONS

Although there are numerous regulations that can be applied to decommissioning by implication and analogy, specific and comprehensive regulatory requirements for decommissioning nuclear plants are presently lacking. For decommissioning a UF₆ conversion plant, a plan should be prepared that includes health, safety, environmental, and safeguards considerations. With regard to safeguards, the plan must cover the accounting procedures for quantities of uranium present and identifiable in the plant, and any measurable quantities recovered or identified in accountable and measurable forms during dismantlement and decontamination of the equipment, facility, and site. These quantities of uranium must be measured, accounted for, and disposed of in a manner acceptable to the NRC.

(a) A Working Level is defined as any combination of short-lived radon decay products in 1 liter of air that will result in the ultimate emission of alpha particles with a total energy of 130 billion electron volts. Studies of indoor background radioactivity at Grand Junction (Colorado), New York, and New Jersey showed that at least 10% of the houses with basements exceed the interim standard's remedial action level of 0.015 WL.⁽²⁵⁾

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6.0 ALTERNATIVE APPROACHES TO FINANCING DECOMMISSIONING

The Nuclear Regulatory Commission (NRC) has for several years been considering possible regulatory changes for the decommissioning of fuel cycle facilities.⁽¹⁾ Possible regulatory changes relating to financial assurance is one of the areas under active consideration. Most financial assurance attention has been directed to power reactors;⁽²⁾ however, other fuel cycles have also received attention. All of the decommissioning study reports prepared by Pacific Northwest Laboratory for the NRC have contained a chapter on assurance of funds for decommissioning. The NRC has issued a staff report on financial assurance of decommissioning funds that contains separate chapters applicable to reactors and other nuclear facilities.⁽³⁾ Regulations have also been issued by the NRC to assure funds for uranium mill decommissioning.⁽⁴⁾

This section identifies the alternatives for assuring the availability of decommissioning funds and discusses the relative merits of these alternatives. In addition, current regulatory developments in the area of decommissioning financing are discussed.

6.1 NEED FOR ASSURANCE OF DECOMMISSIONING FUNDS

The state in which a UF_6 conversion plant is located and the NRC have two principal financial concerns. First, they are concerned that when the facility ceases operation there will be sufficient funds available to perform a thorough decommissioning, and that sufficient funds will be available for care of the property before and after decommissioning until the site is released for unrestricted use. This concern exists for plants that operate throughout their anticipated life as well as for plants that are shut down prematurely. Second, they are concerned that funds will be available to provide for unexpected accidents and contingencies, both during the operating life of the facility and before decommissioning is completed. The principal focus of this section is on the first concern.

The basis of the state concern for availability of decommissioning funds is its inherent obligation to protect the health and safety of its citizens.

The basis of NRC concern is its obligation under the Atomic Energy Act of 1954 to regulate atomic energy and nuclear facilities so as to protect the health and safety of the public. If the UF_6 conversion plant operator defaults on his decommissioning responsibility, conducts decommissioning in a deficient manner, or unreasonably delays the decommissioning process, public health is potentially jeopardized.

6.2 APPROACHES TO PROVIDING FUNDS FOR DECOMMISSIONING AND LONG-TERM CARE

There are three principal alternatives to assure the availability of decommissioning funds for nuclear facilities other than power reactors:⁽⁵⁾

1. Creation of a sinking fund to accumulate sufficient decommissioning funds in a trust account during the facility's operating life,
2. Prepayment of the present value of future decommissioning funds into a trust account fund prior to facility start-up,
3. Purchase of a surety bond or an equivalent guaranteeing funds for decommissioning.

Each alternative is discussed in this section. It is also possible that these options can be combined in an acceptable manner. For example, alternative 1 might be supplemented by a bond guaranteeing the availability of funds in the event of premature facility closure. Internal decommissioning funding (where funds for decommissioning are set aside by the operator or where decommissioning costs are simply paid out of cash flow as costs are incurred) is not considered here because it has a relatively low assurance of providing funds unless it is supplemented by some other method.

The NRC staff has developed five criteria to evaluate the relative effectiveness of the financial assurance alternatives: 1) the degree of assurance provided by the alternative, 2) the cost of providing the assurance, 3) the equity of the alternative, 4) the ability of the alternative to respond to changing financial parameters, and 5) the ability of the alternative to

effectively meet different ownership and jurisdictional arrangements. (3, p.6)
These criteria are used as guidance in the discussion that follows.

6.2.1 Sinking Fund

This alternative contemplates the formation of a sinking fund tied to UF_6 production to generate enough funds during the operating life of the facility to pay anticipated decommissioning costs. Payments would be made into a trust account permanently outside the control of the facility operator and his creditors. This approach is currently used by the three states (Nevada, South Carolina, and Washington) that license and regulate low-level radioactive waste burial grounds.

Payments to the sinking fund would be based on UF_6 production. The charge per unit of product would be determined by estimating total decommissioning and associated administrative costs and total anticipated production over the facility's operating life. An amount would be paid into the fund per unit of product so that the payments, plus compound interest earned by investing the fund during and after the plant operating life, would be sufficient to pay all anticipated costs and provide a reasonable contingency of perhaps 10 to 15% of estimated costs.

The payment per unit of product into the sinking fund could be adjusted regularly to account for changing financial parameters. For example, the rate of return achieved by the fund stewards (i.e., trustee) in relation to decommissioning cost escalation will likely change. The production rate for the facility will not be completely constant over time. The real (i.e., nonescalated) decommissioning cost can also be expected to change with time because of technological innovations, added facilities, and new regulatory requirements. It is also likely that the expected life of a plant will change. All of these changes can be periodically accounted for by adjustments to the sinking fund payment. If such changes are not severe and are regularly reflected in the payments, the value of the sinking fund should be close to the needed funds when the facility is retired.

A variety of entities could be designated to provide stewardship for the sinking fund. Possibilities include state government, the federal government,

or a private financial institution. An independent "Decommissioning Assurance Agency" could be chartered by each state to retain and invest sinking fund payments from various nuclear facilities and perhaps oversee decommissioning activities and disburse funds to those conducting the activities. The pooling of decommissioning funds into such a centralized agency could help to ensure decommissioning performance even if a particular facility operator defaults in some manner. There could also be certain federal income tax advantages if the Agency is a subdivision of state government (see Section 6.5).

An advantage of the annual payment sinking fund approach is that it should generally assure that decommissioning activities actually occur. With funds set aside to cover the costs, the question of who should pay them is alleviated and arguments about responsibility are less likely to occur.

A second advantage of the sinking fund option is that it should improve decision-making when power-generating options are being considered. Currently, future decommissioning costs of nuclear facilities may not be reflected in the electricity prices paid by consumers. If all such facilities were required to have sinking funds to provide for future decommissioning and waste management expenses, these anticipated costs would be evaluated and equitably reflected in the cost of fuel and in power-generation costs and consumers' power bills.

Another advantage of the sinking fund approach is that it is reasonably equitable to consumers. As long as increases in estimated decommissioning costs are reflected in adjusted payment schedules, all consumers should pay their approximately proportional share of costs in dollars of approximately equivalent buying power.

Several potential problems and concerns with the sinking fund option should be recognized. The principal difficulty is the possible shortage of decommissioning funds in the event of premature facility closure. This potential problem is discussed in Section 6.3. Another concern relates to the care and investment of the fund. Professional management of the fund would be desirable. The trustee will be faced with the same problem other conservative investors are; i.e., how can assets be invested with minimal risk to earn a return that

at least matches the rate of cost escalation due to inflation? If the trustee is not able to match the rate of cost escalation, the payments to the fund (in year of start-up dollars) will have to be increased over time at a rate that exceeds the rate of escalation. A third concern is that decommissioning costs must be estimable with reasonable accuracy to provide a basis to calculate an appropriate sinking fund payment. Although revised estimates can be made and reflected in the sinking fund payments later in the facility lifetime, the initial estimate is especially important if the expected operating life is relatively short.

It must also be recognized that establishment and control of a decommissioning trust fund would, by its nature, create some administrative complexities. In addition to the problems of fund management and control, an additional government or quasi-government agency might be required to oversee the operation of one or more funds.

6.2.2 Prepayment of Anticipated Costs

The general framework of the prepayment alternative is similar to the sinking fund option. A trust fund would be established. The trustee would invest the monies until they are required for decommissioning. The difference is that the present value of anticipated decommissioning and administrative costs would be paid into the fund before facility start-up. Adjustments to the fund may be required to account for changes in such factors as the trust fund earnings rate relative to the decommissioning cost escalation rate; facility life; added production capacity; and changing technology, safety, and regulatory requirements.

The principal advantage of the prepayment approach is that it provides the highest degree of assurance that decommissioning funds will actually be available when needed. This is because sufficient money should be available for decommissioning operations even if the facility owner ceases operation prematurely.

One disadvantage of the prepayment option is that it may be the most expensive of the three financial assurance alternatives. The money deposited in the trust fund is likely to earn less than if it were being directly

utilized by the facility owner. This is because the discount rate utilized by a UF₆ conversion facility operator will likely exceed the interest rate obtainable by the trustee. The discount rate favored by the operator will approximate his minimum rate of return on alternative investments. By requiring the operator to prepay the expected decommissioning cost, society loses the productive value represented by the incremental return the operator could realize on the funds over what the trustee could realize.

To the extent debt funds are used to prepay the present value of decommissioning costs, the borrowing capacity of the operator is reduced and consequently his available supply of funds for capital investment is reduced. However, the prepayment approach would increase the amount of funds available for purchase of the conservative government and private security issues that a trustee would be likely to purchase.

One can argue that the prepayment approach unfairly raises the cost of nuclear power. Prepayment of decommissioning costs represents an extraordinary expense not incurred to the same degree in the fuel cycle cost of other power-generating technologies. The prepayment option may also penalize current consumers because future power consumers may not pay their full share of the decommissioning costs. Finally, it can be argued that the prepayment option unfairly discriminates against smaller businesses without the access to capital funds that a larger business would have.

6.2.3 Surety Bond

Under this alternative, the UF₆ conversion facility operator would be required to purchase a surety bond prior to start-up which would guarantee a specific amount of funds for decommissioning performance. A functional equivalent to a surety bond such as certain bank letters of credit are possible substitutes to the bond. It must be determined in advance whether the surety is equally responsible for decommissioning performance with the principal obligor, the UF₆ conversion plant owner, or whether the surety's liability arises only when all attempts to make the principal obligor perform have failed.

The surety bond approach does have precedent. A performance surety bond that guarantees reclamation of lands that are surface-mined for coal is required under the Surface Mining Control and Reclamation Act. Applicable regulations are in 30 CFR 800. Wyoming requires a bond to assure reclamation of any surface-mined lands.⁽⁶⁾ The amount of the bond is adjusted annually to reflect new land reclamation cost estimates. The surety bond alternative was also selected by the NRC as the principal financial assurance mechanism to assure decommissioning of uranium mill sites and the reclamation of any associated tailings or waste disposal areas (see Section 6.4).

A major advantage of the surety alternative is that administration expense is low. The bonds are not inexpensive, however. Depending on the credit rating of the operator and the degree of risk in the decommissioning operation, the cost of a bond, if it can be obtained, will likely be on the order of 1 to 2% per year of the guaranteed amount.⁽⁷⁾

There are several problems with obtaining a surety bond. The principal difficulty is that surety companies may not be interested in selling a long-term bond because of the many uncertainties affecting their obligation. If the bond is renewable at given intervals, the surety may very well decline renewal if the operating company becomes financially weak. Also, the guaranteed amount of the bond will have to be readjusted periodically to reflect revised decommissioning estimates. If the surety does not agree ahead of time to automatic escalation of the guarantee, the usefulness of the bond is again substantially reduced. Designation of appropriate criteria for determining when the facility owner has not performed decommissioning satisfactorily and, correspondingly, when the surety becomes liable may be difficult. An additional problem is that even if a bond can be obtained, financial assurance depends on the solvency of the surety. Finally, collecting on a surety bond could be more difficult (possibly requiring litigation) than utilizing funds previously paid into a trust fund.

It may be possible for a well-financed company to obtain a bond to assure the decommissioning of a UF_6 conversion facility under the DECON decommissioning

alternative. For a weakly financed company, or under the SAFSTOR decommissioning alternative, a bond may be very difficult to obtain and require significant collateral if it is obtainable.

6.3 DECOMMISSIONING FUNDING IN THE EVENT OF PREMATURE CLOSURE

With the sinking fund financial assurance option, the state runs the risk that sufficient funds will not have been collected to cover decommissioning costs if the facility closes prematurely. If the facility operator can and will pay the difference between the fund balance and the needed decommissioning funds, no problem arises. If he is financially unable to do so, the state or possibly the federal government could be forced to provide the needed funds. No special problem exists with the prepayment option because funds should be available whenever closure occurs. This is the principal advantage of the prepayment approach.

If the sinking fund financial assurance alternative is chosen, several options are available to assure the availability of funds in the event of premature closure. The options include one or more of the following:

- an initial extra cash payment to the sinking fund prior to production
- higher per unit sinking fund payments (in real, i.e., constant dollars) during the initial years of operation
- a bond posted by the facility operator
- premature shutdown insurance.

The first two options can be considered variations of the sinking fund and prepayment alternatives.

6.3.1 Initial Cash Payment

This option contemplates that an initial significant cash payment would be made to the trust fund prior to start-up. This money would become part of the trust fund and would thus presumably be outside the reach of the facility owner and his creditors. The size of the payment could be flexible and might

depend on the financial resources of the operator, the probability of premature closure, the extent of anticipated decommissioning problems, the anticipated operating life of the facility, and other factors. In general, however, it seems that an initial payment on the order of at least 10% of total estimated decommissioning costs (in year of start-up dollars) would be appropriate.

The principal advantage of this option is the added assurance it provides that the initial funds, plus sinking fund payments, will be sufficient to cover decommissioning costs. If the prepayment portion is a relatively small portion of total decommissioning cost, there is no significant disadvantage to this option. If it is a significant portion, the difficulties associated with the prepayment option discussed in Section 6.2.2 are relevant.

6.3.2 Higher Initial Sinking Fund Payments

This option contemplates that payments to the sinking fund in constant dollars would be initially higher than average and then would decline with time. The precise sliding scale can be determined by the licensing agency. One variation of this option would be to attempt to have constant payments in nominal dollars over the lifetime of the facility. (Nominal dollars are dollars of the year in which payments are made.) This option also could be utilized in conjunction with an initial cash payment.

The advantages and disadvantages of this option are comparable to those for the initial payment option. The main advantage is the added assurance that sufficient decommissioning funds will be available when needed. It is also reasonably equitable to the operator and his customers because both required sinking fund payments per unit of production and actual decommissioning costs per unit of production (assuming premature facility closure) will decline as the total quantity of UF_6 produced increases.

6.3.3 Surety Bonds

Surety bonds are another alternative for providing funds in the event of premature facility closure. The chief difficulty is the problem of obtaining a long-term commitment from a surety company, as discussed in Section 6.2.3.

If a suitable bond can be obtained, there are two potential advantages. First, it may be a more equitable alternative for the smaller company that is unable to make a significant initial cash payment. Second, it reduces the distortion effect on nuclear power-generation costs of a high initial cash payment.

6.3.4 Premature Shutdown Insurance

Insurance is an additional approach to decommissioning assurance. The policy would provide for payment of the additional funds needed for decommissioning in the event of premature facility closure. An insurance pool with many nuclear facilities participating may be the preferred way to implement this alternative. Participating facilities could be determined by type of facility, by geographic area, or by a combination thereof. Setting appropriate insurance premiums could be difficult, and the insurance pool concept may require federal participation for implementation. Insurance can also be used to provide liability and property damage coverage in case of an accident at the facility. The NRC has retained a contractor to investigate the feasibility of insurance to provide financial assurance for decommissioning a power reactor and funds to rebuild in case of an accident.⁽⁸⁾

6.4 RECENT REGULATORY DEVELOPMENTS

New regulations relating to financial criteria for decommissioning uranium mill sites and non-nuclear hazardous waste disposal sites have recently been issued. Both sets of regulations may be used as guidance for providing financial assurance for decommissioning a UF_6 conversion facility.

6.4.1 Uranium Mill Sites

New licensing requirements that include financial criteria for uranium and thorium milling activities, including tailings and wastes generated from these activities, were issued by the NRC in October 1980 and are found in Appendix A to 10 CFR 40.⁽⁴⁾ The requirements apply to all new or renewal license applications as of November 1980, and are effective for existing licensees in August 1981. Criterion 9 requires mill operators to obtain

financial surety arrangements acceptable to the NRC that will assure decommissioning performance. Arrangements generally acceptable to the Commission are: surety bonds, cash deposits, certificates of deposit, deposit of government securities, irrevocable letters or lines of credit, and various combinations of these. The surety bond seems likely to be the favored approach. The NRC or the cognizant state regulatory agency must be named the beneficiary of the financial assurance mechanism. The surety mechanism must be reviewed annually and the amount of the bond or guarantee must be increased if necessary to assure the availability of adequate funds. The surety agreement should ideally extend throughout the operating life of the facility. If it is for a fixed term that is less than the operating life, and the surety declines to renew, the regulations provide that the bond will be forfeited unless the licensee can locate a replacement surety. The licensee is also required to make a sufficiently large cash deposit with the appropriate regulatory agency such that interest from the deposit will enable long-term site surveillance.

In addition to the requirements for uranium mill licensees, the NRC is also preparing proposed regulations related to low-level waste burial grounds that will include a decommissioning financial assurance element. The proposed regulations are scheduled to be issued during the summer of 1981. Final regulations are scheduled to be issued in 1982 and will appear in 10 CFR 61.

Although the financial assurance requirements for uranium mill licensees emphasize the surety mechanism rather than a sinking fund or prepayment, it is not clear whether this will provide a precedent for other nuclear facilities. The decommissioning regulations applicable to uranium mills and low-level waste burial grounds are being prepared separately from NUREG-0436, the plan for reevaluation of NRC policy on decommissioning of nuclear facilities.^(8, p.3) The surety bond approach to decommissioning financial assurance is likely to be most suitable when expected decommissioning costs are relatively low. An NRC survey of surety companies found none willing to provide the required surety for power reactors,^(3, p.45) which have high decommissioning costs compared to UF₆ plants. Total estimated decommissioning costs at the model uranium mill

utilized in the Final Generic Environmental Impact Statement on Uranium Milling⁽⁹⁾ are approximately equal to or exceed the expected costs of decommissioning the reference UF_6 conversion facility described in this document. If mill operators are successful in obtaining decommissioning performance bonds, this approach may also be acceptable for UF_6 conversion plants if the difficulties discussed in Section 6.2.3 can be resolved.

6.4.2 Non-Nuclear Hazardous Waste Facilities

The Environmental Protection Agency (EPA) has recently issued revised proposed regulations designed to provide financial assurance for closure and post-closure care for non-nuclear hazardous waste disposal facilities.⁽¹⁰⁾ The regulations were issued under the authority of the Resource Conservation and Recovery Act. The facility owner can select from several acceptable financial assurance mechanisms including a trust fund, a surety bond, a letter of credit, or an acceptable combination of these.

The principal distinction of the EPA regulations in relation to the NRC uranium mill regulations is the availability of the trust fund built up with periodic payments over the operating life of the facility as an acceptable financial assurance mechanism. The regulations call for a bank or other financial institution to act as trustee for the fund. A required form of the trust agreement is provided in the regulations. The EPA acknowledges that the trust fund may not provide adequate protection in the event of premature facility closure and indicates that this problem is under further study.⁽¹¹⁾

6.5 FEDERAL INCOME TAX CONSIDERATIONS

The treatment of the various financial assurance alternatives under the Internal Revenue Code (the "Code") is an important element for both the NRC and a facility operator to consider in selecting an appropriate approach. The potential application of the Code to the three principal financing alternatives is discussed in this section.

The surety bond financial assurance mechanism appears to have the most straightforward tax treatment. Premiums for the surety bond should be an allowable deduction for the operator in computing taxable income as an ordinary

and necessary business expense under Section 162 of the Code. Actual decommissioning costs should also be deductible under Section 162 as they are incurred.

The important federal tax issues for the trust fund and prepayment financial assurance options are similar. First, are the periodic trust fund payments deductible from gross income by the facility owner/operator in the tax year they are made to the trustee? Second, is the income earned by the trust fund subject to federal tax, and, if so, who should pay the tax?

In general, it appears that periodic payments to a trust fund would be deductible by the facility operator. As long as the payments are required by a regulatory agency, they should qualify as an ordinary and necessary expense. If the trust funds are subsequently parceled back to the operator by the trustee, as the EPA hazardous waste regulations provide, they would be income to the operator as received. This income, however, would be offset by allowable expenses for actual decommissioning expenses. With the prepayment financing alternative, the tax treatment may be different. There is a reasonable possibility in this case that the Internal Revenue Service would not allow deduction of the full prepayment in the year paid, but rather would require amortization over the expected operating life of the facility. This interpretation would be costly to the operator because the time value of the tax savings realized by taking the deduction for the prepayment in the year paid would be lost.

It is also at least possible that the Internal Revenue Service would not allow current deductions for any payments made to a trust fund. This position would presumably be based on the claim that the monies paid to the trustee are not really an ordinary and necessary expense, but simply a temporary transfer of funds to a custodial trustee. This position would be quite undesirable for the facility owner and also seems inequitable. Several steps can potentially be taken to minimize the likelihood of this claim.^(3, p.20) The most effective step would be discretion on the part of the trustee or regulatory agency to retain someone other than the facility operator to conduct decommissioning operations. In this case the trust fund payments would not be merely a temporary transfer of funds because there is a possibility that the

operator would never see them again. A second step would be to preclude the trustee from investing in the stock or debt issues of the facility owner. A third step would be to assure that the trustee is truly independent of the facility operator. A fourth provision would be to require that any excess trust funds remaining after decommissioning and long-term care would revert to the state.

The interest earned by the trust fund can be exempt from federal taxes if the administering trustee is an agency of state government,⁽¹²⁾ or if the trustee invests in debt obligations issued by state governments or any political subdivision thereof, the District of Columbia, or any U.S. possession.⁽¹³⁾ If these conditions do not apply, there will be federal tax liability. The trust will be a separate tax paying entity and will be taxed in the same manner as an individual.⁽¹⁴⁾ The facility owner/operator is the logical entity to pay the federal taxes as well as any other taxes assessed against the trust account. The potential tax liability must be estimated in advance so that sufficient funds are collected from the operator to make the required tax payments.

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12. Internal Revenue Code, Section 115.
13. Internal Revenue Code, Section 103(a).
14. Internal Revenue Code, Section 641.

7.0 CHARACTERISTICS OF THE REFERENCE URANIUM HEXAFLUORIDE CONVERSION PLANT

The reference uranium hexafluoride (UF_6) conversion plant, the reference site on which it is assumed to be located, and the physiochemical processes used in the plant are described in this section. Estimates are presented of residual radioactivity levels and residual chemical levels on the site and in the plant when conversion operations are terminated.

U_3O_8 concentrate extracted from uranium ore must be converted to the volatile compound uranium hexafluoride (UF_6) for enrichment by the gaseous diffusion process. Two processes are in use for UF_6 production. One process employs a wet chemical solvent extraction step at the head end of the process to prepare a high-purity uranium feed prior to the reduction, hydrofluorination, and fluorination steps. The second process, dry hydrofluorination, consists of continuous successive reduction, hydrofluorination, and fluorination of the ore concentrates, followed by fractional distillation of the crude uranium hexafluoride to obtain a pure product.

The nature of the effluents from the two processes differs. In the wet process, the bulk of the yellowcake impurities is contained as dissolved solids in a solvent extraction waste stream. In the dry process, the bulk of the impurities entering with the crude uranium feed is rejected from the hydrofluor process as solids.

The reference plant used as the basis for this study is based on the wet solvent extraction process. The plant is assumed to have processed 10,000 metric tons (MT) per year of natural uranium and to have been in operation for about 20 years. The reference plant is defined using current information from a combination of existing facilities, and no attempt is made to use all design features from a single plant.

Details of the plant and the plant process descriptions are presented in Appendix A; site description details are given in Appendix B; and the bases for residual radioactivity estimates are presented in Appendix C.

7.1 SITE DESCRIPTION

A reference site is developed to aid in assessing the public safety of conceptually decommissioning the reference plant. The meteorological parameters

and population distributions used for this reference site are taken from the ALAP Study⁽¹⁾ for the river site in the year 2000. The ecological data are taken from environmental information provided for an operating nuclear reactor.⁽²⁾ 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 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 of the site.

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 distant. The closest large city, population 1,800,000, is about 48 km away. The total population in a radius of 80 km is about 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 about 1 km. About 80% of the land surrounding the reference site is used for farming.

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 $5 \times 10^{-8} \text{ sec/m}^3$.⁽²⁾

The reference site is slightly contaminated with radioactive material as a result of deposition from the release of normal operating effluents over the assumed 40-year plant operating life. It is assumed that accidental releases of radioactive material are cleaned up immediately following the event. Estimates of the maximum site contamination levels at the time of plant shutdown and selected times after shutdown are shown in Table 7.1-1 for natural uranium. The site contamination estimates are based on the deposition of predicted normal operating atmospheric releases of particulates.⁽³⁾ The assumptions and calculational methods for relating the normal plant effluents to site surface contamination are found in Appendix B.

TABLE 7.1-1. Estimated Quantities of Radioactive Material Deposited on the Reference Site After a 40-Year Operating Lifetime^(a)

Radionuclide	Deposited Radioactivity ($\mu\text{Ci}/\text{m}^2$) at Plant Shutdown and Selected Times After Shutdown				
	Shutdown	5 Years	10 Years	30 Years	100 Years
^{230}Th	8.8×10^{-4}	1.1×10^{-3}	1.3×10^{-3}	2.1×10^{-3}	5.0×10^{-3}
^{231}Th	6.8×10^{-3}	6.8×10^{-3}	6.8×10^{-3}	6.8×10^{-3}	6.8×10^{-3}
^{234}Th	7.5×10^{-2}	7.5×10^{-2}	7.5×10^{-2}	7.5×10^{-2}	7.5×10^{-2}
^{231}Pa	9.0×10^{-6}	1.1×10^{-5}	1.3×10^{-3}	2.2×10^{-5}	5.3×10^{-5}
$^{234\text{m}}\text{Pa}$	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}
^{234}Pa	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}
^{226}Ra	5.2×10^{-5}	5.2×10^{-5}	5.2×10^{-5}	5.2×10^{-5}	5.2×10^{-5}
^{234}U	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}
^{235}U	4.5×10^{-3}	4.5×10^{-3}	4.5×10^{-3}	4.5×10^{-3}	4.5×10^{-3}
^{238}U	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}
Totals	3.9×10^{-1}	3.9×10^{-1}	3.9×10^{-1}	3.9×10^{-1}	3.9×10^{-1}

(a) Based on the ratio of radionuclides in uranium and the isotopic ratios given in Reference 3.

7.2 PROCESS DESCRIPTION

The reference UF_6 conversion plant is designed to receive yellowcake (U_3O_8) in 0.21-m^3 drums from various uranium mills located in the western United States and to convert the feed stock to uranium hexafluoride (UF_6). The two processes for UF_6 conversion in use today differ only in the method of purification. The major steps in either process are:

1. pre-process handling, weighing, sampling, and storage
2. conversion of the yellowcake (U_3O_8) to uranium trioxide (UO_3) by roasting
3. reduction of the UO_3 to UO_2 with hydrogen
4. hydrofluorination of the UO_2 to UF_4 with hydrogen fluoride
5. fluorination of the UF_4 to UF_6 with elemental fluorine
6. storage of the purified UF_6 in shipping cylinders.

The purification step is added either at the beginning using a solvent extraction process or at the end by fractional distillation of the UF_6 . The solvent extraction purification step (the wet process) results in the radioactive uranium daughters (^{230}Th and ^{226}Ra) and impurities being left in the solvent extraction waste stream. This waste may be buried onsite, solidified and disposed of in a shallow-land burial ground, or returned to the mill for disposal with the tailings (see Figure 7.2-1).⁽⁴⁾ The dry process, however, removes the impurities from the UF_6 product stream by fractional distillation and incorporates them with other waste products for disposal as solid waste in a shallow-land burial ground (see Figure 7.2-2).⁽³⁾

7.2.1 Wet Solvent Extraction Fluorination Process

The reference plant is designed to produce 10,000 MT per year of uranium as UF_6 from concentrates (yellowcake) supplied by various uranium mills. Yellowcake, received in 0.21-m^3 drums, is transferred into a sampling system designed to remove 0.1% of the feed as a representative sample of the material for accounting purposes.⁽⁴⁾ The main yellowcake stream is conveyed either to a redrumming station or to storage hoppers in the digestion area. The yellowcake is dissolved batchwise in hot 40% nitric acid and the resulting

FIGURE 7.2-1. UF_6 Production - Wet Solvent Extraction Fluorination Process Simplified Block Flow Diagram

impure uranyl nitrate solution is processed by counter-current extraction in pumper-decanter using tributylphosphate-hexane solvent. After passing through a two-stage scrub section, the loaded solvent is stripped in a pulse column to recover the uranium values as a pure aqueous solution of uranyl nitrate.⁽⁴⁾

FIGURE 7.2-2. UF_6 Production - Dry Hydrofluorination Process Simplified Block Flow Diagram

This solution is concentrated by a two-step process, first in a single-effect evaporator (after scrubbing with hexane to remove any residual organic), and then in a boildown tank to which sulfuric acid is added to improve product reactivity in subsequent steps. The concentrated solution, containing about 1200 kg of uranium per m^3 , is then decomposed in a trough-type denitrator to uranium trioxide, oxides of nitrogen, and nitric acid vapor. These

volatile nitrogen compounds, along with those generated in the yellowcake digestion step, are recovered as nitric acid for reuse in the process.

The purified uranium trioxide is contacted with cracked ammonia in fluid bed reactors to produce uranium dioxide which in turn is reacted with anhydrous HF in a two-stage fluid bed hydrofluorinator to produce uranium tetrafluoride. The UF_4 is then conveyed to a series of fluorination towers where it is reacted with elemental fluorine to produce high-purity gaseous uranium hexafluoride. The UF_6 gas is cooled and condensed in cold traps as a solid that is periodically melted by heating with steam and drained into approved 9.1-Mg cylinders for shipment to gaseous diffusion enrichment plants.

Off-gas from the reduction fluid beds is passed through sintered metal filters to remove particulates and is then burned before being discharged to the atmosphere.⁽⁴⁾ The resulting gas is released through the main plant stack as a mixture of nitrogen, oxygen, and water vapor containing a small amount of sulfur dioxide. The hydrofluorinator off-gas is also filtered to remove particulates and is then condensed with the aqueous hydrofluoric acid collected and returned to the anhydrous HF supplier. The non-condensable off-gases are scrubbed with water before being vented to the atmosphere through the plant stack and the resulting dilute HF is neutralized with slaked lime.

Process fluorine is produced as needed by the electrolysis of hydrogen fluoride dissolved in a fused salt bath of potassium bifluoride, using medium-temperature water-cooled cells. Both product streams, fluorine as well as hydrogen, are passed through filters to remove any entrained electrolyte, then compressed and cooled to recover the hydrogen fluoride contained in the process gas. The fluorine is fed to the process and the hydrogen is piped to a waste gas burner along with air and any non-condensibles passing through the UF_6 product cold-trap system. The burner off-gas, now containing water vapor and some hydrogen fluoride, is fed into the scrubber used for treatment of the hydrofluorinator off-gases and then to the plant stack.

7.2.2 Effluents and Wastes

Normal operation of the facility produces a variety of solid, liquid, and gaseous wastes that are treated to meet federal and state pollution control

standards before being released to the environment. Liquid wastes containing significant quantities of radioactive materials are impounded in settling lagoons and stored. Other liquid wastes containing traces of radioactivity are diluted and released. All airborne effluents are treated to remove uranium contamination and gases produced in the manufacturing process.

7.2.2.1 Gaseous and Airborne Particulate Waste

Operating areas having the potential for producing dusts, mists, or fumes containing radioactive or other toxic materials are provided with state-of-the-art pollution abatement or ventilation equipment designed to reduce occupational and offsite exposures to levels as low as reasonably achievable.

Trace quantities of gaseous UF_6 pass through the product cold-trap system with the non-condensable gases. This uranium is discharged to the atmosphere as particles of uranyl fluoride after hydrolysis with the water vapor in the plant stack off-gas. Additional natural uranium compounds (UO_3 , UO_2 , and UF_4) are released occasionally as a result of entrainment in ambient air streams from leaking packing glands or spillage during handling and from small amounts of leakage through particulate filters used in solids transfer and vacuum cleaning systems. The total release of uranium-containing particulates from ground-level sources as well as from the 46-m main plant stack during operation of the plant is shown in Table 7.2-1.⁽³⁾

TABLE 7.2-1. Estimated Annual Atmospheric Release of Radionuclides from the Reference UF_6 Conversion Plant^(a)

Radionuclide	Half-Life (days)	Release (Ci/yr)
$^{230}Th^{(b)}$	2.8×10^7	1.4×10^{-3}
$^{234}Th^{(b)}$	2.4×10^1	1.2×10^{-1}
$^{226}Ra^{(b)}$	5.8×10^5	8.3×10^{-5}
$^{234}U^{(c)}$	8.9×10^7	1.6×10^{-1}
$^{235}U^{(c)}$	2.6×10^{11}	7.4×10^{-3}
$^{238}U^{(c)}$	1.6×10^{12}	1.6×10^{-1}
Total		4.5×10^{-1}

- (a) Based on information from Reference 3.
 (b) Based on isotopic ratios in yellowcake.
 (c) Based on the ratio of radionuclides in natural uranium.

In addition to the airborne particulate uranium, the plant produces several off-gases. These off-gases are also treated to minimize their environmental impact. Oxides of nitrogen are generated at three points in the process as well as in the combustion of natural gas for steam production. The process sources are: 1) the primary digesters where the concentrate feed is dissolved, 2) the miscellaneous digester used for dissolution of tower ash and other recycled solids, and 3) the denitrators where uranium trioxide is produced from uranyl nitrate hexahydrate. The digesters are operated at subatmospheric pressure to prevent leakage to the atmosphere. The off-gases from these operations are processed to recover nitric acid, which is returned for reuse in the process. The loss of NO_2 from the nitric acid absorber amounts to about 6.3 kg per hour, while an additional estimated 8.2 kg per hour is produced from natural gas combustion. Both streams are discharged from the top of the main plant stack and, as shown in Table 7.2-2, result in a relatively insignificant increase in the ambient air concentration of this material.

TABLE 7.2-2. Estimated Annual Gaseous Effluents from the Reference UF_6 Conversion Plant

Effluent	kg/day	kg/year
HF	1.5×10^1	5.4×10^3
SO_2	5.6×10^2	2.0×10^5
NH_3	2.4×10^1	8.7×10^3
H_2S	8.6×10^{-3}	6.0×10^0
Hexane	4.4×10^2	1.6×10^5
NO_2	2.6×10^2	9.5×10^4

Other gaseous effluents are also treated to remove undesirable materials before being released to the environment. Unreacted hydrogen and hydrogen sulfide from the reduction towers are burned in air to water vapor and sulfur dioxide before being discharged to the plant stack. Total sulfur dioxide leaving the plant process is estimated to be about 2.3 kg per hour. In addition, waste hydrogen from the fluorine plant is combined with the trace quantities of fluorine and UF_6 from the product cold trap system and fed with combustion

air to a waste gas burner.⁽⁴⁾ The burner product along with the aqueous HF condenser off-gases are scrubbed with water before being vented through the main stack. Fluoride sources other than the HF scrubber include material discharged from the fluorine cell rework area as well as discharges from the fluorine emergency vent. The total combined fluorine and fluoride emission from ground-level sources as well as from the 46-m plant stack is estimated to be about 4.5 kg per day.

7.2.2.2 Liquid Waste

The plant process generates two major liquid waste streams. The first liquid waste stream, the solvent extraction stream leaving the solvent extraction circuit, is primarily a solution of ammonium nitrate, nitric acid, heavy metal salts, and small quantities of uranium and the radioactive daughters of normal uranium decay. This liquid is combined with spent sodium hydroxide from the solvent treatment and the miscellaneous digester scrubber systems along with recovered weak acids. The liquid is neutralized with ammonia and impounded in four solvent extraction waste storage lagoons.

The second liquid waste stream is generated primarily by the hydrofluorination system off-gas scrubber. This fluoride stream is combined with various other plant waste liquids and treated with slaked lime to neutralize the contained acid and precipitate the fluoride as calcium fluoride. The resulting sludge is allowed to settle in a retention pond, and the overflow is treated with sulfuric acid to adjust the pH and precipitate excess calcium as calcium sulfate. The clarified treated waste liquid overflows and is combined with clean intake water, sanitary wastes, and sewage lagoon overflow in a concrete stilling basin before being discharged through a metering weir to the river. The estimated annual liquid effluent from the reference plant is shown in Table 7.2-3.

7.2.2.3 Solid Waste

Various solids containing recoverable uranium values are generated at several points in the process. These include dusts collected on the exit filters from the tower reactors as well as "ash" that escaped conversion to UF_6 in the

TABLE 7.2-3. Estimated Annual Liquid Effluents from the Reference UF_6 Conversion Plant

Effluent	kg/day	kg/year
NO_3^-	7.8×10^0	2.8×10^3
F^-	1.8×10^0	6.6×10^2
SO_4^{--}	2.9×10^2	1.1×10^5
NH_4^+	4.5×10^{-1}	1.6×10^2
Ca^{++}	1.9×10^0	6.9×10^2
Na^+	1.0×10^3	3.6×10^5
K^+	1.3×10^2	4.8×10^4
HCO_3^-	2.6×10^3	9.5×10^5
Cl^-	7.8×10^0	2.8×10^3
Fe	1.6×10^0	5.6×10^2

fluorination step. These materials are recycled to a miscellaneous digester where they are dissolved in an ammonium nitrate solution and fed back through the solvent extraction system. The fluorination tower ash that is left at the end of plant life is processed through the solvent extraction system as part of the final inventory cleanout.

Radioactive waste materials such as scrapped equipment, gloves, respirators, and other contaminated solids are buried onsite or, in the case of drums, are accumulated for disposal through a licensed steel scrap dealer. These are the miscellaneous wastes buried onsite.

Additional solid wastes generated include the contaminated calcium fluoride-calcium hydroxide-calcium sulfate sludge produced by lime neutralization of the HF scrubber products and pH adjustment with sulfuric acid. This material is disposed of by onsite burial in the fluoride waste lagoons.

Another source of radioactive solid wastes is the sludge produced by neutralizing the solvent extraction waste stream with ammonia to precipitate most of the uranium, thorium, and other heavy metals. This sludge settles to the bottom of the solvent extraction waste storage lagoons. Additional treatment of the supernatant liquid with barium chloride precipitates a mixture of barium and radium sulfates that is also allowed to settle, thereby removing most of the residual radioactivity from the liquid.

7.3 PLANT DESCRIPTION

The reference conversion plant, shown in an isometric drawing in Figure 7.3-1, plant is assumed to occupy about 30.4 hectares within the site described in Section 7.1. The plant consists of three buildings containing approximately 11,150 m² of floor area. The buildings are of normal industrial construction, with heavy concrete floors to support equipment. The principal structures are the main building, the solvent extraction facility, a warehouse, a cooling tower, retention lagoons, and other storage areas. The main building, which occupies about 7,400 m², contains the major processing and fluorine generating facilities, the utility and maintenance areas, administrative offices, the laboratory, and the ore concentrate sampling plant. The 46-m plant stack is located near the northwest corner of the main building.⁽⁴⁾

For the wet process, the solvent extraction facility is located in a separate 370-m² building west of the main structure.

A single story warehouse is located north of the main building. A cooling tower is located north of the solvent extraction building, and retention ponds for sanitary sewage, fluoride treatment, clarification, and raffinate storage are located west of the plant buildings.

Additional plant facilities include a large electrical substation, a UF₆ cylinder storage area, a chemical tank farm north of the main building, and a drum storage area at the southwest corner of the building.

7.3.1 Process Building Description

The main process building is a 55-m by 100-m structure. The height of the building is 20 m over the sampling plant area, and most of the roof above the manufacturing area is about 12.5 m above ground. The building is fabricated of 40-mm insulated metal siding attached to a steel framework. The interior walls are constructed of concrete block and sheetrock. The roof consists of a 40-mm insulated corrugated metal deck that is capped with asphalt and gravel.

In the main process building, the plant operations are divided into the following operations:

- sampling facilities
- digestion
- conversion

FIGURE 7.3-1. Isometric Drawing of the Reference UF₆ Conversion Plant

- denitration reduction
- hydrofluorination
- fluorination.

The sampling facilities are separated from the rest of the area by partition walls within the building. The rest of the process areas are generally not partitioned by walls, but the various operations are located in separate areas of the building. The floors in the process areas are concrete or steel plate. The chemical areas and contaminated areas have sealed concrete floors or steel plate floors. Non-production, non-contaminated areas have concrete floors, and offices and laboratories have either tiled or painted floors. The solvent extraction building has a steel frame with metal siding, and the floors are sealed concrete.

7.3.2 Ventilation System Description

Ventilation systems are provided throughout the manufacturing building, with special attention given to those areas having the potential for contamination with radioactive materials. Process areas in the manufacturing building are ventilated by forcing air into the building at a rate sufficient to provide 10 air changes per hour as the air is exhausted through eleven $8.5\text{-m}^3/\text{min}$ powered vents as well as through non-powered roof hatches. Both vents and hatches are sampled to evaluate radioactive discharges to the atmosphere in accordance with 10 CFR 40.65. Fumes from the fluorine cell maintenance area are discharged directly to the atmosphere by means of a $450\text{-m}^3/\text{min}$ exhaust fan. The exhaust from each laboratory fume hood is conveyed to the sampling area roof through separate sampled ducts.

A central, all-purpose dust collection system is provided to service all dry-processing areas within the plant. Dust collection hoods are installed around packing glands, routinely opened equipment, solids transfer areas, and the discharge from special-duty dust collectors or vacuum systems. The size and shape of the hoods and the exhaust air velocities are all designed to obtain efficient collection of the generated dust. The dust collector is a $785\text{-m}^3/\text{min}$ 5-zone plenum pulse (reverse jet), bag-type filter that discharges to the

atmosphere through a monitored stack. Collected solids are drummed and recycled to process. The stack sampler operates continuously with a 24-hour sample collected and analyzed daily.

Four additional vacuum cleaning and dust collection systems providing separate vacuum sources, cyclone separators, and bag filters are used in specialized plant areas. The sampling plant has separate vacuum cleaner and dust collection systems for recovery and direct recycle of spilled yellowcake, with the filtered air from the vacuum cleaner discharged to the sample plant dust collector. Filtered air from this system in turn discharges to the main plant dust collector, thus providing three stages of filtration. Two additional vacuum recovery systems, each consisting of a cyclone separator, bag filter, and vacuum pump in series, are provided, with both bag filters discharging to the main plant dust collector for backup filtration. One system, called the main plant vacuum system, services all areas after yellowcake sampling through the reduction area and discharges collected solids to the miscellaneous digester. The second unit, the UF_4 ash vacuum system, services the hydrofluorination and fluorination areas, with collected solids drummed for transfer to the ash-grinding system. Both systems are used for general plant clean-up in their respective service areas and for removal of uranium materials from equipment components prior to maintenance operations.

The entire ash-grinding system provided for all-purpose crushing, pulverizing, and screening of fluorination tower ash or other materials capable of direct recycle is housed in an enclosure that is vented to the main plant dust collection system.

7.4 RESIDUAL RADIOACTIVITY ESTIMATES

The radionuclide mixtures found in the reference UF_6 conversion plant at the time of plant shutdown and 5, 10, 30, and 100 years after plant shutdown are shown in Table 7.4-1. Using this mixture of radionuclides, Table 7.4-2 summarizes estimated residual radionuclide inventories of natural uranium in major pieces of equipment of the reference plant after final inventory clean-out. Key assumptions and rationale on which the inventory estimates are based

TABLE 7.4-1. Specific Radioactivity of Residual Uranium in the Reference UF₆ Plant at the Time of Plant Shutdown and at Various Times After Shutdown

Radionuclide	Specific Activity ($\mu\text{Ci/g}$ of mixture) ^(a)				
	Shutdown	5 Years	10 Years	30 Years	100 Years
²³⁰ Th	2.9×10^{-3}	3.7×10^{-3}	4.4×10^{-3}	7.2×10^{-3}	1.7×10^{-2}
²³¹ Th	2.2×10^{-2}	2.2×10^{-2}	2.2×10^{-2}	2.2×10^{-2}	2.2×10^{-2}
²³⁴ Th	2.5×10^{-1}	2.5×10^{-1}	2.5×10^{-1}	2.5×10^{-1}	2.5×10^{-1}
²³¹ Pa	3.0×10^{-5}	3.7×10^{-5}	4.4×10^{-3}	7.4×10^{-3}	1.8×10^{-2}
^{234m} Pa	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}
²³⁴ Pa	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}
²²⁶ Ra	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}
²³⁴ U	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}
²³⁵ U	1.5×10^{-2}	1.5×10^{-2}	1.5×10^{-2}	1.5×10^{-2}	1.5×10^{-2}
²³⁸ U	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}
Totals	1.3	1.3	1.3	1.3	1.3

(a) Based on the ratio of radionuclides in uranium and the isotopic ratios given in Reference 3.

and detailed inventories for each system are presented in Appendix C. Final inventory cleanout operations and decommissioning decontamination operations are described in Appendices F and G.

At shutdown and before final inventory cleanout, the uranium inventory in the reference plant is estimated to be about 105,000 kg, while after final inventory cleanout it is estimated to be approximately 46,800 kg. This inventory consists of 8,600 kg in each of the solvent extraction lagoons or about 34,400 kg total, 2,500 in the fluoride lagoons, 9,200 kg in the equipment, and 3,200 kg in the buildings and ventilation system. Final cleanout, done periodically during plant operations between processing campaigns, removes process materials, chemicals, trash, visible quantities of contamination, scrap, scrap solutions, and contaminated solutions from the facility. Empty product, scrap, and waste handling tanks are also rinsed to flush out remaining process solutions. These operational inventory cleanout operations are charged to plant operations and not to decommissioning.

TABLE 7.4-2. Estimated Uranium Inventory in Major Pieces of Equipment in the Reference UF_6 Plant After Final Inventory Cleanout

Equipment	kg of Uranium
<u>Sampling Station</u>	
Drum Elevator	5
Yellowcake Receiving Bin	50
Primary Sampler I	10
Primary Sampler II	10
Secondary Sampler	10
Yellowcake Transfer Screw	40
Bucket Elevator	25
Digest Feed Bin	50
Subtotal	200
<u>Solvent Extraction</u>	
Digesters	700
Adjustment Tanks	200
Pumper Decanters	50
Scrub Decanter	60
Re-extraction Pulse Column	90
Subtotal	1,100
<u>Concentration</u>	
Evaporator	1 000
Boildown Tanks	500
UNH Filter	100
Denitrator	300
Subtotal	2,000
<u>Denitration</u>	
Denitrator Scrubber Cooler	600
Cooling Screw	20
UO_3 Vertical Elevator	30
Pulverizer and Surge Bin	150
Subtotal	800

TABLE 7.4-2. (Contd)

Equipment	kg of Uranium
<u>Reduction</u>	
UO ₃ Storage Bin	50
Reduction Fluid-bed	<u>350</u>
Subtotal	400
<u>Hydrofluorination</u>	
UO ₂ Filter Bin	100
1st Stage Hydrofluorination	800
2nd Stage Hydrofluorination Fluid-bed	600
UF ₄ Seal Hopper	40
UF ₄ - UO ₂ Intermediate Seal Hopper	40
UF ₄ Purge Conveyor	20
Redler Conveyor 509	20
UF ₄ Storage Hopper	400
Redler Conveyor 510	<u>80</u>
Subtotal	2,100
<u>Fluorination</u>	
Primary Fluorination Tower	800
Redler Conveyor 511	20
Primary Filter	60
Primary Cold Trap	110
Product Cylinder	70
Back-up Filter	80
Clean-up Reactor Feed Hopper	100
Clean-up Reactor	600
Redler Conveyor 512	20
Back-up Filter	500
Secondary Cold Trap	<u>240</u>
Subtotal	2,600
Total	9,200

Decommissioning decontamination operations start with thorough flushing and chemical decontamination of all process equipment and piping. Decontamination procedures include spray cleaning, hot corrosive acid flushing, dry cleaning, and handwiping of dry processing equipment. These procedures are estimated to reduce the inventory of uranium in the buildings and equipment to approximately 1,200 kg of natural uranium.

7.5 CHEMICAL INVENTORIES

Several potentially toxic chemical compounds are used in processing and scrap recovery operations in the reference plant. These compounds include anhydrous ammonia, lime, nitric acid, sodium hydroxide, sulphuric acid, hydrofluoric acid, potassium fluoride, and tributylphosphate-hexane solvent. Most process chemicals not planned to be used for decommissioning are assumed to be removed from the plant as part of final inventory cleanout operations. Inventories of these chemicals will therefore be limited to residuals in process equipment and piping at the start of decommissioning. Except for nitric acid, fluoride salts, and degassing agents used in chemical decontamination operations, no significant inventories of toxic chemicals are anticipated in the plant when decommissioning begins.

REFERENCES

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3. U.S. NRC, Environmental Impact Appraisal of the Allied Chemical Corporation Nuclear Service Division Uranium Hexafluoride Conversion Facility, Metropolis, Illinois, Docket no. 40-3392, pp. 4-1 through 4-16, August 1974.
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8.0 SUGGESTED METHODOLOGIES FOR DETERMINING ACCEPTABLE RESIDUAL RADIOACTIVE AND CHEMICAL CONTAMINATION LEVELS FOR THE DECOMMISSIONED REFERENCE UF₆ CONVERSION PLANT

This section contains a discussion of suggested methodologies for determining acceptable levels of residual radioactive and chemical contamination at decommissioned nuclear facilities. Demonstrations of the suggested methodologies are also presented using the reference inventories of radionuclides and chemicals, and the reference site description developed for this study.

Detailed information about the mixture of radionuclides and chemicals found in the reference UF₆ conversion plant and on its site is contained in Appendices B and C. Descriptions of the reference facility and site are presented in Appendices A and B, respectively. Discussions of the radiation dose models and parameters used to determine the acceptable radioactive contamination levels are presented in Appendix E.

8.1 TECHNICAL APPROACH

The ultimate disposition of a decommissioned nuclear facility and its surrounding site depends on the degree and type of radioactive and chemical contamination remaining. This section describes and demonstrates suggested methods for determining acceptable radioactive and chemical contamination levels for unrestricted use of the decommissioned reference plant and its site. The basic premise for these suggested methods is that no member of the public will receive an annual radiation dose or chemical exposure in excess of a specific limit (which is yet to be established by U.S. federal agencies).

Examination of existing guidelines and regulations shows that there is a need for these general methods.^(1,2) Currently, some guidance exists defining the levels of radioactive surface contamination that are acceptable to the NRC for the termination of operating licenses.^(3,4) Other guidance addresses specific types of nuclear facilities, or accident situations involving radioactivity.⁽⁵⁻¹⁰⁾ There are currently no regulations or specific guidance on acceptable chemical exposures to the public from a decommissioned nuclear facility. Current

guidance on chemical exposures that could be useful for decommissioning is specifically written for working environments or drinking water.^(11,12)

None of these guidelines is sufficiently flexible to accommodate the various radionuclide mixtures, chemical forms, or site-specific features found at each unique nuclear facility. This suggests that the methods used to calculate the acceptable levels of residual radioactive or chemical contamination should be based on general concepts that can accommodate either generic or site-specific features. The relationship between these calculated acceptable levels and generic or site-specific studies is shown in Figure 8.1-1.

FIGURE 8.1-1. Relationship of Acceptable Contamination Level Analysis Between Generic and Site-Specific Studies

For residual mixtures of radionuclides, the methodology suggested in this study is to compare an established annual dose limit with calculated annual doses to members of the public to determine acceptable radioactive contamination levels. The contamination levels derived from a maximum annual dose concept take into account the exposure of individuals to contamination remaining at a decommissioned facility or on its site following unrestricted release. The NRC has endorsed using an annual dose limit for determining unrestricted release of decommissioned property.^(13,14) In a Federal Register notice, they stated that because of the variety of facilities and radioactive materials, a single dose limit is difficult to establish.⁽¹³⁾ Thus, the basis of their recommendations is that the doses from residual radioactive materials are to be maintained as low as reasonably achievable (ALARA) within a maximum annual whole-body dose equivalent range of less than 10 mrem per year. The NRC stresses that a range of dose values less than 10 mrem per year gives the needed flexibility for an adequate cost and benefit effectiveness analysis. For special situations, a case-by-case analysis, using a radiation dose pathway model, will determine the exact acceptable dose limit.

For residual chemicals, the methodology suggested in this study is to limit chemical exposures, so that no member of the public is exposed to more than 1% of the most restrictive occupational chemical exposure standard. This procedure results in a safety factor of 100, and should account for the most sensitive individuals in a large population. It should also account for unpredicted long-term or environmental behavior that could increase the toxicity of a chemical.

8.1.1 Terminology and Definitions

The following terminology and definitions are used to develop methods for determining acceptable residual radioactive and chemical contamination levels:

Exposure Pathways. The potential routes by which people may be exposed to the effects of residual radioactive or chemical contamination. Exposure pathways in the environment that are considered in this study include: external or direct contact exposure, ingestion of residual contamination, and inhalation of airborne contamination. Exposure pathways inside the UF₆ conversion plant

are: external or direct contact exposure and inhalation of airborne contamination. Air submersion as a radiation exposure pathway is not considered since previous studies have shown it to be insignificant compared to the others.^(1,15,16)

Organs of Reference. These are the organs of the human body for which radiation doses are calculated. For this study, the organs of reference are the total body, lungs, bone, and lower large intestine (LLI) of the GI-tract. The total body is the head and trunk of the human body and includes active blood-forming organs, eye lenses, and gonads.

Maximum-Exposed Individual. This is the individual who receives the maximum radiation dose to an organ of reference. The maximum-exposed individual is assumed to reside at the location of the highest airborne radionuclide concentration. Maximized exposure pathway parameters are used.

Annual Dose. This is the radiation dose equivalent calculated during any year following the start of continuous exposure. It is the sum of the dose received by an organ of reference during the year of interest from all exposure pathways and the dose received during that year from radionuclides deposited in the organ of reference during the previous year.

Maximum Annual Dose. This is the largest of the annual doses calculated to occur during the 50 years following the start of continuous exposure.

Threshold Limit Values (TLV). TLVs refer to the time-weighted air concentrations of chemical substances under which it is believed nearly all workers may be repeatedly exposed "to day after day without adverse effect. TLVs should be used as guides for the control of hazardous substances, and not as "fine lines" between safe and dangerous air concentrations."⁽¹¹⁾

8.1.2 Definition of Use Categories

During the planning stages of decommissioning, a variety of future uses may be considered for the UF₆ conversion plant and its site. Two general use categories after decommissioning are considered in this study:

- Restricted Use - permits activities at the decommissioned UF₆ conversion plant within a nuclear-license restriction. Since this category requires

a continuation of a nuclear license, the residual radioactive and chemical contamination levels may be similar to those found at other licensed operating nuclear facilities. Therefore, public and occupational exposure are controlled by the restrictions imposed by the nuclear license.

- Unrestricted Use - permits, without license restrictions, public use of the released portions of the decommissioned UF_6 conversion plant. In this category, the potential exposure to members of the public from residual radioactive or chemical contamination levels attributable to the decommissioned facility/site will not exceed either an assumed annual radiation dose limit of 50 mrem to the maximum-exposed individual, or 1% of the most restrictive occupational standard on chemicals. Decommissioning a site will, in general, result in the unrestricted public use of land areas which had been restricted during the operational life of the UF_6 conversion plant.

No attempt is made to define all of the possible specific uses that may fall into these two general categories. The ability to enforce the license restrictions required for the first category for long periods of time requires continuous surveillance. Since license restrictions can only be enforced by a government agency, and since it is not reasonable to assume that this government agency will exist for all time, no calculations are attempted for the restricted use category. Example calculations of the acceptable residual radioactive and chemical contamination levels are performed for the unrestricted use category only.

8.1.3 Acceptable Residual Radioactive Contamination Levels

Determination of acceptable radioactive contamination levels for the reference plant is necessarily linked with other decommissioning considerations. Acceptable radioactive contamination levels are calculated using previously developed methods.^(1,15) The method for determining acceptable radioactive contamination levels is based on the assumption that an annual radiation dose limit is established for decommissioned nuclear facilities.

Currently, there are no unique regulations or specific guidance on acceptable annual dose to individuals living on or near a decommissioned site. Guidance that could be interpreted as recommending annual radiation dose limits for decommissioned facilities or sites includes:

- Federal Register, Vol. 46, No. 27, 1981. "Nuclear Regulatory Commission, 10 CFR Parts 30, 40, 50, 70, and 72, Decommissioning Criteria for Nuclear Facilities: Notice of Availability of Draft Environmental Impact Statement" (NRC).⁽¹³⁾
- Recommendations of the International Commission on Radiological Protection (ICRP), Publication 9.⁽¹⁷⁾
- Surgeon General's Guidelines (DHEW).⁽¹⁸⁾
- Appendix I of 10 CFR 50, Guides for Design Objectives for Light-Water-Cooled Nuclear Power Reactors (NRC).⁽¹⁹⁾
- Proposed Federal Guidance for the Environmental Limits of Transuranium Elements (EPA).⁽²⁰⁾
- 40 CFR 190, Environmental Radiation Protection Standards for Normal Operations of Activities in the Uranium Fuel Cycle (EPA).⁽²¹⁾

Most of this guidance provides limits for operating nuclear facilities. Only the NRC Federal Register notice is specifically written to provide an annual dose limit that defines unrestricted release conditions for decommissioned property. However, the annual dose range of less than 10 mrem per year given by the NRC is a recommendation only, and it has not been formally written into the U.S. Code of Federal Regulations.

It is beyond the scope of this study to recommend annual radiation dose limits for public exposure to radioactive materials. Instead, example acceptable residual radioactive contamination levels are calculated for a single assumed annual radiation dose limit of 10 mrem/yr. The selection of this assumed limit is intended to be consistent with current NRC recommendations. Corresponding levels for any other radiation dose limit can be found through direct ratio. The actual levels achieved at nuclear facilities will be determined based on a cost-benefit study for each facility and site. It is also assumed that any annual dose limit established for decommissioning applies to the maximum annual dose to any organ of reference, thus ensuring that applicable regulatory limits on annual radiation dose will not be exceeded.

The method for determining radioactive contamination levels, based on annual radiation dose, is illustrated in Figure 8.1-2 and is briefly discussed below:

- Calculation of the Maximum Annual Radiation Dose for the Use Category Selected. For this study, the maximum annual radiation dose during 50 years of continuous exposure after decommissioning is calculated using the dose models discussed in Appendix E. Characteristic radionuclide inventories at the reference UF_6 conversion plant, used in the calculations, are presented in Appendices B and C. Site-specific exposure pathway parameters, defined for the reference site in Appendix B, are used in these dose calculations. After decommissioning, unrestricted use of the UF_6 conversion plant and its site is assumed.
- Comparison of the Maximum Annual Dose to the Annual Dose Limit. For this study, since assumed or calculated levels of contamination are used, no direct comparison is made. Rather, the quantities of the radionuclide inventories corresponding to a dose of 10 mrem/yr are calculated to demonstrate the suggested methodology both for the facility and for the site.

FIGURE 8.1-2. Suggested Methodology for Determining Acceptable Residual Radioactive Contamination Levels

In site-specific studies that use measured radioactivity levels, this step can be used as a decision point to determine the need for further decontamination efforts.

- Calculation of Acceptable Levels Based on the Assumed Dose Limit. The acceptable radioactive contamination levels in the decommissioned reference UF_6 conversion plant and on its site are calculated and presented in Sections 8.2.1 and 8.2.2. These reported levels are determined by selecting the largest calculated organ dose derived from all exposure pathways. Acceptable contamination levels are reported in units of $\mu Ci/m^2$ for surface area and pCi/g for soil.

8.1.4 Acceptable Residual Chemical Contamination Levels

Analysis and determination of acceptable chemical contamination levels for a UF_6 conversion plant have not previously been conducted. However, the method used in this study is consistent with the relationship developed in Figure 8.1-1, and with the model developed for decommissioning a reference U-Fab plant.⁽²⁾

There are currently no regulations or specific guidance on acceptable chemical exposures to individuals from a decommissioned nuclear facility. Guidance that could be used includes:

- Threshold Limit Values for chemical substances and physical agents in the workroom environment.⁽¹¹⁾
- EPA interim primary and proposed secondary drinking water standards.⁽¹²⁾
- National Institute of Occupational Safety and Health Bulletins, Recommended Standard for Occupational Exposure to Various Chemicals.
- EPA and state environmental standards.

None of these guides provides specific information on public exposures to residual chemicals remaining in decommissioned facilities. This lack of definitive guidance makes it difficult to determine acceptable residual chemical levels. The following factors further complicate the selection process:

- Unlike radioactive contaminants, for which the combined dose to an individual or population serves as a measure of the effect, chemical contaminants must presently be considered individually, with no accepted common index of accumulated dose.
- Once released to the environment, many chemicals undergo reactions which alter their characteristics and effects.
- Some chemical pollutants can cause a number of different effects, depending on factors such as concentration and duration of exposure.
- Additive and synergistic effects of chemical combinations are poorly understood and difficult or impossible to predict.

It is not within the scope of this study to recommend annual chemical exposure limits for the public. The following limits have been assumed for the purpose of determining acceptable residual chemical contamination levels:

- Inhalation--exposures should not exceed 0.01 TLV. The factor 0.01 is a safety factor which accounts for nonoccupational exposures to most susceptible individuals.
- Ingestion--exposures should not exceed the equivalent of the EPA interim primary and secondary drinking water standards.
- Direct Contact--exposures should not cause any dermatitis or burns upon contact with open wounds or skin.

Based on the above criteria, residual contamination limits are determined as follows:

- Establish Chemical Contamination Characteristics. The chemical contamination characteristics of the plant and site are established by appropriate sampling or record review.
- Compute the Maximum Acceptable Residual Chemical Contamination Levels. The maximum acceptable residual chemical levels are established so the exposure to any individual by the most limiting pathway will not exceed the criteria for that pathway.

8.2 CALCULATION OF EXAMPLE ACCEPTABLE RESIDUAL RADIOACTIVE CONTAMINATION LEVELS FOR THE REFERENCE UF₆ CONVERSION PLANT

The methodology for determining acceptable radioactive contamination levels is best demonstrated by calculating example levels for the reference UF₆ conversion plant and site based on the methods described in Section 8.1.3.

8.2.1 Example Acceptable Residual Radioactive Contamination Levels in the Reference UF₆ Conversion Plant

Example contamination levels in the decommissioned reference plant are calculated using the maximum annual dose methodology presented in Appendix E. The surface contamination inventory of radionuclides in the reference plant is presented in Section 7 and Appendix C. Contamination is assumed to accumulate on exposed surfaces over the entire 40-year operating life of the plant. Surface contamination estimates for the reference UF₆ conversion plant in this study are based on a variety of assumptions. In actuality, contamination levels are specific to the facility design and its operating history. Thus, the levels are best determined by measurements on a case-by-case basis for each facility. It is reasonable, however, to predict the isotopic composition of this contamination. Therefore, surface contamination levels for the facility in this study are normalized to 1 pCi/m² for the dose analysis contained in Appendix E. The actual radioactivity levels and isotopic composition encountered in the facility at shutdown are important in determining the degree of decontamination required; however, only the isotopic composition is necessary in determining acceptable levels.

The residual radioactive contamination levels present during decommissioning operations are assumed to be appropriately monitored and recorded. The decommissioning operations discussed in Section 9 and Appendices F and G are designed to remove surface radioactive contamination until the radiation levels are acceptable for unrestricted use of the facility. These levels for the facility are derived in this section based on radioactive surface contamination, with the assumption that all volumetric wastes generated during decommissioning are disposed of as radioactive wastes.

For the maximum annual dose calculations, airborne radionuclide concentrations in the reference UF_6 conversion plant are calculated using a constant resuspension factor of $5 \times 10^{-6} \text{ m}^{-1}$, as discussed in Section E.3 of Appendix E. Results of actual measurements of airborne radionuclide concentrations in decommissioned facilities could alter the acceptable contamination levels calculated here.

The maximum annual radiation doses to people working in the released decommissioned facility are calculated using a 40-hr week and are listed in Tables E.4-1 and E.4-2 of Appendix E for translocation Class W and Y material, respectively. These tables contain doses calculated for selected organs of reference from the inhalation and direct exposure pathways, and for all radionuclides that contribute more than 1% to the total dose. The doses in Tables E.4-1 through E.4-4 apply for specific times after operation shutdown, out to 100 years. The total residual contamination level does not change significantly from radioactive decay and daughter buildup during the first 100 years. Only two radionuclides, ^{231}Pa and ^{230}Th , actually increase with time, but they do not contribute significantly to the total dose. The corresponding contamination levels are next calculated for an annual dose of 10 mrem. These calculated residual contamination levels, expressed in units of microcuries per m^2 of surface contamination ($\mu\text{Ci}/\text{m}^2$), for translocation Class W and Y material are shown in Table 8.2-1. For the total body and other organs of reference, the dominant radionuclide contributors to the dose are ^{234}U , ^{235}U , and ^{238}U .

8.2.2 Example Acceptable Residual Radioactive Contamination Levels on the Reference UF_6 Conversion Plant Site

Information about the levels and nature of the radionuclide contamination present on the reference site is derived in Appendix B. Acceptable levels are calculated based on the estimated 40-year accumulated depositions on the site from routine annual releases during production operation. Airborne concentrations of radionuclides in the plant environs are calculated using the time-dependent resuspension factor discussed in Section E.3 of Appendix E. The radionuclide inventories, showing the 40-year accumulated ground depositions and the values at each selected decay period, are listed in Table 7.7-1 of

TABLE 8.2-1. Example Acceptable Contamination Level Within the Reference
 UF_6 Conversion Plant (unrestricted use)

Organ of Reference	Maximum Year ^(a)	Translocation Class	Surface Contamination ($\mu Ci/m^2$) Corresponding to a Maximum Annual Dose of 10 mrem ^(b)
Total Body	2	W	2.4
	2	Y	4.6
Bone	2	W	0.28
	4	Y	1.1
Lungs	1	W	0.040
	2	Y	0.0092
LLI	1	W	4.6
	1	Y	4.6

(a) The year in which the maximum annual dose occurs following the start of continuous exposure.

(b) Ingestion pathways from contamination in the released decommissioned facility are nonexistent.

Section 7. A more complete discussion of the calculational methods and assumptions used to relate normal plant effluents to deposited surface contamination on the site are found in Section B.5 of Appendix B.

At plant shutdown, these radionuclides are assumed to be mixed to a depth of 10 mm in the soil with no mechanical mixing or weathering effects. If the site contamination levels are measured to be below acceptable levels at the time of decommissioning, plowing is not required as a decommissioning activity. A dry soil "surface density" factor of 224 kg per square meter mixed to a depth of 150 mm (or soil density of 1.49 g/cm^3) is used to determine soil radioactivity concentrations. Example acceptable soil contamination levels are calculated for surface contamination and for soil contamination uniformly mixed to depths of 10 mm and 150 mm.

It should be noted that the contamination levels assumed for the site are probably higher than might actually be encountered at a UF_6 conversion plant. This is primarily because no credit is taken in the calculations for weathering effects on the radioactive contamination either during the 40-year UF_6 conversion plant operating life or during the subsequent decay times. For specific sites, comprehensive measurements will be necessary at the time of production shutdown to characterize the quantity and mixture of the deposited radioactive contamination.

Maximum annual doses calculated using the reference site radionuclide inventory at plant shutdown are listed in Tables E.4-3 and E.4-4, Appendix E. Again, these tables contain the calculated doses for the environmental exposure pathways considered, the organs of reference, and the radionuclides that contribute 1% or more to the total dose. The corresponding residual contamination levels are calculated for an annual dose of 10 mrem and are listed in Table 8.2-2 for Class W and Y material. The dominant radionuclide contributors to the organ doses are ^{234}U , ^{235}U , and ^{238}U .

TABLE 8.2-2. Example Acceptable Residual Contamination Levels on the Reference UF_6 Conversion Plant Site at Plant Shutdown^(a)

Reference	Year (W/Y) ^(b)	Dominant Radionuclide Contributors To Dose	Acceptable Residual Radioactive Surface Contamination Levels ($\mu Ci/m^2$) ^(c)		Acceptable Soil Contamination Levels			
			Class W	Class Y	Mixed to 10 mm		Mixed to 0.15 m	
					Class W (pCi/g)	Class Y (pCi/g)	Class W (pCi/g)	Class Y (pCi/g)
Total Body	3/50	$^{238}U+D$	30	30	2.0×10^3	2.0×10^3	1.3×10^2	1.3×10^2
Bone	50/50	^{234}U	2.6	2.8	1.7×10^2	1.9×10^2	8.8×10^0	1.3×10^1
Lungs	1/1	$^{238}U+D$	11	2.4	7.6×10^2	1.6×10^2	5.0×10^1	6.8×10^0
LLI	1/1	$^{238}U+D$	16	16	1.0×10^3	1.0×10^3	7.0×10^1	7.0×10^1

(a) Corresponding to an annual dose rate of 10 mrem/yr to specific organs of reference.

(b) The year in which the maximum annual dose occurs following the start of continuous exposure for Class W material and Class Y material (W/Y).

(c) These values can be compared to the acceptable contamination levels listed in NUREG/CR-1266 for a uranium fuel fabrication plant by dividing the values mixed to 0.15 m (in pCi/g) shown in this table by 15.⁽²⁾

(d) The +D notation indicates that after deposition in the organ of reference, the decay energy of the daughter is included with the parent.

8.2.3 Summary of Example Acceptable Residual Radioactive Contamination Levels

The example acceptable residual contamination levels calculated in this study are summarized in Table 8.2-3. For the facility, the inhalation pathway dominates and the lungs are the critical organ receiving the most dose and

TABLE 8.2-3. Summary of Example Acceptable Residual Radioactive Contamination Levels for the Reference UF₆ Conversion Plant and its Sites

			Acceptable Residual Contamination Levels for an Annual Dose of 10 mrem ^(a)		
	Translocation Class	Critical Organ	Surface Contamination ($\mu\text{Ci}/\text{m}^2$)	Soil Contamination Mixed To:	
				10 mm (pCi/g)	150 mm (pCi/g)
Facility	W	Lungs	0.040	-- ^(b)	--
	Y	Lungs	0.0092	--	--
Site	W	Bone	2.6	170	13
	Y	Lungs	2.8	160	6.8

(a) Based on the natural uranium mixtures shown in Table 7.4-1 for the facility and Table 7.1-1 for the site.

(b) A dash indicates that soil contamination levels are not calculated for surface contamination inside the facility.

yielding the most restrictive acceptable contamination levels. For the site, the dominant exposure pathways are ingestion of farm products for Class W material and inhalation of resuspended radioactivity for Class Y material. The critical organs are bone and lungs for Class W and Class Y material, respectively. It should be noted that, because of the time dependent resuspension factor in the Anspaugh⁽²²⁾ model (see Appendix E), the dominant exposure pathway for Class Y material changes from inhalation to ingestion, the critical organ changes from lungs to bone, and the acceptable contamination levels approach those calculated for Class W material. These changes occur because the resuspension factor in the Anspaugh model decreases with time.

8.2.4 Acceptable Contamination Levels on UF₆ Conversion Plant Equipment

The problems associated with releasing equipment used in the nuclear industry for unrestricted public use have been addressed in two recent studies.^(23,24) The first of these studies describes a generic methodology for estimating potential radiation doses to man from recycling radioactively contaminated metals reclaimed during decommissioning a nuclear power plant.⁽²³⁾ The methodology is demonstrated for 27 radionuclides from six recycle pathways with a contamination level of 10 pCi/g. The results in Reference 23 are for a generic case and several key assumptions are made to obtain radiation dose estimates to exposed population groups. However, the methodology presented in Reference 23 should be useful in determining acceptable radioactive contamination levels for the release of contaminated equipment.

The second study is a draft environmental impact statement concerning a possible exemption from licensing of smelted alloys containing residual ⁹⁹Tc and low-enriched uranium.⁽²⁴⁾ Exposure pathways are considered for the recycle of scrap copper, nickel, and iron resulting from improvement of the Department of Energy's gaseous diffusion plants. The primary environmental impacts considered are the doses to foundry and metal fabrication workers, and the doses to members of the public who use items made from the recycled metal. The results obtained in Reference 24 are for six exposure scenarios and indicate that less than one health effect could be expected for recycle of the materials selected.

Release of much of the equipment recovered from the UF₆ conversion plant after decontamination could be covered by standards developed by the ANSI Committee N13.12;⁽¹⁰⁾ however, the NRC has not yet endorsed this standard. The complexities of decontaminating equipment for public release are great and are briefly discussed in Appendix H. Because decommissioning an actual UF₆ conversion plant requires special procedures to release equipment on a piece-by-piece basis, no further effort is made in this report to analyze equipment-release conditions.

8.3 CALCULATION OF EXAMPLE ACCEPTABLE RESIDUAL CHEMICAL CONTAMINATION LEVELS

The methodology for developing chemical contamination levels is based on human health impact potential and compliance with environmental standards. Example acceptable chemical contamination levels for the decommissioned reference plant are calculated using methodology discussed in Section 8.1.4. Contamination is assumed to accumulate both in the facility and on the site over the entire 40-yr operating plant lifetime. The residual contamination levels present during decommissioning are assumed to be known from sensitive analytical techniques similar to those used for occupational exposures to dusts, vapors, and gases. It should be noted that although specifically directed towards removal of radionuclides, the decommissioning tasks discussed in Section 9 and Appendices F and G will also remove or reduce chemical contamination levels.

Because of complex chemical-surface interactions, it is difficult to estimate the concentration of residual chemicals in the air above contaminated surfaces. Thus, no attempt is made in this study to consider diffusion or evaporation as driving forces in the concentration calculations. Instead, a simple resuspension analysis, similar to the analysis done for radionuclides in a facility, is used to calculate example chemical concentrations in air. However, it should be remembered that for some chemicals a diffusion analysis may be more adequate when determining acceptable surface contamination levels in a decommissioned facility. Results of actual measurements in a decommissioned facility could also significantly alter the example chemical contamination levels calculated in this study.

Table 8.3-1 shows the example acceptable surface contamination levels calculated to be in the reference UF₆ conversion plant, based on a resuspension factor of $5 \times 10^{-6} \text{ m}^{-1}$. The data in this table indicate relatively large allowable levels of surface contamination resulting in air concentrations that are a factor of 0.01 below the TLV. The chemicals that are considered in Table 8.3-1 are shown in Tables 7.2-2 and 7.2-3, and are discussed in Appendix B.

An appropriate ingestion exposure pathway must be defined before acceptable chemical surface contamination levels can be developed based on direct ingestion.

TABLE 8.3-1. Example Acceptable Chemical Contamination Levels in the Reference UF₆ Conversion Plant (unrestricted use)--Inhalation, Airborne Pathway

Chemical	TLV (mg/m ³)	Allowable Maximum Surface Contamination (g/m ²) ^(a)
F ^(b)	2.5	5.0
HF	2.0	4.0
SO ₂	13	26
NH ₃	18	36
H ₂ S	15	30
Hexane	360	720
NO ₂	9	18
Uranium (insoluble)	0.2	0.4
Uranium (soluble)	0.2	0.4
H ₂ SO ₄	1.0	2.0 ^(c)
HNO ₃	30	6.0 ^(c)
Nitrate	High	>50
Sulfate	High	>50

- (a) Corresponding to 1% of the TLV as a product of the resuspension factor of $5 \times 10^{-6} \text{ m}^{-1}$.
 (b) Fluoride limit is for inorganic fluoride; no specific limit for CaF₂ is given.
 (c) Not expected to be present as a surface contaminant over any appreciable surface area.

The EPA drinking water standards provide a basis for fluoride, nitrate, and sulfate limits. Acceptable daily ingestion of these three chemicals is assumed to be the amount consumed by an individual drinking 1.2 l/day of water containing the EPA maximum level. Acceptable daily ingestion rates for other chemicals, for which no drinking water limit is published, are found based on an analysis using TLVs. For this analysis, it is assumed that an individual will ingest surface contamination from $8 \times 10^{-4} \text{ m}^2$ of surface during an 8-hour day.⁽²⁵⁾ This amount of ingestion agrees with the assumptions made by Healy in References 25 for the ingestion of removable radioactive surface contamination. It

is further assumed that this amount of ingestion equals the amount the individual would inhale during a 24-hour day (at the rate of $1 \text{ m}^3/\text{hr}$) at an air concentration of 1% of the TLV. The acceptable surface contamination levels calculated for ingestion are intended to be "example" levels showing how to apply the calculational methods, and are not intended to reflect the specific conditions found at an existing facility. The example acceptable chemical surface contamination levels calculated for ingestion are shown in Table 8.3-2.

TABLE 8.3-2. Example Acceptable Chemical Contamination Levels Within the Reference Facility (unrestricted use)--Ingestion Pathway

Chemical	Maximum Acceptable Daily Ingestion (mg/day)	Estimated Acceptable Surface Contamination (g/m^2)(a)
F(b)	2.9(c)	3.6
HF	0.48(d)	0.60
SO ₂	3.1(d)	3.9
NH ₃	4.3(d)	5.4
H ₂ S	3.6(d)	4.5
Hexane	86(d)	110
NO ₂	2.2(d)	2.7
Uranium (insoluble)	0.05(d)	0.06
Uranium (soluble)	0.05(d)	0.06
H ₂ SO ₄	0.24(e)	0.30
HNO ₃	7.2(e)	9.0
Nitrate	12(c)	15
Sulfate	300(c)	380

(a) Assumes a person ingests the contamination from $8 \times 10^{-4} \text{ m}^2$ of surface per day.

(b) Fluoride limit is for inorganic fluoride; no specific limit for CaF₂ is given.

(c) Based on EPA interim primary and proposed primary drinking water standards and a daily water consumption of 1.2 l.

(d) Based on the amount inhaled by a person breathing $1 \text{ m}^3/\text{hr}$ at 1% of TLV for 8-hr/day.

(e) Not expected to be present as a surface contaminant over any appreciable surface area.

Using this method, the acceptable concentrations of chemical contaminants appear to be limited by the ingestion pathway calculations. It should be emphasized that the levels calculated here could change significantly if a different resuspension factor, a diffusion calculation for air concentrations, or different assumptions for ingestion are used. Thus, site-specific information will help determine the magnitude of the chemical contamination problem and will also influence the calculated acceptable contamination levels.

8.4 EXISTING GUIDANCE ON ACCEPTABLE RESIDUAL RADIOACTIVE CONTAMINATION LEVELS

Existing guidance on acceptable radioactive contamination levels for the unrestricted release of decommissioned nuclear reactors is found in Regulatory Guide 1.86,⁽²⁶⁾ in the draft ANSI Standard N13.12,⁽¹⁰⁾ and in a Federal Register notice by the NRC.⁽¹³⁾ The levels reflected in References 26 and 10 are listed in Tables 8.4-1 and 8.4-2. Since the radiation dose basis of 10 mrem/yr and calculations presented in this study are different from the instrumentation capability basis used in Reference 26 and 10, the resulting contamination level

TABLE 8.4-1. Regulatory Guide 1.86 Acceptable Surface Contamination Levels

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

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

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

(c)Measurements of average contaminant should not be averaged over more than 1 m². For objects of less surface area, the average should be derived for each object.

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

(e)The amount of removable radioactive material per 100 cm² of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, and assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface wiped.

TABLE 8.4-2. ANSI N13.12 Surface Contamination Limits

Radionuclide ^(a)	Activity Limit (dpm/100 cm ²)	
	Total	Removable
<u>Group 1:</u>		
Nuclides for which the nonoccupational MPC _a ^(b) is 2 x 10 ⁻¹³ Ci/m ³ or less or for which the nonoccupational MPC _w ^(c) is 2 x 10 ⁻⁷ Ci/m ³ or less; includes Ac-227; Am-241, -242m, -243; Cf-249, -250, -251, -252; Cm-243, -244, -245, -246, -247, -248; I-125, I-129; Np-237; Pa-231; Pb-210; Pu-238, -239, -240, -242, -244; Ra-226, -228; Th-228, -230.	Nondetectable ^(d)	20
<u>Group 2:</u>		
Those nuclides not in Group 1 for which the nonoccupational MPC _a is 1 x 10 ⁻¹² Ci/m ³ or for which the nonoccupational MPC _w is 1 x 10 ⁻⁶ Ci/m ³ or less; includes Es-254; Fm-256; I-126, -131, -133; Po-210; Ra-223; Sr-90; Th-232; U-232.	Nondetectable ^(B,Y) ^(e) 2 000 ^(α)	200
<u>Group 3:</u>		
Those nuclides not in Group 1 or Group 2.	5 000	1 000

- (a) Values presented here are obtained from 10 CFR Part 20. The most limiting of all given MPC values (e.g., soluble vs. insoluble) are to be used. In the event of the occurrence of mixtures of radionuclides, the fraction contributed by each constituent of its own limit shall be determined and the sum of the fractions must be less than 1.
- (b) MPC_a: maximum permissible concentration in air applicable to continuous exposure of members of the public as published by or derived from an authoritative source such as NCRP, ICRP or NRC (10 CFR Part 20 Appendix B Table 2, Column 1).
- (c) MPC_w: maximum permissible concentration in water applicable to members of the public.
- (d) The instrument utilized for this measurement shall be calibrated to measure at least 100 pCi of any Group-1 contaminants uniformly spread over 100 cm².
- (e) The instrument utilized for this measurement shall be calibrated to measure at least 1 nCi of any Group-2 beta or gamma contaminants uniformly spread over an area equivalent to the sensitive area of the detector. NOTE: Direct survey for unconditional release should be performed in areas where the background is <100 c/m. When the survey must be performed in a background exceeding 100 c/m, it may be necessary to use the indirect survey method to provide the additional sensitivity required.

criteria do not directly compare and require analysis. Using the maximum annual dose as the general basis for determining acceptable radioactive contamination levels permits the necessary flexibility for considering the various radionuclide mixtures expected at decommissioned nuclear facilities.

8.5 RESIDUAL CONTAMINATION LEVEL MEASUREMENTS

After final inventory cleanout is completed, residual contamination levels in the reference plant must be measured to determine the amount of effort required to decommission the facility and site. Methods of measurement are discussed in this section.

8.5.1 Radioactivity Measurements

The capability of measuring the maximum acceptable residual contamination levels in Table 8.2-3 is dependent on the sensitivity of the instrumentation utilized and the time available for surveying. The latest scintillation alpha survey meters have the capability to detect as low as 50 d/m per probe area. This detection level corresponds to a contamination level of $0.004 \mu\text{Ci}/\text{m}^2$. The most restrictive acceptable levels for the reference facility shown in Table 8.2-3 is $0.046 \mu\text{Ci}/\text{m}^2$, or about 1,000 d/m per 100 cm^2 . Thus, measurement of activity levels for the reference facility corresponding to an annual dose of 50 mrem is possible using survey instrumentation available commercially.

Inside generally contaminated spaces, in the presence of large contaminated equipment items, or over large generally contaminated surfaces, it may be necessary to use indirect survey methods to measure radiation levels. On hard nonporous surfaces, smears or scrapings may be taken and removed for analysis to a lower-background location or a low-level laboratory counting instrument.

The acceptable surface contamination levels shown in Table 8.2-3 imply that the history of the material or the mixture of radionuclides being measured is known. Sampling for laboratory identification measurements and specific radionuclide contamination levels are desirable even when the characteristics of the contamination are known. Sampling is absolutely essential when such preliminary information is lacking.

Sampling techniques for bulk materials such as soils have many variations. Practicality limits the fraction of any large area that can be sampled and analyzed. A fixed scheme is needed for selecting sampling stations and the number, size, and spacing of sample aliquots at each location. A fixed sampling scheme is desirable not only for appropriate statistical inferences but also for reproducibility and comparability. For soil, variability of overlying vegetation and inclusion of rock and gravel is a problem. Regulatory Guide 4.5⁽²⁷⁾ provides one commonly used scheme that is generally applicable for soil sampling. Adequate sampling of bulk materials requires sampling to depths of 0.3 to 1 m in soil, depending on climate and history.

There is no commonly accepted procedure for translation of surface concentration limits to mass contamination limits or vice versa. However, with reasonable assumptions as to soil bulk density and the volume of soil seen by a portable alpha probe, the value of $12 \mu\text{Ci}/\text{m}^2$, shown in Table 8.2-3, translates to approximately 34,000 pCi/kg, or $\sim 10^3$ times the lower limit of detection (LLD) for laboratory analysis given in Table 8.5-1.^(28,29) For all radionuclides in environmental media, sample radioanalysis can provide the sensitivity required by any of the proposed limits in Section 8.4 or the example radioactive surface contamination levels in Table 8.2-3. The cost will depend on whether chemical separation is required and on the length of counting time needed to measure a particular radionuclide at a given level above instrumental and sample background.

TABLE 8.5-1. Detection Capabilities for Environmental Sample Analysis^(a)

Analysis	Lower Limit of Detection (LLD) ^(b)			
	Water (pCi/l)	Vegetation (pCi/kg, wet)	Soil (pCi/kg, dry)	Air (pCi/m ³) ^(c)
²³⁵ U ^(d)	2	50	30	$6 \times 10^{-5} \mu\text{g}/\text{m}^3$
²³⁸ U ^(d)	2	50	30	$6 \times 10^{-5} \mu\text{g}/\text{m}^3$

- (a) This table is intended to be comparable to a similar table in Regulatory Guide 4.8 reflecting current experience at a commercial radioanalytical laboratory.⁽²⁸⁾
- (b) The normal lower limit of detection is defined in HASL 300, Appendix D, Rev. 8/74, at the 95% confidence level.⁽²⁹⁾ The LLD for radionuclides analyzed by gamma spectrometry will vary according to the number of radionuclides encountered in environmental samples.
- (c) LLD based on 300 m³ sample volume and alpha proportional counter.
- (d) Fluorometric analysis.

Scintillation alpha survey meters are capable of measuring the radioactive contamination levels presented in Table 8.2-3. Sampling of soil, vegetation and ground water would be required to assure compliance with acceptable levels. These samples would be collected based on a statistical design to assure that acceptable levels are met.⁽³⁰⁾ Analytical procedures involve chemical

concentration and separation of the nuclide of interest. Detecting the activity of the concentrated samples is done with gas flow proportional counters or other systems with similar or superior detection capabilities.

Table 8.5-2 summarizes relative advantages and disadvantages for common methods of determining radioactive contamination levels. Further discussion of instrument capabilities may be found in LBL-1.⁽³¹⁾ Further discussion of

TABLE 8.5-2. Comparison of Measurement Methods for Radiation Surveys to Permit Unrestricted Use

Measurement Method	Advantages	Disadvantages
Direct		
Portable Instruments ^(a)	Relatively fast; Relatively inexpensive; Readily available; Able to delineate hot spots.	Limited sensitivity; usually not nuclide-specific; Subject to interferences from high background and surface conditions; For alpha and beta emitters, useful for exposed surfaces only.
Aerial Survey	Extremely fast.	Useful in general for gamma emitters only; ^(b) Insensitive to small areas.
Indirect		
Smears, Scrapings	Avoidance of high background interference.	Not indicative of total activity present, highly variable results; Incomplete coverage of large surfaces; Not applicable to loose or confined materials.
• with direct field count	Relatively fast; Relatively inexpensive.	Not nuclide-specific.
• with laboratory counting	Nuclide identification possible (but more expensive); Greater sensitivity than direct field count.	Relatively slow and expensive.
Sampling and Laboratory Analysis		
Analysis	Nuclide-specific; Highly sensitive.	Relatively slow; Relatively expensive; Applicable only when sample of material can be taken to laboratory; Provides data for only small part of total surface.

(a) See Table 8.5-1 for typical examples and detection levels.

(b) With special calibrations, aerial surveys may be useful for large areas, but not to release levels specified in Section 8.4.

environmental survey techniques may be found in ERDA-77-24⁽³²⁾ and NCRP Report No. 50,⁽³³⁾ as well as in the health physics literature.

8.5.2 Chemical Contamination Measurements

Virtually all of the chemical contaminants would require laboratory analysis to provide detection capability of all the chemical contaminants at or below the unrestricted release criterion. Such methods are currently available and should be considered as part of the routine monitoring done prior to release of the decommissioned UF₆ conversion plant.

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9.0 DECOMMISSIONING ACTIVITIES

As discussed in Section 4, two alternatives for decommissioning the reference uranium hexafluoride conversion (UF_6) plant are selected for evaluation in this study: 1) DECON, and 2) passive SAFSTOR. An outline of the major components of a Master Decommissioning Plan (MDP) for both of these alternatives is presented in this section, together with discussions of the major work items in each plan. In addition, decommissioning activities for onsite disposal of the UF_6 plant lagoon waste is summarized in this section.

9.1 DECOMMISSIONING BY DECON

In choosing to decommission a UF_6 conversion plant by the DECON alternative, the owner trades potential further use of the plant for near-term release of the site for unrestricted use. The program plan and the postulated work schedules and sequences for decontamination, together with a brief discussion of essential systems and services and security, are presented in the following subsections. Details specific to the DECON decommissioning alternative are found in Section G.2 of Appendix G.

9.1.1 Program Plan

Decommissioning of the reference UF_6 conversion plant by the DECON alternative involves four phases: planning and preparation, dismantlement and decontamination, transportation, and the final release survey. Some of these activities will proceed simultaneously in different sections of the facility. To minimize scheduling conflicts and accidents, a well-defined sequence and schedule for dismantlement and decontamination of the various portions of the plant must be created and followed carefully. The major activities considered and scheduled for decontaminating the facility are illustrated in Figure 9.1-1.

The 6 months prior to final plant shutdown are used for planning and preparation. It is estimated that approximately 8 months are required to dismantle and decontaminate the facility and site. Uranium and contaminated materials are removed and transported in parallel. The time and work estimates

FIGURE 9.1-1. Sequence and Schedule of Major Activities for Decommissioning by DECON

assume reasonable success, with a minimum of delays and/or major unanticipated problems. A detailed schedule for the DECON phase is shown in Figure G.2-1 of Appendix G.

An extensive quality assurance (QA) program is carried on throughout the decommissioning effort to assure that: 1) all applicable regulations are met, 2) the work is performed according to plan, and 3) the associated radiation releases do not endanger public and decommissioning worker safety. A more detailed review of the anticipated elements of an appropriate QA program for the decontamination effort is given in Section F.4 of Appendix F.

9.1.1.1 Planning and Preparation

Essential to the results of this study is the assumption that the facility owner/operator becomes the prime contractor of the DECON work; otherwise, a more extensive training program would be necessary to acquaint workers with details of the facility. Approximately 6 months prior to final plant shutdown

work begins in the engineering and operations departments of the parent organization to perform the planning needed to amend the license to permit DECON. The proposed sequence and timing schedule for the planning and preparation phase of decommissioning the facility and site is illustrated in Figure 9.1-2.

An important part of the planning involves a review of all regulations and guides, which may provide guidance to the decommissioning planning. A review of these regulations is presented in Section 5.

Important items to consider in planning are: preparation and submittal of a decommissioning plan for NRC review and approval, preparation of detailed plans and procedures for dismantlement and decontamination of intact systems, detailed sequences for equipment and systems removal, and sectioning and disposal of contaminated equipment. In addition, design, procurement, and testing of

FIGURE 9.1-2. Sequence and Schedule of the Planning and Preparation Phase of DECON

special devices and equipment must be initiated during the 6-month period before final plant shutdown to ensure that work can proceed without undue delay after shutdown.

Creation of a decommissioning organization within the present organization is initiated about 6 months prior to final plant shutdown, with the structure and staffing requirements identified, and commitments obtained from engineering and operating personnel to fill key positions. Orientation and training of personnel identified as members of the decommissioning organization is carried on during the last few months of plant operation. A suggested organizational structure and staffing requirements are given in Section 10.

Selection of the various specialty contractors required for the DECON effort is accomplished during the final few months of plant operation. The types of specialty contractors anticipated to be required are listed in Section 10.1.5.

After termination of routine plant production operations, an extensive inventory cleanout and uranium audit is conducted. These cleanout operations are slightly more extensive than those conducted periodically to audit uranium. Based on plant experience, cleanout operations are estimated to remove one-half of the residual uranium plant inventory. Because these cleanouts are done typically during plant production, they are also considered a part of normal plant operations in this study and are not charged to decommissioning. All recovered uranium products are shipped to offsite locations. Details of the final inventory cleanout and uranium audit are found in Section G.1 of Appendix G.

The final preparatory step is a comprehensive survey of radiation dose rates and contamination levels within the facility. Taken after final plant shutdown and inventory cleanout, this survey provides the baseline data for decisions on chemical and physical decontamination, as well as initial data on radiation dose rates and contamination levels likely to be encountered during the various DECON activities. Physical decommissioning is assumed to start upon completion of the survey and receipt of NRC approval for the decommissioning.

9.1.1.2 Dismantlement and Decontamination

The dismantlement and decontamination phase of the DECON program was summarized previously in Figure 9.1-1. The detailed schedule is provided in Figure G.2-1 of Appendix G. Detailed descriptions of the activities are also contained in Appendix G.

The general approach is to clean up the areas that contain the largest amount of uranium contamination first. The UF_6 conversion, solvent extraction, and waste treatment areas are cleaned up in parallel. Within the UF_6 conversion area, work progresses from sampling and preparation of yellowcake through the conversion process to fluorination of the uranium into the UF_6 product.

Laboratories, the fluoride separation facilities, and other auxiliary facilities are cleaned up near the end of the campaign because their contamination levels are very low.

Dismantlement of the uranium recovery areas (incinerator, temporary radwaste facility, etc.) is saved until the bulk of the site decontamination effort is completed. Also, the other service facilities, such as the temporary decontamination room, are dismantled near the end of the program.

The temporary radwaste treatment facility is the last system to be dismantled, so it can be utilized throughout the program.

The final areas to be dismantled and decontaminated are the laundry and the change rooms.

9.1.1.3 Transportation

The transportation phase of the DECON alternative is initiated during the inventory cleanout stage and is continued throughout all the physical activities. All uranium and radioactively contaminated materials or equipment (most equipment is decontaminated onsite) are transported offsite for disposition at licensed low-level waste burial sites.

Packaging of contaminated materials for disposal is accomplished in accordance with DOT regulations published in 49 CFR, Parts 173 through 178, NRC regulations published in 10 CFR, Part 71, and Regulatory Guide 7.1. Shipping of packaged contaminated materials from the facility to a low-level waste

disposal site is accomplished using trucking companies that are licensed to transport special materials. The volume of these materials to be transported and the number of shipments required are estimated in Appendix H, and costs are summarized in Table H.2-5.

9.1.1.4 Waste Management

Radioactive wastes generated during DECON must be packaged and shipped to a low-level waste burial ground. The radioactive wastes generated include: process and facility equipment and components, pipe, concrete rubble, filters, trash, decontamination liquids, miscellaneous waste buried onsite, and sludge and material from the waste lagoons.

About 1260 m³ of the radioactive waste from the main process building and the solvent extraction building is assumed to be packaged and shipped to low-level waste burial. About 1870 m³ of equipment from the buildings and site are decontaminated and excessed or sent to commercial waste burial.

The solvent extraction lagoon waste (21,600 m³) is removed for offsite disposal and the fluoride lagoon waste (5700 m³) is either buried onsite or removed for disposal offsite. Handling of the lagoon wastes is discussed in Section 9.1.1.5.

Process and Facility Equipment and Piping. Equipment and piping from the process system and the facility is decontaminated and disassembled for shipment. The equipment is packaged in plywood boxes for disposal.

Concrete Rubble. Concrete rubble from decontamination of floors and walls in the main and solvent extraction buildings is packaged in plywood boxes for shipment to an LLW burial ground.

Filters. Roughing filters and HEPA filters from building air ventilation systems and temporary decommissioning tents are packaged for disposal in steel boxes.

Solidified Decontamination Liquids. Decontamination liquids are assumed to be processed by evaporation, in the temporary radwaste system, with the residues solidified with cement. Solidified decontamination liquids are packaged for disposal in 0.21-m³ steel drums.

Trash. Trash is essentially all other kinds of radioactive waste that is generated. It consists of contaminated equipment, tools, rags, wipes, etc. All trash is packaged for disposal in 0.21-m³ steel drums.

9.1.1.5 Lagoon Waste Management

Sludges from the solvent waste extraction and fluoride lagoons must be disposed of for UF₆ plant decommissioning. Several alternative methods of disposal for the lagoon wastes are available. These are: 1) onsite burial, 2) removal of waste to a uranium mill for recovery of residual uranium, 3) removal of waste to an LLW burial ground, and 4) removal of waste to a chemical or commercial waste dump.

Onsite Burial. The solvent extraction waste sludge has a uranium concentration of about 0.28 wt% uranium, which puts it at levels slightly higher than the levels in uranium mill tailings. For the DECON alternative, the extraction wastes must be removed from the site to allow unrestricted release.

The reference plant generates fluoride waste that is nearly insoluble and so low in radioactive material that it is barely distinguishable from ordinary waste containing naturally occurring radioactive materials. These wastes are assumed to be buried onsite for the DECON alternative.

Removal of Waste to a Uranium Mill. For the reference plant, the extraction waste (21,600 m³) is assumed to be processed by a milling contractor to recover the residual uranium. The uranium concentration in the extraction waste for the reference plant is about 0.28%, which is about the level of uranium concentrates in ore being currently mined.

The decision to recover the uranium would be based on an economic study to determine whether the cost of shipping and recovery less the value of the recovered uranium would be less than the cost of burial. Based on current prices for uranium and the inflationary trends in yellowcake cost, it is expected that the economic value of the recovered uranium would make it cost effective to process the extraction waste.

The fluoride waste does not contain sufficient uranium to warrant recovery and that option is not included in this analysis.

Removal of Waste to an LLW Burial Ground. If the extraction waste is not processed to remove the residual uranium, it would have to be disposed of at a low-level waste burial ground at the time of decommissioning. The waste material would be loaded into plywood boxes and shipped to the nearest LLW disposal site.

If fluoride waste cannot be stored onsite it would be shipped to LLW burial. The waste would be shipped in plastic-lined, 10-m³ dump trucks to the nearest LLW burial ground.

Removal of Waste to a Commercial Waste Dump. A potential alternative to low-level waste disposal of the extraction waste would be reduction of the residual activity levels by dilution and storage onsite or shipment to a chemical or commercial waste dump. There is presently no clear definition of what constitutes acceptable levels of uranium in bulk waste.

Several proposals and requests for acceptable values for contamination in bulk materials (CaF₂, NO₃ waste) have been made to the NRC, but no specific regulation has yet been developed. Permission has been given in specific cases to discharge to the environment liquid from CaF₂ and NO₃ wastes containing low levels (2 to 5 ppm) of uranium.^(a,b,c) However, no general limit exists as, for example, in transportation regulations (49 CFR 173.389), where radioactive material is defined as "material in which the estimated specific activity is greater than or equal to 0.002 microcuries per gram."^(d)

Lacking a definitive regulatory lower limit value for uranium in solid waste, for this report, estimates of residual levels of uranium in the extraction waste that would permit disposal as nonradioactive waste are based on

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- (a) NRC License #SNM 1097, Section 1.6.4 of Appendix A. (General Electric Co., Wilmington, NC, plant).
 - (b) Westinghouse Licensing Correspondence, DOCKET #70-1151, August 10, 1979, letter to NRC.
 - (c) Kerr McGee Licensing Correspondence, Docket #40-8027, March 3, 1978, letter to NRC.
 - (d) Westinghouse SNM License #1107, Condition #13 specifically prohibits use of the 49 CFR 173.389 cutoff level (0.002 μ Ci/g) if the activity is a result of contamination with SNM.

guidance developed in the National Environmental Studies Project report AIF/NESP-016 "de minimis Concentrations of Radionuclides in Solid Wastes."⁽¹⁾ Based on a maximum total body dose of 1 mrem/yr, the acceptable level of activity for natural uranium was found to be about 100 picocuries per gram of waste material. At 0.28% uranium content, the activity level of the extraction waste and uranium mixture would be about 920 pCi/g of waste material. Thus, a dilution factor of about 9 parts soil to 1 part extraction waste would be required to satisfy the limit based on Reference 1. The residual uranium concentration in the waste that corresponds to this lower limit would be about 310 ppm, compared to 2800 ppm for the waste stored at the reference UF_6 conversion plant.

Since the uranium concentration is low, burial of fluoride waste in a commercial waste dump would be governed by regulations of the state in which the UF_6 plant is located. The question of chemical or commercial waste burial of extraction and fluoride wastes would also be determined by the regulations of the state where the UF_6 conversion plant is located.

9.1.1.6 Final Release Survey

The final task is to perform a radiation and chemical survey of the UF_6 plant, the auxiliary facilities, and the entire site. Any remaining spots of radioactivity and chemical contamination higher than the amount allowable for unrestricted use are removed and the contaminated materials are disposed of at a licensed low-level waste burial site.

This concludes the DECON responsibilities of the site owner. The NRC then audits the project and sponsors an overcheck survey. In the event of a discrepancy, the owner provides the necessary corrective action.

The NRC then terminates the nuclear license, releases the site for unrestricted use, and discontinues their surveillance and responsibilities at the site. The owner is then free to use the site for unrestricted applications.

9.1.2 Essential Systems and Services

Some of the facility systems and services must remain in place until all radioactive and/or contaminated materials are removed from the site to

ensure that no significant amounts of radioactive or hazardous materials are released to the environs. In addition, some of the systems are needed to facilitate the cleanup and disassembly efforts. As areas within the facility are readied for unrestricted use, the extensions of services into those areas are deactivated and removed, while maintaining continuity of the services to the remaining work areas. The required support systems, together with the justification for retaining each system, are listed in Table 9.1-1.

TABLE 9.1-1. Systems and Services Required During DECON

Systems or Components	Justification
Electrical power, including emergency diesel backup system	Required for HVAC, lighting and radiation monitoring
HVAC Systems	Required for ventilation and contamination confinement
Environmental Surveillance and Safeguards Program	Required to identify and quantify any releases of radioactivity to the environs from dismantlement activities and to identify and safeguard any significant quantities of uranium discovered during dismantlement
Water Supply (service and domestic systems)	Required for decontamination, clean up, fire protection, and general potable water usage
Fire Protection System (direction and suppression)	Required for health and safety
Compressed Air Systems (control and suppression)	Required for operation of pneumatic controls, for operation of pneumatically operated tools
In-Plant Communications Systems (telephones and intercoms)	Required to facilitate and coordinate activities
Radiation Monitoring Systems	Required for protection of personnel
Solid and Liquid Contaminated Waste Systems	Required for treatment and disposal of potentially contaminated liquids and solids
Clean Scrap and Dirty Scrap Recovery Systems	Required for recovery of uranium from liquid, solid non-combustible, and incinerated wastes
Sewage Treatment Plant (septic tanks and sewage lagoon)	Required for sewage treatment

Essentially the same environmental monitoring program carried on during plant operation is continued during the DECON period. This program is to identify and quantify any releases of radioactivity to the surrounding areas resulting from DECON activities. The proposed program, detailed in Section F.5 of Appendix F, is sufficient to permit evaluation of any significant releases. Additional short-term surveillance efforts may be added for emergency situations involving radionuclide releases from events such as fires or malicious acts that may necessitate prompt emergency action.

9.1.3 Security

Protection of the public (often against the consequences of their own actions) is an important dimension of the security program throughout the decommissioning effort. Security during decommissioning is assumed in this study to be similar to, but less stringent than, that needed during plant production operations.

9.2 DECOMMISSIONING BY PASSIVE SAFSTOR

The goal of the passive SAFSTOR alternative is to achieve a condition that ensures that residual radioactivity is kept confined to the UF_6 conversion plant site. Modifications to the facility are limited to those that ensure the security of the buildings and to those required to ensure containment of radioactive material. The passive SAFSTOR alternative allows deferral of the decision regarding final disposition until no further use is found for the plant. To achieve this goal, the facility is left structurally sound. All loose contamination in readily accessible locations is removed. Hoods and equipment are vacuumed and handwiped. Access into contaminated areas is limited. All systems and equipment not required to be in operation during the safe storage period of passive SAFSTOR are deactivated. The preparations for safe storage and the period of safe storage that follows should be recognized as only temporary stages in the total decommissioning process. Current NRC philosophy encourages a decommissioning approach that ends in the termination of the plant's nuclear license and the release of the property for unrestricted use within a finite period of time. Thus, decontamination to unrestricted levels is required eventually.

The major benefits gained from decommissioning the facility by the passive SAFSTOR alternative are: 1) postponement of final decommissioning activities, 2) possible reuse of the facility by the owner, if so desired, and 3) low initial outlay of funds.

The passive SAFSTOR period may vary from a few years to a few tens of years, depending on the needs and desires of the facility owner and the public safety risks.

General work sequences and procedures for passive SAFSTOR are presented in this section. These sequences and procedures are developed under the assumption that the physical activities commence immediately following plant shutdown and final operational inventory cleanout operations. The program plan is discussed below.

9.2.1 Program Plan

The passive SAFSTOR alternative is divided into seven major phases:

- planning and preparation
- waste treatment facilities stabilization
- equipment deactivation
- isolation of contaminated areas
- final preparations for safe storage
- safe storage (security, surveillance, and maintenance)
- deferred decontamination or restart of production.

The approximate schedule for these work phases is given in Figure 9.2-1. It is estimated that the planning and preparation will take about 6 months. About 3 months are required for physical decommissioning activities before the facility and site enters the period of safe storage. Safe storage consists primarily of security, surveillance, and maintenance. At the conclusion of safe storage the site undergoes deferred decontamination to prepare it for

FIGURE 9.2-1. Sequence and Schedule for Major Activities for Passive SAFSTOR

unrestricted use. An estimated 14 months are required to plan and prepare for and to perform deferred decontamination. An unlikely alternative is to restartup the plant for uranium fuel production. The time and work estimates assume reasonable success with a minimum of delays and/or major unanticipated problems.

Most of the time in this decommissioning alternative is spent in safe storage. During that state, the plant condition is one where most of the transportable radioactivity is either removed or confined. The small amount of radioactivity remaining is spread thinly throughout the facility.

The areas listed in the passive SAFSTOR decommissioning plan are kept in the restricted-use category, as defined in Section 8. The areas are in a condition amenable to security, surveillance, and maintenance, but generally unavailable for any other use. Access is limited in accordance with the requirements of 10 CFR 20.103 and the provisions detailed in Section 9.2.1.6.

The onsite restricted areas assumed in this study include:

- the contaminated portions of the main building and the solvent extraction building
- the extraction waste and fluoride waste treatment facilities
- excess equipment storage yard
- uranium storage areas and cylinder storage areas.

Activities at the site during the safe storage period are limited primarily to: 1) operation of the building utility systems and fire prevention systems, 2) system maintenance, 3) building maintenance and radiation monitoring, 4) environmental radiation surveillance, and 5) security. The facility is patrolled by a security contractor on a periodic basis during the safe storage period. Periodic surveillance and maintenance of the facility structures and of passive safety and security-related systems are also required. The outer-perimeter site fence is maintained and no unauthorized entry is permitted. Detailed accounts of the decommissioning operations are stored at the facility and made a part of the public record. These accounts are required for use when final decontamination of the facility is performed.

Discussions of the seven major work phases of passive SAFSTOR are given in the following subsections.

9.2.1.1 Planning and Preparation

The planning and preparation activities for passive SAFSTOR are carried out concurrently within the final 6 months of facility operation. Figure 9.2-2 shows the time sequence for the planning and preparation phase of decommissioning for passive SAFSTOR. Work begins in the engineering and operations departments of the company organization to prepare the analyses and documentation needed to amend the operating license following final plant shutdown.

FIGURE 9.2-2. Sequence and Schedule of the Planning and Preparation Phase of Passive SAFSTOR

The initial steps include gathering data for the environmental impact assessment and the technical specification revisions. The data are used in the submittal of a decommissioning plan for NRC review and approval.

The decommissioning plan submitted to the NRC includes the following types of information:

- facility current status description
- general description of the overall plan
- description of measures taken to contain radioactivity
- proposed changes to the technical specifications
- necessary disassembly/retirement activities to be performed
- safety analysis of activities
- inventory of radioactive materials and their location in the facility
- security plan for total decommissioning program.

As stated earlier, a review of regulations and guides applicable to decommissioning is given in Section 5.

Detailed work plans and procedures are prepared to accomplish the physical activities in the most efficient and safe manner. Work is divided into easily manageable tasks. Available decommissioning techniques are reviewed and decisions are made on the general techniques to be used to accomplish each task. Detailed procedures are developed, including those for the inventory cleanup of the facility. Equipment and material requirements, manpower estimates, cost estimates, and work schedules are prepared. The plan is documented in detail, safety analysis reports are prepared, and all necessary documents are submitted for approval of plant management and appropriate regulating agencies.

Design, procurement, and testing of any special devices and equipment needed for decommissioning is also accomplished. This will assure that work can proceed without undue delay after plant shutdown.

A decommissioning organization within the company is initiated, with the structure and staffing requirements identified, and commitments are obtained from key engineering and operating personnel to fill strategic positions. Orientation and training of personnel identified as members of the decommissioning staff are carried on during the final 2 months of plant operation. The decommissioning staff draws on their own experience, as well as on the experience of the operations staff to assist in the planning activities.

It is assumed that most of the planning and the actual decommissioning activities are performed by plant operating and maintenance personnel. The various specialty contractors required for the decommissioning effort are selected during the final 2 months of plant operation.

Upon termination of routine plant production operations, an extensive final inventory cleanout and uranium audit are conducted. These cleanout/audit operations are similar, but slightly more extensive, than those conducted periodically to audit uranium content. Based on plant experience, clean-out operations are estimated to remove one-half of the residual uranium plant inventory. Because these cleanouts are done typically during plant production, they are also considered a part of normal plant operations in this study and are not charged to decommissioning. Details of the final inventory cleanout and uranium audit are found in Section G.1 of Appendix G.

A variety of other activities are carried out as part of plant shutdown and inventory cleanout operations. These activities include: 1) reduction of inventories of process chemicals and nonessential materials and equipment, and 2) an engineering review of effluent control and safety systems necessary for decommissioning.

Final preparatory steps to decommissioning are confirmation of radioactive materials inventories and a comprehensive survey of radiation dose rates at contaminated areas within the facility. These steps are taken immediately after uranium inventory cleanout and plant shutdown. In addition, a comprehensive radiation survey of the site is performed.

9.2.1.2 Waste Treatment Facilities Stabilization

The waste treatment lagoons for extraction waste and fluoride all require stabilization. These effluents contain uranium, which settles out in the lagoons. The extraction waste and fluoride lagoons are also drained and plastic covers with weights are placed over the remaining solids.

The remainder of the waste treatment systems is made up of pipes, pumps, valves, tanks, etc., that provide a closed system that contains any of the residual radioactivity left after being flush-rinsed several times.

9.2.1.3 Equipment Deactivation

Essential safety systems (such as lighting, utilities, radiation detection alarms, security monitors, and fire detection and portable fire fighting equipment) remain in operation during the safe storage period. All other equipment and systems are placed in a condition that provides maximum safety with minimum maintenance. When possible, equipment is left in a condition that permits startup or salvage at a later date.

The first step in equipment deactivation is a safety audit of all pumps and pipes used for radioactive materials or chemicals to ensure that hazardous or corrosive materials are removed. Electrical service is disconnected from all pumps not required to be in operation during the safe storage period.

Deactivation and isolation techniques include closing and securing installed valves; sealing hoods; capping ventilation exhaust stacks; installing blank flanges; and disconnecting electrical power, compressed air, and other utilities. A safety audit of all systems is performed to ensure that all flammable and other potentially hazardous materials are removed. All deactivated equipment and systems are tagged for identification and status.

In general, all systems not necessary to prevent the spread of contamination are deactivated. (See Section 9.2.1.6 for systems retained.) All equipment, valves, circuit breakers, etc., are tagged when deactivated. These tags identify the piece of equipment, the system it is in, and its condition.

Systems inside the building are deactivated by a variety of methods. Many piping systems are isolated using the installed valves, with handles or valve operators removed. Pipes that lead from uncontaminated to contaminated zones are blanked where flanges are readily accessible. Some uncontaminated systems are drained and left open to the atmosphere. All cranes are disabled by removal of their circuit breakers to prevent their unauthorized use during the safe storage period. Other electrical equipment that should not be operated during the safe storage period is disabled in a similar manner. Electrical service is disconnected from instrumentation not required to be in operation during the safe storage period.

9.2.1.4 Isolation of Contaminated Areas

Portions of the UF_6 conversion plant containing uranium contamination are isolated by the installation of high-security locks on entryways. Indirect access routes, however unlikely, are investigated from as-built drawings and secured. Such routes may include (but are not limited to) access through ventilation ductwork, roof plugs, or pipe trenches. Temporary barriers are constructed to block potential pathways for unauthorized entry. Warning signs are posted. The same steps are taken for the incinerator and waste treatment facilities. Fences around the waste treatment lagoons and storage areas are also secured, and posted with warning signs.

9.2.1.5 Final Preparations for the Safe Storage Period of Passive SAFSTOR

Final preparations for safe storage are:

- installing and/or upgrading monitoring systems and radiation alarms
- installing or relocating intrusion alarms
- performing a comprehensive radiation survey of both the restricted and unrestricted areas at the site; spots of excessive contamination in the unrestricted areas are decontaminated
- shipping all recovered uranium materials offsite for disposition
- training of personnel and contractors employed during the storage period
- conducting final survey.

9.2.1.6 Safe Storage Period of Passive SAFSTOR

Activities at the site during the safe storage period are limited primarily to services operations, security, building and equipment maintenance, radiation monitoring, and environmental radiation surveillance. The facility is not manned on a continuous basis after being placed in safe storage. Periodic surveillance and maintenance of the facility structures and of active and passive safety and security-related systems are required. The outer perimeter site fence is maintained and no unauthorized entry is permitted. Detailed accounts of the decommissioning operations are stored at the facility and made a part of the public record. These accounts are required for use when final decontamination of the facility is performed.

Surveillance and Maintenance Activities. The surveillance and maintenance programs are structured so that personnel inspect various portions of the facility on a routine basis. Radiation monitoring is done at each pre-established surveillance point at least monthly. These checks are staggered so that the monitoring takes place on different days of each month. Preventive maintenance activities and routine equipment inspections are also distributed throughout the safe storage period. Warning signs and physical barriers are inspected routinely and repaired as necessary. Electrical distribution systems, fire alarms, and radiation and intrusion alarms are operated and monitored continuously by an offsite contractor. Routine inspections of these systems, which were performed by outside experts during plant operation, continue on a reduced frequency during safe storage.

Environmental Surveillance Activities. A somewhat abbreviated version of the environmental monitoring program conducted during plant operation is carried out during the safe storage period; this is to identify and quantify any releases of radioactivity to the environment. This surveillance program is adequate for evaluating most potential nonroutine or accidental releases. For situations involving releases from events such as fire or malicious acts that may require prompt emergency actions to minimize public risk, special surveillance requirements are added. This program is discussed in more detail in Section F.5 of Appendix F.

Security Activities. The protection of the public, principally against the consequences of their own actions, is an important dimension of the security program used for the safe storage period. Conventional security detection and notification systems normally used to protect the plant owner against loss or damage are augmented by audible alarms. These alarms, strategically located outside secured radiation zones, loudly warn an intruder of his potential danger. Silent sensors simultaneously alert offsite security personnel.

Routine patrol checks are carried out by offsite guards. A reputable private security agency is contracted to ensure adequate surveillance and prompt response to alarms. Liaison with local law enforcement agencies is maintained and their assistance is called for when necessary.

Security is provided during the safe storage period by two basic methods: offsite guards and security systems. Locks on the fence around the decommissioned facility provide the first line of security. The fence is maintained in good condition throughout the surveillance and maintenance period. Facility security is maintained at all times by intrusion alarms and high-security locks on exterior doors. Intrusion, fire, and radiation alarms are monitored continuously by an offsite security firm. Depending on the situation indicated by the alarms, offsite security agency personnel are available to respond immediately.

Physical security to prevent inadvertent radiation exposure of surveillance and maintenance personnel is provided by locked barriers, which make it extremely difficult for unauthorized access to areas where radiation or contamination is present.

The facility manager is responsible for controlling authorized access into and movement within the facility. The facility manager is further charged with the responsibilities of appropriate actions and notifications regarding breaches of security, upkeep of plant surveillance, and maintenance programs. He is also charged with administrative reporting of these events, as required by state and federal regulations. He is responsible for health physics work, uranium accountability, and record keeping.

Essential Systems and Services. The support systems requiring surveillance and maintenance during the safe storage period are listed in Table 9.2-1. These systems remain in operation throughout the safe storage period. These systems, in combination with inherent facility structural integrity, provide the primary means for minimizing the release of hazardous material to the environment. The equipment in these systems is inspected and renovated to ensure adequate equipment reliability before the surveillance and maintenance period begins. In addition, the intrusion alarm system within the facility and on the perimeter fence are both modified to provide offshift surveillance capability by a commercial security agency.

TABLE 9.2-1. Systems and Services Required During the Safe Storage Period of Passive SAFSTOR

<u>Systems or Components</u>	<u>Justification</u>
Electric power	Normal and emergency power are maintained for: radiation monitoring systems and alarms, lighting circuits, fire protection systems and alarms, and surveillance monitoring systems and alarms. Switchboards are aligned so that no electrical power is fed to deactivated systems.
Fire Protection System (detection and suppression)	Portable fire extinguishers remain at selected locations and fire detection systems remain in operation as required for safety.
In-Plant Communication Systems	Required for normal communication.
Radiation Monitoring Systems	Radiation monitors and alarms remain in operation at strategic locations throughout the facility sections. The locations of some devices may be installed to ensure that important areas are adequately covered. Selected monitoring programs are also continued.
Security Systems	Security devices and alarms (provided with both normal and backup emergency power) are maintained by a security agency subcontractor. In addition to intrusion system monitoring and maintenance, it is postulated that the security agency responds appropriately to intrusions.

9.2.1.7 Deferred Decontamination

The deferred decontamination phase is essentially the same as the DECON decommissioning case discussed in Section 9.1.

As defined in Section 4, deferred decontamination is the final stage of decommissioning when passive SAFSTOR is utilized. The facility and site must be shown to have residual radioactivity levels sufficiently low to permit unrestricted use when decontamination is complete.

The same basic operations are performed during deferred decontamination as were performed during DECON. The primary decontamination done for the safe

storage period does not need repeating. A small amount of additional manual decontamination and cleanup effort is performed to collect loose smearable contamination that may have moved during the safe storage period. Some sealed areas need to be unsealed. The same disassembly techniques and contamination control methods employed during DECON are required.

It is anticipated that a new staff is needed for deferred decontamination. The hiring could be done by the facility operator or the decommissioning contractor. Extensive training and familiarization of this staff with the facility is necessary, because of dispersal of personnel from the operations staff during the extended period of safe storage. Additional effort is required to restore some services needed for decontamination and to remove the various enclosures, doors, locks, and temporary barricades used to secure the facility from unauthorized entry during safe storage.

In view of these considerations, it is reasonable to assume that a work force of about the same size as was used for DECON is required for deferred decontamination and over approximately the same period. Essentially, the same volumes of contaminated and uncontaminated materials must be removed and transported to an authorized burial site, except for those materials removed during decontamination for safe storage. The items that increase the costs of deferred decontamination with respect to DECON are the labor costs associated with training a decommissioning staff, removing covers from the lagoons, and restoring services needed for decontamination.

The program plan outlined for DECON in Section 9.1 is assumed to be valid for deferred decontamination, except in the planning and preparation phase. Portions of the environmental impact assessment and technical specifications have to be revised. The final inventory cleanout need not be repeated. Other elements in the planning and preparation phase (Figure 9.1-2) are likely to take slightly more manpower and time due to lack of experienced personnel.

9.3 DECOMMISSIONING BY DECON OF THE PLANT AND EQUIPMENT WITH ONSITE DISPOSAL OF LAGOON WASTES

DECON of the plant and equipment is covered in Section 9.1. Decommissioning activities for onsite disposal of the lagoon wastes is summarized in this section. Details of onsite disposal are contained in Section G.4 of Appendix G.

Decommissioning of the lagoon wastes begins with planning and preparation activities that include a radiological survey to determine the radiological condition of the waste lagoons. 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 stabilization of the lagoon wastes, the following procedures are assumed. The lagoon is covered with a 50-mm-thick layer of asphalt. This asphalt layer is then covered with 1 m of soil. The soil is mounded slightly at the center of the lagoon to allow water to drain from the soil cover and to prevent the accumulation of runoff from rainfall or snow melt. After compaction and contouring of the soil cover, the area is seeded with grass.

REFERENCES

1. W. A. Rodger, et al., "de minimus" Concentrations of Radionuclides in Solid Wastes, National Environmental Studies Project, AIF/NESP-016, Atomic Industrial Forum, Washington, DC, April 1978.

10.0 DECOMMISSIONING COSTS

This section presents estimates of the costs for decommissioning the reference uranium hexafluoride conversion (UF_6) plant. Cost estimates are made for DECON, for passive SAFSTOR including preparations for safe storage, safe storage, and deferred decontamination and for DECON of the plant buildings with onsite burial of the lagoon wastes. The costs are based on decommissioning procedures developed in detail in Appendix G and summarized in Section 9. Costs are included for direct support and decommissioning worker labor, equipment and materials, contaminated waste packaging, transportation and disposal, utilities and other miscellaneous owner expenses, and specialty contractors. All costs are in 1981 dollars.

The basic cost estimates presented in this section assume relatively efficient performance of the decommissioning activities. A 25% contingency is added to the cost estimate totals as an allowance for unforeseen problems or scheduling delays that may arise during the decommissioning. The total costs presented are believed to be representative of actual expenses that would be incurred to decommission the reference facility using the methods described in this report.

10.1 COST ESTIMATES FOR DECON

The estimated costs for DECON of the reference plant are summarized in Table 10.1-1. DECON is estimated to require about 8 months (plus 6 months for planning and preparation) at a cost of approximately \$5.91 million. If disposal of lagoon waste at a commercial low-level waste (LLW) burial ground is necessary, the cost is estimated to be about \$21 million.

Manpower costs include both support staff and decommissioning workers and represent about 57% of the total cost of DECON. In Table 10.1-1, manpower costs are shown separately for the planning and preparation and the decommissioning phases of DECON. These costs include onsite labor for packaging radioactive waste materials for shipment. Labor costs related to radioactive waste transportation are included in waste management costs.

TABLE 10.1-1. Summary of Estimated DECON Costs

Cost Category	Cost in Millions of 1981 Dollars ^(a)		
Manpower			
Planning and Preparation	0.304		
Decommissioning	1.749		
Equipment and Supplies	0.149		
Disposal of Radioactive Material	0.509		
Miscellaneous Owner Expense	0.740		
Subtotal	3.451		
Lagoon Waste Management Options	Ship to Mill	Ship to LLW Burial Ground	Ship to Commercial Dump
Extraction Waste	1.219	10.011	1.219 ^(b)
Fluoride Waste	0.055 ^(c)	3.301	1.011
Subtotal (includes lagoon waste)	4.725	16.763	5.681
25% Contingency	1.181	4.191	1.420
Total Costs	5.906	20.954	7.101

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

(b) Not feasible to ship extraction waste to commercial dump. Cost is for extraction waste shipped to mill.

(c) Not feasible to ship fluoride waste to mill. Cost is for fluoride waste buried onsite.

Waste management accounts for about 38% of the \$5.91 million cost of DECON. Waste management costs include shipping container costs, transportation charges, and fees for waste disposal at a commercial LLW burial ground.

10.1.1 Manpower Requirements and Costs for DECON

Estimates are made for the work force required to plan and execute the decommissioning activities for DECON described in Section 9.1. These work force estimates are used, together with the unit manpower costs given in Section H.1 of Appendix H, to estimate DECON manpower costs. The bases for these manpower estimates and the results in terms of decommissioning manpower costs are described in this section.

10.1.1.1 Manpower Requirements

The decommissioning work force organizational chart for DECON is shown in Figure 10.1-1. The work force is described in two parts: 1) the decommissioning support staff that plans, supervises and provides support for the

^(a) ASSIGNED TO WORK CREWS AS WORK SITUATION DEMANDS. COUNTED ONLY ONCE.

FIGURE 10.1-1. Postulated Organization Chart for DECON

decommissioning activities, and 2) the decommissioning workers who perform the actual decommissioning activities. The seven general types of functions performed during decommissioning are described briefly below:

- Project Management - prepare and implement the decommissioning plan in a safe and cost-effective manner.
- Quality Assurance (QA) - develop the QA plan and monitor the safety and performance of the decommissioning activities.
- Decommissioning Operations - develop the decommissioning plan and carry out the actual decommissioning activities.
- Plant Operations and Maintenance - operate and maintain plant equipment that must be operated during the decommissioning.
- Safety Protection - develop methods to assure the safety of the public and decommissioning workers.
- Security - provide protection for the site and facility against unauthorized entry.
- Support Services - providing accounting, procurement and stores, secretarial and clerical services in support of the decommissioning activities.

Decommissioning operations are carried out by crews that consist of a foreman, four to twelve decommissioning technicians, and health physics technicians and craftsmen who are added to the crews as the work situation demands.

A key assumption in estimating the manpower and time for the basic events is that the decommissioning work force is composed primarily of former plant operating and maintenance personnel. The decommissioning workers are, therefore, familiar with plant facilities and equipment and experienced with radiation work procedures. The duties and experience of the members of the basic decommissioning crew are outlined below.

Foreman. This person supervises the performance of all decommissioning activities. He coordinates with the engineering staff through the operations supervisor to plan and execute each day's activities. He assembles the crew

and equipment required to perform these activities and instructs the crew on procedures and safety precautions to be followed. The foreman is assumed to perform some of the actual decommissioning activities as well as supervise other members of his crew. It is anticipated that the foreman would have been employed in a position comparable to a process shift supervisor or maintenance supervisor during plant operations, so that he has detailed knowledge of plant systems and equipment.

Decommissioning Technicians. These people perform the bulk of the actual decommissioning operations. They are assumed to possess a variety of skills either through past experience in the plant or through specialized training prior to or during the decommissioning. The technicians are assumed to be employed in positions comparable to process operators, maintenance technicians or mechanical technicians during plant operations. It is anticipated that they would be qualified in several craft disciplines, including operation of much of the plant equipment.

Health Physics Technician. This person is assigned to the work crews as the work situation demands to provide instruction in radiation and industrial safety precautions to be followed for each task and to monitor compliance with written radiation work procedures for the task. He performs on-the-job radiation measurements and has the authority to stop work on the job if any potentially unsafe situation arises.

Craftsmen. These are people who are added to the basic crew to carry out particular tasks that require assistance of a/an:

- welder
- pipefitter
- carpenter
- electrician
- instrument technician
- millwright
- machinist.

The decommissioning staff is assembled during the planning and preparation phase. Initial management staff consists of the project manager, project engineer, quality assurance supervisor, and operations supervisor. Other staff personnel are added as their services are required during the planning and preparation phase. Planning and preparation activities take place during the final 6 months of plant operation. Therefore, support activities such as plant maintenance and plant security are available as part of plant operations and are not charged to decommissioning during the planning and preparation phase.

The decommissioning staff is generally sized and structured on a 1-shift, 5-day week. Certain operations such as security are carried out on a 4-shift, 7-day week. Decommissioning activities require that workers wear protective clothing, and in some cases, respiratory protection. Because of the inconveniences of the physical environment in which decommissioning tasks are carried out, manpower requirements are developed on the basis of an assumed worker time efficiency of 75%. During the approximately 8-month period of dismantlement and decontamination, the staff size is estimated to remain approximately constant.

10.1.1.2 Manpower Costs

Table 10.1-2 shows manpower and associated cost estimates for the planning and preparation phase of DECON and Table 10.1-3 shows manpower requirements and costs for the dismantlement and decontamination phase of DECON. A total of about 6 man-years is estimated to be required for planning and preparations, at a labor cost of about \$304,000. A total of about 43 man-years is estimated to be required to decontaminate and remove contaminated materials from the facility, at a labor cost of about \$1.75 million. The total labor cost for DECON is estimated to be about \$2.1 million without contingencies. Manpower costs shown in Table 10.1-3 include labor costs for packaging radioactive waste materials for shipment. These costs do not include specialty labor costs discussed in Section 10.1.3 for transportation and in Section 10.1.5 for other specialty contractors.

It is recognized that completion of decommissioning activities may take longer than the estimated times, resulting in higher labor costs. These cost

TABLE 10.1-2. Summary of Manpower Utilization and Staff Costs for Planning and Preparation Phase of DECON

Title or Function	Man-Years	Cost (\$ thousands) ^(a,b)
Project Manager	0.65	58
Project Engineer	0.50	38
Health and Safety Supervisor	0.20	12
Contracts and Accounting Specialist	0.50	24
Radioactive Shipment Specialist	0.20	8
Q.A. Engineer	0.50	24
Planning Engineer	0.50	26
Engineering Technician	0.50	20
Operations Supervisor	0.50	26
Foreman	0.85	38
Secretary	<u>1.25</u>	<u>30</u>
Total Man-Years	6.15	
Total Cost		304

- (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.

TABLE 10.1-3. Summary of Manpower Utilization and Staff Costs for the Dismantlement and Decontamination Phase of DECON

Title or Function	Man-Years	Cost (\$ thousands) ^(a,b)
Project Manager	0.75	67
Project Engineer	0.75	57
Health and Safety Supervisor	0.75	45
Health Physics Technician	1.25	38
Security Force Supervisor	0.75	42
Security Patrolman	3.0	76
Contracts and Accounting Specialist	0.75	35
Radioactive Shipment Specialist	0.75	30
Q.A. Engineer	0.75	35
Planning Engineer	0.75	39
Engineering Technician	0.75	29
Maintenance and Crafts Supervisor	1.50	70
Custodian	0.75	23
Craftsman	3.41	109
Operations Supervisor	0.75	39
Foreman	3.29	147
Technician	21.17	832
Secretary	<u>1.50</u>	<u>36</u>
Total Man-Years	43.37	
Total Cost		1749

- (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.

increases may be offset by reducing the labor force after the most labor-intensive tasks are completed. Final deactivation and cleanup activities, for example, may be accomplished by a relatively small work force.

10.1.2 Material and Equipment Requirements and Costs for DECON

Estimates of material and equipment requirements and costs for DECON are shown in Table 10.1-4. Equipment requirements are based on decommissioning

TABLE 10.1-4. Estimated Material and Equipment Requirements and Costs for DECON

Description	Quantity	Estimated Unit Cost (\$ thousands)(a)	Estimated Total Cost (\$ thousands)(a)
Oxyacetylene Torch	4 ea.	1.0	4.0
Guillotine Pipe Saw	2 ea.	4.0	8.0
Tube Cutter	2 ea.	0.41	0.82
Ratcheting Pipe Cutter	6 ea.	0.07	0.42
Reciprocating Saw	4 ea.	0.80	3.2
Nibbler	2 ea.	1.0	2.0
High-Velocity Liquid Jet	1 ea.	6.9	6.9
Low-Velocity Liquid Jet	2 ea.	2.8	5.6
Hydraulic Concrete Surface Spalling Device	1 ea.	5.0	5.0
Concrete Drill	3 ea.	0.28	0.84
Electric Pneumatic Hammer	2 ea.	0.70	1.4
Portable A-Frames	2 ea.	4.1	8.2
Portable Wash Sinks	2 ea.	2.8	5.6
Portable Spray Clean Booth	1 ea.	5.5	5.5
Portable Greenhouse Erection Kit	1 ea.	2.8	2.8
Portable Powered Brushes	20 ea.	0.21	4.2
HEPA Filter	5 ea.	0.20	1.0
Roughing Filter	20 ea.	0.10	2.0
Decontamination Chemicals			11.0
Cleaning Supplies			22.1
Expendable Tools			11.0
Protective Clothing (including laundry)			19.9
Office Supplies: Planning and Preparation			11.0
Decommissioning			<u>6.6</u>
Total(b)			149.1

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

(b) Total is shown as direct addition of prior numbers to retain calculational information. Precision is less than shown. Contingency of 25% is not included.

procedures described in detail in Appendices F and G and summarized in Section 9.1. Costs of decontamination chemicals are calculated on the basis of quantities required for decontamination and unit costs given in Section H.1 of Appendix H. Cleaning supplies represent a major cost item and include assorted cleaning agents, rags, mops, brushes, plastic bags, plastic sheeting, etc. The cost of protective clothing includes the cost of laundering the clothing onsite and is estimated to be about \$620 per week. The total cost of material and equipment for DECON of the reference plant is estimated at about \$149,000 without contingency.

10.1.3 Waste Management Requirements and Costs for OECON

Waste management requirements and costs for DECON are described in this section. Estimates are made of quantities of radioactive wastes generated during OECON of the reference plant and of packaging, transportation and disposal requirements and costs for managing these wastes. These estimates are based on decontamination procedures described in Section 9.1, and on unit waste management costs given in Appendix H.1.

10.1.3.1 Waste Management Requirements

Radioactive wastes generated during OECON must be properly packaged and shipped to an LLW burial site. Radioactive wastes generated during DECON include:

- contaminated process equipment, tanks, and piping;
- concrete rubble from the mechanical decontamination of contaminated floors and walls;
- HEPA and roughing filters;
- sections of ventilation ductwork;
- combustible and noncombustible trash (protective clothing, contaminated tools, rags, paper, plastic, metal scrap, etc.);
- sludge, liners, and soil from the waste lagoons.

All 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 LLW burial sites. Assumed waste shipping volumes for contaminated process equipment, tanks, piping, ducts, etc., postulated to require disposal at an LLW burial site are given in Table H.2-2 of Appendix H.

All shipments of decommissioning wastes are assumed to be made in compliance with federal, state and local regulations, as described in Section F.3.2 of Appendix F.

Table 10.1-5 gives estimated weights and volumes of decommissioning wastes from DECON of the reference plant, together with the type of packaging and the number of shipments required for these wastes. Detailed information about waste quantities is given in Section H.2 of Appendix H.

TABLE 10.1-5. Waste Disposal Packaging and Shipping Data for DECON

<u>Waste Category</u>	<u>Shipping Weight (kg)^(a)</u>	<u>Shipping Volume (m³)^(a)</u>	<u>Type of Container</u>	<u>Number of Shipments</u>
<u>To Low-Level Waste Burial:</u>				
Equipment and Components	380 000	855	Plywood Boxes	24
Pipe, Conduit, Duct, Trays, Fixtures, etc.	219 200	195	Plywood Boxes	12
Concrete Rubble	64 000	99	Steel Drums	7
Disposal of Solid Waste Buried Onsite	48 000	80	Steel Drums	5
Miscellaneous	17 500	12	Steel Drums	1
Micellaneous	<u>12 000</u>	<u>18</u>	Plywood Boxes	<u>2</u>
Total LLW	740 700	1 259		51
<u>Lagoon Wastes</u>				
Extraction Waste	37 260 000	22 270	--(b)	1 840
Fluoride Waste	<u>18 200 000</u>	<u>5 700</u>	--(b)	<u>552</u>
Total Lagoon Waste	55 460 000	27 970		2 392

(a) Number of significant figures shown is for computational accuracy and does not imply precision to three or more significant figures.

(b) Lagoon waste packaging is covered in Section 10.1.3.3.

10.1.3.2 Waste Management Costs

The estimated costs for containers, transportation, and disposal of the radioactive wastes from buildings and for miscellaneous waste buried onsite during operations for DECON of the reference plant are summarized in Table 10.1-6.

TABLE 10.1-6. Estimated Waste Management Costs for DECON

Waste Category	Costs in 1981 Dollars ^(a)			
	Container	Transportation	Burial	Total ^(b)
<u>To Low-Level Waste Burial:</u>				
Equipment and Components	23 500	31 560	262 660	317 700
Pipe, Conduit, Duct, Trays, Fixtures, etc.	5 400	15 780	59 900	81 100
Concrete Rubble	14 160	8 510	30 410	53 100
Disposal of Solid Waste Buried Onsite ^(c)	11 430	6 580	24 580	42 590
Miscellaneous	1 170	1 320	3 690	6 700
Miscellaneous	<u>500</u>	<u>2 430</u>	<u>5 530</u>	<u>8 500</u>
Totals	56 700	66 180	386 770	509 700

(a) Number of figures is for computational accuracy and does not imply precision to three or more significant figures.

(b) Contingency of 25% is not included.

(c) This waste is exhumed onsite and shipped to LLW burial.

Cost estimates are based on projected packaging and shipping data summarized in Table 10.1-5 and on waste management cost data in Section H.2 of Appendix H. Waste management cost details are also given in Section H.2. The waste management cost exclusive of lagoon wastes for DECON is estimated to be about \$510,000 without contingency. The cost of management for the lagoon wastes is covered in the following section on lagoon waste management costs.

About 1260 m³ of the total waste volume of 31,100 m³ is assumed to be shipped to LLW burial. The remainder is assumed to be decontaminated and sent to commercial waste disposal, buried onsite, or processed to recover residual uranium.

10.1.3.3 Lagoon Waste Management Costs

The sludges from the solvent extraction waste lagoon and the fluoride lagoons must be disposed of as part of decommissioning. The alternatives are discussed in Section 9. Cost estimates for the alternatives are provided in this section in Table 10.1-7.

TABLE 10.1-7. Cost of Lagoon Waste Management Options for DECON

Method of Waste Disposal	Cost in 1981 Dollars		Total Cost
	Extraction Waste	Fluoride Waste	
Ship to Uranium Mill for Processing	1 218 720 ^(a)	54 900 ^(b)	1 397 700
Ship to LLW Burial Ground	10 011 100	3 301 020	13 312 000
Ship to Commercial Waste Dump	1 218 720 ^(c)	1 011 340	2 343 540

(a) Includes \$270,600 for contaminated soil and liner shipped to LLW burial ground.

(b) Not feasible to ship fluoride waste to mill; cost is for onsite burial.

(c) Not feasible to ship extraction waste to commercial dump; cost is for shipment to mill.

Solvent Extraction Waste Disposal Cost. Shipment of extraction waste to a uranium mill costs about \$1,218,700, including \$270,600 for shipment of liner and contaminated soil material to an LLW burial ground. It also includes a credit of \$2,156,900 for the value of the residual uranium, which is assumed to be 95% reclaimed at a value of \$66 per kilogram of uranium.

Shipment of extraction waste to an LLW burial ground is estimated to cost \$10,011,100. Shipment to a commercial waste dump would require that the extraction waste be mixed with soil at a ratio of 9 parts soil to 1 part waste. This results in a very large cost and is not considered to be viable.

Fluoride Waste Disposal Cost. Fluoride waste disposed of by onsite burial is estimated to cost \$65,500. It is not feasible to ship the fluoride waste to

a mill for reprocessing since the uranium content is very small. The cost of shipment of fluoride waste to an LLW burial ground is estimated to cost about \$3,301,000 and shipment to a commercial waste dump is estimated at \$1,011,300.

The waste disposal costs summarized in this section are specific to the reference plant and should not necessarily be considered representative of other UF_6 plants. Waste volumes generated at other UF_6 facilities may be different and waste disposal costs could be significantly different from those discussed in this section.

10.1.4 Miscellaneous Owner Expenses for DECON

Estimated miscellaneous owner expenses for DECON are given in Table 10.1-8.

TABLE 10.1-8. Estimated Miscellaneous Owner Expenses for DECON

<u>Cost Category</u>	<u>Cost in Thousands of 1981 Dollars (a,b)</u>
Utilities	340
Taxes	180
Inspections and License Amendments	70
Insurance	<u>150</u>
Total	740

(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.

The annual inspection fees for safety inspections at the operating UF_6 plant is estimated to be \$20,000.⁽¹⁾ In addition, fees for license amendments for decommissioning could total \$50,000 or more. Thus, the license-related costs during the first year following shutdown of operations are estimated to be about \$70,000.

The cost of nuclear liability insurance for a facility being decommissioned has also not been determined. An allowance of \$150,000 is included for the annual insurance premium for nuclear liability and conventional insurance.

10.2 ESTIMATED COSTS OF THE PREPARATIONS FOR SAFE STORAGE PERIOD OF PASSIVE SAFSTOR

The estimated costs of preparing the reference plant for the safe storage period are summarized in Table 10.2-1. These decommissioning activities are estimated to require about 6 months at a cost of approximately \$881,000.

TABLE 10.2-1. Summary of Estimated Costs of Preparations for Safe Storage

<u>Cost Category</u>	<u>Cost in Thousand of 1981 Dollars (a,b)</u>	<u>Percent of Total</u>
Manpower		
Planning and Preparation	169	24.0
Decommissioning	205	29.1
Equipment and Supplies	125	17.7
Disposal of Radioactive Material	5	0.7
Miscellaneous Owner Expense	<u>201</u>	<u>28.5</u>
Subtotal	705	100.0
25% Contingency	<u>176</u>	
Total Costs of Preparations for Safe Storage	881	

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

Manpower represents about 53% of the total cost of preparations for safe storage. Manpower costs include both support staff and decommissioning worker labor. Breakdown of manpower costs are shown separately for planning and preparation activities and for decommissioning activities described later in the text. The annual costs of safe storage are described in Section 10.3.

10.2.1 Manpower Requirements and Costs of Preparations for Safe Storage

Estimates are made of the work force required to plan and execute the preparations for safe storage described in Section 9.2. These work force estimates are used, together with the unit manpower costs given in Section H.1 of Appendix H, to estimate manpower costs. The bases for these manpower estimates and the results in terms of decommissioning manpower costs are described in this section.

10.2.1.1 Manpower Requirements

The organizational chart of the decommissioning work force for preparations for safe storage is shown in Figure 10.2-1. This work force includes the support staff that plans, supervises and provides support for decommissioning activities and the workers who perform the actual decommissioning activities. Job description details for key individuals in the decommissioning work force are given in Section H.2 of Appendix H.

The decommissioning staff is assembled during the planning and preparation phase that takes place during the final 6 months of plant operation. Initial management staff consists of the project manager, project engineer, quality assurance engineer and operations supervisor. Other staff personnel are added as their services are required during the planning and preparation phase.

Actual decommissioning activities require approximately 2 months following plant shutdown.

10.2.1.2 Manpower Costs

Table 10.2-2 shows manpower and associated cost estimates for the planning and preparation phase, and Table 10.2-3 shows support staff and worker manpower requirements and costs for the active decommissioning phase of preparations for safe storage. About 3 man-years are estimated to be required for planning and preparations, at a labor cost of about \$169,000. About 5 man-years are estimated to be required to deactivate the facility, at a labor cost about \$205,000. The total labor cost for placing the facility in safe storage is estimated to be about \$374,000 without contingencies.

FIGURE 10.2-1. Postulated Organization Chart for Preparations for Safe Storage

**TABLE 10.2-2. Summary of Manpower Utilization and Staff Costs
for Planning and Preparation Phase of SAFSTOR**

Title or Function	Man-Years	Cost (\$ thousands)(a,b)
Project Manager	0.50	45
Project Engineer	0.50	38
Health and Safety Supervisor	0.25	15
Contracts and Accounting Specialist	0.25	12
Q.A. Engineer	0.25	12
Radioactive Shipment Specialist	0.25	10
Operations Supervisor	0.25	13
Maintenance and Crafts Supervisor	0.25	12
Secretary	0.50	12
Total Man-Years	3.00	
Total Cost		169

- (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.

**TABLE 10.2-3. Summary of Manpower Utilization and Staff Costs
for Decommissioning Phase of SAFSTOR**

Title or Function	Man-Years	Cost (\$ thousands)(a,b)
Project Manager	0.25	22
Project Engineer	0.25	19
Health and Safety Supervisor	0.25	15
Health Physics Technician	0.35	11
Security Force Supervisor	0.25	14
Security Patrolman	0.07	2
Contracts and Accounting Specialist	0.25	12
Q.A. Engineer	0.35	16
Radioactive Shipment Specialist	0.17	7
Maintenance and Crafts Supervisor	0.25	12
Custodian	0.17	5
Craftsman	0.67	22
Operations Supervisor	0.25	13
Technician	0.75	29
Secretary	0.25	6
Total Man-Years	4.50	
Total Cost		205

- (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.

10.2.2 Estimated Material and Equipment Requirements and Costs of Preparations for Safe Storage

Estimates of material and equipment requirements and costs of preparing the reference plant for safe storage are shown in Table 10.2-4. Equipment requirements are based on decommissioning procedures described in detail in Appendices F and G and summarized in Section 9.2. The total cost of material and equipment for placing the reference plant in safe storage is estimated at about \$125,000 without contingency.

TABLE 10.2-4. Estimated Costs of Equipment and Supplies for Preparations for Safe Storage

Description	Estimated Total Cost (\$ thousands)(a,b)
Decontamination Chemicals	2.0
Cleaning Supplies	2.5
Expendable Tools	2.0
Protective Clothing (including laundry)	4.0
Intrusion Alarm System	100.0
Miscellaneous	7.0
Office Supplies	
Planning and Preparation	5.0
Decommissioning	<u>3.0</u>
Total	125

- (a) Total is shown as direct addition of the prior numbers to retain calculational information. Precision is less than shown.
(b) Contingency of 25% is not included.

10.2.3 Miscellaneous Owner Expenses for Preparations for Safe Storage

Estimated miscellaneous owner expenses for preparing the reference plant for safe storage are shown in Table 10.2-5. These expenses are calculated on the same bases as were similar expenses for DECON (see Section 10.1.4), except that the time period is only 2 months.

TABLE 10.2-5. Estimated Miscellaneous Owner Expenses
During Preparations for Safe Storage

<u>Cost Category</u>	<u>Cost in Thousands of 1981 Dollars (a,b)</u>
Utilities	68
Taxes	30
Regulatory Fees	70
Insurance	<u>33</u>
Total	201

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

(b) Contingency of 25% is not included.

In estimating the applicable safety and safeguards inspection fees during preparations for safe storage, it is assumed that the full operating fees are paid, plus the fees for license amendments. These fees are assumed to be the same as for DECDN (\$70,000) during the first year. The annual insurance premium for nuclear liability and conventional insurance is assumed to be at the same rate as for DECON.

10.3 COST ESTIMATES FOR THE SAFE STORAGE PERIOD OF PASSIVE SAFSTOR

This section presents estimates of the annual manpower and material requirements and costs for the safe storage period of passive SAFSTOR.

Activities carried out at the plant to assure the continued protection of the public safety during this phase include:

- monitoring of alarm systems
- periodic radiation surveys of the facility
- periodic environmental surveys
- maintenance of alarm systems and protective barriers
- inspection of facility structures, protective barriers and alarm systems
- site and facility security
- fulfillment of regulatory requirements.

The annual costs of safe storage are summarized in Table 10.3-1. Safe storage is estimated to cost about \$95,000 annually in 1981 dollars. Staff labor costs represent about 40% of this total.

TABLE 10.3-1. Estimated Annual Costs of the Safe Storage Period

<u>Cost Category</u>	<u>Annual Cost in Thousands of 1981 Dollars^(a,b)</u>	<u>Percent of Total</u>
Staff Labor ^(b)	30	39.5
Supplies and Equipment	4	5.3
Security Contractor	12	15.8
Maintenance Contractor	10	13.2
Annual Allowance for Repairs	4	5.3
Utilities	7	9.2
Taxes	3	3.9
Insurance	5	6.5
Regulatory Fees	<u>1</u>	<u>1.3</u>
Subtotal	76	100.0
25% Contingency	<u>19</u>	
Total Annual Cost	95	

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

(b) See Table 10.3-2 for cost details.

10.3.1 Manpower Requirements and Costs for Safe Storage

Annual manpower requirements and costs for safe storage are described in this section. The work force required to perform safe storage activities at the decommissioned UF₆ conversion plant is shown in Figure 10.3-1.

Surveillance and maintenance activities are supervised by a facility manager who reports directly to corporate headquarters. This person directs routine and corrective maintenance and radiation and environmental surveys;

FIGURE 10.3-1. Postulated Organization Chart for the Safe Storage Period

performs routine physical inspections of the facility; arranges for third-party inspections of facility structures and equipment; assures that all regulatory requirements are fulfilled; and makes routine reports to corporate headquarters and regulatory agencies.

Third-party inspections are assumed to be carried out by a two-man team on a semi-annual basis.

Table 10.3-2 shows annual manpower requirements and costs for safe storage. Manpower requirements are based on the safe storage staff organization chart shown in Figure 10.3-1. Costs are based on unit cost data in Section H.1 of Appendix H. Third-party inspection costs are based on an assumed cost of \$600 per man-day.

10.3.2 Security and Surveillance Costs for Safe Storage

The cost for the security and surveillance contractor is estimated to be about \$11,600 per year based on information from Reference 2 with a cost escalation factor added.

TABLE 10.3-2. Estimated Annual Staff Requirements for Safe Storage

<u>Title or Function</u>	<u>Man-Years/ Year</u>	<u>Annual Cost 1981 Dollars^(a,b)</u>
Facility Manager	0.25	19 000
Secretary	0.25	6 000
Third-Party Inspection Team ^(c)	<u>0.03</u>	<u>4 700</u>
Total Man-Years	1.03	
Total Costs/Year		29 700

(a) The number of figures carried is for computational accuracy and does not imply precision to the nearest dollar.

(b) Contingency of 25% is not included.

(c) Third-party inspection costs are based on an assumed \$600 per man-day.

10.3.3 Material and Equipment Requirements and Costs for Safe Storage

An annual allowance of \$4,000 for equipment and supplies is included in the material and equipment cost estimate. This allowance includes funds for monitoring supplies, secretarial supplies, etc.

Repairs to monitoring instruments, security alarm systems, etc., are assumed to be made by outside contractors. An annual allowance of \$4000 for facility repairs is included in the cost estimate.

10.3.4 Miscellaneous Owner Expenses During Safe Storage

Miscellaneous owner expenses during the safe storage period include the costs of taxes, regulatory fees and insurance. Except for the regulatory fees, annual costs for these items are assumed to be about 2% of the annual costs during DECON because it is an inactive facility. Cost of utilities was estimated at \$7,000, based on an estimate from Reference 2, since only the alarm systems are operating.

10.3.5 Maintenance Contractor

The annual cost of a maintenance contractor is estimated to be about \$10,000, including \$2,000 for supplies. The maintenance contractor performs

routine maintenance and minor repairs to the safety and security systems. The cost is based on an estimate of 0.25 man-years required for maintenance on the plant during safe storage.

10.4 COST ESTIMATES FOR THE DEFERRED DECONTAMINATION PERIOD OF PASSIVE SAFSTOR

The estimated costs of deferred decontamination following safe storage of the reference plant are summarized in Table 10.4-1.

TABLE 10.4-1. Summary of Estimated Deferred Decontamination Costs

<u>Cost Category</u>	<u>Cost in Millions of 1981 Dollars</u> ^(a)		
Manpower			
Planning and Preparation	0.589		
Decommissioning	1.826		
Equipment and Supplies	0.151		
Disposal of Radioactive Material	0.509		
Miscellaneous Owner Expense	<u>0.740</u>		
Subtotal	3.815		
<u>Lagoon Waste Management Options</u>	<u>Ship to Mill</u>	<u>Ship to LLW Burial Ground</u>	<u>Ship to Commercial Dump</u>
Extraction Waste	1.332	10.011	1.332 ^(b)
Fluoride Waste	<u>0.055</u> ^(c)	<u>3.301</u>	<u>1.011</u>
Subtotal (includes lagoon waste)	5.202	17.127	6.158
25% Contingency	<u>1.301</u>	<u>4.282</u>	<u>1.540</u>
Total Costs	6.503	21.409	7.698

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

(b) Not feasible to ship extraction waste to commercial dump. Cost is for extraction waste shipped to mill.

(c) Not feasible to ship fluoride waste to mill. Cost is for fluoride waste buried onsite.

Deferred decontamination is estimated to require about 8 months (plus 6 months for planning and preparation), at a cost of approximately \$6.5 million.

Manpower costs represent about 55% of the total cost of deferred decontamination. Manpower costs include both support staff and decommissioning worker labor costs. In Table 10.4-1, manpower costs are shown separately for the planning and preparation and the decommissioning phases of deferred decontamination. Details of manpower requirements and costs for deferred decontamination are given in Section 10.4.1.

Material and equipment costs represent about 3.5% of the total cost of deferred decontamination. Details of material and equipment requirements and costs are given in Section 10.4.2.

Waste management costs represent about 36% of the total cost of deferred decontamination. Details of waste management requirements and costs are given in Section 10.4.3.

Miscellaneous owner expenses are assumed to be the same for deferred decontamination as for DECON, measured in 1981 dollars.

10.4.1 Manpower Requirements and Costs for Deferred Decontamination

Estimates are made of the work force required to plan and execute the decommissioning activities for deferred decontamination. These work force estimates are used, together with unit manpower costs given in Section H.1 of Appendix H, to estimate deferred decontamination manpower costs. It is assumed that the work force organizational chart for deferred decontamination is similar to that for DECON, as shown in Figure 10.1-1. The same basic operations are performed during deferred decontamination as are performed during DECON, with the following exceptions:

- Lagoon covers will have to be removed, but not having to drain the lagoons will compensate.
- Additional time and manpower (about 14 man-weeks) will be required to remove seals and barricades erected during preparations for safe storage, to restore services, and to perform a small amount of manual decontamination and cleanup.

- Because deferred decontamination occurs 10 to 30 years after plant shutdown, training of the decommissioning staff will be necessary during the planning and preparation phase.

Table 10.4-2 shows manpower requirements and costs for planning and preparation, and Table 10.4-3 shows manpower requirements and costs for the decommissioning phase of deferred decontamination. A total of about 12 man-years is estimated to be required for planning and preparation, at a labor cost of about \$589,000. The decommissioning staff is generally sized and structured on a 1-shift, 5-day week.

TABLE 10.4-2. Summary of Manpower Utilization and Staff Costs for Planning and Preparations Phase of Deferred Decontamination

<u>Title or Function</u>	<u>Man-Years</u>	<u>Cost (\$ thousands)^(a,b)</u>
Project Manager	1.00	89
Project Engineer	0.75	57
Health and Safety Supervisor	0.67	40
Contracts and Accounting Specialist	0.67	32
Radioactive Shipment Specialist	0.38	15
Q.A. Engineer	0.67	31
Planning Engineer	0.67	35
Engineering Technician	0.67	26
Maintenance and Crafts Supervisor	0.67	31
Operations Supervisor	0.67	35
Foreman	1.50	67
Technician (training)	2.00	79
Craftsman (training)	0.50	16
Secretary	<u>1.50</u>	<u>36</u>
Total Man-Years	12.20	
Total Cost		589

(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.

TABLE 10.4-3. Summary of Manpower Utilization and Staff Costs for Decommissioning Phase of Deferred Decontamination

<u>Title or Function</u>	<u>Man-Years</u>	<u>Cost (\$ thousands)^(a,b)</u>
Project Manager	0.75	67
Project Engineer	0.75	57
Health and Safety Supervisor	0.75	45
Health Physics Technician	1.50	45
Security Force Supervisor	0.75	42
Security Patrolman	3.00	76
Contracts and Accounting Specialist	0.75	35
Radioactive Shipment Specialist	0.75	29
Q.A. Engineer	0.75	35
Planning Engineer	0.75	39
Engineering Technician	0.75	29
Maintenance and Crafts Supervisor	0.75	35
Custodian	0.75	23
Craftsman	3.00	96
Operations Supervisor	0.75	39
Foreman	3.50	155
Technician	24.0	943
Secretary	<u>1.50</u>	<u>36</u>
Total Man-Years	45.5	
Total Cost		1826

(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.

A total of about 45 man-years is estimated to be required for decontamination activities, at a labor cost of about \$1.8 million. The total labor cost for deferred decontamination is estimated to be about \$2.4 million without contingencies.

10.4.2 Material and Equipment Requirements and Costs for Deferred Decontamination

Estimates of material and equipment requirements and costs for deferred decontamination are shown in Table 10.4-4. Material and equipment costs are expected to be about the same for deferred decontamination as they are for DECON. A small additional cost is required for training supplies. The total material and equipment costs are estimated at about \$151,000.

TABLE 10.4-4. Estimated Material and Equipment Requirements and Costs for Deferred Decontamination

Description	Quantity	Estimated Unit Cost (\$ thousands)	Estimated Total Cost (\$ thousands)
Oxyacetylene Torch	4 ea.	1.0	4.0
Guillotine Pipe Saw	2 ea.	4.0	8.0
Tube Cutter	2 ea.	0.4	0.8
Ratcheting Pipe Cutter	6 ea.	0.07	0.4
Reciprocating Saw	4 ea.	0.8	3.2
Nibbler	2 ea.	1.0	2.0
High-Velocity Liquid Jet	1 ea.	6.9	6.9
Low-Velocity Liquid Jet	2 ea.	2.8	5.6
Hydraulic Concrete Surface Spalling Device	1 ea.	5.0	5.0
Concrete Drill	3 ea.	0.3	0.9
Electric/Pneumatic Hammer	2 ea.	0.7	1.4
Portable A-Frames	2 ea.	4.1	8.2
Portable Wash Tanks	2 ea.	2.8	5.6
Portable Spray Clean Booth	1 ea.	5.5	5.5
Portable Greenhouse Erection Kit	1 ea.	2.8	2.8
Hand Powered Brushes	20 ea.	0.2	4.0
HEPA Filter	5 ea.	0.2	1.0
Roughing Filter	20 ea.	0.1	2.0
Decontamination Chemicals			11.0
Cleaning Supplies			22.1
Expendable Tools			11.0
Protective Clothing (including laundry)			19.9
Office Supplies: Planning and Preparation			13.2
Decommissioning			6.6
Total			151.0 ^(a,b)

(a) Total is shown as direct addition of prior numbers to retain calculational information. Precision is less than shown.

(b) Contingency of 25% is not included.

10.4.3 Waste Management Requirements and Costs for Deferred Decontamination

The estimated weights, volumes, and number of shipments of decommissioning wastes from deferred decontamination are the same as from DECON of the plant, as given in Table 10.1-5. The estimated costs of containers, transportation and disposal of the radioactive wastes from buildings and of the miscellaneous waste buried onsite during operation for deferred decontamination are the same as from DECON, which are summarized in Table 10.1-6. Cost estimates are based on projected packaging and shipping requirements in Table 10.1-5 and on waste management cost data in Section H.2 of Appendix H. The total waste management cost for deferred decontamination of the buildings is estimated to be \$509,000 without contingencies. Lagoon waste management costs for deferred decontamination are the same as for DECON and are summarized in Table 10.1-7.

10.5 COST ESTIMATES FOR DECON OF PLANT AND EQUIPMENT WITH ONSITE DISPOSAL OF LAGOON WASTES

The estimated cost for DECON of the reference plant and equipment with onsite burial of the lagoon waste is summarized in Table 10.5-1. The total decommissioning cost is estimated to be \$4.6 million, with an annual cost for long-term care estimated to be \$11,000.

The cost of onsite burial of the lagoon waste is summarized in Table 10.5-2. If the solvent extraction waste is buried onsite with site stabilization provided, the cost is estimated to be \$210,200 including contingency, with an annual cost for long-term care of \$8,000. The cost of onsite burial of the fluoride waste is \$68,600 including contingency, with an annual cost of long-term care of \$3,000.

The total cost of onsite burial of the lagoon waste is \$278,800 including contingency, with an annual cost for long-term care of \$11,000.

TABLE 10.5-1. Summary of Estimated Cost for DECON of Plant with Onsite Burial of Lagoon Waste

<u>Cost Category</u>	<u>Cost in Millions of 1981 Dollars^(a)</u>
Manpower	
Planning and Preparation	0.304
Decommissioning	1.749
Equipment and Supplies	0.149
Disposal of Radioactive Material	0.509
Miscellaneous Owner Expense	<u>0.740</u>
Subtotal	3.451
<u>Lagoon Waste Management Options</u>	<u>Onsite Burial</u>
Extraction Waste ^(b)	0.168
Fluoride Waste ^(c)	<u>0.055</u>
Subtotal (includes lagoon waste)	3.674
25% Contingency	<u>0.919</u>
Total Costs	4.593

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

(b) Not feasible to ship extraction waste to commercial dump. Cost is for extraction waste shipped to mill.

(c) Not feasible to ship fluoride waste to mill. Cost is for fluoride waste buried onsite.

TABLE 10.5-2. Cost of Onsite Burial of Lagoon Waste^(a)

<u>Type of Waste</u>	<u>Cost in 1981 Dollars</u>	
	<u>Onsite Burial with Stabilization</u>	<u>Annual Care</u>
Solvent Extraction	210 200	8 000
Fluoride	<u>68 600</u>	<u>3 000</u>
Total Costs	278 800	11 000

(a) Includes 25% contingency.

REFERENCES

1. Title 10, Code of Federal Regulations, Parts 140.31 and 170.32, September 1978.
2. R. I. Smith, G. J. Konzek and W. E. Kennedy, Jr., Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0130, U.S. Nuclear Regulatory Commission Report by Pacific Northwest Laboratory, June 1978.

11.0 PUBLIC AND OCCUPATIONAL SAFETY DURING DECOMMISSIONING OF THE REFERENCE UF₆ CONVERSION PLANT

Public and occupational safety impacts from decommissioning activities at the reference UF₆ conversion plant are summarized in this section. The safety evaluation includes a consideration of the impacts of public radiation exposure, occupational radiation exposure, industrial accidents, and chemical pollutants. This evaluation utilizes current data and state-of-the-art methods to estimate the information required. A conservative approach, using parameters that tend to maximize the consequences, is used to evaluate the safety impacts of each decommissioning task. Safety assessment details are provided in Appendix I.

The safety evaluation is divided into three major parts: public safety, occupational safety, and transportation safety. Within each of these major parts are discussions of the radiological and nonradiological impacts of both routine and accident situations. Public radiological considerations are determined by using the atmospheric release scenarios in Appendix I and the radiation dose evaluation methods presented in Appendix E. Occupational radiation doses are estimated using information about expected dose rates and man-hour job requirements discussed in Section 10 and Appendices C, G and H.

The decommissioning alternatives selected for analysis are OECON and passive SAFSTOR, as discussed in Sections 4 and 9. The radiological safety evaluation is accomplished by calculating radiation doses to the public from airborne radionuclide releases and to the decommissioning and transportation workers from direct external exposure.

The fifty-year committed dose equivalent to the public from routine airborne releases during DECON (transportation not included) is calculated to be about 4.0 man-rem to the lungs, and the 50-year committed dose equivalent to the maximum-exposed individual from the worst postulated accident is calculated to be about 0.13 rem to the lungs. Radiation doses to the public resulting from preparations for safe storage are significantly less than those calculated for DECON since no major decontamination or demolition tasks are required.

The estimated occupational radiation doses are calculated to be about 62 man-rem for OECON, about 0.12 man-rem for preparations for safe storage, about 0.087 man-rem for each year of safe storage, and about 62 man-rem for deferred decontamination (the same as for DECON).

Radioactive waste material transportation activities associated with decommissioning are estimated to give a radiation dose to the total population along the transport route of about 1.7 man-rem for DECON or deferred decontamination. Occupational doses from the transportation of radioactive wastes are estimated to be about 1.7 man-rem from DECON or deferred decontamination shipments. These doses include the impacts of 1877 shipments of wastes recovered from the extraction waste pond and the fluoride lagoons.

11.1 TECHNICAL APPROACHES

The safety evaluation is divided into two areas of interest: radiological safety and nonradiological safety. Radiological safety is evaluated using a three-part technical approach. First, a description of the reference facility is developed (see Section 7). Second, the radionuclide inventories and external dose rates within the facility are characterized and quantified (see Appendix C). Finally, reference decommissioning tasks are defined for each alternative to permit calculation of radiation exposures (discussed in Appendix I.) The nonradiological safety evaluation is based on industrial and transportation accidents that result in injuries or fatalities. The technical approach is divided into two parts. First, the total labor requirements for each decommissioning alternative are analyzed and divided into categories of effort (discussed in Appendices G and I); second, injuries and fatalities are calculated based on statistical information from the literature on accident frequencies for the different categories of effort.

Key assumptions are made during the safety evaluation to coordinate the parts of each of the technical approaches. Some of the major assumptions are:

1. The maximum potential radiological consequences of a given decommissioning task are associated with performing that task in the area of the UF_6 conversion plant with the highest radionuclide inventory.

2. The plant equipment areas are kept relatively free of radioactive contamination during the operating lifetime to permit operational maintenance. As a result, expected radioactive contamination levels are generally modest and are reasonably consistent with the quality of operation expected in modern UF_6 conversion plants.
3. The maximum release from a specific decommissioning task applies to that task whenever it is used in the facility. In performing the dose calculations for releases of radionuclides from routine tasks, the estimated total releases for the entire decommissioning period are released at a uniform rate during a 1-year period. This process results in a conservatively large estimate of releases and doses, but this conservatism is used to compensate for uncertainties in the analysis.
4. Radiation protection techniques applied conform to the principle of keeping occupational radiation dose as low as reasonably achievable (ALARA).
5. All radioactive wastes are shipped 800 km by truck to a commercial shallow-land burial ground.
6. For natural uranium, inhalation of airborne radionuclides is the dominant radiation exposure pathway to members of the public during decommissioning.
7. The dominant radiation exposure pathway to the decommissioning worker is the external radiation received during normal decommissioning operations. Workers are assumed to wear adequate respiratory protection gear to prevent significant internal deposition of radionuclides.
8. For onsite waste management from the extraction pond and fluoride lagoons, the only option considered for public and occupational dose calculations is the exhumation, packaging, and offsite disposal option. Methods similar to those defined in Reference 1 are used to estimate atmospheric releases and occupational doses from waste exhumation and packaging tasks.

Other specific assumptions for calculating public and occupational safety impacts for DECON or SAFSTOR tasks are discussed in Appendix I.

11.2 PUBLIC SAFETY ASPECTS OF DECOMMISSIONING THE REFERENCE UF₆ CONVERSION PLANT

The consequences of atmospheric releases of radioactivity during routine decommissioning tasks 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 routine atmospheric releases are direct external exposure, inhalation, and ingestion of food products. The consequences of postulated accidents are determined by calculating inhalation radiation doses to the maximum-exposed individual. The radiation dose calculations for both the routine and accidental releases use the environmental information discussed in Appendix B and the radiation dose models and parameters discussed in Appendix E. These calculations use current data and methodology to quantify the atmospheric releases and obtain results that are useful in comparing the decommissioning alternatives discussed in this study. The following sections contain summaries of the calculated radiation doses to the public during DECON and passive SAFSTOR.

11.2.1 Public Radiation Doses from Routine Decommissioning Tasks

Loss of confinement of radioactive materials resulting in public radiation exposure is a primary safety concern during decommissioning. The primary sources of radioactive effluents from routine decommissioning operations are the release of contaminated vaporized metal during cutting and equipment removal, the release of contaminated concrete dust during decontamination or removal of concrete structures, and the release of contaminated aerosols during chemical decontamination. Equipment and concrete removal operations are minimal during preparations for safe storage.

A complete discussion of the methods used to calculate atmospheric releases during decommissioning is found in Appendix I. The calculated atmospheric releases for DECON and preparations for safe storage are shown in Table I.2-4. Complete listings of the calculated radiation doses to the maximum-exposed individual and to the public residing within 80 km of the site are given in Tables I.2-5 and I.2-6. These doses are listed for each area and general method considered. Summaries of these doses are shown in Tables 11.2-1 and 11.2-2 for the maximum-exposed individual and the population, respectively. The doses

TABLE 11.2-1. Summary of Calculated Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine Decommissioning Tasks^(a)

Alternative/Area	First-Year Dose (mrem)		Fifty-Year Committed Dose Equivalent (mrem)	
	Total Body	Lung	Total Body	Lung
DECON				
Main Process Building	1.3×10^{-5}	2.8×10^{-3}	1.7×10^{-5}	1.3×10^{-2}
Onsite Waste Exhumation	3.3×10^{-3}	6.7×10^{-1}	5.6×10^{-6}	3.3×10^{-1}
Totals	3.3×10^{-3}	6.9×10^{-1}	5.9×10^{-3}	3.3×10^0
Preparation for Passive Safe Storage				
Main Process Building	3.1×10^{-7}	6.3×10^{-5}	5.4×10^{-7}	3.1×10^{-4}
Onsite Waste Stabilization	-- ^(b)	--	--	--
Totals	3.1×10^{-7}	6.3×10^{-5}	5.4×10^{-7}	3.1×10^{-4}

(a) More detailed information for each area and general decommissioning method is shown in Table I.2-5.

(b) Dashes indicate no doses resulted for this area and alternative.

TABLE 11.2-2. Summary of Calculated Radiation Doses to the Population from Atmospheric Releases During Routine Decommissioning Tasks^(a)

Alternative/Area	First-Year Dose (man-rem)		Fifty-Year Committed Dose Equivalent (man-rem)	
	Total Body	Lung	Total Body	Lung
DECON				
Main Process Building	1×10^{-5}	3×10^{-3}	2×10^{-5}	1×10^{-2}
Onsite Waste Exhumation	3×10^{-3}	8×10^{-1}	6×10^{-3}	4×10^0
Totals	3×10^{-3}	8×10^{-1}	6×10^{-3}	4×10^0
Preparation for Passive Safe Storage				
Main Process Building	3×10^{-7}	8×10^{-3}	5×10^{-7}	3×10^{-2}
Onsite Waste Stabilization	-- ^(b)	--	--	--
Totals	3×10^{-7}	8×10^{-5}	5×10^{-7}	3×10^{-4}

(a) More detailed information for each area and decommissioning method is shown in Table I.2-6. All calculated population doses are rounded to one significant figure, and are for the population of 3.5 million people residing within an 80-km radius of the site.

(b) Dashes indicate no doses were calculated.

shown in these tables are for total body and lung, for both the first-year dose and the fifty-year committed dose equivalent. The calculated doses for DECON are about a factor of 10^4 larger than the doses for preparations for safe storage. The radiation doses shown in Tables 11.2-1 and 11.2-2 are extremely small by comparison to the range of annual radiation dose to an individual from natural background in the United States (from 80 to 170 mrem per year).⁽²⁾

The release of uranium during safe storage is expected to be negligible compared to the releases during either DECON or preparations for safe storage. This is because of the steps taken during preparations for safe storage to isolate the uranium within the plant, and steps taken during safe storage to limit human activities within the plant. Thus, no public doses are calculated for safe storage. Since the radionuclides in the uranium mixture are generally long-lived, for this study the doses from deferred decontamination are assumed to be equal to those calculated for DECON.

11.2.2 Public Radiation Doses from Postulated Accidents During Decommissioning

The consequences of postulated decommissioning accidents that result in atmospheric releases of radioactivity are determined by calculating the inhalation dose to the maximum-exposed individual. For DECON tasks, postulated accidents are discussed and analyzed in Section I.2.2 of Appendix I. Using engineering judgment, a general estimate of the frequency of occurrence of the level of atmospheric release is made for each accident. The frequency of occurrence is judged to be "high" if the occurrence of a release of similar magnitude per year is greater than 10^{-2} , "medium" if between 10^{-2} and 10^{-5} , and "low" if less than 10^{-5} . While it is beyond the scope of this study to evaluate every potential accident for each decommissioning alternative, an attempt is made to identify the most significant potential accidents associated with DECON tasks. Accidents during preparations for safe storage are determined by direct comparison with DECON, with no attempt at further analysis. Thus, several of the accidents postulated for DECON do not apply to preparations for safe storage, since the latter does not involve the removal of activated concrete or components.

A summary of the postulated accidents considered in this study is given in Table 11.2-3. These accidents are listed in order of decreasing magnitude of atmospheric release. First-year radiation doses and fifty-year committed radiation dose equivalents are listed for the total body and lung of the maximum-exposed individual. The accident that is postulated to result in the largest atmospheric release of radioactivity is the oxyacetylene explosion during the removal of piping and equipment in Area 2, the ADU slurry receiving and storage area. This explosion is assumed to occur in the contamination control envelope, releasing the uranium collected in the control envelope HEPA filters. It is calculated that 1.8×10^{-3} Ci of uranium could be released. The frequency of occurrence of this accident is estimated to be "medium." Transportation accidents are included in Table 11.2-3 for comparison purposes. These accidents are discussed in Section 11.4.

11.2.3 Nonradiological Public Safety Evaluation

Chemical pollutants that could be released during decommissioning activities are examined and the quantities released are found to have an insignificant safety impact on the public. Potentially hazardous chemicals are found to come from two sources: 1) residuals from UF_6 conversion plant production operations, and 2) chemicals employed to chemically and physically decontaminate the plant. From the relatively small quantities of hazardous chemicals used, the low likelihood of their dispersal into the environs, and the dilution factors involved in the dispersal of hazardous material from the plant to the environs, it can be concluded that chemical pollutants from decommissioning operations do not pose a significant public hazard. A more detailed discussion of the potential chemicals and their impacts is given in Section I.6 of Appendix I.

TABLE 11.2-3. Summary of Radiation Doses to the Maximum-Exposed Individual from Postulated UF₆ Conversion Plant Decommissioning Accidents

Incident	Frequency of Occurrence (a)	Total Atmospheric Release (Ci/h) (b)	DECON				Preparations for Safe Storage			
			First-Year		Fifty-Year		First-Year		Fifty-Year	
			Total Body	Lung	Total Body	Lung	Total Body	Lung	Total Body	Lung
Oxyacetylene Explosion	Medium	1.8×10^{-3}	1.5×10^{-4}	5.1×10^{-2}	8.3×10^{-4}	1.3×10^{-1}	-- (c)	--	--	--
Contamination Control Envelope Rupture	High	2.9×10^{-5}	2.5×10^{-6}	8.1×10^{-4}	2.9×10^{-6}	2.0×10^{-3}	--	--	--	--
Vacuum Filter Bag Rupture	Medium	2.4×10^{-5}	2.0×10^{-6}	6.7×10^{-4}	2.4×10^{-6}	1.7×10^{-3}	2.0×10^{-6}	6.7×10^{-4}	2.4×10^{-6}	1.7×10^{-3}
Severe Transportation Accident	Low	1.1×10^{-5}	4.3×10^{-4}	1.4×10^{-2}	5.3×10^{-5}	3.5×10^{-2}	4.3×10^{-4}	1.4×10^{-2}	5.3×10^{-5}	3.5×10^{-2}
Contaminated Metal Cutting	High	7.2×10^{-6}	6.1×10^{-7}	2.0×10^{-4}	7.2×10^{-7}	5.0×10^{-4}	--	--	--	--
Minor Transportation Accident	Low	2.8×10^{-7}	1.1×10^{-5}	3.6×10^{-4}	1.3×10^{-5}	9.0×10^{-4}	1.1×10^{-5}	3.6×10^{-4}	1.3×10^{-5}	9.0×10^{-4}
Gross Loop Chem. Decon. Leak	Low	1.9×10^{-7}	1.6×10^{-8}	5.3×10^{-6}	1.9×10^{-8}	1.3×10^{-5}	--	--	--	--
Contaminated Solid Waste Spill	High	1.3×10^{-7}	1.1×10^{-8}	3.6×10^{-6}	1.3×10^{-8}	9.1×10^{-6}	--	--	--	--
Explosion of LPG Leaked from Front-End Loader	Low	1.0×10^{-7}	8.5×10^{-9}	2.8×10^{-6}	1.0×10^{-8}	7.0×10^{-6}	--	--	--	--
Contaminated Sweeping Compound Fire	Medium	2.0×10^{-8}	1.7×10^{-9}	5.6×10^{-7}	2.0×10^{-9}	1.4×10^{-6}	1.7×10^{-9}	5.6×10^{-7}	2.0×10^{-9}	1.4×10^{-6}
Combustible Waste Fire	High	2.0×10^{-9}	1.7×10^{-10}	5.6×10^{-8}	2.0×10^{-10}	1.4×10^{-7}	1.7×10^{-10}	5.6×10^{-8}	2.0×10^{-10}	1.4×10^{-7}

(a) The frequency of occurrence considers not only the probability of the accident, but also the probability of an atmospheric release of similar magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release is $>10^{-2}$ per year, "medium" if between 10^{-2} and 10^{-5} , and "low" if $<10^{-5}$.

(b) All releases are assumed to occur in a 1-hour period for comparison purposes.

(c) Dashes indicate that the accident shown does not apply for the alternative shown.

11.3 OCCUPATIONAL SAFETY EVALUATION OF DECOMMISSIONING THE REFERENCE UF₆ CONVERSION PLANT

Occupational safety impacts for DECON and SAFSTOR are evaluated for both radiological and nonradiological events. The analysis considers routine radiological events and postulated nonradiological accidents.

Radiation doses to workers are calculated based on estimated radiation dose rates for various areas of the reference UF₆ conversion plant and on the estimated labor requirements to perform the decommissioning work. Summaries of the appropriate detailed information in Appendices C and I are given in this section. An estimate of worker injuries and fatalities resulting from decommissioning tasks is also presented, based on nuclear industry experience.

11.3.1 Radiological Safety Evaluation of Routine Decommissioning Tasks

Summaries of the estimated worker hours and occupational radiation exposures for DECON and SAFSTOR are given respectively in Tables 11.3-1 and 11.3-2. These tables summarize tasks in each area of the reference UF₆ conversion plant, man-hour estimates for each area, and man-rem estimates of the accumulated external radiation dose to all workers in each area. Detailed estimates of occupational radiation doses, showing worker categories and average dose rates, are found in Section I.4 of Appendix I.

The radiation doses to decommissioning workers are calculated using the manpower requirements estimated for each area and estimates of the average radiation dose rates associated with each area. The dose rate estimates are based on the data given in Reference 3 for an operating uranium fabrication facility. The radiation doses computed in Appendix I are based on constant values of dose rates for a work area, regardless of where the worker is located within the area.

The major radionuclides within the facility have long half-lives and do not decay significantly within the few years considered in this study. Therefore, occupational radiation doses do not decrease with time.

TABLE 11.3-1. Estimated Occupational Radiation Dose for DECON

Area	Estimated Total (man-hours)(a)	Estimated Total Occupational Dose (man-rem)
Area 1: Drum Receiving	180	6.6×10^{-3}
Area 2: ADU Slurry	255	8.4×10^{-1}
Area 3: Sampling Station	285	9.5×10^{-1}
Area 4: Laboratory	720	2.7×10^{-2}
Area 5: UF ₄ Slurry	1275	4.1×10^0
Area 6: Ore Dissolution	1740	5.8×10^0
Area 7: Solvent Extraction	2700	7.7×10^0
Area 8: Uranyl Nitrate	2315	2.4×10^{-1}
Area 9: Denitration	2460	2.7×10^{-1}
Area 10: Nitric Acid	1260	1.4×10^{-1}
Area 11: Reduction Systems	2315	2.2×10^{-1}
Area 12: Hydrofluorination	1905	2.1×10^{-1}
Area 13: Fluorination	2263	2.5×10^{-1}
Area 14: Cylinder Loading	1410	2.1×10^{-2}
Area 15: F ₂ Production	630	2.4×10^{-2}
Area 16: F ₂ Cell	510	1.8×10^{-2}
Area 17: Instrument Repair	1050	3.9×10^{-2}
Area 18: Ammonia Cracking	735	2.7×10^{-2}
Area 19: Waste Gas Disposal	525	1.9×10^{-2}
Area 20: H ₂ Condensing	477	1.8×10^{-2}
Area 21: Temporary Radwaste	300	3.1×10^{-2}
Area 22: DECON Area	180	2.1×10^{-2}
Area 23: Incinerator	660	2.5×10^{-2}
Area 24: Laundry	555	2.1×10^{-2}
Area 25: Overall Cleanup	180	3.1×10^{-2}
Building Area Totals	2.7×10^4	2.1×10^1

TABLE 11.3-1. (Contd)

Area	Estimated Total (man-hours) ^(a)	Estimated Total Occupational Dose (man-rem)
Site Areas: ^(b)		
Extraction Waste Pond	5670	1.8×10^1
Fluoride Waste Lagoons	6060	2.0×10^1
Misc. Buried Waste	735	2.4×10^0
Drum Storage	495	1.8×10^{-2}
Cylinder Storage	1275	4.7×10^{-2}
Plant Stack	660	2.5×10^{-2}
Overall Cleanup	1095	4.0×10^{-2}
Site Area Totals	1.6×10^4	4.1×10^1
DECON Totals	4.3×10^4	6.2×10^1

(a) Based on a 6-hour effective work day.

(b) Only the waste exhumation, packaging, offsite disposal option is considered.

The total occupational radiation dose for in-plant decommissioning tasks for DECON is estimated to be about 21 man-rem. For onsite waste exhumation, an additional occupational dose of 41 man-rem is calculated; thus the total occupational dose for DECON is about 62 man-rem. Specific areas that result in the highest accumulated worker doses are: 1) onsite waste relocation, 2) Area 7, solvent extraction, 3) Area 6, ore concentrate dissolution, 4) Area 5, UF_6 slurry processing, 5) Area 3, sampling station, and 6) Area 2, ADU slurry receiving and storage. The estimated occupational dose for tasks during preparations for safe storage is about 0.12 man-rem. During safe storage, the radiation doses are estimated to be about 8.7×10^{-2} man-rem per year, and since most of the radionuclides present have long half-lives, the doses for deferred decontamination are considered to be the same as those estimated for DECON.

The estimated external occupational radiation doses for decommissioning the reference UF_6 conversion plant are summarized in Table 11.3-3. The total

TABLE 11.3-2. Estimated Occupational Radiation Dose for SAFSTOR

Phase/Task	Estimated Totals (man-hours) ^(a)	Estimated Total Occupational Dose (man-rem)
Preparations for Safe Storage:		
Equipment Deactivation		
Audit of Pumps and Pipelines	180	1.9×10^{-2}
Secure Valves, Hoops, Conveyors	210	2.3×10^{-2}
Disconnect Unneeded Services	270	9.9×10^{-3}
Tag Equipment for Status	210	7.7×10^{-3}
Safety Audit	210	7.7×10^{-3}
Isolation of Areas		
Install Locks and Signs	120	4.4×10^{-3}
Building Systems Check	180	6.6×10^{-3}
Install Fence and Locks	90	6.6×10^{-4}
Final Preparations		
Complete Offsite Shipments	225	6.1×10^{-3}
Install Intrusion Alarms	300	1.1×10^{-2}
Install Rad. Monitoring System	240	8.8×10^{-3}
Comprehensive Rad. Survey	270	1.8×10^{-2}
Building Area Totals	2.5×10^3	1.2×10^{-1}
Onsite Waste Stabilization	540	6.5×10^{-2}
Totals for Preparations	3.0×10^3	1.8×10^{-1}
Safe Storage Period ^(b)	1170	8.7×10^{-2}
Deferred Decontamination ^(c)	4.3×10^4	6.2×10^1

(a) Based on a 6-hour effective work day.

(b) Labor and dose estimates per year during safe storage.

(c) Labor and dose estimates same as for DECON.

TABLE 11.3-3. Summary of Estimated Collective Occupational Radiation Doses at the Reference UF₆ Conversion Plant

<u>Decommissioning Alternative</u>	<u>Time After Facility Shutdown (Years)</u>	<u>Estimated Dose (man-rem) (a)</u>
DECON	0	62
Preparations for Safe Storage	0	0.18
Safe Storage Period	10	0.87
	30	2.6
	50	4.4
Deferred Decontamination	0 to 50	62
Totals for Passive SAFSTOR	10	63
	30	65
	50	67

(a) Rounded to two significant figures.

occupational dose is given for DECON, and a breakdown of passive SAFSTOR activities into preparations for safe storage, the safe storage period, and deferred decontamination is presented.

Estimates of occupational radiation dose are sensitive to management philosophy and to the decommissioning methods utilized. Administrative controls are assumed to be in place that monitor radiation records for each individual and assure that no one worker exceeds recommended limits. Estimates contained in Table 11.3-3 are based on decommissioning methods that utilize technicians who are highly trained in effective radiation work procedures. Different basic assumptions, decommissioning procedures, or increased manpower may change the occupational radiation dose estimates significantly.

11.3.2 Evaluation of Industrial Accidents

As a result of decommissioning activities, the potential exists for worker injuries and fatalities. As with any industrial operation, proper management and industrial safety practices will minimize the potential for worker accidents.

The following estimates of worker injuries and fatalities are based on data provided by the U.S. AEC for the period 1943 to 1970.⁽⁴⁾ Table 11.3-4 lists the estimates of worker injuries and activities that are conducted during DECON and during preparations for safe storage.⁽⁵⁾ As shown in the table, about 0.25 lost-time injuries and 0.002 fatalities are calculated for DECON and about 0.02 lost-time injuries and 0.0001 fatalities are calculated for preparations for safe storage.

TABLE 11.3-4. Estimated Occupational Lost-Time Injuries and Fatalities During DECON and Preparations for Safe Storage

Activity	Frequency ^(a) (Accidents Per 10 ⁶ man-hours)		DECON			Preparations for Safe Storage		
	Lost-Time Injuries	Fatalities	Estimated Man-Hours ^(b)	Lost-Time Injuries	Fatalities	Estimated Man-Hours ^(c)	Injuries	Fatalities
Heavy Construction	10.0	4.2×10^{-2}	7.8×10^3	7.8×10^{-2}	3.3×10^{-4}	5.6×10^2	5.6×10^{-3}	2.4×10^{-5}
Light Construction	5.4	3.0×10^{-2}	2.1×10^4	1.1×10^{-1}	6.3×10^{-4}	1.5×10^3	8.1×10^{-3}	4.5×10^{-5}
Operational Support	2.1	2.3×10^{-2}	2.9×10^4	6.1×10^{-2}	6.7×10^{-4}	2.0×10^3	4.2×10^{-3}	4.6×10^{-5}
Totals			5.8×10^4	2.5×10^{-1}	1.6×10^{-3}	4.1×10^3	1.8×10^{-2}	1.2×10^{-4}

(a) The frequency of injury and fatality accidents is taken from Reference 4.

(b) Man-hour estimates for DECON are from Table H.2-1 in Appendix H.

(c) Man-hour estimates for preparations for safe storage are from Table H.3-1 in Appendix H.

Estimates of the number of injuries and fatalities that could occur to the surveillance and maintenance staff from industrial-related accidents for safe storage activities are given in Table 11.3-5. As shown in this table, about 0.08 lost-time injuries and 5.6×10^{-4} fatalities are estimated over 10 years of safe storage.

TABLE 11.3-5. Estimated Occupational Lost-Time Injuries and Fatalities During Safe Storage

Occupation ^(a)	Estimated Man-Hours/yr	Storage for 10 Years		Storage for 30 Years		Storage for 50 Years	
		Lost-Time Injuries	Fatalities	Lost-Time Injuries	Fatalities	Lost-Time Injuries	Fatalities
Facility Manager	500	1.1×10^{-2}	1.2×10^{-4}	3.3×10^{-2}	3.6×10^{-4}	5.5×10^{-2}	6.0×10^{-4}
Maintenance and Operations Man	1000	5.4×10^{-2}	3.0×10^{-4}	1.6×10^{-1}	9.0×10^{-4}	2.7×10^{-1}	1.5×10^{-3}
Secretary	500	1.1×10^{-2}	1.2×10^{-4}	3.3×10^{-2}	3.6×10^{-4}	5.5×10^{-2}	6.0×10^{-4}
Three-Party Inspection Team	60	1.3×10^{-3}	1.4×10^{-5}	3.9×10^{-3}	4.2×10^{-5}	6.5×10^{-3}	7.0×10^{-5}
Totals	2060	7.7×10^{-2}	5.6×10^{-4}	2.3×10^{-1}	1.7×10^{-3}	3.9×10^{-1}	2.8×10^{-3}

(a) All lost-time and fatality estimates are based on the operational support accident frequency data shown in Table 11.3-4, except for the maintenance and operations category whose estimates are based on the light construction accident frequencies. The labor estimates (in man-hours/yr) are from Table 10.3-2.

11.4 TRANSPORTATION SAFETY

Radioactive wastes collected during decommissioning tasks at the reference UF_6 conversion plant are assumed to be packaged and shipped offsite as part of planned operations. These wastes are assumed to be shipped by truck to a low-level waste burial ground located 800 km away. The methods used to estimate radiation doses to transportation workers and members of the public along the transportation route are based on information in Reference 6. The discussion of transportation accidents resulting in atmospheric releases of radioactivity is based on information contained in Reference 7. The following sections contain a summary of the radiation dose calculations discussed in Section I.5 of Appendix I, and estimates of casualties resulting from traffic accidents during waste transport. Radiation doses received by workers unloading the radioactive waste at a repository or disposal site are not estimated in this study since they are assumed to occur at a separate licensed facility. For onsite wastes recovered from the extraction pond and fluoride lagoons, the only option considered is the removal, packaging, and offsite disposal option.

11.4.1 Radiation Doses from Routine Decommissioning Transportation Tasks

Department of Transportation (DOT) regulations⁽⁸⁾ set the following exposure limits for shipments of radioactive material:

- 1000 mR/hr at 1 m from the external surface of any package transported in a closed vehicle
- 200 mR/hr at the external surface of the vehicle
- 10 mR/hr at any point 2 m from the vehicle
- 2 mR/hr at any normally occupied position in the vehicle.

Each shipment of wastes collected within the main process building is assumed to contain enough radioactive material to result in the maximum exposure rates allowed by these DOT regulations. Wastes collected onsite from the pond and lagoons are assumed to have dose rates that are only 10% of these limits.

For DECON, the amount of radioactive waste collected is shown in Table 10.1-5 to require 51 shipments from the main process building and 1877 shipments from recovered onsite wastes. The calculated radiation doses from routine waste transportation tasks are listed in Table 11.4-1. Total doses for DECON are calculated to be 17 man-rem to transport workers and 1.7 man-rem to members of the public. If the 1877 shipments of exhumed onsite waste are omitted, disposal of wastes from the main process building result in 3.6 man-rem to transport workers and 0.35 man-rem to members of the public.

TABLE 11.4-1. Calculated Radiation Doses from Routine Radioactive Waste Transportation

Alternative/Group	Radiation Dose per Shipment (man-rem) ^(a)	Number of Shipments	Total Population Dose per Group (man-rem) ^(b)
<u>DECON-Building Wastes</u>			
Truck Drivers	6.7×10^{-2}	51	3.4
Garagemen	3.3×10^{-3}	51	0.17
Total Transport Worker Dose			3.6
Onlookers	5.0×10^{-3}	51	0.26
General Public	1.8×10^{-3}	51	0.092
Total Public Dose			0.35
<u>DECON-Site Wastes</u>			
Truck Drivers	6.7×10^{-3}	1877	13
Garagemen	3.3×10^{-4}	1877	0.62
Total Transport Worker Dose			14
Onlookers	5.0×10^{-4}	1877	0.94
General Public	1.8×10^{-4}	1877	0.38
Total Public Dose			1.3
<u>Totals for DECON</u>			
Truck Drivers	-- ^(c)	1928 ^(d)	16
Garagemen	--	1928 ^(d)	0.79
Total Transport Worker Dose	--		17
Onlookers	--	1928 ^(d)	1.2
General Public	--	1928 ^(d)	0.47
Total Public Dose			1.7

(a) Based on one-way trips of 800 km.

(b) All doses are rounded to two significant figures.

(c) Dashes indicate that weighted average dose rates were not separately calculated.

(d) Total shipments are found by adding building waste plus site waste shipments.

11.4.2 Radiation Doses from Postulated Transportation Accidents

Transportation accidents have a wide range of severities. Most accidents occur at low vehicle speeds and have relatively minor consequences. In general, as speed increases, accident severity also increases. However, accident severity is not a function of vehicle speed only. Other factors such as the type of accident, the kind of equipment involved, and the location of the accident can have an important bearing on accident severity.

Furthermore, damage to a package in a transportation accident is not directly related to accident severity. In a series of accident of the same severity, or in a single accident involving a number of packages, damage to packages may vary from none to extensive. In relatively minor accidents, serious damage to packages can occur from impacts on sharp objects or from being struck by other cargo. Conversely, even in very severe accidents, damage to packages may be minimal.

Probabilities of rail and truck accidents and the calculation of airborne concentrations of radioactivity from such accidents are discussed in Section I.5.3 of Appendix I. Most of the information about moderate and severe accidents is obtained from Reference 9. Two realistic accidents involving combustible radioactive wastes in Type A packages are defined. Both, however, are judged to have a low frequency of occurrence. The calculated radiation doses to the lung of the maximum-exposed individual resulting from these accidents are shown in Table 11.2-3. These transportation accidents are ranked with the other postulated decommissioning accidents by order of magnitude of atmospheric release.

The severe transportation accident is assumed to involve rupture and fire in 40 waste containers, each containing about 5.5×10^{-4} Ci of uranium. The total fifty-year committed radiation dose equivalent to the lung of the maximum-exposed individual from the total release of 1.1×10^{-5} Ci is calculated to be 3.5×10^1 mrem. For the minor accident, only one container is assumed to rupture and burn. In this case, 2.8×10^{-7} Ci of uranium are released, resulting in a fifty-year committed dose equivalent to the lung of the maximum-exposed individual of about 9.0×10^3 mrem. A more complete discussion of the methods used to calculate these doses is found in Section I.5.3 of Appendix I.

11.4.3 Casualties from Traffic Accidents

As with any transportation task, a certain potential for accidental injury or death exists from traffic accidents during decommissioning tasks.⁽⁶⁾ A summary of the casualties calculated to result during the transportation tasks considered in this study is shown in Table 11.4-2. As shown in this table, 1.5 injuries and 9.0×10^{-2} fatalities are estimated for DECON shipments of radioactive wastes. This total is dominated by shipments associated with the offsite disposal of exhumed wastes.

TABLE 11.4-2. Estimated Casualties from DECON Transportation Accidents

Transportation Task	Accident Frequency ^(a) (Accidents per Vehicle Kilometer)	Injuries Per Accident	Fatalities Per Accident	Total Kilometers ^(b) (Round Trips)	Estimated Casualties ^(c)	
					Injuries	Fatalities
DECON:						
Building Wastes	1.0×10^{-6}	0.51	0.03	8.2×10^4	4.2×10^{-2}	2.5×10^{-3}
Wastes Exhumed Onsite	1.0×10^{-6}	0.51	0.03	3.0×10^6	1.5×10^0	9.0×10^{-2}
Totals for DECON				3.1×10^6	1.5×10^0	9.0×10^{-2}

(a) Accident frequencies are from Reference 6.

(b) Assuming a truck transportation distance of 1600 km round-trip, with 51 trips required for building wastes and 1877 trips required for exhumed wastes.

(c) Rounded to two significant figures.

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12.0 DISCUSSION OF STUDY RESULTS

There are no commercial UF_6 facilities for which decommissioning is reported in the open literature and for which direct comparisons can be made with the results of this study. DECON is the most likely decommissioning alternative for commercial UF_6 plants. DECON of a facility requires that contaminated components, building surfaces and materials either be: 1) decontaminated to unrestricted release levels, or 2) packaged and shipped to an authorized disposal site.

SAFSTOR would be an acceptable decommissioning alternative for facilities where the owner wished to keep open the option of later reactivating the facility or if waste disposal space is not available at the time of decommissioning.

If the owner wants to maintain the site for nuclear use, DECON of the plant with stabilization and long-term care of the waste lagoon site would be the least expensive method of decommissioning. However, the acceptability of this method by the NRC would have to be negotiated on a case-by-case basis.

Major conclusions of this study related to technology, cost, and safety are discussed in the following sections.

12.1 DECOMMISSIONING TECHNOLOGY

No major technical impediments exist to the successful decommissioning of uranium containment facilities. Decommissioning of UF_6 plants can be accomplished using technology and equipment that are in common industrial use.

The technology for the removal of radioactivity from contaminated surfaces is based on a cleaning technology that has been established for several decades. Almost all of the procedures and chemicals used in decontaminating nuclear equipment and facilities were first developed and used for cleaning equipment and facilities in non-nuclear industries.

Although no UF_6 plants have been decommissioned, experience has been gained in decommissioning fuel cycle facilities and decontamination techniques

are well documented.⁽¹⁻⁴⁾ Decommissioning involves many of the same procedures and techniques that are used to decontaminate a facility during its operating lifetime.

Decommissioning of the waste lagoons may involve onsite burial with site stabilization procedures or it may involve the removal of radioactive waste and contaminated soil. A variety of techniques exist for stabilizing the residues in the lagoons to prevent radionuclide transport. These techniques are described in detail in another report⁽⁵⁾ in this series. The removal of radioactive materials such as the waste lagoon residues can be accomplished using standard earthmoving techniques and equipment.

12.2 DECOMMISSIONING COSTS

The decontamination of uranium-contaminated facilities is a labor-intensive, hands-on effort. Thus, labor is a major fraction of the total decommissioning costs. Development of facility and equipment designs and decontamination systems and techniques that can minimize labor could reduce overall decommissioning costs for UF_6 facilities.

The cost of handling, packaging, transporting, and disposing of radioactive waste materials is a significant fraction of the total decommissioning cost. Development of facility designs and decontamination techniques that minimize the quantities of contaminated material that must be disposed of as radioactive waste could reduce overall decommissioning costs and the waste management burden.

One of the most significant waste management items for the reference UF_6 plant is the disposal of the waste lagoon residues. The solvent extraction wastes contain uranium that can be potentially valuable to the plant owner. It is estimated that the uranium residues will have a value of about \$2.8 million by the end of the 40-year plant life. Thus, there is an economic incentive to process the waste and recover the residual uranium in addition to the advantage of removing of the waste material, thereby completing the DECON alternative.

12.3 PUBLIC AND OCCUPATIONAL SAFETY

Workers engaged in the decommissioning of fuel cycle facilities generally experience the same levels of radiation exposure as workers engaged in normal facility operations. An exception exists for decommissioning operations that result in the production of significant quantities of airborne radioactivity. Operations in an environment with the potential for high inhalation exposure to radiation may require worker use of protective respiration equipment. The use of this equipment could result in a reduction of the inhalation dose by one or two orders of magnitude.

Decommissioning studies of reference fuel cycle facilities have shown that the 50-year committed dose equivalent to the populace living within 80 kilometers of the site, from airborne releases during normal decommissioning operations, is a small fraction of the dose that results from naturally occurring radiation. Potential radiation doses to the public from postulated decommissioning accidents are also found to be small. The impact on public safety from decommissioning of UF_6 plants is therefore estimated to be negligible.

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13.0 CONSIDERATIONS FOR THE FACILITATION OF DECOMMISSIONING

Title 10 Code of Federal Regulations, Part 50, Appendix F.4 describes the Nuclear Regulatory Commission's position regarding facilitation of decommissioning of fuel reprocessing plants: "A design objective shall be to facilitate decontamination and removal of all significant radioactive wastes at the time the facility is permanently decommissioned." Application of this NRC objective to other fuel cycle facilities is a logical extension of the intention of this regulation. In addition, NRC Regulatory Guide 8.8, Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations be as Low as is Reasonably Achievable (ALARA), explicitly points out that "design concepts and station features should reflect consideration of the activities of station personnel (including decontamination and decommissioning) that might be anticipated."

This study on decommissioning of a uranium hexafluoride conversion (UF_6) plant describes activities that can be used to conceptually decommission a reference facility. With this study as a basis, insights have been gained as to plant design characteristics that could simplify the task of decommissioning. This section summarizes some of these potential plant and equipment design features.

It is recognized that some of the considerations intended to enhance decommissioning may not always be compatible with those plant characteristics that are desirable for normal production operations. Some of the characteristics may be also be prohibitively expensive. However, the aim of this discussion is to point out design characteristics that would expedite and simplify the decommissioning task. These findings regarding desirable features for decommissioning are presented with no attempt to rank their relative importance or to determine impacts on the rest of the plant designs, on plant production operations, or on plant process performance. These insights are aimed only at areas that present obvious complexity or difficulty to decommissioning a UF_6 conversion plant; they are not all-inclusive, and do not consider details, side effects, or variations of the alternatives. Such an analysis would require an in-depth study beyond the scope of this report.

The general criteria used in selecting design features for consideration are based on the effect they might be expected to have on decreasing decommissioning cost, improving occupational or public safety, reducing total decommissioning time, creating less radioactive waste, and the general ease of performing the decommissioning. In evaluating new design features for future decommissioning application, appropriate balance must be maintained between designs that meet these criteria and potential negative effects on plant construction and operating costs and operating characteristics. For the considerations given below, qualitative comments are made about the possible effects a given design feature might have in satisfying the selection criteria.

Three periods in which specific decommissioning facilitation activities may be implemented are design and construction, operation, and decommissioning. The benefits of various facilitation activities within these periods are discussed in the following subsections.

13.1 DESIGN AND CONSTRUCTION

The design and construction period affords an early opportunity to take steps to facilitate decommissioning. Facilitation steps that can be taken during the two phases of this period are discussed below.

13.1.1 Design

Careful attention to design of the equipment and facility can help reduce efforts required for routine maintenance and eventual decommissioning. Standardization of plant design would allow direct application of experience gained in the operation and decommissioning of one facility to other similar plants. Improvement of access to contaminated equipment would reduce worker exposure and effort during maintenance and decommissioning. Installation of removable wall and ceiling panels could be used for this purpose. Minimizing crevices in process equipment and hoods would reduce holdup of radioactive materials, allowing easier decontamination. Hoods and process equipment designed for easy disassembling without cutting would facilitate maintenance and decontamination, possibly allowing extended use of the equipment for salvage as scrap upon decommissioning.

13.1.2 Construction

Accurate as-built drawings and photos should be kept to facilitate later maintenance, decontamination, and decommissioning efforts. Scale models of the facility with details for concrete pours and penetrations, and placement of reinforcing steel, and dry mock-ups of equipment would aid in developing better maintenance, decontamination and decommissioning procedures. Exposure and effort of workers could be reduced as a result.

Pretreatment of all surfaces subject to radioactive contamination and/or corrosion prior to operation of the facility would reduce effort required for later decontamination by minimizing the holdup of materials, making surfaces more corrosion resistant, and reducing the amount of surface material to be removed during decontamination.

Pre-polishing of metal piping and process equipment may be accomplished by vibratory finishing followed by electropolishing. Vibratory finishing is a process in which parts are placed in a bath containing abrasive particles in a cleaning solution and vibrated at high frequency. Electropolishing is essentially an electroplating process operated in reverse. When used together, these processes produce a very smooth surface which resists material holdup and corrosion. These processes can be most fully exploited by installing equipment for vibratory finishing and electropolishing at the facility during construction. The capability will then exist for use of these surface treatment techniques during all three periods of the plant life cycle.

Concrete or brick floors, walls, and ceilings may be made much more contamination-resistant by sealing or covering them. Concrete and brick surfaces may be sealed and covered using epoxy or phenolic coatings. Linoleum or steel plate may be used to cover areas of concrete surface. These coverings are preferable to tile, which may allow contaminants to leak into cracks. The surface of contaminated concrete usually must be removed to a depth of 4 inches for decommissioning purposes. Effective methods for preventing surface contamination of concrete and brick will greatly aid cleanup efforts, and reduce the amount of contaminated material that must be removed during decommissioning.

The capability to flush process equipment, piping, tanks, and hoods should be provided to facilitate routine clean-up, maintenance, and eventual decommissioning. Covered and/or sealed ceilings, walls, floors, and auxiliary equipment should be spray rinsable. Spray rinsing equipment should be installed to take advantage of this feature, and appropriate drains or sumps, especially in dry processing areas, should be provided. These provisions would minimize time-consuming manual cleaning techniques.

Provisions for remote maintenance of equipment may be desirable in some instances from the standpoint of reduction of personnel exposure and effort required to perform some maintenance tasks due to the reduced amount of pre-maintenance cleanup required. If such provisions are included, the role of the use of this equipment for decommissioning should be incorporated into their design.

13.2 OPERATION

Routine maintenance and decontamination activities may lead to the development of special tools or techniques that will reduce the effort required to complete a specific task and therefore reduce worker exposure. These tools or techniques may later prove to be useful during decommissioning. Information regarding their design and use should be recorded in detail.

Maintenance procedures providing for routine decontamination of plant areas and equipment subject to holdup and/or spills will help minimize the amount of material that must be removed during decommissioning. Frequent recovery or disposal of materials stored in settling ponds and effluent lagoons would reduce the amount of waste to be disposed of during decommissioning.

The total volume of radioactive waste due to routine operation and decommissioning can be reduced by practicing various radwaste reduction techniques. Combustible solids and some liquids may be reduced in volume by incineration, with a typical volume reduction of at least 5 to 1. Process equipment, piping, tools, auxiliary equipment, and hoods may be partially or completely decontaminated by vibratory finishing and electropolishing. These techniques are

effective in removing surface deposits and corrosion-containing radioactive contamination. Where only corrosion and surface contamination exist, vibratory finishing and electropolishing may remove enough of these materials so the part may be returned to active service, or disposed of with other noncontaminated waste. The amount of contaminated material remaining to be disposed of is thus reduced to the corrosion and surface materials the processes removed.

Computerized data systems can be used to store information, and make it more readily available. Information concerning routine operation and maintenance procedures, unusual techniques, and experiences from other facilities would be useful in developing better procedures during operation and decommissioning. Scale models should be updated to show areas of high radiation dose and system modification, information which would be useful during operation, maintenance, and decommissioning to reduce workers exposure and effort.

13.3 DECOMMISSIONING

Features designed into the facility and equipment, techniques, and procedures developed during the operation of the facility can be exploited to reduce worker exposure and effort during decommissioning.

Some special decommissioning tools have been developed for removal of concrete and steel. Methods for cutting and/or demolishing concrete include thermic lancing, explosive cutting, and heating followed by wrecking. Methods for cutting or sectioning steel components include the plasma arc torch, arc saw, oxyacetylene torch, explosive cutting, and thermic lancing.

Information stored in computerized data systems can facilitate decommissioning, particularly in the case of deferred decontamination. In that case, operating personnel will probably not be available for consultation, and accurate information concerning maintenance procedures and experiences recorded during the decommissioning of similar facilities will be helpful.

Reduction of radioactive waste volume during decommissioning should be a primary goal. Methods for achieving this goal are available for different kinds of waste. In many cases, combustible solid and some liquid wastes can be

incinerated for a significant volume reduction. For surface contaminated process equipment, vibratory finishing and electropolishing can often be used to remove corrosion and surface deposits to the extent that the equipment can be released as uncontaminated. In such cases, only the contaminated surface material removed will require disposal at a licensed burial site. Concrete that has surface contamination can be surface-spalled, to a depth of 50 to 90 mm, so that only the spalled material must be disposed of at a licensed burial site. Concrete that has been sealed or covered may not require spalling or licensed burial.

Recovery of stored materials, such as the reprocessing of uranium-bearing muds of settling ponds and storage lagoons, can reduce the amount of stored wastes that must be disposed of at licensed burial sites, and at the same time result in the recovery of valuable materials.

14.0 GLOSSARY

Abbreviations, acronyms, symbols, terms, and definitions used in this study and directly related to decommissioning work and related technology are defined and explained in this section. The section is divided into two parts, with the first part containing abbreviations, acronyms, symbols, and a conversion table to International System of Units (SI), and the second part containing terms and definitions (including those used in a special sense for this study). Common terms covered adequately in standard dictionaries are not included.

14.1 ABBREVIATIONS, ACRONYMS, SYMBOLS, AND SI UNITS

Abbreviations and Acronyms

AEC	Atomic Energy Commission
ALARA	As Low As is Reasonably Achievable ^(a)
CFR	Code of Federal Regulations ^(a)
Ci	Curie ^(a)
DF	Decontamination Factor ^(a)
DOT	Department of Transportation
DPM	Disintegrations per Minute ^(a)
EDTA	Ethylenediamine tetraacetic acid
FSAR	Final Safety Analysis Report
GB	Glove Box
HEPA	High Efficiency Particulate Air (Filters)
HP	Health Physicist ^(a)
HVAC	Heating, Ventilation and Air Conditioning
LWR	Light Water Reactor
mR	Milliroentgen ^(a)
mrad	Millirad ^(a)
mrem	Millirem, see rem also
MT	Metric Ton ^(a)

(a) See Section 14.2 for additional information or explanation.

MTHM	Metric Ton of Heavy Metal
MWd/MTU	Thermal Megawatt-day per Metric Ton of Uranium, the Burnup ^(a)
NRC	Nuclear Regulatory Commission
Q.A.	Quality Assurance ^(a)
Q.C.	Quality Control ^(a)
R	Roentgen ^(a)
rad	Radiation Absorbed Dose ^(a)
rem	Roentgen Equivalent Man ^(a)
SNM	Special Nuclear Material ^(a)
SS	Stainless Steel
SX	Solvent Extraction
$T_{1/2}, T_R$	Half Life, Radiological ^(a)
UF	Urea-formaldehyde
UF ₆	Uranium Hexafluoride Conversion

Symbols

α	Alpha Radiation ^(a)
β	Beta Radiation ^(a)
γ	Gamma Radiation ^(a)
χ	Chi, Concentration, pCi/m ³
Q	Released Quantity of Radioactive Material, Ci
Q'	Release Rate of Radioactive Material, Ci/sec
$\bar{\chi}/Q'$	Chi-bar/Q prime, normalized annual average air concentration (pCi/m ³) per Ci/sec released, also written sec/m ³). Also called the annual average atmospheric dilution factor.

SI Units

SI units for use with radioactivity and ionizing radiations are as follows:

Quantity	New Named Unit and Symbol	In Other SI Units	Old Special Unit and Symbol	Relationship New to Old Units
Exposure	--	coulomb/kg (C/kg)	röntgen (R)	1 C/kg $\approx$ 3876 R
Absorbed dose	gray (Gy)	joule/kg (J/kg)	rad (rad)	1 Gy = 100 rad
Dose equivalent	sievert (Sv)	J/kg	rem (rem)	1 Sv = 100 rem
Activity	becquerel (Bq)	seconds ⁻¹ (s ⁻¹)	curie (Ci)	1 Bq $\approx$ 2.70 x 10 ⁻¹¹ Ci

(a) See Section 14.2 for additional information or explanation.

14.2 GLOSSARY DEFINITIONS

Actinides:	A series of heavy radioactive metallic elements of increasing atomic number (Z) beginning with actinium (89) or thorium (90) through element hahnium of atomic number 105.
Activity:	See Radioactivity.
Airborne Radioactive Material:	Radioactive particulates, mists, fumes, and/or gases in air.
ALARA:	A philosophy to maintain exposure to radiation <u>As Low As is Reasonably Achievable</u> .
Alpha Decay:	Radioactive decay in which an alpha particle is emitted. This transformation lowers the atomic number of the nucleus by two and its mass number by four.
Alpha Particle:	A positively charged particle emitted by certain radioactive materials. It is made up of two neutrons and two protons, hence it is identical with the nucleus of a helium atom. It is the least penetrating of the three common types of radiation (alpha, beta and gamma) emitted by radioactive material.
Alpha Emitter:	A radionuclide that undergoes transformation by emission of alpha particles.
Atomic Number (Z):	The number of protons in the nucleus of an atom; also its positive charge. Each chemical element has its characteristic atomic number, and the atomic numbers of the known elements form a complete series from 1 (hydrogen) through 105 (hahnium).
Background:	That level of radioactivity from sources other than the one directly under consideration, in this case those existing without the presence of the U-Fab plant.
Bag Out:	Term used to describe the techniques for transferring objects into and/or out of glove boxes without loss of confinement, utilizing various types of containers, sealing and packaging techniques.
Beta Decay:	Radioactive decay in which a beta particle is emitted or in which an orbital electron capture occurs.

Beta Particle:	An electron, of either positive or negative charge, which has been emitted by an atomic nucleus in a nuclear transformation.
Burial Grounds:	Areas designated for storage of packaged radioactive wastes in soils just below the surface.
Burnup, Specific:	The total energy released per unit mass of a nuclear fuel. It is commonly expressed in megawatt-days per metric ton of fuel material. (Also called fuel irradiation level.)
Byproduct Material:	Any radioactive material (except special nuclear material) yielded in or made radioactive by exposure to the radiation incident to the process of producing or utilizing special nuclear material.
Calcine:	To heat a substance to a high temperature, but below its melting point, causing loss of volatile constituents such as moisture. Material produced by this process is also called Calcine.
Cask:	A heavily shielded shipping container for radioactive materials. Some casks weigh as much as 100 metric tons.
Chemical Limits:	Maximum concentrations or quantities imposed upon chemical releases to the environment in gaseous or liquid effluents discharged from a facility, and consistent with known air or water quality standards.
Code of Federal Regulations (CFR):	The Code of Federal Regulations is a documentation of the general rules by the Executive departments and agencies of the Federal Government. The Code is divided into 50 titles that represent broad areas subject to Federal regulation. Each title is divided into Chapters that usually bear the name of the issuing agency. Each Chapter is further subdivided into Parts covering specific regulatory areas.
Contact Maintenance:	"Hands-on", or maintenance performed by direct contact of personnel with the equipment. It includes maintenance with protective equipment or clothing, such as through gloves in glove boxes. Most nonradioactive maintenance is contact maintenance.

Contamination:	Undesired materials that have been deposited on the surfaces, or are internally ingrained into structures or equipment, or that have been mixed with another material.
Critical:	<p>A condition wherein a medium is capable of sustaining a nuclear chain reaction at a constant rate.</p> <p>Prompt critical is being capable of sustaining a chain reaction without the aid of delayed neutrons</p>
Critical Mass:	The mass of fissionable material that will support a self-sustaining nuclear chain reaction.
Curie:	<p>A special unit of radioactivity. One curie equals 3.7×10^{10} nuclear transformations per second. (Abbreviated Ci.) Several fractions of the curie are in common usage:</p> <ul style="list-style-type: none"> • Millicurie. One-thousandth of a curie. Abbreviated mCi (3.7×10^7 d/s). • Microcurie. One-millionth of a curie. Abbreviated μCi (3.7×10^4 d/s). • Nanocurie. One-billionth of a curie. Abbreviated nCi (37 d/s). • Picocurie. One-millionth of a microcurie. Abbreviated pCi; replaces the term μCi (0.037 d/s).
Custodial SAFSTOR:	A minimum cleanup and decontamination preparation followed by safe storage and terminated by deferred decontamination. The active protection systems (i.e., ventilation, utilities, fire) are kept in service, the site is secured against intrusion by physical barriers and by guards, and use of the facility and site is limited to nuclear activities.
Decay, Radioactive:	A spontaneous nuclear transformation in which a particle, gamma radiation, or x-ray radiation are emitted.
Decommissioning:	The retirement from active service of nuclear facilities, including all activities to remove the radioactive material to levels that allow unrestricted release of the facility and its site.

DECON:	Those actions required immediately after shutdown to remove sufficient radioactive or contaminated materials from the facility and site, to permit release of the property for unrestricted use.
Decontamination:	Those activities employed to reduce the levels of contamination in or on structures, equipment and materials. Also used to infer decontamination to levels corresponding to unrestricted release.
Decontamination Agents:	Those chemical materials used to effect decontamination.
Decontamination Factor (DF):	The ratio of the initial concentration of an undesired material to the final concentration resulting from a treatment process. The term may also be used as a ratio of quantities.
Deferred Decontamination:	Those actions required after the safe storage period of SAFSTOR to disassemble and remove sufficient radioactive or contaminated materials from the facility and site, to permit release of the property for unrestricted use.
Design Basis Accident:	A postulated accident believed to have the most severe expected impacts on a facility. It is used as the basis for safety and structural design.
Discount Rate:	The rate of return on capital that could have been realized in alternative investments, if the money were not committed to the plan being evaluated, i.e., the opportunity costs of alternative investments. This cost is equivalent to the weighted average cost of capital.
Disintegration, Nuclear:	The transformation of the nucleus of an atom from one element to another, characterized by a definite half-life and the emission of particles or radiation.
Disintegration Rate:	The rate at which disintegrations occur, characterized in units of inverse time; i.e., disintegrations per minute (dpm), etc.
Dismantlement:	Those actions required to disassemble and/or remove radioactive or contaminated materials from the facility and site.

Dispersion:	A process of mixing one material within a larger quantity of another. For example, the mixing of material released to the atmosphere with air causes a reduction in concentration with distance from the source.
Disposal:	The disposition of materials with the intent that the materials will not enter man's environment in sufficient amounts to cause a health hazard.
Dose, Absorbed:	The mean energy imparted to matter by ionizing radiation per unit mass of irradiated material at the place of interest. The unit of absorbed dose is the rad. One rad equals 0.01 joules/kilogram in any medium (100 ergs per gram).
Dose, Equivalent:	Expresses the amount of radiation that is effective in the human body, expressed in rems. Modifying factors associated with human tissue and body are considered. Equivalent dose is the product of absorbed dose multiplied by a quality factor multiplied by a distribution factor. Referred to as Dose in this report.
Dose, Occupational:	The exposure of an individual to radiation as a result of his employment, expressed in rems.
Dose Rate:	The radiation dose delivered per unit time and measured, for instance, in rems per hour.
Dosimeter:	A device, such as a film badge or ionization chamber, that measures radiation dose.
Enrichment:	The ratio (usually expressed as a percentage) of fissile isotope to the total amount of the element (e.g., the % of ^{235}U in uranium.)
ENTOMB:	The encasement of radioactive materials in concrete or other structural materials sufficiently strong and durable to assure retention of the radioactivity until it has decayed to levels that permit unconditional release of the site.
Exposure:	A measure of the ionization produced in air by x or gamma radiation. It is the sum of the electrical charges on all ions of one sign produced in air when all electrons liberated by photons in a volume element of air are completely

stopped in air, divided by the mass of air in the volume element. The special unit of exposure is the roentgen. (See Roentgen.)

Facility:	The physical complex of buildings and equipment within a site.
Fission:	The splitting of a heavy atomic nucleus into two lighter parts (atomic nuclides of lighter elements), accompanied by the release of a relatively large amount of energy and generally one or more neutrons. Fission can occur spontaneously but usually it is caused by nuclear absorption of gamma rays, neutrons, or other particles.
Fission Products:	The lighter atomic nuclides (fission fragments) formed by the fission of heavy atoms. It also refers to the nuclides formed by the fission fragments' radioactive decay.
Fluoride Waste:	Liquid waste generated primarily by the hydrofluorination system off-gas scrubber. This waste collected in the fluoride waste lagoons.
Food Chain:	The pathways by which any material (such as radioactive material from fallout) passes through man's environment through edible plants and/or animals to man.
Fuel Assembly:	A grouping of fuel elements (hollow rods filled with nuclear fuel for LWRs) that supply the nuclear heat in a nuclear reactor. A fuel element or rod is the smallest structurally discrete part of a reactor or fuel assembly that has nuclear fuel as its principal constituent.
Fuel Cycle:	<p>The series of steps involved in supplying fuel for nuclear power reactors, handling the spent fuel and the radioactive waste, including transportation.</p> <p>Head end: Mining, milling, conversion, enrichment, and fabrication of fuel.</p> <p>Back end: Includes reactors, spent fuel storage, spent fuel reprocessing, mixed-oxide fuel fabrication and waste management.</p>
Fuel Element:	A rod, tube, or other form into which nuclear fuel is fabricated to use in a reactor.

Gamma Rays:	Short-wavelength electromagnetic radiation. Gamma radiation frequently accompanies alpha and beta emissions and always accompanies fission. Gamma rays are best stopped or shielded against by dense materials such as lead or uranium. These rays usually originate from within the nucleus of the atom.
Gaseous:	Material in the vapor or gaseous state, but can include entrained liquids and solids. A gas will completely fill its container regardless of container shape or size.
Glove Box:	A box, usually made of stainless steel and large panes of glass or transparent rigid plastic, in which workers using gloves attached to, sealed, and passing through, openings in the box can safely handle radioactive materials from the outside by inserting their hands into the gloves and manually performing manipulations.
Greenhouse:	In nuclear terms, a temporary structure, frequently constructed of wood and plastic film, used to provide a confinement barrier between a radioactive work area and a nonradioactive area.
Guard:	An individual whose primary duty is the guarding and protection of material against theft and/or the protection of the facility against vandalism or undesired intruders.
Half-Life, Biological:	The time required for a biological system, such as a man or animal, to eliminate by natural processes, half the amount of a substance that has been absorbed by it.
Half-Life, Effective:	The time required for a radionuclide contained in a biological system, such as a man or animal, to reduce its radioactivity by half as a combined result of radioactive decay and biological elimination.
Half-Life Radioactive:	The time in which half the atoms of a particular radioactive substance disintegrate to another nuclear form. Each radionuclide has a unique half-life. Measured half-lives vary from millionths of a second to billions of years.
Health Physicist:	A person trained to perform radiation surveys, oversee radiation monitoring, estimate the degree of radiation hazard, and advise on operating procedures for minimizing radiation exposures.

Health Physics:	The science concerned with recognition, evaluation, and control of health hazards from radiation.
Heavy Metal:	Jargon used in reference to metals with atomic numbers 90 and greater. It usually refers to nuclear fissile or fertile fuels such as thorium, uranium, and plutonium.
Hood:	Vented containment space, enclosed on 5 sides, with the sixth side covered by a movable glass window to allow access and to maintain sufficient in-flow of air and splash control to protect the worker from the hazardous materials handled inside.
Hot Spots:	Areas of radioactive contamination higher than average.
Immobilization:	Treatment and/or emplacement of materials (e.g., radioactive contamination) so as to impede its movement. Concepts for interim storage include bulk or compartmented storage of solid, liquid and gaseous wastes or other materials.
Intrusion Alarm:	A means of detecting intrusion of individuals into a protected area utilizing an electro-mechanical, electro-optical, electronic, mechanical or similar device with a visible or audible alarm signal.
Ion Exchange:	A chemical process involving the selective absorption or desorption of various chemical ions in a solution onto a solid material, usually a plastic or resin. The process is used to separate and purify chemicals, such as fission products from plutonium or "hardness" from water (i.e., water softening).
Lagoon Waste:	Waste in the solvent extraction waste lagoons or the fluoride waste lagoons.
Layaway:	See Custodial SAFSTOR.
Licensed Material:	Nuclear source material, special nuclear material, or nuclear by-product material received, possessed, used, or transferred under a license issued by the Nuclear Regulatory Commission.

Long-Lived Nuclides:	For this study, radioactive isotopes with long half-lives typically taken to be greater than about ten years. Most nuclides of interest to waste management have half-lives on the order of one year to millions of years.
Management (Waste):	The planning, execution, and surveillance of essential functions related to radioactive waste, including treatment, solidification, packaging, interim or long-term storage, transportation and disposal.
Man-rem:	A measure of radiation dose distributed to a population. To calculate radiation dose to the population, the dose equivalent in rem received by each person in the population is summed.
Mass Number:	The number of nucleons (protons and neutrons) in the nucleus of an atom. (Symbol: A).
Maximum Exposed Individual:	The hypothetical member of the public who receives the maximum radiation dose to an organ of reference. For the common case where exposures from airborne radionuclides result in the highest radiation exposure, this individual resides at the location of the highest airborne radionuclide concentration and eats food grown at that location.
Megawatt-day:	A unit for expressing the energy generated in a reactor; specifically, the number of millions of watt-days of heat output per metric ton of fuel in the reactor. Also, the net electrical output in millions of watts of electrical energy averaged over one day.
Megawatts per Metric Ton of Uranium:	Amount of thermal megawatts produced per metric ton of uranium.
Megawatt Days per Metric Ton of Uranium:	Amount of thermal megawatt-days produced per metric ton of uranium; also called burnup. (See also specific power.)
Metric Ton:	1000 kilograms (See Tonne.)
Mixed Oxide:	A mixture of uranium dioxide and plutonium dioxide.
Monitoring:	Taking measurements or observations for recognizing the status or adequacy, or significant changes in conditions or performance of a facility or area.

Mothball:	See Passive SAFSTOR
Normal Operating Conditions:	Operation (including startup, shutdown, and maintenance) of systems within the normal range of applicable parameters of an operating facility.
Nuclear Reaction:	A reaction involving a change in an atomic nucleus, such as fission, fusion, particle capture, or radioactive decay.
Offsite:	Beyond the boundary line marking the limits of plant property.
Onsite:	Within the boundary line marking the limits of plant property.
Operable:	Capable of performing the required function.
Overpack:	Secondary (or additional) external containment or cushioning for packaged materials.
Package:	The packaging plus the contents of radioactive materials.
Packaging:	The assembly of radioactive material in one or more containers and other components necessary to assure compliance with prescribed regulations.
Passive SAFSTOR:	A partial cleanup and decontamination preparation followed by safe storage and terminated by decontamination. All systems are deactivated, the structures are secured by rigid physical barriers and continuous remote monitoring, and the plant is limited to nuclear use only, while the site may have non-nuclear uses.
Plant:	The physical complex of buildings and equipment, including the site.
Present Value of Money:	The present value of a future stream of costs or payments is the present investment necessary to secure or yield the future stream of payments with compound interest at a given discount or interest rate.
Primary Wastes:	Wastes that are generated as a part of the principal operation of a facility. Secondary wastes are generated from supporting operations, such as waste treatment.

Process Cells:	Shielded rooms housing (radioactive) processing systems.
Process Equipment:	The functional equipment items or systems associated directly with the operation of a chemical or mechanical operation.
Protective Clothing:	Special clothing worn by a person in a radioactively contaminated area to minimize the potential for contamination of his body or personal clothing.
Protective Storage:	See Passive SAFSTOR.
Protective Survey:	An evaluation of the radiation and its hazards incidental to the production, use or existence of radioactive materials. It normally includes a physical survey of the arrangement and use of equipment and measurements of the radiation dose rates under expected conditions of use. Also called protection survey.
Quality Assurance:	The systematic actions necessary to provide adequate confidence that a material, component, system, process, or facility performs satisfactorily, or as planned, in service.
Quality Control:	The quality assurance actions that control the attributes of the material, process, component, system, or facility in accordance with predetermined quality requirements.
Rad:	A unit of absorbed dose. The energy imparted to matter by ionizing radiation per unit mass of irradiated material at the place of interest. One rad equals 0.01 joule/kilogram of absorbing material.
Radiation:	(1) The emission and propagation of radiant energy: for instance, the emission and propagation of electromagnetic waves, or of sound and elastic waves. (2) The energy propagated through space or through a material medium. for example, energy in the form of alpha, beta, and gamma emissions from radioactive nuclei.

Radiation Area:	Any area, accessible to personnel, in which there exists radiation at such levels that a major portion of the body could receive in any one hour a dose in excess of 5 millirem, or in any 5 consecutive days a dose in excess of 100 millirems. (10 CFR 20.202)
Radiation Background:	See background.
Radiation, Leakage (Direct):	All radiation coming from a source housing except the useful beam.
Radioactive Material:	Any material or combination of materials which spontaneously emit ionizing radiation and which has a specific radioactivity in excess of 0.002 microcuries per gram of material. (49 CFR 173.389 (e)).
Radioactive Series:	A succession of nuclides, each of which transforms by radioactive disintegration into the next until a stable nonradioactive nuclide results. The first member is called the "parent," the intermediate members are called "daughters," and the final stable member is called the "end product."
Radioactivity:	The number of nuclear transformations occurring in a given quantity of material per unit of time with the emission of particles, gamma radiation, or x-ray radiation. Often shortened to "activity."
Radioactivity, Natural:	The property of radioactivity exhibited by more than fifty naturally occurring radionuclides.
Radiological Protection:	Protection against the effects of internal and external exposure to radiation and to radioactive materials.
Regulatory Guides:	Regulatory Guides are issued by the NRC, to describe and make available to the public, methods acceptable to the NRC staff, for implementing specific parts of the NRC's regulations, to delineate techniques used by the staff in evaluating specific problems or postulated accidents, or to provide other guidance to applicants for nuclear operations. Guides are not substitutes for regulations and compliance with them is not explicitly required. Methods and solutions different from those set out in the guides may be acceptable if they provide

	a basis for the findings requisite to the issuance or continuance of a permit or license by the NRC.
Rem:	A unit of radiation dose equivalence. The radiation dose equivalence in rems is numerically equal to the absorbed dose in rads multiplied by the quality factor, the distribution factor, and any other necessary modifying factors.
Remote Maintenance:	Maintenance by remote means, i.e., the operator is separated by a shielding wall from the item being maintained.
Reporting Levels:	Those levels or parameters called out in the Environmental Technical Specifications, the Decommissioning Order, and/or the Amended License that do not limit decommissioning activities, but which may indicate a measurable impact on the environment.
Repository (Federal):	A site owned and operated by the Federal Government for long-term storage or disposal of radioactive materials.
Restricted Area:	Any area to which access is controlled for protection of individuals from exposure to radiation and radioactive materials.
Roentgen:	A unit of exposure to ionizing radiation. It is that amount of gamma or x-rays required to produce ions carrying one electrostatic unit of electrical charge (either positive or negative) in one cubic centimeter of dry air under standard conditions. One roentgen equals 2.58×10^{-4} coulombs per kilogram of air. (See also Exposure.)
SAFSTOR:	The decommissioning alternative wherein a preparatory period after shutdown is followed by safe storage, which, in turn, is terminated by deferred decontamination.
Safe Storage:	A period of time starting after the initial decommissioning activities cease and wherein surveillance and maintenance takes place. The duration of time can vary from a few years to more than 100 years; called "continuing care" in some other NRC decommissioning reports.

Safety-Related:	Structures, systems, and components whose functions tend to prevent or mitigate the exceeding of safety limits, as defined in Regulatory Guide 3.6, and set forth in Technical Specifications that are part of the Operating License for a nuclear power plant.
Scarfiging:	A technique used to mechanically decontaminate concrete by chipping, cutting, jackhammering, or blasting the surface layer(s) away.
Secondary Wastes:	Forms and quantities of all wastes that result from treatment of primary wastes or effluents.
Security Officer:	A guard or watchman whose primary duty is the protection of material and property.
Shield:	<p>A body of material used to reduce the passage of particles or electromagnetic radiation. A shield may be designated according to what it is intended to absorb (as a gamma ray shield or neutron shield), or according to the kind of protection it is intended to give (as a background, or thermal shield).</p> <p>It may be required for the safety of personnel or to reduce radiation enough to allow use of counting instruments for research or for locating contamination or airborne radioactivity.</p>
Short-Lived Radionuclides:	For this study, those radioactive isotopes with half-lives less than about 10 years.
Shutdown:	The time during which a facility is not in productive operation.
Site:	The geographic area upon which the facility is located that is subject to controlled public access by the facility licensee (includes the restricted area as designated in the NRC license).
Solid Radioactive Waste:	Material that is essentially solid and dry but may contain sorbed radioactive fluids in sufficiently small amounts as to be immobile.
Solidification:	Conversion of radioactive wastes (gases or liquids) to dry, stable solids.

Solvent Extraction Waste:	Liquid Waste from the solvent extraction system. Primarily a solution of ammonium nitrate, nitric acid, heavy metal salts and uranium and daughter products of normal uranium decay. This waste collected in the solvent extraction waste lagoons.
Special Nuclear Material:	Plutonium, uranium enriched in the isotopes 233 or 235, and any other material as defined in 10 CFR 51 by the NRC.
Specific Power (of Fuel Assemblies):	Commonly expressed in units of thermal megawatts per metric ton of uranium (MW/MTU). It represents the rate at which thermal energy is extracted from the fuel; burnup, commonly expressed in thermal megawatt-days per metric ton of uranium (MWd/MTU), represents the total integrated energy extracted. For MOX fuel, the unit of fuel is a metric ton of heavy metal (MTHM); i.e., a metric ton of (U + Pu).
Surface Contamination:	Contamination that is the result of the deposition and attachment of foreign materials to a surface.
Surveillance:	Those activities necessary to assure that the site remains in a safe condition (including inspection and monitoring of the site, maintenance of barriers to access to radioactive materials left on the site, and prevention of activities on the site that might impair these barriers).
Survey:	An evaluation of the radiation hazards incident to the production, use, release, disposal or presence of radioactive materials or other sources of radiation under a specific set of conditions.
Technical Specifications:	Requirements and limits that encompass nuclear safety but are simplified to facilitate use by plant operation and maintenance personnel. They are prepared in accordance with the requirement of 10 CFR 50.36, and are incorporated by reference into the amended license issued by the NRC.
Tonne:	A metric ton, or 1000 kg, or 2204.6 lb.
Transuranic Elements:	Elements with atomic number (Z number) greater than 92.

Transuranic Waste:	Any waste material measured or assumed to contain more than a specified concentration (i.e., proposed as 10 nanocuries of alpha emitters per gram of waste, or more presently proposed as 100 nanocuries/cm ³ of waste ²³⁹ U) of transuranic elements.
Underground Solid Waste Storage Area:	Area within an exclusion area where radioactive solid waste is stored by burial.
Wastes, Radioactive:	Equipment and materials (from nuclear operations) that are radioactive and for which there is no further known use.
Wastes, Low-Level:	Wastes containing types and concentrations of radioactivity such that little or no shielding to minimize personnel exposure is required.
Wastes, High-Level:	Wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, in a facility for reprocessing irradiated reactor fuels (10 CFR 50, App. F.2). It is also applied generally to radioactive wastes of other origins, where the rate of heat evolution becomes of concern in waste disposal or the external radiation dose rates are extremely high.
X-ray:	A penetrating form of electromagnetic radiation emitted either when the inner orbital electrons of an excited atom return to their normal state (characteristic x-rays) or when a metal target is bombarded with high speed electrons. X-rays are always non-nuclear in origin; i.e., they originate external to the nucleus of the atom.

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APPENDICES

APPENDIX A

REFERENCE URANIUM HEXAFLUORIDE CONVERSION FACILITY DESCRIPTION

Major features of the reference uranium hexafluoride (UF_6) conversion plant are presented in Section 7. This appendix describes in more detail the processes used for UF_6 conversion and the portions of the facility and site that are important to decommissioning. The detailed reference site description is presented in Appendix B.

The reference plant used in this study is representative of contemporary UF_6 conversion plants in the United States. Existing plants are expected to require decommissioning in the future.

Much of the detailed plant design information in this appendix has been extracted from the Special Nuclear Materials License Application for the Kerr-McGee Hexafluoride facility in Sequoyah County, Oklahoma,⁽¹⁾ from the Final Environmental Impact Statement for the Kerr-McGee Sequoyah Hexafluoride Facility,⁽²⁾ and from the Environmental Impact Appraisal of the Allied Chemical Plant at Metropolis, Illinois.⁽³⁾

This appendix is divided into two main sections. Section A.1 presents details of the process description. Section A.2 presents details of the plant structures and the equipment in the main process, waste treatment, and supporting facilities.

A.1 PROCESS DESCRIPTION

The production method used at the reference plant involves: a) feed preparation, b) dissolution of the ore concentrate in nitric acid, c) purification of the uranium solution by solvent extraction, d) denitration of the uranyl nitrate to prepare uranium trioxide, e) hydrogen reduction of the uranium trioxide to uranium dioxide, f) conversion of the uranium dioxide to uranium tetrafluoride by reaction with anhydrous hydrogen fluoride, and g) formation of uranium hexafluoride by contacting the uranium tetrafluoride with elemental fluorine.

Pulse columns and pumper decanters are used for solvent extraction; fluid beds or mechanically agitated beds for denitration, reduction, and hydrofluorination; and flame reactors for fluorination.

Figures A.1-1 and A.1-2 respectively show a block diagram and schematic flowsheet of the reference plant production process.

FIGURE A.1-1. UF_6 Production - Wet SX - Fluorination
Simplified Block Flow Diagram

FIGURE A.1-2. Schematic Flow Sheet of the Reference Plant

After sampling, ore concentrate is dissolved in nitric acid and the uranium extracted with TBP-hexane in a series of pumper decanters. The extract is passed through pulse columns to reextract the uranium into water. This solution is evaporated and then heated in a continuous trough denitrator to form UO_3 . The raffinate from the extraction step is pumped to holdup tanks for sampling and then to holding ponds. Before recycling, the solvent is treated with ammonium sulfate-caustic to remove any traces of uranium and TBP degradation products.

After grinding, the UO_3 is converted to UF_6 in three successive dry processing steps. The powder is fed first to a two-stage fluid-bed where it is contacted with H_2 to form UO_2 . This product is then charged to series-connected stirred fluid-bed reactors in which UF_4 is formed by reaction with anhydrous HF. Any excess HF along with the water formed by the reaction is condensed to produce an aqueous acid which is neutralized with lime. The UF_4 is then converted to UF_6 by reaction with F_2 in a flame-type reactor. The gaseous UF_6 formed is collected by desublimation in refrigerated heat-exchangers or cold traps. These units are operated on a batch-cycle and are periodically taken off-line and heated to liquify and drain the UF_6 into product cylinders. For economy, fluorine vent gases are passed through a secondary or cleanup reactor to assure complete usage of gas.

Elemental fluorine required for the final processing step is generated by the electrolysis of a molten KF-HF electrolyte. Appropriate gas feed systems, dust collection devices, and disposal systems are provided. The process also includes grinding of fluorination ash and recycling it to the fluorination system.

A.1.1 Production Systems Descriptions

Descriptions of the production systems involved are presented in the following subsections.

A.1.1.1 Sampling the Dry Ore Concentrate

Dry concentrate (yellowcake) is received from uranium mills in 0.21-m^3 steel drums which are weighed at the mill and the sampling plant. The drums

of ore concentrate are weighed and sampled by a dry splitter system. Powder discharged from the falling-stream sampler is conveyed to the digester feed hopper.

The drums are emptied one at a time through a falling-stream sampling unit. This unit consists of two samplers in series, each taking a small cut. This produces an initial sample weight of about 1% or less of the total lot. The sample is split down again by a factor of about 50 and is collected in trays. The material collected in the trays is processed to a final sample pulp by the operations of drying, pulverizing, riffing, sieving, and blending.

The sampled ore concentrate is usually fed directly into the digester feed hopper. The sampled concentrate can also be redrummed if desired.

A.1.1.2 Ore Concentrate to Purified $\text{UO}_2(\text{NO}_3)_2$

The process of purifying ore concentrate sufficiently for production of pure $\text{UO}_2(\text{NO}_3)_2$ involves nitric acid digestion and solvent extraction.

A.1.1.3 Digestion of Dry Yellowcake

Yellowcake and 40% nitric acid are fed to three 15-m^3 digestors which operate on a batch basis. After digestion, the digester tank contents are transferred to two 15-m^3 adjustment tanks where adjustment chemicals are added. Overall residence time varies from 12 to 24 hours.

A.1.1.4 Unloading, Sampling, and Digestion of Wet Yellowcake

Wet yellowcake slurry delivered from a uranium mill contains approximately 38 wt% of water. The slurry is transported in a stainless steel cargo tank. A special yellowcake slurry receiving area is provided. This area contains facilities and equipment for dissolving the ADU slurry with nitric acid, unloading the cargo tank, and sampling the uranyl nitrate solution (see Figure A.1-5).

Tanks for receiving, weighing, and sampling the uranium product are enclosed in a building which rests on a concrete curbed foundation with sufficient volume to contain spillage in case of accidental release. There are three tanks provided, one 3.8-m^3 tank and two 37.8-m^3 tanks. The 3.8-m^3 tank is used for

FIGURE A.1-3. ADU Slurry Receiving and Storage Flow Sheet

non-routine, small operations and for calibration of the load cells on the 37.8-m³ receiving and weigh tank. The second 37.8-m³ tank is used for measured storage. The receiving and transfer pumps are contained in a small building with a curbed foundation that drains to the large curb in case of spillage in the pump house. The area where the cargo tank is parked is also curbed adequately to contain the entire tank volume.

The cargo tank will arrive at the facility ~60% full, with blind flanges on all openings except the pressure relief valve (tank rated at 207 kPa). The cargo tank will be "spotted" and the blinds removed. Three equally spaced pump-discharge-bayonets are inserted to within 76 mm of the bottom of the tank, and two pump suction lines are attached to two diptubes, one at the bottom and one one-third tank diameter from the top. The vent line and a fail-safe level switch are attached.

A predetermined amount of either 60 or 40% HNO₃ is pumped into the receiving tank. The receiving tank, cargo tank, and receiving pump are valved to allow pumping acid into the cargo tank. The cargo tank vent valve is opened. The acid is pumped into the cargo tank until the level switch indicates proper level. (High level shuts down the upper outlet in the cargo tank.) The pump is restarted, providing circulation of acid in the cargo tank. When acid has been consumed, the contents of the cargo tank are pumped back to the receiving and weighing tank. The dissolving operation is repeated until the slurry is totally dissolved. The contents of the cargo tank are pumped to the receiving tank using the lower outlet. The cargo tank may be acid rinsed before being disconnected and prepared for return to the shipper.

The uranyl nitrate in the receiving tank is mixed, weighed, sampled, and transferred to the 37.8-m³ storage tank. The storage tank contents are pumped to the process tankage used to prepare the feed for solvent extraction purification.

A.1.1.5 Unloading, Sampling, and Digestion of Wet UF₄

The UF₄ slurry as received is approximately 50% water. It is contained in heavy polyethylene drums, each containing approximately 0.2-m³ of slurry.

The UF_4 slurry receiving building rests on a curbed foundation with sufficient volume to contain the contents of the digestion tank in case of an accidental release. The UF_4 slurry processing equipment is shown in Figure A.1-4.

The full drums are placed on a roller conveyor using a fork truck. The drum lids are removed and the drums are emptied by inversion over the process feed hopper using a mechanical drum dumper. The slurry is pumped into the digester containing a pre-mixed quantity of HNO_3 - $\text{Al}(\text{NO}_3)_3$ solvent. After digestion, the resulting solution is mixed, weighed, and sampled. It may then be pumped to the storage tank or to process tankage for preparation as feed for the solvent extraction operation.

A.1.1.6 Purification

Recovery of the uranium from the digester product is accomplished by extraction with 30% tributyl phosphate (TBP) in hexane, using a series of 1.2-m-diameter by 1.8-m-high pumper-decanters. The loaded solvent is scrubbed in mixer settlers to remove residual impurities and entrained aqueous solution. The scrubber liquid and impurities are returned to the feed stream where they become part of the waste product after further processing. The uranyl nitrate is reextracted into water using pulse columns as contactors. The purified solution is subsequently boiled down and denitrated to UO_3 .

After uranium has been removed from the solvent, it is treated with ammonium sulfate-caustic to remove the residual uranium and TPB degradation products. The waste from the solvent extraction step (pumper decanters) is stored in onsite ponds.

A.1.1.7 Preparation of Uranium Trioxide

The aqueous uranyl nitrate prepared in the solvent extraction step is converted to dry uranium trioxide by a two-step procedure. The solution is first concentrated by evaporation to form uranyl nitrate hexahydrate (UNH) which is then heated to yield uranium trioxide. Continuous processes utilize stirred trough denitrators.

FIGURE A.1-4. UF₄ Slurry Processing

A.1.1.8 Evaporation and Denitration

The purified uranyl nitrate is concentrated by evaporation in a recompression evaporator followed by batch boildown tanks and converted to UO_3 by heating in agitated trough denitrators.

The boildown tanks are heated by steam. The boiling temperature is controlled at levels ranging from 113 to 123.5°C. Solution freezing points range from 140 to 200°F, thus requiring heating on all transfer lines and storage tanks. In general, nitric acid is recovered from the denitrator off-gas, and condensed streams containing uranium are recycled.

The concentrated UNH is denitrated to UO_3 by a horizontal heated and agitated trough.

UNH is introduced through four feed pipes extending vertically from the top of the trough to within about 75 mm of the bottom. The uranium trioxide produce overflows through a weir into a collection bin below the reactor. The uranyl nitrate hexahydrate feed rate is regulated to control the bed temperature at a set value, usually about 274°C.

Trough outer wall temperatures are controlled at 482 to 592°C. The uranium trioxide overflow is conveyed to a hammer mill. Off-gases go through a wet scrubber and then to a nitric acid absorber.

A.1.1.9 Reduction of Uranium Trioxide to Uranium Dioxide

The UO_3 from the denitrators is pulverized and then converted to UO_2 and UF_4 by successive reactions with H_2 and HF. Two stages of 0.46-m-diameter fluid-bed reduction and two stages of 0.762-m-diameter fluid-bed hydrofluorination are furnished.

The fluid-bed reduction is a two-stage system operated with series powder flow and parallel gas flows. With this arrangement, the desired use of staging to prevent short-circuiting of raw feed powder to the product discharge is accomplished. The parallel gas flows permit reduction of the reactor cross-sectional area, since this parameter is determined by the design gas velocity. Both stages are heated electrically and are provided with external air cooling coils.

Uranium trioxide is screw-fed from a weighed storage hopper into the reducer near the top of the bed. The outlet gases carry the uranium dioxide product into a collection hopper and are vented through sintered metal filters. The filter tubes are pulsed periodically with nitrogen to dislodge powder which drops into the hopper. The uranium dioxide is transferred to a seal bin.

The hydrogen gas is supplied by cracking ammonia. A mixture of 75 mole% hydrogen and 25 mole% nitrogen is formed by dissociation of ammonia and is passed through the fluid beds. The off gases after filtering are vented through a burner.

A.1.1.10 Hydrofluorination of Uranium Dioxide to Uranium Tetrafluoride

The fluid-bed hydrofluorinator consists of a 406-mm-diameter by 3.7-m-long feed screw reactor with 762-mm-diameter fluid beds arranged in series. The 152-mm-deep powder bed in each reactor is agitated slowly with a wall scraping type stirrer. Uranium dioxide powder is screw fed into the screw reactor near the top of the bed. The partially reacted product powder and the outlet gases are discharged into an intermediate hopper which feeds the second reactor. The outlet gases vent through carbon filters. The second fluid bed functions like the first, with the exception that the outlet gases carry the powder to the uranium tetrafluoride product hopper and are then introduced without filtering to the bottom of the first reactor. Both reactors are provided with electrical heating and external forced-air cooling.

Hydrogen fluoride economy in the fluid-bed system can be accomplished in several ways. A procedure is to sell dilute acid. The hydrogen fluoride flow to the reactor is adjusted to give the desired outlet concentration, and the aqueous acid is collected in tanks after condensation in a heat exchanger. When there is no market for the acid, the hydrogen fluoride is neutralized with lime.

A.1.1.11 Fluorination and Condensation

The UF_4 is converted to UF_6 by reaction with F_2 in a system consisting of primary and cleanup flame reactors. Primary reactors operate with an excess of F_2 gas to consume as much UF_4 as possible, while the cleanup units use an excess of UF_4 powder to remove the F_2 from the vent gas stream. The primary reactors are 203 mm in diameter by 3.7 m long. Primary and secondary cold traps are provided for UF_6 collection.

A.1.1.12 Fluorination of Uranium Tetrafluoride to Uranium Hexafluoride

The production process use flame reactors (fluorinating towers) for the conversion of uranium tetrafluoride to uranium hexafluoride.

In principle, a primary reactor is used in which uranium tetrafluoride is essentially converted completely to uranium hexafluoride using an excess of fluorine. The unused fluorine is then passed through a secondary or cleanup reactor which operates with an excess of uranium tetrafluorine and consumes the fluoride. A single cleanup reactor can service a bank of primary reactors.

The fluorinator consists of a vertical 203-mm-diameter, 3.7-m-long pipe, provided with external cooling coils which are continuously welded to the shell and are used to remove the large amount of heat released by the reaction. Steam is used as the coolant. Uranium tetrafluoride is charged by a screw conveyor and drops through a powder disperser into the top of the reactor. Dispersion is accomplished by a spined shaft rotating at 1,750 rpm. Fluorine is introduced through an annulus at the bottom of the disperser where it reacts essentially instantaneously with the uranium tetrafluoride. When viewed through a sight port, a white-hot flame is observed just below the point at which the reactants are charged. If gas flows are controlled in the desirable range, 95% or more of the uranium tetrafluoride is consumed. In fact, under closely controlled pilot-plant conditions, more than 99% of the uranium tetrafluoride reacts. Most of the remaining solids are collected in an ash receiver mounted directly below the tower. The gases and a small part of unreacted uranium tetrafluoride are discharged through a gas cooler and a sintered metal filter. Solids collected in the filter usually contain a high percentage of non-volatile impurities and are reprocessed through a wet uranium recovery system. Ash in the tower receiver consists of uranium fluorides and impurities and is ground and recycled.

The gas leaving the primary fluorinator is passed through a heat exchanger (cold trap) which operates at -17°C and removes a large portion of the uranium hexafluoride by desublimation. The gas is compressed and flows to the cleanup reactor.

In the cleanup reactor, an excess of uranium tetrafluoride is contacted with the vent gas in a flame-type unit which has heat supplied to hold the wall

temperature above a minimum of 454°C. Cooling coils are also provided as in the primary towers. At this temperature level, the fluorine is consumed, and uranium hexafluoride and other uranium fluorides are present. The hot gas and solid stream are cooled before filtering. As the stream is cooled, the reaction of the uranium hexafluoride with the uranium tetrafluoride forms low-melting compounds. Caking is prevented by cooling in a horizontal section equipped with a wall-scraping ribbon-screw. The solid discharged from the cooler is an uranium fluoride material which is recycled to the fluorination reactor.

A.1.1.13 UF₆ Condensation

Uranium hexafluoride in the gases leaving the cleanup reactor is removed in a refrigerated cold trap operating at minus 24°C. The uranium hexafluoride content of the gas leaving the cold trap is about 50 ppm. This gas is then vented to the waste gas disposal system.

The UF₆ product is collected in two different types of heat exchangers. The first or primary cold trap operates at approximately 4.4°C (40°F) and removes most of the UF₆. This trap is of tube and shell construction.

U-tubes are used to minimize thermal stresses and to allow easy removal of the core from the containing vessel. The heat exchange surface consists of flat plate fins attached to the tubes. Baffles are provided to give side-to-side flow. A water-glycol solution is circulated through the tubes during the cooling cycle. The trap is heated to liquify and drain the UF₆. Heated glycol is passed through the tubes when the trap is heated.

The secondary traps heating and cooling system is similar to the primary traps except that they operate at -50°C. Freon is recirculated and used as the coolant.

After collection, the UF₆ is heated to above its triple point of 64°C, filtered, and drained into an evacuated shipping cylinder.

A.1.1.14 Fluorine Production

The production system consists of 60 electrolytic cells with two conditioning systems. Fluorine is produced from HF by the electrolysis of HF dissolved in a fused saltbath of KF-HF in medium-temperature, water-cooled cells. Hydrogen is produced as a byproduct and subsequently burned.

A metered continuous flow of gaseous hydrogen fluoride is fed to the electrolyte to maintain the optimum composition and also to hold sufficiently high liquid level to prevent mixing of the hydrogen and fluorine.

The fluorine is compressed with a centrifugal unit and is charged to the uranium hexafluoride production reactors. Hydrogen flows to a gas burner. Both the hydrogen and fluorine headers are controlled at about 25 mm of water positive pressure on the cell side. Close control of the pressure of both these streams is provided to prevent mixing fluorine and hydrogen.

Both the hydrogen and the fluorine leaving the cell contain 10% or more hydrogen fluoride. This concentration is reduced before use in the uranium hexafluoride manufacturing systems. The fluorine stream is passed through condensers which collect hydrogen fluoride as a liquid and reduce the concentration to about 4%. The condensed anhydrous hydrogen fluoride is reused. Reduction to the 4% level is adequate to prevent deposition of hydrogen fluoride in subsequent uranium hexafluoride collection cold traps.

A.1.2 Effluent and Waste Disposal

Normal operation of the UF_6 plant produces waste heat along with a variety of solid, liquid, and gaseous wastes which are treated to meet federal and state pollution control standards before being released to the environment. Liquid wastes containing significant quantities of radioactive materials are currently impounded in settling basins and stored. Other liquid wastes which contain traces of radioactivity are diluted and released. All airborne effluents are treated to remove uranium contamination and noxious gases produced in the manufacturing process.

A.1.2.1 Liquid Waste Treatment and Disposal Systems

The plant process generates two major liquid waste streams of varying composition which are treated. The solvent extraction waste treatment flow sheet is shown in Figure A.1-5.

The waste stream from the extraction system is primarily a solution of ammonium nitrate, nitric acid, metallic salts, and minute quantities of uranium and the radioactive daughter products of normal uranium decay. This stream

FIGURE A.1-5. Solvent Extraction Waste Treatment System Flow Sheet

is combined with spent sodium hydroxide from the solvent treatment and miscellaneous digester scrubber systems along with any recovered weak acids. This liquid is neutralized by ammonia and impounded in storage ponds. Neutralization of the extraction waste removes residual uranium and thorium.

A second liquid waste stream is generated by the hydrofluoric acid scrubber. This fluoride waste stream is combined with waste sodium carbonate solutions originating in the fluorine cell rework area, with any acid spilled in the HF vaporizer room sump, and with laboratory wastes. The combined stream is treated with lime which neutralizes acids and precipitates fluorides as calcium fluoride. The fluoride waste treatment system flow sheet is shown in Figure A.1-6.

The alkaline sludge is permitted to settle in a retention basin to permit flocculation and sedimentation. The flow is treated with sulfuric acid to adjust the pH and precipitate excess calcium. It is then permitted to clarify. The clarified treated waste overflows and is combined with clean bypass water and the sewage lagoon overflow. A concrete stilling basin at the point of combination allows for mixing of the flow and controlled release through a flume so that the rate of discharge can be measured.

Other plant streams which are combined with the treated fluorine effluent before being discharged to the river include sanitary and domestic waste liquids.

A.1.2.2 Solid Waste Treatment and Disposal

Solid wastes generated at the facility originate from operations in the facility or the treatment of liquid waste. Each material generated must be treated in its own particular fashion in order to ensure no spread of radioactive materials to the environment.

Combustible Wastes. Combustible wastes are examined for contamination and disposed of in two ways. Contaminated combustible wastes are buried onsite at a designated location near the rear of the active plant area, while non-contaminated combustible wastes are burned in an open pit constructed in accordance with air pollution regulations.

FIGURE A.1-6. Fluoride Waste Treatment Flow Sheet

Non-Combustible Wastes. Non-combustible wastes consist of: 1) used anodes from the fluorine generation system, 2) scrap contaminated drums resulting from transportation of yellowcake to the site, and 3) maintenance waste generated by the repair or removal of equipment in the plant. The fluorine anodes composed of graphite are not contaminated with radioactive material and constitute a waste of very low volume. It is contemplated that these will be buried in accordance with the solid waste disposal regulations. Contaminated solids are buried onsite in accordance with 10 CFR 20.301 at a designated location at the rear of the active plant area.

Solids Resulting from Liquid Waste Treatment. As described in previous sections, liquid wastes are treated by the removal of chemical toxicity and subsequently disposed of. The selected treatment generally results in the precipitation of a solid material slightly contaminated with radioactive elements that must be further disposed of.

Fluoride Treatment Wastes. Fluoride-contaminated liquids are treated with lime in order to precipitate the fluoride as the insoluble calcium fluoride. This procedure also results in the formation of calcium hydroxide and calcium carbonates in excess of those used for the reaction. These materials precipitate at the bottom of a sludge pond and are buried onsite.

Solvent Extraction Wastes. Extraction waste is currently neutralized with ammonia, which results in the precipitation of heavy metal oxides and hydroxides. These precipitates, along with any acid-insoluble contents of the concentrate, settle to the bottom of the raffinate storage ponds where they accumulate as sludge. Further treatment of the liquid portion with BaCl results in the precipitation of barium radium sulfate, thus eliminating radioactivity from the liquid wastes.

A.1.2.3 Gaseous and Particulate Control

Ventilation systems are provided throughout the manufacturing facility. Special attention is given to those areas having the potential for contamination with radioactive materials. Process areas in the manufacturing building are ventilated by forcing air into the building at a rate sufficient to provide

10 air changes per hour as the air is exhausted through eleven $8.5\text{-m}^3/\text{min}$ vents as well as through non-powered roof hatches. Both vents and hatches are sampled to evaluate radioactive discharges to the atmosphere in accordance with 10 CFR 40.65.

Fumes from the fluorine cell maintenance area are discharged directly to the atmosphere by means of a $450\text{-m}^3/\text{min}$ exhaust fan.

The exhaust from each laboratory fume hood is conveyed to the sampling area roof through separate sampled ducts.

Equipment used to remove particulates from gas streams include cyclones, bag filters, and porous carbon or porous metal filters. A central, all-purpose dust collection system is provided to service all dry-processing areas within the plant. Dust collection hoods are installed around packing glands, routinely opened equipment, solids transfer areas, and the discharge from special-duty dust collectors or vacuum systems. The size and shape of the hoods and the exhaust air velocities are all designed to obtain efficient collection of the generated dust. The dust collector is a $785\text{-m}^3/\text{min}$, 5-zone plenum pulse (reverse jet), bag-type filter which discharges to the atmosphere through a monitored stack. Collected solids are drummed and recycled to process. The stack sampler operates continuously, with a 24-hour sample collected and analyzed daily.

Four additional vacuum cleaning and dust collection systems providing separate vacuum sources, cyclone separators, and bag filters are used in specialized plant areas. The sampling plant has separate vacuum cleaner and dust collection systems for recovery and direct recycle of spilled yellowcake, with the filtered air from the vacuum cleaner discharged to the sample plant dust collector. Filtered air from this system in turn discharges to the main plant dust collector, thus providing three stages of filtration. Two additional vacuum recovery systems, each consisting of a cyclone, bag filter, and vacuum pump in series, are provided, with both bag filters discharging to the main plant dust collector for backup filtration. One system services all areas after yellowcake sampling through the reduction area and discharges collected solids

to the miscellaneous digester. The second unit, the UF_4 ash vacuum system, services the hydrofluorination and fluorination areas, with collected solids drummed for transfer to the ash grinding system. Both systems are used for general plant clean-up in their respective service areas and for removal of uranium materials from equipment components prior to maintenance operations.

The entire ash grinding system provided for all-purpose crushing, pulverizing, and screening of fluorination tower ash or other materials capable of direct recycle is housed in an enclosure which is vented to the main plant dust collection system.

The primary vent system for the digesters provides for removal of entrained dust and liquid from the off-gas vapor with return to the digestion tanks. The vent system discharges to the nitric acid recovery system where oxides of nitrogen are removed. The system is maintained at a slight negative pressure to prevent leakage to the atmosphere. Off-gas from the miscellaneous digester is processed through a water scrubbing system and a caustic scrubber to remove nitrogen oxides, HF vapor, and entrained uranium or fluoride bearing liquids or solids. After scrubbing, the off-gas is routed to the nitric acid recovery system. Spent scrubber liquids are disposed of with the solvent extraction waste. Denitrator off-gases are scrubbed with nitric acid and then cooled before being fed to the nitric acid absorber.

Off-gases from the reduction system are vented through a sintered metal filter and a backup filter for 98% removal of 0.7-micron particles. Excess hydrogen is burned and vented to the steam boiler stack.

Gases passing through the hydrofluorinator back-up filter and the HF condenser are combined with the off-gases from the waste gas burner servicing the fluorine plant and the fluorination system off-gases and are scrubbed with water before being vented to the atmosphere. The scrubber liquor is discharged to the fluoride treatment ponds.

The fluorination system uses a once-through fluorine process, with the UF_6 leaving the primary reactor through sintered metal filters and passing through a back-up filter before entering a primary cold trap. Residual gas is fed to a

cleanup reactor, with the UF_6 leaving this unit passing through two sintered metal filters and then into the secondary cold trap.

A.1.3 Ventilation Systems

Ventilation systems are provided for the various areas of the facility. Special attention is given to obtain adequate air changes in areas having air contamination potentials.

Ducts in the solvent extraction and fluorine areas are of aluminum. Galvanized steel ducts are used in the office and laboratory, shops, and maintenance facility and utility areas.

Air for the heating and ventilation of the main process area is provided by two fan-coil units (total capacity $2600 \text{ m}^3/\text{min}$), one at each end of the process area. Ventilating air is introduced to maintain a minimum of six air changes per hour. Ten air changes is typical in the main process area.

The fluorine cell area is ventilated by four fans, for a total of about $1020 \text{ m}^3/\text{min}$ of exhaust air, resulting in approximately 31 air changes per hour.

The ventilation of the solvent extraction building provides for a minimum of 12 air changes per hour by a supply fan-coil unit with a $680 \text{ m}^3/\text{min}$ capacity.

A.1.3.1 Main Plant Dust Collection System

A central, all-purpose dust collection system is provided. The system services all "dry-processing" areas within the plant, sampling/digestion and UO_3 pulverizing through ash grinding. The dust collection hoods servicing the areas are installed around packing glands, routinely opened equipment, solids transfer areas, and the discharge from special-duty dust collectors or vacuum systems. Size and shape of hoods and exhaust air volume are designed to efficiently collect the dust normally generated by the source.

The $785\text{-m}^3/\text{min}$, 5-zone, "plenum pulse" (reverse jet), bag-type dust collector discharges through a monitored stack above the plant to the atmosphere. The collector is periodically cleaned (back-blown) stepwise, one zone at a time.

It can also operate with one or two zones out of service for maintenance. The solids are drummed off and reprocessed. The stack sampler operates continuously. A 24-hour sample is collected and analyzed for radioactivity daily.

Vacuum Systems for Recovering Soiled Solids. The sampling plant has a separate vacuum for recovery and direct recycle of spilled yellowcake. Two other vacuum recovery systems are provided. Both are used for general plant clean-up and for "vacuum gulping" of uranium compounds from equipment components prior to maintenance servicing. The main plant vacuum system services all areas after yellowcake sampling to hydrofluorination. The UF_4 ash vacuum system services all areas from hydrofluorination through fluorination. The main plant system discharges to the miscellaneous digester. The UF_4 ash system discharges to drums for transfer to the ash grinding system. Each vacuum system has one vacuum source with a cyclone separator and bag filter in series.

The general cleanup piping system is provided with hose connections at convenient points. Piping sizing provides a minimum gas velocity of 91.4 k/min.

The UF_4 ash system depends on screens in the ash grinding system to protect conveyors and screws when it is re-fed to the primary process stream.

A portable vacuum cleaner is also provided.

Ash Grinding System for Direct Recycle of Fluorination. An ash grinding system is provided for all-purpose crushing, pulverizing, and screening of fluorination tower ash, or any other material capable of direct recycle. The pulverized product is normally re-fed directly to fluorination through the UF_4 storage bin. The product can be redrummed. The system consists of a jaw-crusher, a hammer mill pulverizer, and a vibrating screener. Material is fed to the system via a drum elevator. The entire system is housed in an enclosure which is vented to the main plant dust collection system.

The hammer mill is protected from pieces of upstream equipment or other foreign matter by 25-mm screen in the feedstream, after the crusher. Collected solids from the UF_4 ash vacuum system are also recycled through the ash grinding system.

A.1.3.2 F₂ Cell Maintenance Area Fume Exhaust

Fumes from the dip tank and electrolyte area are exhausted by a 450-m³/min exhaust fan to the atmosphere.

Laboratory System. The exhaust from each laboratory fume hood is conveyed to the roof of the sampling area through separate ducts. Sampling of these ducts is done until ample data shows it is no longer necessary.

Off-Gas Handling. Off-gas handling systems are described below.

Digestion. The primary vent system for the digestion vessels provides for removal of entrained dust and solution from the vapor stream and return to the digesters. The primary vent system for the digestion vessels exhausts to the plant nitric acid recovery system, where NO₂ and NO are scrubbed out. The digestion system is maintained to assure that all air flow is inward.

Off-gas from the miscellaneous digester is processed through a water scrubbing system and a caustic scrubber to remove nitrogen oxides, HF vapor, and entrained uranium or fluoride bearing solutions or solids. After scrubbing, the off-gas is routed to the HNO₃ recovery system. Spent scrubbing liquor is disposed of with the raffinate.

Denitrator off-gases are wet-scrubbed with nitric acid and then pass through a gas cooler before passing on to the nitric absorber column.

Gases from reduction to UO₂ are vented through a sintered metal filter then through a back-up filter (nominal 10-micron pore size for 98% removal of 0.7-micron particles). Excess hydrogen is burned before venting. Burned gases are vented via the boiler stack.

Hydrofluorination. Only small amounts of gases containing traces of HF are vented from the HF condensers except when purging the reactors. In the latter case, very little HF should be present. The vent gases are combined with other similar gases and scrubbed with water prior to being vented through a pipe associated with the main boiler stack. The waste scrub liquor is discharged to the fluoride treatment ponds.

Fluorination. The system uses a once-through fluorine process. The UF_6 -laden gas leaving the sintered metal filters from the primary reactor passes through a back-up filter and then through a primary cold trap. Gas passes through a tower-type cleanup reactor. The UF_6 gas leaving this reactor is removed in the secondary cold trap and the gas is sent to off-gas disposal. Under normal conditions with the fluorine clean-up reactor performing properly, little F_2 is discharged from the system. Any waste F_2 is burned with H_2 from the fluorine plant to form HF , which is quenched and scrubbed with water. The scrubbed waste gases are discharged to the atmosphere through pipe associated with the boiler stack. The waste scrub liquor is discharged to the fluoride treatment ponds.

A.2 PLANT DESCRIPTION

This section provides details of the reference UF_6 conversion plant to supplement the general plant description given in Section 7.3 of Section 7. The plant is assumed to be located on the reference site described in Section 7.1 and Appendix B. The plant area, shown in Figure A.2-1, includes an area of about 30.4 hectares that contains the main process building waste treatment facilities, equipment and materials storage pads, and other supporting facilities.

The sanitary lagoons are used for holding discharge from the plant non-radioactive laundry, showers, sinks, and toilets. The sewage treatment plant and sanitary lagoon have very little contamination and are not considered to be part of the decommissioning activities.

Figures A.2-2 shows a plan of the main building layout. An isometric drawing of the plant is shown in Figure 7.3-1 of Section 7.

A.2.1. Buildings

The main plant building occupies 7500 m^2 of manufacturing, laboratory, maintenance, decontamination, storage, and office floor space. Contaminated waste incineration operations occupy another 220 m^2 of floor space in a separate building located 30 meters to the west of the main building. Other auxiliary facilities include fluoride and extraction waste treatment lagoons, liquid

FIGURE A.2-1. Site Layout of the Reference Plant

FIGURE A.2-2. Main Building Layout of Reference Plant

chemical waste treatment lagoons, a sanitary waste treatment plant, a tank and pump station, equipment storage yards, uranium storage yards, an electrical substation, and a warehouse.

A.2.1.1 Main Building

The main building is an 55-m by 100-m structure fabricated of 38-mm insulated metal siding attached to a steel framework. The interior walls are constructed of 200- by 200- by 410-mm concrete block and 13-mm sheet rock in the office, laboratory, and control room areas and sheet metal in the process areas. 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 35-mm insulated corrugated metal deck that is capped with 38 mm of asphalt and gravel.

The processing areas or rooms within the main building are generally partitioned with painted sheet-metal walls, or in some instances, with painted wall board. The floors of the production areas are generally of painted concrete or steel plate. 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.

A.2.1.2 Solvent Extraction Building

The solvent extraction building has a steel frame with metal sidings and sealed concrete floors.

A.2.2 Detailed Plant Descriptions

The following sections describe the process and laboratory areas of the plant. The descriptions include tabulations of the equipment items located in each area. Estimates of shipping volumes and weights are included with these tabulations.

A.2.2.1 Sampling Station

Table A.2-1 lists the equipment in the sampling station room of the main building.

TABLE A.2-1. Sampling Station Equipment and Laboratory

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping ₃ Volume (m ³)	Estimated Weight (Mg)
Drum Elevator	2	5D length	12.8	4.4
Yellowcake Receiving Bin	2	1.5 x 1.5 x 2.8	12.5	6.3
Primary Sampler I	1	1.7 x 0.9 x 2	2.7	2.6
Primary Sampler II	1	1.7 x 0.9 x 2	2.7	2.6
Secondary Sampler	1	1.5 x 0.9 x 2	2.5	1.8
Yellowcake Transfer Screw	2	33 length	4.2	1.5
Bucket Elevator	2	30 length	4.8	1.1
Digester Feed Bin	2	1 x 1 x 2.6	2.6	3.1
Hood and Ductwork	--	--	4.1	4.8
Stacker and Forklift	--	--	12	5.0
Storage Racks	--	--	23	5.0

The receiving and sampling area covers about 165 m² and contains sampling equipment. The room contains elevators, bins, samplers, and transfer equipment.

A.2.2.2 ADU Slurry Receiving and Storage

Table A.2-2 lists the equipment in the ADU slurry receiving and storage area.

TABLE A.2-2. Wet Yellowcake System Equipment

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping ₃ Volume (m ³)	Estimated Weight (Mg)
ADU Storage Tank	2	38 m ³	76	12.6
Receiving and Weight Tanks	1	3 m ³	3.8	3.0
Piping	--	--	2.6	2.0
Miscellaneous Equipment	--	--	2.0	3.0
Receiving Pumps	2	0.9 x 0.9 x 1.4	3.0	4.0
Transfer Pump	2	0.7 x 0.7 x 1.4	2.8	3.5

A.2.2.3 UF₄ Slurry Processing

Table A.2-3 lists the equipment in the UF₄ slurry sampling and digestion area.

The ADU receiving and storage area covers 50 m² and contains facilities and equipment for dissolving ADU slurry with nitric acid including tanks, piping, pumps, and equipment.

A receiving building for the UF₄ is located adjacent to the west wall of the yellowcake slurry receiving area. The UF₄ slurry receiving building rests on a curbed foundation with volume to contain the tank contents. The equipment includes a process feed hopper, a mechanical drum dumper, a digester tank, a scrubber, a pump, and process piping.

TABLE A.2-3. UF₄ Slurry Equipment

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping ₃ Volume (m ³)	Estimated Weight (Mg)
Process Feed Hopper	1	2 x 2 x 3	12.0	3.0
Mechanical Drum Dumper	1	--	3.0	2.0
Digester Tank	1	17 m ³	17.0	4.5
Scrubber System	1	--	6.8	1.4
Pumps	1	1.1 x 1.1 x 1.6	2.3	1.0

A.2.2.4 Dissolution of Ore Concentrate

Table A.2-4 lists the equipment in the ore concentrate dissolution system.

The digestion area covers 200 m² of the main building and contains digestion tanks, adjustment tank piping and pumps.

TABLE A.2-4. Ore Concentrate Dissolution Equipment

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping ₃ Volume (m ³)	Estimated Weight (Mg)
Digestion Tanks	3	2.5 dia. x 2.7	44	12.0
Adjustment Tanks	2	2.6 dia. x 2.7	30	8.0
Piping	--	--	4.6	2.0

A.2.2.5 Solvent Extraction

Table A.2-5 lists the equipment in the solvent extraction system.

The solvent extraction building covers 377 m² and contains decanters, pulse columns, holding tanks and piping and pumps.

TABLE A.2-5. Solvent Extraction Equipment

<u>Equipment Item</u>	<u>Quantity</u>	<u>Unit Dimension L x W x H (m)</u>	<u>Shipping₃ Volume (m³)</u>	<u>Estimated Weight (Mg)</u>
Pumper Decanters	6	1.2 dia. x 1.9	14	8.3
Scrub Decanters	2	1.2 dia. x 1.9	4.3	3.5
Re-extraction Pulse Column	2	0.8 dia. x 2.5	2.5	6.0
Raffinate Holding Tank	3	2.8 dia. x 2.8	36	16.5
Piping	--	--	6.4	4.0
Miscellaneous Equipment	--	--	5.1	3.0

A.2.2.6 Concentration of Uranyl Nitrate

Table A.2-6 lists the equipment in the concentration area of the main building. The area covers 130 m² and includes the evaporator tanks, filters, and piping.

TABLE A.2-6. Uranyl Nitrate Concentration Equipment

<u>Equipment Item</u>	<u>Quantity</u>	<u>Unit Dimension L x W x H (m)</u>	<u>Shipping₃ Volume (m³)</u>	<u>Estimated Weight (Mg)</u>
Evaporator	1	2.4 x 1.8 x 1.8	7.9	6.8
Boildown Tanks	3	3.0 dia. x 3.0	42	15.0
UNH Filter	3	0.8 x 0.8 x 0.8	1.5	1.5
Piping	--	--	3.4	2.1
Miscellaneous Equipment	--	--	0.9	0.3
Hood and Ductwork	--	--	6.3	7.3

A.2.2.7 Denitration Area

Table A.2-7 lists the equipment in the denitration area of the main building. The area covers 130 m² and includes denitrators, scrubbers, elevators, bins, piping, pumps, and equipment.

TABLE A.2-7. Denitration Equipment

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping Volume (m ³)	Estimated Weight (Mg)
Denitrator	4	--	27.2	6.4
Denitrator Scrubber Cooler	4	0.6 x 0.6 x 1.2	2	5.0
Cooling Screw	4	--	5.4	4.0
UO ₃ Vertical Elevator	2	--	4.3	1.8
Hammer Mill	2	1.9 x 1.9 x 2.2	16.2	4.0
Surge Bin	2	--	19.0	2.9
Piping	--	--	6.8	5.4
Hood and Ductwork	--	--	7.4	1.8
Miscellaneous Equipment	--	--	1.5	0.9

A.2.2.8 Reduction Area

Table A.2-8 lists the equipment in the reduction area of the main building. The area covers 270 m² and includes storage bins, reactor feeder and piping, and equipment.

TABLE A.2-8. Uranium Trioxide to Uranium Dioxide Reduction Equipment

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping Volume (m ³)	Estimated Weight (Mg)
UO ₃ Storage Bin	2	3 x 3 x 3.8	68.4	12.6
Screw Feeder	2	--	5.2	1.2
Fluid Bed Reduction Reactors	4	0.5 dia. x 4.6	34.0	30.5
Piping	--	--	20.0	7.6
Miscellaneous Equipment	--	--	5.0	1.7
Hoods and Ductwork	--	--	50.0	7.7
Ammonia Dissociater	2	2.1 x 2.1 x 2.8	24.6	18.8

A.2.2.9 Hydrofluorination Area

Table A.2-9 lists the hydrofluorination equipment in the main building. The area covers 260 m² and includes hydrofluorinators, storage hoppers, conveyors, piping, and equipment.

TABLE A.2-9. Hydrofluorination Equipment

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping Volume (m ³)	Estimated Weight (Mg)
Screw Feeder	4	--	19.5	3.2
UO ₂ Filter Bin	4	--	24.5	4.1
1st Stage Fluid Bed Hydrofluorinator	4	0.8 dia. x 9.0	16.3	112
2nd Stage Fluid Bed Hydrofluorinator	4	0.8 dia. x 9.0	16.3	112
UF ₄ Seal Hopper	4	--	10.8	2.4
Interstage Hopper	4	--	10.8	2.4
UF ₄ Purge Conveyor	4	36 length	5.2	4.4
Redler Conveyor (509)	4	36 length	5.2	4.4
UF ₄ Storage Hopper	4	--	10.8	2.4
Redler Conveyor (510)	4	36 length	5.2	4.4
Piping	--	--	4.6	4.0

A.2.2.10 Fluorination Area

Table A.2-10 lists equipment in the fluorination area. The area covers 270 m² of the main building. Equipment in the area includes fluorination towers, conveyors, filters, storage hoppers, clean-up reactors, and piping.

A.2.2.11 Fluorine Generation Area

The fluorine generation equipment located in the main building is listed in Table A.2-11. The area covers 1500 m² and includes cells, tanks, filters, condenser tanks, and piping and equipment.

TABLE A.2-10. Fluorination Equipment

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping ₃ Volume (m ³)	Estimated Weight (Mg)
Primary Fluorinator Tower	5	1.1 dia. x 7.6	36.1	7.0
Redler Conveyor (511)	2	36 length	2.5	3.9
Primary Filter	4	0.9 x 0.9 x 1.2	12.2	2.2
Primary Cold Trap	6	2.4 x 2.4 x 3.0	60.5	38.2
Product Cylinder	2	--	34.4	12.6
Back-up Filter	2	--	2.4	1.6
Cleanup Reactor Feed Hopper	2	--	5.2	6.2
Cleanup Reactor	2	1.1 dia. x 7.6	14.6	2.8
Redler Conveyor (512)	2	36 length	2.5	3.9
Backup Filter	2	--	2.4	1.6
Secondary Cold Trap	4	1.8 x 1.8 x 2.0	25.9	5.1
Piping	--	--	22.0	10.2
Hoods and Ductwork	--	--	6.8	4.2

TABLE A.2-11. Fluorine Generation Equipment

Equipment Item	Quantity	Shipping ₃ Volume (m ³)	Estimated Weight (Mg)
Waste Gas Burner	2	22.5	12.0
Fluorine Cells	60	160	95.0
Fluorine Surge Tank	2	18.0	120
Fluorine Filter	4	5.6	2.1
HF Condenser	2	12.6	4.3
HF Backup Filter	4	8.4	2.2
Hydrogen Surge Tank	2	16.4	12.5
Fluorine Filter	2	7.2	3.1
HF Storage Tank	2	9.5	4.2
Piping	--	8.4	6.5
Miscellaneous Equipment	--	3.2	1.6
Instruments and Controls	--	4.6	1.1

A.2.2.12 Nitric Acid Recovery Area

Table A.2-12 lists the nitric acid recovery equipment.

TABLE A.2-12. Nitric Acid Recovery Equipment

<u>Equipment Item</u>	<u>Quantity</u>	<u>Unit Dimension L x W x H (m)</u>	<u>Shipping₃ Volume (m³)</u>	<u>Estimated Weight (Mg)</u>
Tanks	2	(1.3 dia. x 1.5 length)	8.0	8.6
Piping	--	--	6.4	2.1
Equipment	--	--	3.2	1.8
Pumps	2	0.7 x 0.7 x 1.4	4.7	3.1

A.2.2.13 Instrument Repair Shop

Table A.2-13 lists the equipment located in this shop. This shop has about 110 m² of floor area and about 515 m³ of volume. The shop contains equipment for doing maintenance on electrical and electronic equipment.

TABLE A.2-13. Equipment in the Instrument Repair Shop

<u>Equipment Item</u>	<u>Quantity</u>	<u>Unit Dimension L x W x H (m)</u>	<u>Shipping₃ Volume (m³)</u>	<u>Estimated Weight (Mg)</u>
Scopes and Testers	4	0.6 x 0.5 x 0.9	1.0	0.4
Benches and Tables	4	1.8 x 0.9 x 0.9	10.2	1.2
Braze and Solder Booth	1	1.5 x 0.9 x 2.4	0.6	0.3
Cabinets	6	1.2 x 0.6 x 1.8	8.4	1.5
Cleaning Hood	1	1.5 x 0.9 x 2.1	3.0	0.3
Miscellaneous	--	--	3.0	0.6

A.2.2.14 Temporary Radwaste Facility

The equipment in the temporary radwaste facility is listed in Table A.2-14. Tanks and a centrifuge take up the bulk of the volume of the equipment.

TABLE A.2-14. Equipment in the Radwaste Room

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping ₃ Volume (m ³)	Estimated Weight (Mg)
Quarantine Tank	1	4.1 dia. x 6.1	81.1 ^(a)	2.1
Slab Tanks	1	3.7 x 3.7 x 3.7	48.9 ^(a)	18.6
Centrifuge	1	1.2 x 0.9 x 1.5	1.7	0.9
Miscellaneous (pumps, small tanks, etc.)	--	--	3.6	1.1

(a) Noncompacted volume.

A.2.2.15 Decontamination Facility

The list of equipment in the facility is provided in Table A.2-15.

TABLE A.2-15. Equipment in the Decontamination Facility

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping ₃ Volume (m ³)	Estimated Weight (Mg)
Decontamination Spray Booth	1	3.7 x 3.1 x 3.1	6.8	2.6
Cleaning Sinks	2	1.2 x 0.9 x 1.1	2.4	0.3
Sorting Tables	1	3.7 x 0.9 x 0.8	3.1	0.7
Box Dumper	1	1.2 x 1.2 x 1.8	0.5	0.2
Band Saw	1	1.2 x 0.9 x 2.7	0.6	0.4
Compacting and Packaging	2	0.9 x 0.9 x 1.5	2.0	0.4
Radiation and Uranium Scanner	1	1.2 x 1.2 x 1.2	1.8	0.5
Miscellaneous	--	--	1.8	0.5

A.2.2.16 Laundry Facility

The equipment in the laundry room is listed in Table A.2-16. This room has about 830 m² of floor space and 4,050 m³ of room volume.

TABLE A.2-16. Equipment in the Laundry

<u>Equipment Item</u>	<u>Quantity</u>	<u>Unit Dimension L x W x H (m)</u>	<u>Shipping₃ Volume (m³)</u>	<u>Estimated Weight (Mg)</u>
Washing Machines	2	1.5 x 1.5 x 1.2	5.7	0.9
Dryers	2	2.4 x 1.2 x 2.4	14.5	1.7
Sorting and Folding Tables	4	3.1 x 1.2 x 0.8	2.3	0.9
Miscellaneous	--	--	3.6	1.1

A.2.2.17 Change Rooms

The equipment in these rooms is listed in Table A.2-17. The change rooms are maintained essentially contamination-free. Like the laundry room, there is about 830 m² of floor space and 4,050 m³ of room volume.

TABLE A.2-17. Equipment in the Change Rooms

<u>Equipment Item</u>	<u>Quantity</u>	<u>Unit Dimension L x W x H (m)</u>	<u>Shipping₃ Volume (m³)</u>	<u>Estimated Weight (Mg)</u>
Hand and Foot Counters	3	0.8 x 0.8 x 2.1	3.7	0.9
Miscellaneous Survey	--	--	1.8	0.4
Miscellaneous	--	--	2.0	0.4

A.2.2.18 Incinerator Facility

The Incinerator Facility is located immediately west of the main plant, as shown in Figure A.1-2. A list of equipment in the Incinerator Facility is given in Table A.2-18.

The Incinerator Facility has a floor area of 150 m².

A.2.2.19 Piping, Ductwork, Trays, and Light Fixtures

Table A.2-19 provides an estimate of the piping, ductwork, trays, and light fixtures in the areas of the plant to be decommissioned. Piping includes process piping, service piping, independent electrical conduit and instrument lines. Ventilation ducts and hood ducts are included in the ductwork estimates. Special and separate ductwork has already been included in previous room discussions in

this report. The trays include hangers and trays with piping and conduit, which is batched and supported by the trays. Light fixtures include all main overhead lighting, most of which are fluorescent lamps.

TABLE A.2-18. Equipment in the Incinerator Facility

Equipment Item	Quantity	Unit Dimension L x W x H (m)	Shipping ₃ Volume (m ³)	Estimated Weight (Mg)
Input and Output U-Scanners	2	1.2 x 1.2 x 1.2	3.6	0.8
Shredder	1	2.4 x 0.9 x 1.8	4.1	1.3
Blow Tube Unit	1	1.5 x 0.6 x 0.6	0.6	0.2
Vortex Incinerator Chamber	1	2.4 x 2.4 x 2.4	14.5	8.9
Offgas Scrubber w/Demister	1	2.4 x 1.2 x 1.2	3.6	0.7
Scrubber Ancillaries (Tank, etc.)	1	1.2 x 1.2 x 0.9	1.4	0.3
Ash Recovery System	1	1.2 x 0.6 x 2.4	1.8	0.4
Packaging Equipment	2	1.2 x 1.2 x 1.2	3.6	0.8
Blower Unit	1	1.5 x 1.5 x 1.5	3.5	0.6
HEPA Filter Unit	1	1.5 x 1.5 x 1.8	4.3	1.0
Miscellaneous	--	--	3.6	1.1

TABLE A.2-19. Volumes and Weights of Piping, Ductwork, Trays and Light Fixtures

Location	Piping and Conduit		Duct and Supports		Trays and Light Fixtures	
	Compacted Volume (m ³)	Weight (Mg)	Non-Compacted Volume (m ³)	Weight (Mg)	Compacted Volume (m ³)	Weight (Mg)
Sampling Room	0.3	0.5	15.2	0.9	4.5	5.2
Laboratory	4.1	4.8	60.6	3.1	11.5	10.3
UF ₆ Slurry Processing Area	3.1	3.5	44.6	2.5	4.5	5.2
Digestion Area	1.6	1.8	27.7	2.7	6.2	7.2
Solvent Extraction Bldg.	54.4	74.6	125.0	12.3	68.0	70.5
Concentration Area	7.8	9.0	109.0	5.5	20.0	22.1
Demonstration Area	9.3	10.7	50.4	9.8	13.3	15.4
Hydrofluorination Area	7.8	5.9	50.8	2.3	11.8	13.6
Loading Area	1.4	1.6	19.4	0.9	2.5	2.9
F ₂ Cell	0.4	0.4	32.4	1.5	2.5	2.9
Cell Rework	1.7	2.0	5.5	0.3	3.0	3.5
Instrument Room	0.5	0.5	3.7	0.2	1.5	1.7

REFERENCES

1. Special Nuclear Material License Application for Kerr-McGee Sequoyah Facility, License No. 50B-1010, Docket No. 40-8027, Kerr-McGee Nuclear Corporation, July 1, 1975.
2. U.S. Nuclear Regulatory Commission, Final Environmental Impact Statement Related to Sequoyah Uranium Hexafluoride Plant of Kerr-McGee Nuclear Corporation, Docket No. 40-8027, February 1975.
3. U.S. Nuclear Regulatory Commission, Environmental Impact Appraisal of the Allied Chemical Corporation Nuclear Service Division Uranium Hexafluoride Conversion Facility, Metropolis, Illinois, Docket No. 40-3392, pp. 4-1 through 4-16, August, 1979.

APPENDIX B

REFERENCE SITE DESCRIPTION

A reference site is used to assess the public safety and potential environmental effects of conceptually decommissioning various nuclear fuel cycle facilities.⁽¹⁻³⁾ A summary of the site description found in References 1 through 3 is presented in this appendix. Only information directly relating to the radiation and chemical exposure pathway analysis, required in determining acceptable residual contamination levels and in calculating radiation doses for the public radiological safety analysis, is included here. The meteorological parameters and population distribution for the year 2000 are based on information presented in Reference 4. Other necessary site information is based on data reported in the site description of an operating nuclear power station.⁽⁵⁾ Estimates of the annual radioactive and chemical releases during operation of the reference UF_6 conversion plant are also included in this appendix. The remainder of the information is obtained from a variety of sources or is developed specifically for this study. All of the information is representative of potential sites for nuclear fuel cycle facilities in the midwestern or middle southeastern United States.

Individual features of an actual site (for a specific facility) may vary from those of this reference site. However, it is believed that this approach (i.e., use of a reference site) results in a more meaningful overall analysis of the potential impacts associated with decommissioning most nuclear fuel cycle facilities. A site-specific assessment is required for a similar analysis of a specific UF_6 conversion plant.

B.1 SITE LOCATION AND SIZE

The reference site is located in a rural area with characteristics similar to those found in the midwestern or middle southeastern United States. The site occupies about 4.7 km^2 in a rectangular shape of about 2 km by 2.35 km. A moderate-size river flows through one corner of the site.

B.2 DEMOGRAPHY

The site is located in a rural area with a relatively low population density, with highest population densities occurring at distances of 20 to 60 km. Population distribution data for the year 2000 is given in Table B.2-1. The total population residing within an 80-km radius of the site is about 3.52 million.

TABLE B.2-1. Population Distribution Around the Reference Site for the Year 2000⁽⁴⁾

Distance From Site Boundary (km)	Population Density (Persons/km ²)	Total Population In Annulus (a,b)	Cumulative Population
0 to 2	--(c)	10	10
2 to 3	136	2 130	2 140
3 to 5	104	5 230	7 370
5 to 6	230	7 940	15 300
6 to 8	133	11 700	27 000
8 to 20	85	89 300	116 000
20 to 30	239	375 000	491 000
30 to 50	175	878 000	1 370 000
50 to 60	298	1 030 000	2 400 000
60 to 80	127	1 120 000	3 520 000

(a) It is assumed that the population in each annulus is uniformly divided in each of 16 uniform 22.5 degree sectors.

(b) Totals are rounded to three significant figures.

(c) Indicates a population density less than 1.0/km².

B.3 LAND USE

Use of any part of the total site area for anything other than UF₆ conversion operations is assumed to be prohibited during the 40-year plant operational lifetime. The major plant facilities are located inside a 30.4-hectare fenced portion of the site. The minimum distance from the point of airborne release to the outer site boundary is 1 km. The outer site boundary is fenced and marked.

About 80% of the land within 16 km of the site is used for farming. The main crops are soybeans (60%), corn, oats, and other grains (30%), and hay (10%). It is expected that this area will remain largely agricultural, and that the population will not change significantly because of the operation of the UF₆ conversion plant.

A wildlife refuge and a state forest and campground are located about 14 km from the site. A state park is located about 10 km from the site in the opposite direction.

There are large truck gardens in the area. The nearest dwelling (the residence location of the maximum-exposed individual for the public safety analysis of decommissioning) is a farm located about 1.3 km from the site. A milk cow is kept at this farm and is maintained on fresh pasture 7 months of the year. A family garden with a growing season of 5 months is kept for fresh vegetables. River water is used to irrigate the crops on this farm.

B.4 METEOROLOGY

The reference site has a typical continental climate. It is characterized by wide variations in temperature, modest winter precipitation, normally ample spring and summer rainfall, and a general tendency to extremes in all climatic features. January is the coldest month and July is the warmest. Table B.4-1 shows monthly meteorological statistics.

TABLE B.4-1. Monthly Meteorological Statistics at the Reference Site

	<u>Jan</u>	<u>Feb</u>	<u>Mar</u>	<u>Apr</u>	<u>May</u>	<u>Jun</u>	<u>Jul</u>	<u>Aug</u>	<u>Sep</u>	<u>Oct</u>	<u>Nov</u>	<u>Dec</u>
<u>Air Temperature (°C)</u>												
Maximum	-6.1	-4.4	3.3	12.8	20	25	28.3	26.7	22.2	15	4.4	-3.3
Minimum	-16.1	-14.4	-6.7	1.7	7.8	13.3	16.1	15	10	3.9	-4.4	-12.2
Mean	-11.1	-9.4	-1.7	7.2	13.9	18.9	22.2	21.1	16.1	9.4	0.0	-7.8
Extreme Maximum	15	16.1	27.8	32.8	40.6	39.4	41.7	40	40.6	32.2	23.9	17.2
Extreme Minimum	-38.9	-36.7	-34.4	-15.6	-6.7	0.6	5.6	3.3	-5.6	-13.3	-27.8	-33.9
<u>Mean Relative Humidity (%)</u>												
	74	75	73	66	62	66	68	70	70	66	73	78

On the average, 12 days per year have maximum temperatures of 32°C and above. Annually, an average of 168 days have minimum temperatures of 0°C and below, with 40 of them at -18°C or below. The January average relative humidities at 7:00 a.m., 1:00 p.m., and 7:00 p.m. are 76, 68, and 70%, respectively. The corresponding relative humidities for July are 86, 55, and 55%.

The average annual rainfall in the area is 610 mm. The months of May through September have the greatest amounts of rainfall, with an average during this period of 432 to 457 mm (70% of the annual total). The maximum 24-hr total rainfall for the period 1894 through 1965 was 127 mm and occurred in May. Thunderstorms, with an average annual frequency of 36, are the chief source of rain from May through September. Snowfall in the area averages 1070 mm annually, with occurrences recorded in all months except June, July, and August. The extremes in annual snowfall of record are 152 mm minimum and 2235 mm maximum.

Annually, the winds are predominantly bimodal. This bimodal distribution is characteristic of the seasonal wind distributions as well. The average wind speed for spring is 11 km/hr and for the other seasons is about 16 km/hr. The maximum reported wind speed of 148 km/hr was associated with a tornado. Tornadoes and other severe storms occur occasionally. Eight tornadoes were reported from 1961 through 1967 in the county where the site is located. The probability of a tornado striking a given point in this area is about 5×10^{-4} per year. For design purposes a wind velocity of 480 km/hr is assumed to be associated with tornadoes.

Natural fog that restricts visibility to 0.4 km or less occurs about 30 hr/yr. Icing caused by freezing rain can occur between October and April, with an average of one to two storms per year. The mean duration of icing on utility lines is 36 hr.

Diffusion climatology comparisons with other locations indicate that the site is typical of the region, with relatively favorable atmospheric dilution conditions prevailing. Thermal inversions occur about 32% of the year, and the frequency of thermal stabilities is 19% slightly stable, 27% stable, 20% neutral, and 34% unstable.

Data from a number of river sites used for nuclear power reactors are used to calculate the "typical" annual atmospheric dispersion pattern in an average 22.5-degree sector around the site. Dispersion factors at selected distances for the average sector are determined from joint-frequency distributions for each site. This is done by calculating the dispersion factor, $\bar{\chi}/Q'$, for each sector at selected downwind distances and then calculating the average dispersion factor at each distance. In other words, the dispersion factors in those sectors corresponding to overland trajectories are added without regard to direction and divided by the number of sectors involved. Thus, an average dispersion factor is obtained for each selected downwind distance for all 16 sectors, based on data given in Reference 4.

Standard groups of meteorological data are interpolated from the specific site data. The groupings provide four stability classes based on vertical temperature gradient and five wind speed classes based on the Beaufort wind scale.⁽⁶⁾ The stability classes are based on Reference 7 information, with Pasquill Classes A, B, and C classified as B (unstable); Pasquill Class D (neutral); Pasquill Class E (slightly stable); and Pasquill Classes F and G classified as F (moderately stable).

Some of the wind-speed data is available for only one height. In this event, the measured values are extrapolated to the 10-m level for ground level release calculations and to the 150-m level for reference stack release calculations. Where measurements at two heights are available, the highest is extrapolated to 150 m and the lowest to 10 m, using a standard power-law extrapolation procedure.⁽⁶⁾

The ratio of the maximum sector dispersion factor to the average sector dispersion factor is 2.5. This value is used for all release heights in this study. Investigation of the change in this ratio with increasing distance from the site shows that the ratio remains essentially constant. The dispersion factors for the average sector as a function of height of release and downwind distance are shown in Figure B.4-1.

FIGURE B.4-1. Average ($\bar{X}/Q'$) Values Versus Distance in a Sector from the Reference Site

No credit for plume rise from either momentum or buoyancy is taken in this study. Where large volumes of heated air are being ejected, the plume rise constant for momentum is estimated to be about $50 \text{ m}^2/\text{sec}$. Assuming an annual average wind speed of 2 to 3 m/sec, the increase in effective stack height due to momentum would be about 15 to 25 m. Plume rise from buoyancy (heat effect) would add at least another 25 to 100 m of effective stack height, depending on the temperature of the exhaust gases. Thus, the $\bar{X}/Q'$ values illustrated in Figure B.4-1 are larger than they would be if credit had been taken for momentum and buoyancy.

B.5 ESTIMATES OF RESIDUAL RADIOACTIVITY

In order to make decisions on alternative uses for a site, information is needed on the levels and nature of the radioactive contamination present at the time of decommissioning. This section discusses the potential site radioactive contamination, the origin of this contamination, and its expected levels at the time of plant shutdown.

Naturally occurring radionuclides and radionuclides from nuclear weapons testing will be present on any nuclear facility site consistent with the level generally found in the local region. Naturally occurring radiation results from both extraterrestrial and terrestrial sources. Extraterrestrial radiation originates in outer space as cosmic radiation which interacts with particles in the upper atmosphere to emit radiation that exposes people on the planet surface. A typical dose rate at sea level from this source is about 30 mrem/yr. The dose rate from this source is altitude-dependent, and it is about 120 mrem/yr in Colorado.⁽⁸⁾ Terrestrial radiation is emitted by naturally occurring radionuclides in the soil, rocks, or atmosphere. Terrestrial radioactivity results in an internal dose rate of about 20 mrem/yr (mostly from ^{40}K), and in an external dose rate of about 50 mrem/yr. Radioactive contamination levels resulting from atmospheric nuclear weapons testing vary markedly from region to region. Radiation from deposited ^{137}Cs is the main long-term contributor to external dose, while radionuclides like ^{90}Sr , ^{137}Cs , ^{129}I , ^{131}I , and plutonium isotopes contribute to internal dose rates.

Radioactive contamination is also expected to be present on the site as a result of effluents released during normal operations over the anticipated operating UF_6 conversion plant life of 40 years.

Accidental releases during plant operation are not expected to significantly increase these levels because it is assumed that normal operating practice is to clean up contaminated areas immediately following such events. Estimates of the ground contamination levels are based on the deposition of predicted normal operating releases of gaseous effluents.

The estimated annual atmospheric release of radionuclides from the reference UF_6 conversion plant are listed in Table B.5-1.⁽⁹⁾

TABLE B.5-1. Estimated Annual Atmospheric Release of Radionuclide from the Reference UF_6 Conversion Plant^(a)

Radionuclide	Half-Life (days)	Release (Ci/yr)
$^{230}Th(b)$	2.8×10^7	1.4×10^{-3}
$^{234}Th(b)$	2.4×10^1	1.2×10^{-1}
$^{226}Ra(b)$	5.8×10^5	8.3×10^{-5}
$^{234}U(c)$	8.9×10^7	1.6×10^{-1}
$^{235}U(c)$	2.6×10^{11}	7.4×10^{-3}
$^{238}U(c)$	1.6×10^{12}	1.6×10^{-1}
Total		4.5×10^{-1}

(a) Based on information given in Reference 9.

(b) Based on isotopic ratios in yellow-cake.

(c) Based on the ratio of radionuclides in natural uranium.

The concentration of deposited radionuclides is estimated using an NRC computer model X0Q00Q.⁽¹⁰⁾ Constant annual releases are assumed over the 40-year plant life. The release point is assumed to be at ground level (10 m), with no credit taken for plume rise due to either buoyancy or momentum. Meteorological data on joint frequency of wind speed and wind direction by atmospheric stability are not available for the reference site. Therefore, data are taken from the Barnwell Environmental Report.⁽¹¹⁾ Deposition values reported are representative of the plant facilities area (area within a radius of 1000 m from the point of release). Often the contamination levels would be generally lower.

The procedure used to estimate relative deposition rates in the NRC model is based on numerical solutions to the flux-gradient ("K-Theory") diffusion equation.⁽¹²⁾ The effluent is not allowed to diffuse beyond a height of 200 m in a stable condition and 1000 m in neutral and unstable conditions. At the ground surface, a partial sink boundary condition involving the deposition velocity is assumed. The wind and eddy diffusivity profiles required as input to the diffusion equation are those presented by Markee.⁽¹²⁾ The deposition velocity is allowed to vary with wind speed in accordance with an empirical equation⁽¹³⁾ and assumes an areal grass density of approximately 70 g/m². Resulting deposition velocities are 1.2, 12, and 8.8 mm/sec for stable, neutral, and unstable conditions, respectively.

The estimated surface concentrations of the individual radionuclides averaged over the entire site are shown in Table B.5-2. The effect of radioactive decay and daughter buildup of the residual contamination is demonstrated by site contamination estimates for four decay times following plant shutdown: 5, 10, 30, and 100 years.

The radioactivity levels calculated for the site are expected to be maximum values, since no allowance is made for normal weathering such as runoff, soil, and vegetational coverage, or for resuspension of the deposited material. Except for random concentrations of the contamination, the radioactivity level of the radionuclides is expected to be some fraction of the values in Table B.5-2.

TABLE B.5-2. Estimated Quantities of Radioactive Materials Deposited on the Reference Plant Site After a 40-Year Operating Lifetime^(a)

Radionuclide	Shutdown	Deposited Radioactivity ($\mu\text{Ci}/\text{m}^2$) at Plant Shutdown and Selected Times After Shutdown			
		5 Years	10 Years	30 Years	100 Years
^{230}Th	8.8×10^{-4}	1.1×10^{-3}	1.3×10^{-3}	2.1×10^{-3}	5.0×10^{-3}
^{231}Th	6.8×10^{-3}	6.8×10^{-3}	6.8×10^{-3}	6.8×10^{-3}	6.8×10^{-3}
^{234}Th	7.5×10^{-2}	7.5×10^{-2}	7.5×10^{-2}	7.5×10^{-2}	7.5×10^{-2}
^{231}Pa	9.0×10^{-6}	1.1×10^{-5}	1.3×10^{-3}	2.2×10^{-5}	5.3×10^{-5}
$^{234\text{m}}\text{Pa}$	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}
^{234}Pa	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}	1.0×10^{-4}
^{226}Ra	5.2×10^{-5}	5.2×10^{-5}	5.2×10^{-5}	5.2×10^{-5}	5.2×10^{-5}
^{234}U	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}
^{235}U	4.5×10^{-3}	4.5×10^{-3}	4.5×10^{-3}	4.5×10^{-3}	4.5×10^{-3}
^{238}U	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}	1.0×10^{-1}
Totals	3.9×10^{-1}	3.9×10^{-1}	3.9×10^{-1}	3.9×10^{-1}	3.9×10^{-1}

(a) Based on the ratio of radionuclides in uranium and the isotopic ratios given in Reference 9.

B.6 ESTIMATES OF RESIDUAL CHEMICALS

The release of toxic chemicals during normal operations of the reference UF_6 conversion plant is of primary concern. Since uranium is a valuable commodity, the nuclear industry uses sophisticated technology to minimize losses during chemical processing. The uranium process streams contain excess reagents which can escape from the plant as airborne or liquid releases. This section contains a discussion of the chemical effluents expected from the operating UF_6 conversion plant.

B.6.1 Gaseous Chemical Releases

Discharges that contain nitrogen and fluorine are considered to be the most significant, since these materials account for the major part of the total waste discharge. Table B.6-1 contains a listing of the annual gaseous effluents from the reference UF_6 conversion plant. The gaseous effluents in Table B.6-1 are a summary of literature information. (9,14-16)

TABLE B.6-1. Estimated Annual Gaseous Effluents from the Reference UF_6 Conversion Plant

<u>Effluent</u>	<u>kg/day</u>	<u>kg/yr</u>
HF	1.5×10^1	5.4×10^3
SO_2	5.6×10^2	2.0×10^5
NH_3	2.4×10^1	8.7×10^3
H_2S	8.6×10^{-3}	6.0×10^0
Hexane	4.4×10^2	1.6×10^5
NO_2	2.6×10^2	9.5×10^4

The atmospheric concentrations of these releases are all estimated to be below existing standards for air quality. Information on the environmental concentrations of the gaseous chemical releases is not available at this time; however, it is felt that none of the chemicals listed in Table B.6-1 will result in any significant residual contamination of the reference site.

B.6.2 Liquid Chemical Releases

Estimates of the annual releases of liquid chemicals from the reference UF_6 conversion plant are listed in Table B.6-2. (9,14-16) The major liquid effluent consists of several combined fluoride treatment streams containing a weak hydrofluoric acid solution. The fluoride effluent stream is slaked with lime (calcium hydroxide) to neutralize the acid and raise the pH to 12. The resulting slurry flows to a fluoride sludge pond where most of the excess lime and calcium fluoride precipitate settles out. Overflow from the settling pond is neutralized to a pH of 7 by adding sulfuric acid. The resulting solution

TABLE B.6-2. Estimated Annual Liquid Effluents from the Reference UF_6 Conversion Plant

<u>Effluent</u>	<u>kg/day</u>	<u>kg/year</u>
NO_3^-	7.8×10^0	2.8×10^3
F^-	1.8×10^0	6.6×10^2
SO_4^{--}	2.9×10^2	1.1×10^5
NH_4^+	4.5×10^{-1}	1.6×10^2
Ca^{++}	1.9×10^0	6.9×10^2
Na^+	1.0×10^3	3.6×10^5
K^+	1.3×10^2	4.8×10^4
HCO_3^-	2.6×10^3	9.5×10^5
Cl^-	7.8×10^0	2.8×10^3
Fe	1.6×10^0	5.6×10^2

is drained to a clarifying pond where most of the remaining calcium fluoride and calcium sulfate settle out. The overflow from the clarifying pond is combined with "clean" effluent water and sewage lagoon overflow prior to discharge from the facility to the receiving river.

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APPENDIX C

RESIDUAL RADIONUCLIDE INVENTORY ESTIMATES

This appendix contains details of estimates of the radioactivity remaining in the reference facility following plant shutdown and final inventory cleanout. The final inventory cleanout is considered to be part of production operations in this study and includes collection of the more accessible amounts of uranium for accounting purposes and chemical decontamination in some areas. The residual radionuclide estimates developed in the study are used to calculate radiation exposure to workers and to the public during decommissioning.

C.1 KEY ASSUMPTIONS

Residual uranium estimates in various areas of the plant are calculated based on several assumptions. Residual uranium left on the surfaces of process areas has been evaluated in a study showing that PuO_2 and/or UO_2 powder is visible on stainless steel surfaces polished to 600 grit at a quantity of 7.5 g Pu/m^2 .⁽¹⁾ Using conventional wiping techniques to clean this polished stainless steel typically reduces the contamination level by a factor of 10.⁽¹⁾ In this study, internal surfaces of stainless steel are assumed to be contaminated with uranium compounds and materials that are barely visible. To account for the greater roughness of the stainless steel in the UF_6 plant over the roughness level used in Reference 1, the quantity of oxide powder estimated to be barely visible is assumed to be increased about 100%, to 20 g U/m^2 . For moderately contaminated areas, the quantity is assumed to be 100 g U/m^2 and for highly contaminated areas, the quantity is assumed to be 1 kg U/m^2 .

For stainless steel surfaces in wet processing areas that are acid flushed during inventory cleanout, the uranium contamination levels are assumed to be decreased by a factor of 10 from those in the dry processing system, to about 1 g U/m^2 . The above numbers apply to flat surfaces having no crevices or other properties that may alter the amount of contamination retained.

To calculate the residual uranium contamination levels, or inventory, in equipment and dry and wet processing systems, a correction factor is applied to account for the potential order of magnitude differences in contamination on different types of surfaces. These correction factors are derived for each piece of equipment or system cleaned, using engineering judgement based on the process operations performed, the system design, and the physiochemical forms of the uranium present. The range of the correction factors is 0.5 to 20. The estimated total surface area to be cleaned and the grams of uranium per unit area are multiplied by the correction factor to estimate the residual radionuclide inventory.

Key assumptions used to estimate the residual radionuclide inventories in various plant process areas are:

- Residual uranium contamination levels are 11 g U/m^2 for stainless steel and 1 g U/m^2 for acid-flushed stainless steel. These are values for flat surfaces. For areas where corrosion has occurred, correction factors up to 20 apply.
- Major contamination incidents or episodes are reasonably well-cleaned up immediately following the event.
- The final inventory cleanout, to account for the uranium inventory, involves routine nondestructive decontamination operations. As a result, contamination levels are generally expected to be relatively low throughout the plant. The quantity and mixture of radioactivity left in the plant after final inventory cleanout is an accumulation of fairly difficult-to-clean contamination that has accumulated at the rate of 1/40 per year during the assumed 40 years of plant operations.

C.2 REFERENCE RADIONUCLIDE MIXTURE FOR THE UF_6 CONVERSION PLANT

Table C.2-1 presents the specific radioactivity levels for the radionuclide mixture present at the UF_6 conversion plant at the time of plant shutdown and 5, 10, 30, and 100 years after plant shutdown. This mixture is used to estimate public radiation doses from decommissioning operations and to determine acceptable

residual contamination levels in the released decommissioned UF₆ conversion plant. The radionuclides in Table C.2-1 are the radionuclides in natural uranium mixtures and the isotopic ratios given in Reference 2.

TABLE C.2-1. Specific Radioactivity of Residual Uranium in the UF₆ Plant at the Time of Plant Shutdown and at Various Times After Shutdown

Radionuclide	Specific Activity, $\mu\text{Ci/g}$ of Mixture ^(a)				
	Shutdown	5-Year	10-Year	30-Year	100-Year
²³⁰ Th	2.9×10^{-3}	3.7×10^{-3}	4.4×10^{-3}	7.2×10^{-3}	1.7×10^{-2}
²³¹ Th	2.2×10^{-2}	2.2×10^{-2}	2.2×10^{-2}	2.2×10^{-2}	2.2×10^{-2}
²³⁴ Th	2.5×10^{-1}	2.5×10^{-1}	2.5×10^{-1}	2.5×10^{-1}	2.5×10^{-1}
²³¹ Pa	3.0×10^{-5}	3.7×10^{-5}	4.4×10^{-3}	7.4×10^{-3}	1.8×10^{-2}
^{234m} Pa	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}
²³⁴ Pa	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}	3.3×10^{-4}
²²⁶ Ra	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}	3.0×10^{-4}
²³⁴ U	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}
²³⁵ U	1.5×10^{-2}	1.5×10^{-2}	1.5×10^{-2}	1.5×10^{-2}	1.5×10^{-2}
²³⁸ U	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}	3.3×10^{-1}
Totals	1.3	1.3	1.3	1.3	1.3

(a) Based on the ratio of radionuclides in uranium and the isotopic ratios given in Reference 2.

C.3 ESTIMATES OF RESIDUAL RADIOACTIVITY LEVELS IN THE UF₆ PLANT

Table C.3-1 contains the estimates of the amounts of uranium in the facility after the final inventory cleanout (final inventory cleanout terminates plant operations), and after chemical decontamination operations (part of decommissioning).

After final inventory cleanout, the uranium inventory remaining is expected to be difficult-to-remove material. Final inventory cleanout operations are assumed to be the same as those conducted between processing campaigns. Final inventory cleanout operations and decommissioning decontamination operations are described in Appendix F, Section F.1.

As shown in Table C.3-1, the inventory of uranium in the process buildings and equipment, after final inventory cleanout, is estimated to be approximately 9,200 kg, based on the above method of calculating the inventory. By combining the decontamination factor of 10 for flat polished stainless steel surfaces and correcting for crevices, the overall chemical decontamination factor is reduced to approximately 2 before conventional chemical decontamination techniques become non-beneficial.⁽¹⁾ Thus, after chemical decontamination and before disassembly of equipment, the building inventory is reduced to approximately 4,100 kg of uranium. Further reductions in the building uranium inventory occur as the equipment is disassembled and either packaged for disposal at a federal repository or further decontaminated.

It should be emphasized that these estimates, based on the assumptions used in this study and for any specific case, are highly dependent upon the operating philosophy of the plant owner. The estimates of residual uranium are believed to be reasonable for the assumptions used.⁽³⁾ The strict accounting methods required for special nuclear materials are believed to effectively limit the material accumulations to the ranges estimated.

TABLE C.3-1. Estimated Inventory of Uranium in Process Areas of the Reference UF₆ Plant

Location	Equipment	Estimated Contaminated Surface Area (m ²)	Process Operation	Form of Uranium Contamination	Approximate Chemical Decontamination Correction Factor	Estimated Uranium Concentration Level After Final Operational Cleanout (g/m ²)	Estimated kg of Uranium (kg of U)	
							After Operational Cleanout	After Decontamination
Sampling Station	Drum Elevator	37.0	Sampling	U ₃ O ₈	20	20	5	2.5
	Yellowcake Receiving Bin	37.5	Sampling	U ₃ O ₈	20	20	50	25
	Primary Sampler I	8.1	Sampling	U ₃ O ₈	20	20	10	5
	Primary Sampler II	8.1	Sampling	U ₃ O ₈	20	20	10	5
	Secondary Sampler	7.5	Sampling	U ₃ O ₈	20	20	10	5
	Yellowcake Transfer Screw	12.6	Sampling	U ₃ O ₈	20	20	40	20
	Bucket Elevator	18.4	Sampling	U ₃ O ₈	20	20	25	12.5
	Digest Feed Bin	8.8	Sampling	U ₃ O ₈	20	20	50	25
	Piping	20.0	Sampling	U ₃ O ₈	1	20	5	2.5
	Hood and Ductwork	12.3	Sampling	U ₃ O ₈	20	20	10	5
							215 Subtotal	107.5 Subtotal
Solvent Extraction Building	Digestors	132.0	Solvent Extraction	U ₃ O ₈	20	1000	700	350
	Adjustment Tanks	90.0	Solvent Extraction	U ₃ O ₈	20	1000	100	50
	Pumper Decanters	42.0	Solvent Extraction	U ₃ O ₈	20	100	50	25
	Scrub Decanter	12.9	Solvent Extraction	U ₃ O ₈	20	20	60	30

TABLE C.3-1. (Contd)

Location	Equipment	Estimated Contaminated Surface Area (m ²)	Process Operation	Form of Uranium Contamination	Approximate Chemical Decontamination Correction Factor	Estimated Uranium Concentration Level After Final Operational Cleanup (g/m ²)	Estimated kg of Uranium (kg of U)	
							After Operational Cleanup	After Decontamination
Solvent Extraction Building (Contd)	Re-extraction Pulse Column	7.5	Solvent Extraction	U ₃ O ₈	20	1000	90	45
	Extraction Waste Holding Tanks	108.0	Solvent Extraction	U ₃ O ₈	20	100	100	50
	Piping	19.2	Solvent Extraction	U ₃ O ₈	10	100	10	5
							1110 Subtotal	550 Subtotal
Concentration Area	Evaporator	24.0	Concentration	U ₃ O ₈	20	1000	1000	500
	Boil-down Tanks	126.0	Concentration	U ₃ O ₈	20	1000	500	250
	Filter	3.5	Concentration	U ₃ O ₈	20	1000	100	50
	Piping	10.2	Concentration	U ₃ O ₈	10	100	5	2.5
	Hood and Ductwork	18.9	Concentration	U ₃ O ₈	10	20	5	2.5
							1710 Subtotal	855 Subtotal
Denitration Area	Denitrator	81.6	Denitration	U ₃ O ₈	20	1000	300	150
	Denitrator Scrubber Cooler	24.0	Denitration	U ₃ O ₈	20	1000	600	300
	Cooling Screw	16.2	Denitration	U ₃ O ₈	20	20	20	10
	UO ₃ Vertical Elevator	12.9	Denitration	U ₃ O ₈	20	20	30	15
	Pulverizer and Surge Bin	57.0	Denitration	U ₃ O ₈	2	1000	150	75

TABLE C.3-1. (Contd)

Location	Equipment	Estimated Contaminated Surface Area (m ²)	Process Operation	Form of Uranium Contamination	Approximate Chemical Decontamination Correction Factor	Estimated Uranium Concentration Level After Final Operational Cleanout (g/m ²)	Estimated kg of Uranium (kg of U) After	
							Operational Cleanout	Decontamination
Denitration Area (Contd)	Piping	60.0	Denitration	U ₃ O ₈	10	1	5	2.5
	Hoods and Ductwork	150.0	Denitration	U ₃ O ₈	3	10	5	2.5
							1110 Subtotal	555 Subtotal
Reduction Area	UO ₃ Storage Bin	205.2	Reduction	UO ₃	10	20	50	20
	Reduction Fluid Bed	102.0	Reduction	UO ₃	20	1000	300	150
	Piping	60.0	Reduction	UO ₃	10	1	40	20
	Hoods and Ductwork	120.0	Reduction	UO ₃	10	10	20	10
							400 Subtotal	200 Subtotal
Hydrofluorination Area	UO ₂ Filter Bin	73.5	Hydrofluorination	UF ₄	10	1000	100	50
	1st Stage Hydrofluorination	48.9	Hydrofluorination	UF ₄	20	1000	800	600
	2nd Stage Hydrofluorination Fluid Bed	48.9	Hydrofluorination	UF ₄	20	1000	600	300
	UF ₄ Seal Hopper	32.4	Hydrofluorination	UF ₄	20	20	40	20
	UF ₄ - UO ₄ Intermediate Seal Hopper	32.4	Hydrofluorination	UF ₄	20	20	40	20

Table C.3-1. (Contd)

Location	Equipment	Estimated Contaminated Surface Area (m ²)	Process Operation	Form of Uranium Contamination	Approximate Chemical Decontamination Correction Factor	Estimated Uranium Concentration Level After Final Operational Cleanout (g/m ²)	Estimated kg of Uranium (kg of U)	
							After Operational Cleanout	After Decontamination
Hydrofluorination Area (Contd)	UF ₄ Purge Conveyor	15.6	Hydrofluorination	UF ₄	20	20	20	10
	Redler Conveyor 509	15.2	Hydrofluorination	UF ₄	20	20	20	10
	UF ₄ Storage Hopper	32.4	Hydrofluorination	UF ₄	20	1000	400	200
	Redler Conveyor 510	15.6	Hydrofluorination	UF ₄	20	100	80	60
	Piping	13.8	Hydrofluorination	UF ₄	10	1	5	2
							2100 Subtotal	1050 Subtotal
Fluorination Area	Primary Fluorination	108.3	Fluorination	UF ₆	20	1000	800	400
	Redler Conveyor 511	7.5	Fluorination	UF ₆	20	20	20	10
	Primary Filter	36.6	Fluorination	UF ₆	20	100	60	30
	Primary Cold Trap	181.5	Fluorination	UF ₆	20	1000	110	55
	Product Cylinder	103.2	Fluorination	UF ₆	20	100	70	35
	Backup Filter	7.2	Fluorination	UF ₆	20	100	80	40
	Cleanup Reactor Feed Hopper	15.6	Fluorination	UF ₆	20	1000	100	50
	Cleanup Reactor	43.8	Fluorination	UF ₆	20	1000	600	300
	Redler Conveyor 512	7.5	Fluorination	UF ₆	10	20	20	10

TABLE C.3-1. (Contd)

Location	Equipment	Estimated Contaminated Surface Area (m ²)	Process Operation	Form of Uranium Contamination	Approximate Chemical Decontamination Correction Factor	Estimated Uranium Concentration Level After Final Operational Cleanout (g/m ²)	Estimated kg of Uranium (kg of U)	
							Estimated kg of Uranium (kg of U) After Operational Cleanout	After Decontamination
Fluorination Area (Contd)	Backup Filter	7.2	Fluorination	UF ₆	20	1000	500	250
	Secondary Cold Trap	77.1	Fluorination	UF ₆	20	1000	200	100
	Piping	66.0	Fluorination	UF ₆	10	20	5	2.5
	Hoods and Ductwork	19.4	Fluorination	UF ₆	10	20	5	2.5
							2600 Subtotal	1300 Subtotal
Fluorine Generation Area	Waste Gas Burner	36.0	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
	Cells	160.0	Fluorine Generation	UF ₄	1	1	0.16	0.08
	Fluorine Surge Tank	36.0	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
	Fluorine Filter	16.8	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
	Fluorine - HF Condenser	37.8	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
	HF Backup Filter	25.2	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
	Hydrogen Surge Tank	49.2	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
	Fluorine Filter	21.6	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
	Aqueous HF Storage Tank	28.5	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
	Ammonia Disassociator	7.2	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
	Piping	25.2	Fluorine Generation	UF ₄	1	1	>0.01	>0.005
							0.2 Subtotal	0.1 Subtotal
							9200 Total	4100 Total

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APPENDIX D

PAYMENTS NEEDED TO FINANCE DECOMMISSIONING

The purpose of this appendix is to illustrate the payment schedules required to provide funds for decommissioning of the reference UF₆ conversion plant. Payments are computed for the prepayment and sinking fund financial assurance options discussed in Section 6. The annual cost of the third financial assurance option, the use of a surety bond that guarantees decommissioning performance, is likely to be between 1 and 2% of the guaranteed amount, as discussed in Section 6.

D.1 FORMULAS FOR COMPUTATION OF PAYMENTS

For the prepayment financial assurance option, an amount of money is deposited into a fund at the time of plant startup that, when compounded at an annual fractional rate of return, r , will provide the required funding for decommissioning. If S is the cost of decommissioning estimated in dollars of the year of plant startup, the required prepayment, P , is obtained from:

$$S (1+j)^n = P (1+r)^n \quad (D.1)$$

Where n is the anticipated plant operating life, and j is the anticipated annual fractional cost escalation rate.

For the sinking fund financial assurance option, the money deposited into the fund plus interest on the money compounded annually over the plant operating life must equal the anticipated decommissioning cost. If the initial annual payment is R_0 , the value of the sinking fund after n years is:

$$S (1+j)^n = R_0 \sum_{a=1}^n (1+j)^{a-1} (1+r)^{n-a} \quad (D.2)$$

The initial payment, R_0 , is given by:

$$R_0 = \left\{ \frac{S (1 + j)^n}{\sum_{a=1}^n (1 + j)^{a-1} (1 + r)^{n-a}} \right\} \quad (D.3)$$

and the annual payment in the k th year is

$$R_k = R_0 (1 + j)^{k-1} \quad (D.4)$$

The estimated costs of decommissioning (in 1981 dollars) as given in Section 10 are: for DECON, \$4.85 million; for SAFSTOR, \$0.94 million initially, \$0.22 million annually for safe storage, and a final decontamination cost of \$5.30 million.

D.2 COMPUTATION OF PAYMENTS

For purpose of making illustrative payment calculations, several simplifying assumptions are made. It is assumed that the plant initiates production on January 1, 1981 and that it permanently ceases production on December 31, 2020 after a 40-year operating life. It is further assumed that decommissioning costs escalate in nominal (i.e., current) dollars at the rate of 10% per year and that the trustee realizes an annual return of 10% by investing in trust funds. In actual practice it can be expected that the trustee will be able to earn 1 to 3% more than the general inflation rate by investing in debt instruments whose interest is subject to federal taxes.⁽¹⁾ There is also no clear reason to expect decommissioning costs to escalate at a rate that exceeds the general inflation rate. However, a portion of the incremental return above decommissioning cost escalation will be needed for administrative expenses and may also be needed for payment of taxes (see Section 6). Consequently, no incremental return is assumed.

The prepayment financial assurance alternative is the most straightforward. Under DECON, payment of \$4.85 million into a trust fund prior to startup will provide sufficient funds if costs escalate at 10% and the trustee earns 10% annually after expenses.

Using the sinking fund option, with a 40-year plant operating life followed by DECON, the initial year's payment (from Equation D.3) is \$0.132 million in 1981 dollars, assuming an estimated DECON cost of \$4.85 million in 1981 dollars. Subsequent annual payments are escalated at a 10% annual rate until the final payment is \$5.43 million in dollars of that year. The summation of all annual payments is equal to the cost of DECON in escalated dollars, \$217.2 million. Obviously, if the SAFSTOR alternative is selected this value will be larger due to the larger total cost of SAFSTOR and the additional escalation during the safe storage period.

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APPENDIX E

RADIATION DOSE MODELS AND CALCULATED MAXIMUM ANNUAL DOSES

This appendix contains information to support the discussions in Sections 8 and 11. Terminology, models, equations and parameters used in radiation dose calculations, as well as tabulated maximum annual doses to the maximum-exposed individual, are presented. The tabulated maximum annual doses are based on the reference radionuclide inventory detailed in Appendices B and C. These doses are used to determine the acceptable contamination levels for public release of the decommissioned reference UF_6 conversion plant. It should be noted that the occupational radiation doses during decommissioning operations are not calculated using the dose models presented in this appendix.

E.1 RADIATION DOSE TERMINOLOGY AND DEFINITIONS

The following terminology and definitions apply for the radiation dose calculations made for this study:

Routine Release. An atmospheric release of radionuclides from routine decommissioning tasks that is assumed to occur at a constant rate for 1 year.

Acute Release. A short-duration (usually less than 8 hours) release of radionuclides. The accident analysis, discussed in Section 11 and Appendix I, is based on acute releases.

Organs of Reference. The organs of the human body for which radiation doses are calculated. For this study, the organs of reference are the total body, lungs, bone, and lower large intestine (LLI) of the GI-tract. The total body is the head and trunk of the human body and includes active blood-forming organs, eye lenses, and gonads.

Exposure Pathways. The potential routes by which people may be exposed to radionuclides or radiation. Radiation exposure pathways in the environment that are considered in this study are: external exposure to contamination

deposited on the ground, ingestion of food products containing radionuclides, and inhalation of airborne radionuclides. Radiation exposure pathways inside the UF_6 conversion plant are: external exposure from contaminated room surfaces and inhalation of airborne radionuclides. External exposure from airborne radionuclides (air submersion) is not considered, since previous decommissioning studies have shown this exposure pathway to be insignificant compared to the others.⁽¹⁻³⁾

Maximum-Exposed Individual. The individual who receives the maximum radiation dose to an organ of reference. The maximum-exposed individual is assumed to reside at the location of the highest airborne radionuclide concentration. Maximized exposure pathway parameters are used.

Radiation Dose Equivalent. A quantity used to express the absorbed dose from all radiations on a common scale in units of rem. It is defined as the product of the absorbed dose in rads, a quality factor intended to allow for the effect of the microscopic distribution of absorbed energy, and other modifying factors as defined by the International Commission on Radiological Protection (ICRP).

Committed Radiation Dose Equivalent. The dose equivalent accumulated over a specific number of years from radionuclides deposited in the body from the various radiation pathways during a given year.

Collective Dose. The summation of radiation dose equivalents received by all individuals in the population of concern. It is calculated by multiplying the dose to the average individual by the population distribution for the site listed in Appendix B. Average individual exposure parameters values are used for the exposure pathways.

First-Year Dose. The radiation dose equivalent accumulated during the year of the release period.

Fifty-Year Committed Dose Equivalent. The committed radiation dose equivalent for 50 years following a 1-year exposure to radionuclides.

Annual Dose. The radiation dose equivalent calculated during any year following the start of continuous exposure. It is the sum of the dose received

by an organ of reference during the year of interest from all exposure pathways and the dose received that year from radionuclides deposited in the organ of reference during previous years of exposure.

Maximum Annual Dose. The largest of the 50 annual doses calculated to occur during the 50 years following the start of continuous exposure.

Biological Half-life. The time required for the body to eliminate half of any substance by regular body elimination processes.

Solubility Classes W and Y. Radionuclides are classified according to the rate of which they are dissolved into body fluids in the lung after inhalation. The two classifications used in this appendix are: Class W materials, with a maximum biological half-life ranging from a few days to a few months; and Class Y materials, with a maximum biological half-life of from 6 months to a few years.⁽⁴⁾

E.2 GENERAL MODELS FOR ESTIMATING RADIATION DOSE

The methodology and parameters necessary to calculate the first-year dose or the fifty-year committed dose equivalent are presented in this section. The fundamental relationship for calculating radiation doses to people from any radionuclide or exposure pathway is given in Equation E.1.⁽⁵⁾

$$R_{ipr} = C_{ip} U_p D_{ipr} \quad (E.1)$$

where:

- R_{ipr} • first-year dose or fifty-year committed dose equivalent from radionuclide i via exposure pathway p to organ r
- C_{ip} • concentration of radionuclide i in the media of exposure pathway p ; for calculations involving airborne radionuclides, C_{ip} is replaced with the term $\bar{X}_i$, which represents the average airborne concentration of radionuclide i
- U_p • Usage parameter (exposure rate or intake rate) associated with exposure pathway p

- D_{ipr} • Radiation dose equivalent factor or fifty-year committed radiation dose equivalent factor (for a given radionuclide i , exposure pathway p , and organ r) that converts the concentration and usage parameters to the first-year dose or fifty-year committed dose equivalent.

Each of the terms in the general dose equation (Equation E.1) are discussed in detail below:

C_{ip} . An analysis of the radiation doses from separate exposure pathways requires a calculation of the radionuclide concentrations associated with each pathway. An external exposure analysis requires a soil concentration or surface contamination level. The radionuclide concentration in food crops grown in contaminated soil is directly related to the radionuclide concentration in the soil plow layer (a depth of 0.15 m). Exposure pathways that require the concentration of radionuclides in air from airborne radionuclide releases use dilution factors discussed in the site description in Appendix B. They are: $7.2 \times 10^{-8} \text{ sec/m}^3$ for the maximum-exposed individual located 1 km from the site and $2.3 \times 10^{-11} \text{ sec/m}^3$ for the average member of the population residing within 80 km of the site.

U_p . Each radiation exposure pathway has an associated exposure rate or intake rate. Parameters used to calculate radiation doses from the consumption of food products grown in contaminated soil are listed in Table E.2-1. For the inhalation pathway, the intake rate (or ventilation rate) is assumed to be $2.7 \times 10^{-4} \text{ m}^3/\text{sec}$ for members of the public and $3.3 \times 10^{-4} \text{ m}^3/\text{sec}$ for workers with a 40-hour work week.

D_{ipr} . The radiation dose equivalent factors and the fifty-year committed radiation dose equivalent factors that are used in this study for internal exposure pathways are obtained from several sources. Those for ingestion are obtained primarily from References 5, 7, 8, 9, and 10, which are derived from information given by the ICRP for the body burden and the maximum permissible concentration of each radionuclide.⁽¹¹⁾ Ingestion factors not obtainable from the references are calculated by the method given in Reference 10. In this

TABLE E.2-1. Parameters Used for Calculation of Radiation Doses from Consumption of Foods^(a)

Food	Growing Period (days)	Yield (kg/m ²)	Holdup (days) ^(b)	Up, Maximum Individual Consumption (kg/year) ^(c)	Up, Average Individual Consumption (kg/yr)
Leafy Vegetables	90	1.5	1	30	15
Other above-Ground Vegetables	60	0.7	1	30	15
Potatoes	90	4.0	10	110	71
Other Root Vegetables	90	5.0	1	72	47
Berries	60	2.7	1	30	6
Melons	90	0.8	1	40	8
Orchard Fruit	90	1.7	10	265	50
Wheat	90	0.72	10	80	72
Other Grain	90	1.7	1	8	8
Eggs	90	0.84 ^(d)	2	30	20
Milk	30	1.3 ^(d)	2	274 ^(e)	230 ^(e)
Beef	90	0.84 ^(d)	15	40	40
Pork	90	0.84 ^(d)	15	40	30
Poultry	90	0.84 ^(d)	2	18	8

(a) Based on information in Reference 6.

(b) Time between harvest and consumption.

(c) Only that fraction of the diet grown locally (and, therefore, potentially contaminated) is listed.

(d) Yield of animal feeds (i.e., grain or pasture grass).

(e) Units of ℓ /year.

method, effective decay energies for the radionuclides are calculated from the ICRP model, which assumes that all of a given radionuclide is in the center of a spherical organ with an appropriate effective radius. Dose factors for inhalation are calculated using the computer code DACRIN.⁽¹²⁾ This code incorporates the ICRP Task Group Lung Model to calculate the dose to the lung and other organs of reference. A particle size of 1 micrometer in diameter is assumed. Radiation dose equivalent factors for internal pathways have units of mrem per year per pCi taken into the body during that same year. Committed radiation dose equivalent factors have units of mrem/50-yr per pCi taken into the body in the first year. External exposure can arise from either contaminated land or contaminated building

surfaces. Contaminated ground resulting from deposition of airborne radionuclides is assumed to be a large, uniform, thin (approximately 20 mm deep) plane source for the purpose of calculating doses from external exposure. Dose rate factors for converting surface contamination to gamma dose, one meter above a uniformly contaminated plane have been estimated.^(5,9) Dose rate factors for exposure to contaminated soil have units of mrem/hr per pCi/m² of surface. These dose rate factors are adjusted for a finite plane source by multiplying by 0.3 when determining acceptable residual contamination levels inside the UF₆ conversion plant.

Discussion of the various radiation exposure pathways considered in this study, together with appropriate modifications to the general dose equation and its parameters, are given in the following sections.

E.2.1 External Exposure

Calculation of the radiation dose resulting from external total body exposure to surface contamination is done with no modification to Equation E.1. Soil concentrations are calculated using a simple cloud-depletion model, as is described for the food ingestion pathway analysis in the next section. For routine releases from decommissioning operations, the maximum-exposed individual is assumed to be exposed to contaminated soil for 12 hours a day for a full year. The average individual is assumed to be exposed for 6 hours a day for a full year. The fifty-year committed dose equivalent for external exposure to surface contamination is equal to the first-year dose, since there is no internal radionuclide deposition.

E.2.2 Ingestion

The radiation dose equivalent or committed radiation dose equivalent to a person consuming vegetation grown on the contaminated site is given in Equation E.2

$$R_{vr} = \sum_{i=1}^n C_{iv} U_v D_{ir} \quad (E.2)$$

Similarly, the radiation dose equivalent or committed radiation dose equivalent to a person consuming a contaminated animal product is given in Equation E.3.

$$R_{ar} = \sum_{i=1}^n C_{ia} U_a D_{ir} \quad (E.3)$$

Where:

- R_{vr}, R_{ar} • radiation dose equivalent or committed radiation dose equivalent from vegetation (v) or animal (a) food product pathways, mrem or mrem/50 yr
- C_{iv}, C_{ia} • concentration of radionuclide i in vegetation (v) or animal product (a), pCi/kg or pCi/l
- U_v, U_a • annual consumption of contaminated vegetable (v) or animal products (a), kg or l
- D_{ir} • radiation dose equivalent factor or committed radiation dose equivalent factor, mrem/pCi or mrem/50 yr per pCi.

Specific values of U_v and U_a are shown in Table E.2-1.

E.2.2.1 Deposition on Food Products or Ground Surfaces

Equation E.4 is used to describe the deposition of airborne particulate radionuclides directly on food products or ground surfaces.

$$d_i = 86,400 \bar{X}_i V_{di} \quad (E.4)$$

where:

- d_i • deposition rate of flux of radionuclide i, pCi/(m²-day)
- 86,400 • dimensional conversion factor, seconds/day
- V_{di} • deposition velocity of radionuclide i, m/second
- $\bar{X}_i$ • annual average air concentration from airborne releases of radionuclide i, pCi/m³.

Specific values of the deposition velocity, V_{di} , are obtained from Reference 13.

E.2.2.2 Concentration in Vegetation

The concentration of radioactive material in vegetation resulting from direct deposition onto plant foliage and uptake of prior depositions from the soil is given in Equation E.5.

$$C_{iv} = d_i \left[\frac{r T_v (1 - e^{-\lambda_{Ei} t_e})}{Y_v \lambda_{Ei}} + \frac{B_{vi} (1 - e^{-\lambda_i t_b})}{P \lambda_i} \right] e^{-\lambda_i t_h} \quad (E.5)$$

where:

- C_{iv} • concentration of radionuclide i in the edible portion of the vegetation (v), pCi/kg
- d_i • deposition rate or flux of radionuclide i , pCi/(m²-day)
- r • fraction of deposition retained on the vegetation (dimensionless), assumed to be 0.25 in this study
- T_v • factor for translocation of externally deposited radionuclides to the edible parts of the vegetation (dimensionless). For simplicity it is assumed in this study to be independent of the radionuclide and is 1.0 for leafy vegetables and fresh forage, and 0.1 for all other produce, including grain. (Reference 8 lists values of this parameter for various radionuclides.)
- λ_i • radiological decay constant for radionuclide i , day⁻¹
- λ_{Ei} • effective removal constant for radionuclide i , day⁻¹; $\lambda_{Ei} = \lambda_i + \lambda_w$, where λ_i is as defined above and λ_w is the weathering removal constant for vegetation, day⁻¹, assumed to be $0.693/14 = 5 \times 10^{-2}$ day⁻¹
- Y_v • yield of vegetation (v), kg (wet weight)/m²
- B_{vi} • concentration factor for uptake of radionuclide i from the soil in vegetation (v), pCi/kg (net weight) per pCi/kg (dry soil)

- t_b • time for buildup of radionuclides in the soil, days; assumed in this study to be 40 years of routine UF_6 conversion plant operation at 75% efficiency
- t_e • time of exposure of above-ground vegetation to contamination during growing season, days
- t_h • holdup time between harvest and food consumption, days
- P • soil "surface density," $kg (dry\ soil)/m^2$; a value of $224\ kg/m^2$ is used assuming the contaminated ground is plowed to a depth of 0.15 m after decommissioning.

The first term inside the brackets of Equation E.5 relates to the concentration resulting from direct deposition on foliage during the growing season. The second term relates to the uptake from the soil and reflects the deposition throughout the UF_6 conversion plant operating life. Specific values for the parameters in Equation E.5 are obtained from References 6, 7, and 13.

E.2.2.3 Concentration in Animal Products

The radionuclide concentration in animal products such as meat, milk, or eggs is dependent on the amount of contaminated forage or feed eaten by the animal. This concentration is described by Equation E.6.

$$C_{ia} = S_{ia} C_{iF} Q_F \quad (E.6)$$

where:

- C_{ia} • concentration of radionuclide i in animal products (a), pCi/kg or pCi/l
- S_{ia} • transfer coefficient of radionuclide i to the edible portion of animal product (a) from daily intake by the animal, pCi/l (milk) per pCi/day or pCi/kg (animal product) per pCi/day.
- C_{iF} • concentration of radionuclide i in feed or forage F , pCi/kg; calculated from Equation E.5
- Q_F • animal consumption rate of contaminated feed or forage F , kg/day.

Specific values for the parameters used in Equation E.6 are obtained from References 6, 7, and 13. Where data are lacking, comparisons are made with biological data from chemically similar elements.

E.2.3 Inhalation

The radiation dose equivalent or the committed radiation dose equivalent via inhalation is calculated using Equation E.7.

$$R_{ir} = \sum_{i=1}^n \bar{X}_i B T D_{ir} \quad (E.7)$$

where:

- R_{ir} • radiation dose equivalent or committed radiation dose equivalent to organ r from radionuclide i , mrem or mrem/50 yr
- $\bar{X}_i$ • annual average airborne concentration of radionuclide i , pCi/m³
- B • ventilation rate of exposed individual:
for a chronic release, $B = 2.7 \times 10^{-4}$ m³/sec;
for an acute release, $B = 3.3 \times 10^{-4}$ m³/sec⁽¹¹⁾
- T • time of exposure to the airborne radionuclide concentration, sec
- D_{ir} • radiation dose equivalent factor or committed radiation dose equivalent factor, mrem/pCi or mrem/50 yr per pCi.

The inhalation dose model presented in Equation E.7 is consistent with the ICRP Task Group Lung Model.

E.3 MODEL FOR ESTIMATING ANNUAL DOSE

The determination of acceptable residual contamination levels at the decommissioned reference UF₆ conversion plant (discussed in Section 8) is made by direct comparisons of calculated annual doses with an assumed annual dose limit. The annual dose to internal body organs from internally deposited radionuclides tends to increase for a time (after the start of continuous exposure

to a radioactively decaying source) until a maximum is reached. The annual dose then tends to decrease with time because of radioactive decay, decrease in the radionuclide concentrations of specific exposure pathways, and biological elimination of radionuclides deposited in the organ. For continuous exposure to a radioactively decaying source, the year in which the annual dose is a maximum depends on the chemical and physical characteristics of the radionuclides in the source, the organ of reference, and the exposure pathway.

The calculated first-year dose from ingested or inhaled radionuclides will most likely underestimate the maximum annual dose during continuous exposure, and therefore does not provide the best comparison to annual dose limit. Nor is it appropriate to compare a fifty-year dose (accounting for 50 years of committed radiation dose equivalent) to an annual dose limit. Therefore, an extension of the general dose relationship given in Equation E.1 is required to calculate annual doses for each year of a continuous exposure period. The maximum annual dose occurring in the continuous exposure period is then compared to the annual dose limit to determine acceptable residual contamination levels.

Calculation of the annual dose to an organ of reference requires the dose equivalent from exposure during the year of interest. It also requires a detailed accounting of the committed dose equivalents to that organ from radionuclides deposited during previous years of intake. The general expression for annual dose calculations is deduced by inspecting the annual dose equations for the first 3 years of continuous exposure. The annual dose for the first year to an organ of reference is simply the summation of the radiation dose equivalents from all internal and external exposure pathways. For the second year, the annual dose is calculated by the following expression:

$$A_2 = R_2^* + (R_{1,2} - R_{1,1}) \quad (\text{E.8a})$$

where:

- A_2 • the annual dose during the second year from all exposure pathways to the organ of the reference, mrem

- R_2^* • the radiation dose equivalent in the second year to the organ of reference from all internal and external exposure pathways from intake and exposure in the second year of continuous exposure, mrem
- $R_{1,2}$ • the committed dose equivalent to the organ of reference for the first two years from radionuclides internally deposited during intake from exposure pathways in the first year, mrem
- $R_{1,1}$ • the radiation dose equivalent to the organ of reference for the first year from radionuclides internally deposited during intake from exposure pathways in the first year (no external component to the dose equivalent), mrem.

The second-year annual dose to an organ of reference (A_2 in Equation E.8a) is the summation of the radiation dose equivalents from all exposure pathways during the second year and the dose equivalent resulting from the radionuclides internally deposited in that organ during the first year. The term in parentheses in Equation E.8a is the expression for the dose equivalent to the organ of reference from radionuclides deposited in that organ in the first year. It is found by subtracting the first-year dose equivalent from internally deposited radionuclides from the two-year committed dose equivalent.

The mathematical expression for the annual dose to an organ of reference in the third year of continuous exposure is shown in Equation E.8b.

$$A_3 = R_3^* + (R_{1,3} - R_{1,2}) + (R_{2,2} - R_{2,1}) \quad (\text{E.8b})$$

where:

- A_3 • the annual dose during the third year from all exposure pathways to the organ of reference, mrem
- R_3^* • the radiation dose equivalent in the third year to the organ of reference from all internal and external exposure pathways from intake and exposure in the third year of continuous exposure, mrem.

The terms $R_{1,3}$, $R_{1,2}$, $R_{2,2}$, and $R_{2,1}$ are of similar form, each having two subscripts. The first subscript defines the year of intake or exposure after the start of continuous exposure, and the second defines the number of years used in calculating the committed dose equivalent.

The quantity in the first parentheses in Equation E.8b is the dose equivalent to the organ of reference in the third year from radionuclides deposited in the first year of continuous exposure (i.e., the difference between the 3-year committed dose equivalent and the 2-year committed dose equivalent). The quantity in the second parentheses is the dose equivalent in the third year to the organ of reference from radionuclides deposited in the second year of continuous exposure (i.e., the difference between the 2-year committed dose equivalent and the first-year committed dose equivalent).

The general expression for calculating the annual dose to an organ of reference during any year after the start of continuous exposure can be expressed as:

$$A_t = R_t^* + \sum_{i=1}^{t-1} [R_{i,(t-i+1)} - R_{i,(t-i)}] \quad (E.9)$$

where:

- A_t • the annual dose during the year t from all exposure pathways to the organ of reference, mrem
- R_t^* • the radiation dose equivalent in year t to the organ of reference from all internal and external exposure pathways from intake and exposure in the year t , mrem.

The summation term represents the dose equivalent delivered to the organ of reference in year t from radionuclides deposited in the organ from intake in all previous years since the start of continuous exposure.

The summation term in Equation E.9 is valid only for integer values of t greater than 1. For values of t equal to or less than 1, the subscripts define a non-real case, and the summation term is set equal to zero.

The annual dose, A_t , to the organ of reference is calculated for each value of t from 1 to 50, and the maximum annual dose is determined by inspection. The radionuclide inventories are adjusted for radionuclide decay and daughter-product buildup during the calculation.

Resuspended surface or ground contamination is assumed to be the only source of airborne radionuclides following decommissioning of the reference UF_6 conversion plant. Airborne radionuclide concentrations for the decommissioned UF_6 conversion plant are calculated using a resuspension analysis. The average airborne concentration is the product of a resuspension factor and the surface contamination level as shown in Equation E.10.

$$\bar{X} = S_f S_A \quad (E.10)$$

where:

- $\bar{X}$ • average airborne concentration, Ci/m³
- S_f • resuspension factor, m⁻¹
- S_A • surface radioactivity deposition, Ci/m².

Inside the decommissioned UF_6 conversion plant, airborne concentrations are calculated using a constant resuspension factor of $5 \times 10^{-6} \text{ m}^{-1}$. This value is one-tenth of the value suggested by the IAEA for surfaces with removable contamination.⁽¹⁵⁾ This lower value is assumed since unrestricted release of nuclear facilities should not be contemplated with readily removable contamination remaining on accessible surfaces. Actual facility surface resuspension factors may be considerably lower than this value.

Airborne radionuclide concentrations on the reference site are calculated using a time dependent suspension factor to account for the environmental "aging" of radionuclides. The relationship for this factor is given in Equation E.11.⁽¹⁶⁾

$$S_f = \left(10^{-4} e^{-\lambda \sqrt{t}} \right) + 10^{-9} \quad (E.11)$$

where:

- S_f • resuspension factor, m^{-1}
- 10^{-4} • resuspension factor at time $t = 0$, m^{-1}
- λ • effective decay constant controlling the availability of material for resuspension, $0.15 \text{ day}^{-1/2}$
- t • time after deposition, days
- 10^{-9} • resuspension factor after 17 years, m^{-1} .

The second term in Equation E.11 (10^{-9}) is added based on the assumption that there is no further measureable decrease in the resuspension factor process after 17 years, which is the longest period for which data are available. This is deemed appropriate since this model is derived empirically to simulate experimental measurements and contains no fundamental understanding of the resuspension process.⁽¹⁶⁾ The time dependence of the site resuspension factor is illustrated in Figure E.3-1.

FIGURE E.3-1. Time-Dependent Surface Resuspension Factor for the Site

After decommissioning, unrestricted use of the site will allow activities such as unlimited farming. Therefore, the amount of surface contamination available for resuspension must be defined. For the calculation of the maximum annual doses, it is assumed that the site surface is plowed after decommissioning to a depth of 0.15 m, and the original surface contamination is assumed to be uniformly mixed into that volume of soil. This reduces the amount of material available for resuspension.

E.4 CALCULATED MAXIMUM ANNUAL DOSES AT THE DECOMMISSIONED REFERENCE UF₆ CONVERSION PLANT

This section contains the assumptions, specifies the radionuclide inventories, and presents listings of the resultant calculated maximum annual doses. These maximum annual doses are used in determining the acceptable residual contamination levels for the decommissioned reference UF₆ conversion plant that are presented in Section 8. Doses are calculated and reported for the maximum-exposed individual working in the UF₆ conversion plant or residing on its site after decommissioning.

E.4.1 Assumptions for Calculating Maximum Annual Dose

The following assumptions are used in calculating the maximum annual doses at the decommissioned UF₆ conversion plant:

- The radionuclide inventories are undisturbed, except for radioactive decay after plant shutdown. Doses are calculated for the mixtures at shutdown, and for decay periods of 10, 30, 50, and 100 years. Since the radionuclide inventory contains radionuclides with very long half-lives, the inventory composition does not change significantly with time. Thus, doses are calculated for the inventory at shutdown only.
- The maximum-exposed individual is considered to have realistically maximized exposure pathway parameters.
- The maximum-exposed individual working in the facility receives continuous exposure for 40 hours a week, 50 weeks a year.
- The maximum-exposed individual residing on the site is continuously exposed.

- The airborne radionuclide concentrations are calculated using a resuspension factor of $5 \times 10^{-6} \text{ m}^{-1}$ inside the facility, and the time dependent factor from Equation E.11 on the site.

E.4.2 Radionuclide Inventories for Calculating Maximum Annual Doses

The radionuclide inventory used to calculate the maximum annual doses at the decommissioned UF_6 conversion plant is presented and discussed in Appendix C. Table C.2-1 contains the reference radionuclide mixture that characterizes surface contamination inside the reference UF_6 conversion plant. Table B.5-2 contains the radionuclide inventory that characterizes the contamination deposited on the reference site during the lifetime of plant operations.

E.4.3 Maximum Annual Dose Listings

Calculated maximum annual doses to the maximum-exposed individual using the decommissioned UF_6 conversion plant or its site are listed in Tables E.4-1 through E.4-4. These doses are calculated using the models, parameters, and assumptions presented earlier in this appendix. Dose pathways of concern in the decommissioned reference UF_6 conversion plant are inhalation of resuspended surface contamination and external exposure to surface contamination. The calculated maximum annual doses to an individual working in the decommissioned facility for 40 hours a week, 50 weeks a year, are listed in Tables E.4-1 and E.4-2 for translocation classes W and Y, respectively. Dose pathways of concern on the decommissioned reference site are inhalation of resuspended ground contamination, external exposure to ground contamination, and ingestion of food products raised on the site. The calculated maximum annual doses to an individual residing on the reference site 24 hours a day, 52 weeks a year, are listed in Tables E.4-3 and E.4-4 for translocation classes W and Y, respectively. These tables contain doses calculated using the radionuclide mixtures present at plant shutdown. They are used to determine the acceptable residual contamination levels for unrestricted release of the decommissioned UF_6 conversion plant, which are presented and discussed in Section 8.

TABLE E.4-1. Calculated Maximum Annual Radiation Doses to the Maximum-Exposed Individual Working in the Released Decommissioned UF₆ Conversion Plant - Class W Material^(a)

Organ of Reference	Radionuclide ^(b)	Maximum Annual Dose (mrem) Via:		
		Inhalation	External ^(c)	All Pathways
Total Body at 2 Years ^(d)	²³⁴ U	1.2×10^{-6}	3.3×10^{-8}	1.2×10^{-6}
	²³⁵ U + D ^(e)	5.2×10^{-8}	2.6×10^{-7}	3.2×10^{-7}
	²³⁸ U + D	1.1×10^{-6}	1.6×10^{-6}	2.7×10^{-6}
	Totals	2.4×10^{-6}	1.9×10^{-6}	4.2×10^{-6}
Bone at 2 Years	²³⁴ U	1.7×10^{-5}	3.3×10^{-8}	1.7×10^{-5}
	²³⁵ U + D	7.8×10^{-7}	2.6×10^{-7}	1.0×10^{-6}
	²³⁸ U + D	1.5×10^{-5}	1.6×10^{-6}	1.7×10^{-5}
	Totals	3.3×10^{-5}	1.9×10^{-6}	3.5×10^{-5}
Lungs at 1 Year	²³⁴ U	1.3×10^{-4}	3.3×10^{-8}	1.3×10^{-4}
	²³⁵ U + D	5.7×10^{-6}	2.6×10^{-7}	6.2×10^{-6}
	²³⁸ U + D	1.1×10^{-4}	1.6×10^{-6}	1.1×10^{-4}
	Totals	2.5×10^{-4}	1.9×10^{-6}	2.5×10^{-4}
LLI at 1 Year	²³⁴ Th + D	1.3×10^{-7}	-- ^(f)	1.3×10^{-7}
	²³⁴ U	1.1×10^{-7}	3.3×10^{-8}	1.4×10^{-7}
	²³⁵ U + D	5.7×10^{-9}	2.6×10^{-7}	2.7×10^{-7}
	²³⁸ U + D	9.7×10^{-8}	1.6×10^{-6}	1.7×10^{-6}
	Totals	3.4×10^{-7}	1.9×10^{-6}	2.2×10^{-6}

(a) The mixture of radionuclides at shutdown normalized to 1 pCi/m² is used.

(b) Only radionuclides that contribute more than 1% of the total dose of any pathway are listed.

(c) External dose from floor contamination.

(d) The time the annual dose peaks after exposure begins.

(e) The + D notation indicates that after deposition in the organ of reference, the decay energy of the daughter is included with the parent.

(f) A dash indicates no external dose component.

TABLE E.4-2. Calculated Maximum Annual Radiation Doses to the Maximum-Exposed Individual Working in the Released Decommissioned UF₆ Conversion Plant - Class Y Material^(a)

Organ of Reference	Radionuclide ^(b)	Maximum Annual Dose (mrem) Via:		
		Inhalation	External ^(c)	All Pathways
Total Body at 2 Years ^(d)	²³⁴ U	1.9×10^{-7}	3.3×10^{-8}	2.2×10^{-7}
	²³⁵ U + D ^(e)	8.3×10^{-9}	2.6×10^{-7}	2.7×10^{-7}
	²³⁸ U + D	1.6×10^{-7}	1.6×10^{-6}	1.7×10^{-6}
	Totals	4.3×10^{-7}	1.9×10^{-6}	2.2×10^{-6}
Bone at 4 Years	²³⁰ Th	2.2×10^{-6}	1.8×10^{-9}	2.2×10^{-6}
	²³⁴ U	2.6×10^{-6}	3.3×10^{-8}	2.7×10^{-6}
	²³⁵ U + D	1.2×10^{-7}	2.6×10^{-7}	3.8×10^{-7}
	²³⁸ U + D	2.3×10^{-6}	1.6×10^{-6}	3.9×10^{-6}
	Totals	7.2×10^{-6}	1.9×10^{-6}	9.1×10^{-6}
Lungs at 2 Years	²³⁴ U	5.9×10^{-4}	3.3×10^{-8}	5.9×10^{-4}
	²³⁵ U + D	2.6×10^{-5}	2.6×10^{-7}	2.6×10^{-5}
	²³⁸ U + D	5.0×10^{-4}	1.6×10^{-6}	5.0×10^{-4}
	Totals	1.1×10^{-3}	1.9×10^{-6}	1.1×10^{-3}
LLI at 1 Year	²³⁴ Th + D	1.5×10^{-7}	-- ^(f)	1.5×10^{-7}
	²³⁴ U	1.1×10^{-8}	3.3×10^{-8}	4.4×10^{-7}
	²³⁵ U + D	5.7×10^{-9}	2.6×10^{-7}	2.7×10^{-7}
	²³⁸ U + D	9.9×10^{-8}	1.6×10^{-6}	1.7×10^{-6}
	Totals	2.6×10^{-7}	1.9×10^{-6}	2.2×10^{-6}

(a) The mixture of radionuclides at shutdown normalized to 1 pCi/m² is used.

(b) Only radionuclides that contribute more than 1% of the total dose of any pathway are listed.

(c) External dose from floor contamination.

(d) The time the annual dose peaks after exposure begins.

(e) The + D notation indicates that after deposition in the organ of reference, the decay energy of the daughter is included with the parent.

(f) A dash indicates no external dose component.

TABLE E.4-3. Calculated Maximum Annual Radiation Doses to the Maximum-Exposed Individual Residing on the Released Decommissioned UF₆ Conversion Plant Site - Class W Material

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose (mrem) Via:			
		Ingestion	Inhalation	External Ground ^(b)	All Pathways
Total Body at 3 Years ^(c)	²³⁴ U	4.0×10^{-2}	1.2×10^{-3}	8.5×10^{-4}	4.2×10^{-2}
	²³⁵ U + 0 ^(d)	1.7×10^{-3}	5.3×10^{-5}	6.5×10^{-3}	8.2×10^{-3}
	²³⁸ U + 0	3.4×10^{-2}	1.1×10^{-3}	4.2×10^{-2}	7.7×10^{-2}
	Totals	7.6×10^{-2}	2.4×10^{-3}	4.9×10^{-2}	1.3×10^{-1}
Bone at 50 Years	²³¹ Pa	1.8×10^{-3}	5.8×10^{-4}	1.9×10^{-4}	2.6×10^{-3}
	²³⁰ Th	1.2×10^{-1}	2.7×10^{-2}	4.6×10^{-5}	1.5×10^{-1}
	²³⁴ U	6.5×10^{-2}	6.0×10^{-4}	8.5×10^{-4}	6.5×10^{-2}
	²³⁵ U + 0	2.9×10^{-2}	2.5×10^{-5}	6.5×10^{-3}	3.4×10^{-2}
	²³⁸ U + 0	5.6×10^{-1}	5.3×10^{-4}	4.2×10^{-2}	6.0×10^{-1}
	Totals	1.4×10^0	2.9×10^{-2}	5.0×10^{-2}	1.5×10^0
Lungs at 1 Year	²³⁴ U	-- ^(e)	1.6×10^{-1}	8.5×10^{-4}	1.6×10^{-1}
	²³⁵ U + 0	-- ^(e)	6.5×10^{-3}	6.5×10^{-3}	1.3×10^{-2}
	²³⁸ U + 0	-- ^(e)	1.4×10^{-1}	4.2×10^{-2}	1.8×10^{-1}
	Totals	--	3.1×10^{-1}	4.9×10^{-2}	3.5×10^{-1}
LLI at 1 Year	²³⁴ Th + 0	9.6×10^{-2}	1.6×10^{-4}	--	9.6×10^{-2}
	²³⁴ U	5.5×10^{-2}	1.4×10^{-4}	8.5×10^{-4}	5.5×10^{-2}
	²³⁵ U + 0	3.1×10^{-3}	6.5×10^{-6}	6.5×10^{-3}	9.6×10^{-3}
	²³⁸ U + 0	4.7×10^{-2}	1.2×10^{-4}	4.2×10^{-2}	8.9×10^{-2}
	Totals	2.0×10^{-1}	4.3×10^{-4}	4.9×10^{-2}	2.5×10^{-1}

(a) Only radionuclides that contribute more than 1% of the total dose of any pathway are listed.

(b) External dose from ground contamination plowed to a depth of 0.15 m.

(c) The time the annual dose peaks after exposure begins.

(d) The + 0 notation indicates that after deposition in the organ of reference, the decay energy of the daughter is included with the parent.

(e) A dash indicates no organ data is available for the exposure pathway and radionuclide shown.

TABLE E.4-4. Calculated Maximum Annual Radiation Doses to the Maximum-Exposed Individual Residing on the Released Decommissioned UF₆ Conversion Plant Site - Class Y Material

Organ of Reference	Radionuclide ^(a)	Maximum Annual Dose (mrem) Via:			
		Ingestion	Inhalation	External ^(b) Ground	All Pathways
Total Body at 50 Years ^(c)	²²⁶ Ra	3.4×10^{-4}	4.2×10^{-8}	9.6×10^{-6}	3.5×10^{-4}
	²³⁰ Th	3.7×10^{-3}	3.4×10^{-4}	4.6×10^{-5}	4.1×10^{-3}
	²³¹ Pa	2.7×10^{-5}	1.0×10^{-5}	1.9×10^{-4}	2.2×10^{-4}
	²³⁴ U	4.0×10^{-2}	1.6×10^{-5}	8.5×10^{-4}	4.1×10^{-2}
	²³⁵ U + D ^(d)	1.6×10^{-3}	6.5×10^{-7}	6.5×10^{-3}	8.1×10^{-3}
	²³⁸ U + D	3.4×10^{-2}	1.4×10^{-5}	4.2×10^{-2}	7.6×10^{-2}
	Totals	8.0×10^{-2}	3.8×10^{-4}	5.0×10^{-2}	1.3×10^{-1}
Bone at 50 Years	²³⁰ Th	1.2×10^{-1}	1.1×10^{-2}	4.6×10^{-5}	1.4×10^{-1}
	²³¹ Pa	1.9×10^{-3}	2.4×10^{-4}	1.9×10^{-4}	2.3×10^{-3}
	²³⁴ U	6.5×10^{-1}	2.5×10^{-4}	8.5×10^{-4}	6.5×10^{-1}
	²³⁵ U + D	2.8×10^{-2}	1.1×10^{-5}	6.5×10^{-3}	3.4×10^{-2}
	²³⁸ U + D	5.8×10^{-1}	2.3×10^{-4}	4.2×10^{-2}	6.2×10^{-1}
	Totals	1.4×10^0	1.2×10^{-2}	5.0×10^{-2}	1.4×10^0
Lungs at 3 Years	²³⁴ U	-- ^(e)	8.5×10^{-1}	8.5×10^{-4}	8.5×10^{-1}
	²³⁵ U + D	-- ^(e)	3.6×10^{-2}	6.5×10^{-3}	4.2×10^{-2}
	²³⁸ U + D	-- ^(e)	7.4×10^{-1}	4.2×10^{-2}	7.9×10^{-1}
	Totals	--	1.6×10^0	4.9×10^{-2}	1.7×10^0
LLI at 1 Year	²³⁴ Th + D	9.6×10^{-2}	1.8×10^{-4}	--	9.6×10^{-2}
	²³⁴ U	5.5×10^{-2}	1.4×10^{-4}	8.5×10^{-4}	5.6×10^{-2}
	²³⁵ U + D	3.1×10^{-3}	6.5×10^{-6}	6.5×10^{-3}	9.6×10^{-3}
	²³⁸ U + D	4.7×10^{-2}	1.2×10^{-4}	4.2×10^{-2}	8.9×10^{-2}
	Totals	2.0×10^{-1}	4.4×10^{-4}	4.9×10^{-2}	2.5×10^{-1}

(a) Only radionuclides that contribute more than 1% of the total dose of any pathway are listed.

(b) External dose from ground contamination plowed to a depth of 0.15 m.

(c) The time the annual dose peaks after exposure begins.

(d) The + D notation indicates that after deposition in the organ of reference, the decay energy of the daughter is included with the parent.

(e) A dash indicates no organ data is available for the exposure pathway and radionuclide shown.

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APPENDIX F

GENERAL DECOMMISSIONING CONSIDERATIONS

This appendix gives general descriptions of chemical and physical decontamination and surface cleaning procedures employed in decommissioning a uranium hexafluoride conversion (UF_6) plant and of techniques for the removal of contaminated structural material. Special equipment items used in these activities are described. Methods and procedures for packaging and shipment offsite of radioactive wastes generated during decommissioning are also discussed. Quality assurance considerations important to decommissioning are addressed. Finally, the scope of environmental surveillance programs needed for DECON and for long-term care during passive SAFSTOR are outlined. Detailed descriptions of specific activities for the DECON and passive SAFSTOR decommissioning alternatives are given in Appendix G.

F.1 GENERAL DECONTAMINATION METHODS

Removal of radioactive material from contaminated surfaces can be accomplished in three basic ways: 1) by physical cleaning of the surface, 2) by the use of chemicals that dissolve surface films containing the radioactive material, and 3) by removal of the surface itself.

Physical cleaning methods include sweeping and vacuuming, handwiping, sandblasting, and washing with various cleansing agents. Chemical decontamination involves the use of acid solutions to dissolve residual contamination from surfaces and sometimes part of the surfaces themselves. It is typically applied to wet processing systems, pumps, piping, storage tanks, etc. If physical cleaning and chemical decontamination techniques do not reduce contamination levels on equipment and building surfaces to acceptable radioactivity release levels, it will be necessary to use more extensive methods such as physical methods to remove major surface layers in which radioactivity has become embedded. Surface removal techniques employed in this study are abrasive blasting for metal surfaces and jack hammering and surface spalling for concrete surface layers.

F.1.1 Surface Cleaning Procedures

Removal of contamination from sealed surfaces such as painted walls and floors, tank exteriors, etc., can be accomplished using a variety of techniques. For small quantities of loose contamination on floors, vacuuming or simple sweeping compounds are often effective. For more tenacious contaminants, various cleansing compounds combined with handwiping, handscrubbing, and power scrubbing techniques can be used. Several proprietary decontamination solutions are available. Ordinary household detergents are quite effective but produce sizable quantities of waste water to process. Aerosol-type foaming cleansers have proven effective and eliminate the problem of large volumes of waste water, but their use produces sizable quantities of contaminated wiping material.

Freon, acetone, alcohol and other solvents are effective degreasing agents that can be used in removing contaminated films from equipment surfaces. Organic solvents have an advantage of not being corrosive to equipment and electrical connections. However, their use generates organic solutions that must be processed and disposed of. These organic solutions can degrade plastic material, can require use of sizable quantities of contaminated wiping material, and can increase the risk of fire in the contaminated material.

Variable pressure, high- or low-velocity liquid jets can be effective for some types of surface decontamination work. The devices can be operated by one man, at pressures up to 30,400 kPa (300 atm) using a hand-held jet lance. On those contaminants for which the liquid jet is effective, the jet is a very rapid decontamination method. Its use can produce significant volumes of contaminated liquid that must be processed for disposal.

F.1.2 Chemical Decontamination

In the reference plant, the principal systems to be chemically decontaminated are the pipelines, tanks, process equipment and the chemical processing system.

During the decommissioning planning and preparation stage, procedures and results from routine chemical decontamination efforts carried out during plant operation are reviewed to obtain maximum benefit from previous

experience. Current as-built drawings are reviewed to identify system deadlegs and to facilitate the planning of any system modifications required to achieve decontamination. Existing procedures are reviewed for applicability to the present effort and modified as necessary. A detailed step-by-step procedure is developed for the decommissioning decontamination campaign, with checklists for valve settings, etc., to assure proper operation of the systems.

Some typical chemical solutions that might be used for chemical decontamination of the plant are given below:

- 1) A solution of 0.025 M KF - 0.1 M AlNO_3 may be used to remove uranium deposits.
- 2) Many sludge deposits may be removed by using a solution of 20% HNO_3 - 6% AlF_3 .
- 3) A solution of 10 M HNO_3 - 0.1 wt% CaF may be used to remove uranium contamination from stainless steel surfaces with only modest surface corrosion.
- 4) A tri-sodium phosphate (TSP) solution.

The chemical solution chosen must be compatible with the materials used in the system.

Chemical decontamination is postulated to be accomplished with batches of acid flush solution. Up to five flushes may be required, with recirculation times averaging 3 to 4 days per flush.

F.1.3 Concrete Removal Techniques

There will likely be concrete surfaces in the plant that are contaminated to a depth of a few centimeters and that cannot be cleaned to acceptable release levels by surface wiping or washing techniques. In these instances, the concrete surfaces must be physically removed and packaged for disposal. Several criteria must be considered in selecting a concrete removal method for a particular location. The method should facilitate control of airborne contamination and minimize any spread of contamination. The method should

minimize the potential for personnel exposure to radioactivity, and the size and weight of removed materials must be controlled to facilitate packaging and shipping for disposal.

The various methods available for removal of concrete and reinforcing steel are listed in Table F.1-1, together with the advantages and disadvantages of each. Methods of particular interest to this study are the surface-spalling and rock-splitting techniques for localized removal of concrete surfaces. In this study, the modified rock-splitting tool described in Section F.2.3 is the principal device used for removal of contaminated concrete surfaces. Use of this device permits localized removal of concrete surfaces at depths of 5 to 8 cm with no explosions and relatively little dust. The principal source of dust is the drilling of the hole in the concrete into which the splitting tool is

TABLE F.1-1. Concrete and Reinforcing Steel Removal Methods

Removal Method	Advantages	Disadvantages
Flame Cutting	<ul style="list-style-type: none"> • Concrete, steel liner, and rebar may be cut in one operation • Fast cutting speed • Adaptable for remote operation • Adaptable for pivoted circumferential cutting • Adaptable for cutting on irregular surfaces • No vibration/shock • Low noise level • Controlled rate of material removal 	<ul style="list-style-type: none"> • Generates copious quantities of toxic gases and smoke • Consumes large quantity of gas • Smoke could potentially spread gross contamination throughout contamination control envelope unless adequately controlled • Requires a through starter-hole made by a core drill to provide an outlet gas-flow path
Thermic Lance Cutting	<ul style="list-style-type: none"> • Cuts both concrete and steel without difficulty • Fast "hole punching" speed • Ideally suited for cutting irregular surfaces • Remote operation is possible for up to approximately 3 meters • No vibration/shock • Low noise level • Controlled rate of material removal 	<ul style="list-style-type: none"> • Generates moderate quantities of toxic gases and smoke • Requires need for rock jack to break out concrete sections • Consumes large quantity of bottled oxygen • Smoke could potentially spread gross contamination throughout contamination control envelope unless adequately controlled
Core Boring and Rock-Splitting	<ul style="list-style-type: none"> • Proven technique used extensively in conventional mining operations • Low noise level • Low vibration/shock level • No toxic gases generated • Water cooling minimizes airborne particulate material • Moderate-to-slow cutting speeds • Controlled rate of material removal 	<ul style="list-style-type: none"> • Cutting through steel rebar or structural steel slows cutting speed and tears diamonds off core drill • Requires additional operation of oxyacetylene torch for rebar • Cooling water required for drilling may have to be filtered and recycled • Not ideally suited for operation on irregular surfaces or in cramped quarters

inserted. Use of portable vacuum cleaners with collection at the tool location can minimize scatter of contaminated dust. When decontaminating wall surfaces, work starts at the top of the contaminated area to prevent potential recontamination of the clean surface by falling dust or debris. The material removed is packaged for disposal.

Electric or pneumatic hammers may also be used, particularly on floor surfaces. These devices have the disadvantage of creating significant quantities of dust and small fragments that tend to recontaminate the areas being cleaned unless special precautions are taken.

Two techniques for cutting concrete and steel reinforcing bars in a single operation are flame cutting and thermic lance cutting. Flame cutting utilizes commercially available 3-m-long by 19-mm-diameter thin-walled conduit burning bars. Each bar contains up to seven dissimilar metals in the form of strips or strands and is encased in an outer tube into which oxygen is forced during the burning process. Once ignited, a bar is consumed as it burns. After 4 to 5 minutes and with about 0.6 m of bar left, the oxygen supply is shut off and the exothermic chemical reaction stops. The final 0.6 m of the first bar can be attached to a second bar and the burning process repeated, thus minimizing the use of bar material.

The thermic lance uses powders of material similar to those in the strands of the burning bar. The powders are forced through a tube or nozzle with oxygen and ignited to form a high-temperature cutting medium. In both of these techniques, copious quantities of toxic gases and/or smoke are produced and extensive provisions must be made for adequate ventilation of the work area and control of the associated dust.

F.2 SPECIAL EQUIPMENT REQUIREMENTS

Some of the decommissioning activities described in the previous section require special tools or equipment. Some of this equipment may be available as part of the facility's normal operating equipment complement. Other equipment will require offsite procurement or manufacture.

Several activities are involved in meeting the need for special tools and equipment. These activities include: the development of functional and operational requirements for devices needed in the decommissioning work; the preparation of detailed designs for noncommercial devices and modifications to commercially available equipment; procurement or fabrication of the identified equipment items; and the establishment of a mock-up facility for use in testing and development of operational parameters for remotely operated equipment.

A listing of special tools and equipment needed for plant decommissioning activities, together with the functions of the equipment, is given in Table F.2-1. Descriptions of some of the more unique devices are presented in this section.

F.2.1 High-Speed Cutting Devices

For rapid cutting of steel, the oxyacetylene torch will be used. Since this torch is a relatively common device, it will not be discussed here.

More sophisticated cutting devices such as the plasma-arc torch⁽¹⁾ and arc saw⁽²⁾ can also be used.

F.2.2 Mechanical Cutting Devices

Four mechanical cutting techniques for sectioning contaminated components are used to dismantle the reference plant. The first technique uses a conventional tubing cutter to section mild or stainless steel pipe of up to 0.10 m O.D. into appropriate lengths. The tubing cutter is placed on the pipe and a hardened sharp circular blade is gradually tightened and revolved about the pipe until the pipe is sectioned. This sectioning technique is only useful for smaller diameter pipe and requires complete accessibility to all sides of the pipe.

The second cutting technique uses a heavy-duty reciprocating saw with blades that are 0.30 to 0.60 m long to section stainless steel pipe of up to 0.20 m O.D. The saw is equipped with a pivot bracket that uses the weight of the saw to provide the necessary cutting force. A typical cutting time for a 0.20-m-diameter stainless steel Schedule 10 pipe is approximately 5 minutes. Cutting oil can be used to help cool the reciprocating saw blade and to contain the contamination inside the pipe.

TABLE F.2-1. Special Tools and Equipment for Dismantlement

Item	Estimated Unit Cost (\$ Thousands)	Functions
Oxyacetylene torch	1	Sectioning of hoods, tanks, piping, support structures, etc.
Guillotine pipe saw	4	Pipe and tube cutting
Tube cutter	0.41	Pipe and tube cutting
Ratcheting pipe cutter	0.07	Pipe and tube cutting
Reciprocating saw	0.8	Pipe and tube cutting
Nibbler	1	Cutting ductwork and other thin gauge materials
Assorted tools such as impact wrenches, bolt cutters, etc.	6.9	Disassembly, handling and packaging of contaminated materials
High-velocity liquid jet	6.9	Surface decontamination of piping, tanks and equipment
Low-velocity liquid jet	2.76	
Hydraulic concrete surface spalling device	5	Removal of contaminated concrete surfaces
Concrete drills	0.28	Drilling holes in concrete as required for spalling and volume blasting
Electric/pneumatic hammers	0.70	Breaking up concrete
Portable A-frames	4.10	Lift and transport large equipment items and components
Portable wash tanks	2.76	Used to wash, scrub and rinse for decontamination of equipment components, pipes, etc.
Portable greenhouse erection kit	2.76	Contain spread of contamination from equipment disassembled in dusty plant areas and for outside decontamination activities
Portable spray cleaning booth	5.52	Used to clean large pieces of equipment in original rooms
Portable power brushes	0.21	An assortment of sizes and types of brushes which are powered to improve productivity in decontaminating equipment, walls, ceilings, etc.
Portable abrasive blasting unit	27.6	Abrasive clean corroded metal surfaces in large tanks, pipes, etc.

The third cutting method uses a hand-held reciprocating saw with short (approximately 75-mm) blades. The tool is used to cut small-diameter pipe (less than 50 mm O.D.). Saw blade life is shorter and the work is harder for the operators than with other techniques.

The fourth cutting technique uses a nibbler device with a scissor action to section large-diameter, contaminated steel ducting into pieces suitable for decontamination and compaction.

Studies have shown that mechanical cutting techniques can be used to safely and effectively section pipe, ducting, and other contaminated components into sizes and geometries suitable for decontamination and/or packaging.⁽³⁾ Typical sectioning times are 10 minutes or less and contamination can be restricted to the saw blades and the immediate cutting area.

F.2.3 Tools for Concrete Surface Removal

Instances may occur throughout the facility where concrete surfaces are contaminated to a depth such that surface washing with decontamination solutions is ineffective, and removal of the surface layers is the only effective decontamination method.

Concrete spallers (rock splitters) have been developed to facilitate this type of decontamination effort. A schematic of this device is illustrated in Figure F.2-1. This device permits localized concrete removal to depths of 50 to 75 mm with no explosions and very little dust. (The principal source of dust is the drilling of the hole into which the splitting tool is inserted.) A dust shield with a vacuum attachment minimizes the spread of contaminated dust and can be used to collect all but the largest pieces of rubble.

The concrete spaller consists of three basic parts: a hydraulic cylinder, a push rod, and a bit with expanding wedges. The hydraulic cylinder is attached at one end and activates the push rod, which is installed inside the bit.

The spaller is operated by inserting the bit into a predrilled hole about 55 mm deep and 25 mm in diameter. The hydraulic cylinder is then activated, forcing the push rod toward the end of the bit. The wedges in the bit are

FIGURE F.2-1. Schematic of Concrete Spaller

forced radially outward into the walls of the hole. Then, when the tip of the push rod reaches the bottom of the hole, it forces the wedges away from the bottom and causes the concrete to be spalled off.

The spaller is small, lightweight, and fully portable, and can be readily adapted to remote operation. It is connected to the hydraulic power source with hydraulic hoses. The spaller is about 0.4 m long, with a mass of about 10 kg.

For rapid removal of large surface areas, several devices can be ganged together with a corresponding set of concrete drills and operated as a unit. The spacing between holes and the pattern of the holes (square or triangular array) are important parameters in the effectiveness of this technique for removing concrete surfaces. Arrangement of the concrete drills and the splitting tools into a fixed-geometry array assures a relatively uniform removal pattern. Combining these ganged units with a vacuum transfer system for rubble removal would result in a fast and dust-free concrete removal method, one ideally suited to nuclear facility decommissioning applications.

F.2.4 Portable Equipment

To provide adequate control and productivity in the field processing of equipment, a number of portable items are needed.

Much of the equipment in the UF_6 conversion plant is large. It will be necessary to have A-frame units which can handle the equipment and large components during disassembly, decontamination and handling operations. Many of the areas within the plant are not served by overhead cranes.

Portable tanks will be used to decontaminate equipment, piping, etc., as it is disassembled in each room. The tanks will enable washing, scrubbing and rinsing of the components which will in turn save packaging, shipping, and burial costs for radioactive wastes ($\approx \$530$ to $\$706/m^3$). The components can then be sold as scrap or hauled to a local dump for disposal.

A portable spray cleaning booth can be used to clean large equipment and components in the field. It can be used in conjunction with the portable A-frames for the cleaning operation.

Portable greenhouses can be erected in dusty areas (e.g., disassembling hammer mills) or outside (e.g., decontaminating pipes).

Portable powered brushes will expedite cleanup of tight spaces (e.g., tooth brush size) to large areas (e.g., large rotary brushes) for decontaminating ceilings and walls.

Portable abrasive blasting equipment can be used to clean painted or corroded surfaces where at least partial removal of the surface is necessary.

F.3 PACKAGING AND SHIPPING OF RADIOACTIVE MATERIALS

Large quantities of radioactive wastes are removed during the dismantlement of a UF_6 conversion plant. These materials must be properly packaged and shipped to an authorized disposal site.

Radioactive material removed during dismantlement include:

- contaminated process equipment and tanks

- contaminated piping, ducts, and fixtures
- concrete rubble from the mechanical decontamination of contaminated floors, walls, and abutments
- contaminated lagoon liners and soil
- miscellaneous non-combustible materials

The bulk of the material that must be packaged for disposal will be contaminated with uranium. Contact radiation dose rates from waste materials generated during dismantlement operations will vary from a few micro-roentgen per hour to a few milli-roentgen per hour. The radioactive wastes are assumed to be disposed of at licensed shallow-land burial sites.

F.3.1 Packaging of Radioactive Materials

Only solid wastes are assumed to be transported to a commercial burial ground.

Waste from decommissioning operations is packaged for shipment in 0.21-m³ steel drums or in plywood boxes. Packages of non-TRU waste qualify as low specific activity (LSA) material^(a) and can be shipped without shielding.

F.3.2 Shipping of Radioactive Materials

All shipments of radioactive materials must be made in compliance with federal, state and local regulations. Federal transportation regulations of

(a) Radioactive materials are classified for transportation purposes into one of seven transport groups according to their potential hazard if released to the environment. Transport Group I is the most restrictive. Plutonium and other transuranic elements are in this transport group. Transport Group VII is the least restrictive. Shipments that pose a negligible risk to the public health may be classified as low specific activity (LSA) material 10 CFR 71.4(g). If the radioactivity is essentially distributed uniformly, with a concentration of not more than 0.1 $\mu\text{Ci/g}$ of Group I material, or 5 $\mu\text{Ci/g}$ of Group II material, or 300 $\mu\text{Ci/g}$ of Group III or IV material, the waste qualifies as LSA material. Externally contaminated nonradioactive materials may be considered as low specific activity provided that the radioactive contamination averaged over 1 m² does not exceed 0.1 $\mu\text{Ci/cm}^2$ for Group I radionuclides or 1.0 $\mu\text{Ci/cm}^2$ for others. Basically, only strong, tight packaging that will not leak in normal transport is required for the shipment of LSA material.

DOT and NRC establish container requirements, dose rate limits and handling procedures that ensure the safety of the public and transportation workers during shipment of radioactive materials.⁽⁴⁾ Federal regulations applicable to the transport of radioactive materials are:

- Title 49 Code of Federal Regulations Parts 170-199 (49 CFR 170-199) -- Department of Transportation regulations governing the transport of hazardous materials.
- 10 CFR 71--NRC regulations governing the packaging and shipment of radioactive materials.

In addition, for highway transport, state agencies regulate vehicle sizes and weights and, in some cases, transportation routes and times of travel.

Dose rates, for shipments in exclusive-use, closed transport vehicles, must not exceed the following values (DOT, 49 CFR 173.393):

- 1) 1000 millirem per hour at 1 m (3 ft) from the external surface of the package
- 2) 200 millirem per hour at any point on the external surface of the vehicle
- 3) 10 millirem per hour at 2 m from the external surface of the vehicle
- 4) 2 millirem per hour at any normally occupied position in the transport vehicle.

These dose rate limits are illustrated in Figure F.3-1 for truck transport.⁽⁵⁾ All of these criteria must be met on a given shipment, with an exclusive-use vehicle, properly placarded.

Shipments of non-TRU waste are transported to licensed commercial low-level waste burial grounds. Because none of the currently licensed commercial burial grounds has a rail siding, it is assumed that all non-TRU waste shipments are made by truck in exclusive-use vans.

FIGURE F.3-1. Radiation Dose Limits for Exclusive-Use Vehicles

F.4 QUALITY ASSURANCE

Quality assurance (QA) planning is required from the earliest stages of the decommissioning effort. During the planning and preparation phase of decommissioning, as each detailed procedure is developed, the QA portions are included. While not specifically related to UF_6 conversion plants or to the decommissioning of nuclear facilities, the following documents provide valuable information about QA requirements and procedures:

- Regulatory Guide 1.58, Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel for the Construction Phase of Nuclear Power Plants^(a)

(a) Although not written specifically for the decommissioning phase, the QA principles and objectives found herein would, however, apply to all activities of decommissioning.

- Regulatory Guide 1.88, Collection and Storage of Nuclear Power Plant Quality Assurance Records
- 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants^(a)
- MIL-Q-985A (Military QA Standard), for which a handbook (Handbook 50) helps in the interpretation of QA requirements and guides the user
- ANSI N45.2, Quality Assurance Program Requirements for Nuclear Power Plants.

In this study, the QA engineer is assumed to report directly to the project manager. Thus, the QA staff is organizationally independent of the project staff, up to the highest possible level of the decommissioning organization.

Examples of where quality assurance will be applied to the decommissioning project are:

- reviewing decommissioning plans for quality assurance involvement
- preparing inspection/test procedures as work plans are developed
- reviewing designs of test equipment for quality input
- ordering any inspection/test equipment required to perform quality assurance/quality control function
- receiving procured equipment, verifying acceptance
- qualifying suppliers for fabrication of radioactive shipping containers
- preparing inspection/test procedures to be imposed on subcontractors
- preparing inspection plans for shipment of radioactive materials, containers, trucks, etc.
- finalizing formal Quality Assurance Plan.

QA efforts during the actual dismantlement period involve:

- performing QA functions for procurements
- auditing all program activities

(a) Although not written specifically for the decommissioning phase, the QA principles and objectives found herein would, however, apply to all activities of decommissioning.

- monitoring performances of specialists, utility operators, laborers, craftsmen, and health physics technicians for compliance with work procedures
- verifying compliance of radioactive shipments with appropriate procedures and regulations
- performing dimensional, visual, nondestructive examinations, or other required inspection services to assure compliance with work plans
- maintaining auditable files on the QA audits
- preparing a final report on overall performance of the decommissioning program with regard to the QA function.

F.4.1 Methods Review

Whenever a new procedure is written, a documented peer review of the procedure is assumed to be held. The review assures that the planning includes suitable controls for application to physics, radiation, safety, accident, security, thermal, stress, QA/QC, and hydraulic analyses. The controls may include other checks, tests, and formal design reviews as necessary. Other aspects of the planning that are reviewed are reliability criteria, the choice of test methods, and the choice of processes or materials. The conservatism in margins of safety established for procedures, equipment, or structures also receives documented review.

F.4.2 Procurement Document Control

The project's suppliers, including fixed-price contractors and the shipping cask vendors, are selected from a QA list of Qualified Suppliers. QA inspections of the vendor's plant are performed as needed to place new firms on the qualified list.

Procurement Document Control is a QA procedure to verify that any QA requirements for specified hardware or services are identified in procurement documents or purchase orders. This includes references to applicable drawings, specifications, standards, codes, regulations, required records, test data, certifications, and qualifications of personnel or procedures. Special handling,

shipping, or packaging requirements are also identified to vendors. This control is applied to any supplier of the specific devices and equipment vital to the project, such as environmental monitoring equipment, and safety devices, radiation detection equipment, chemicals and explosives.

F.4.3 Work Instructions, Procedures, and Drawings

Formal documentation of work instructions, procedures, and drawings, and of the sequence of action for preparation, review, approval, and control of information is required. The people responsible for the sequence are clearly identified. These procedures verify the proper completion of activities or steps in the decommissioning process; for example, the disassembly sequences which ensure that vital services remain intact. The latest revisions of drawings, plans and work instructions are required to be at the work place. Also included are work instructions for vendors to review hardware for quantitative (dimensions, tolerances, or limits) and for qualitative (finish, workmanship) acceptability criteria. Changes in planning receive the same level of review and approval as original planning.

F.4.4 Document Control

Document Control verifies that all essential documentation receives review, approval, release, change and distribution control. A procedure is used to purge the system of obsolete or erroneous information.

F.4.5 Identification and Control of Procured Items

A program of supplier surveillance or source inspection for vendors of critical items is established. (Also see the Inspection and Test Section.) Records, such as certification of conformance, are required from vendors to show they meet the QA requirements. Inspection of purchased materials or items is performed upon receipt of the items. Items must be sufficiently identified to allow proper selection for use, completion of processing, or segregation of nonconforming items. The shelf life of chemicals must be clearly stated.

F.4.6 Special Processes

Decommissioning may have several tasks involving new or very unusual work procedures (e.g., large-scale chemical decontamination work). Special provisions are made to identify and control any unusual processes, using reviews, procedures, and audits.

F.4.7 Inspection and Test

All items produced for the project are properly inspected or verified. Inspections in the vendor's plant may be necessary for some items. Adequate inspection instructions are prepared, and inspection records are prepared and retained.

Verification procedures assure inspection of each essential operation, as well.

F.4.8 Calibration

All measurements require the use of properly calibrated equipment. Therefore, a system of calibration control is used to assure instrument reading validity. The accuracy requirements and tolerances for the project measurements are identified. Examples of equipment with calibration requirements include radiation monitoring devices, portable radiation detection equipment used in dose rate surveys of shipping packages, and devices used to monitor for releasability as cleaned or uncontaminated scrap. Instruments for monitoring environmental effluents and vital services probably have calibration procedures already in place. For this equipment, the existing program is maintained.

F.4.9 Shipping and Receiving

Special instructions for handling, packaging, shipping, receiving, storing and marking of components for the project are prepared. Included are radioactive shipments that require written QA procedures for each package type, specifying all steps in the package inspection, preparation, loading, and closing. Signed checklists are used. Seals are carefully verified on contaminated units for which no additional outside packaging is used.

F.4.10 Operating Status

Sufficient identification is applied to prevent the inadvertent operation of systems no longer functional or needed, or the use of discrepant items. Restricted use may be necessary in many cases for some systems, such as pumps, that may not have been fully removed from service.

F.4.11 Nonconformance and Corrective Action

All nonconformances are identified, and the records are retained as part of the project records.

F.4.12 Special Quality Records

An index of records and their required retention periods is maintained. These records include operating logs, results of reviews, inspections, training and qualifications of personnel, procedures, drawings, specifications, criteria documents, procurement documents, shipping records, nonconformance and corrective action reports, and audit reports. Extensive records of dose rate surveys are also maintained.

F.4.13 Quality Auditing

Regular audits are performed to verify that each of the QA procedures is being followed. Examples include witnessing radioactive shipments, calibration audits, verification that the correct procedures are at all work locations, vendor inspection, receiving and storage inspections, and audits for complete QA procedures and record retention.

F.5 ENVIRONMENTAL SURVEILLANCE PROGRAM

The levels of environmental surveillance needed for decontamination operations and for the period of safe storage are different. The scopes of the necessary programs for these two cases are outlined in the following sections.

F.5.1 Environmental Surveillance Program During DECON

The objectives of an environmental surveillance program for operating nuclear facilities are generally recognized as valid and that rationale can be applied to facilities undergoing decommissioning as well. The following objectives are taken from ICRP No. 7.⁽⁶⁾

- Detection of sudden changes and evaluation of long-term trends of concentration in the environment, with the intent to detect failure or lack of adequate control of releases and to initiate appropriate actions.
- Assessment of the actual or potential exposure of man to radioactive materials or radiation present in his environment, or the estimation of the probable upper limits of such exposure.
- Determination of the fate of contaminants released to the environment, especially with the intent of detecting previously unconsidered mechanisms of exposure.
- Demonstration of compliance with applicable regulations and legal requirements concerning releases to the environment.

Basic radiation exposure criteria and radioactive effluent release criteria applicable to population exposure are given in 10 CFR 20, Sections 20.1, 20.105, 20.106, and 20.3D3. For nonradioactive contaminants, consideration must be given to applicable standards such as Water Quality Criteria⁽⁷⁾ and ambient air quality standards listed by the Environmental Protection Agency (EPA).⁽⁸⁾ Local or state air quality criteria would presumably also apply on a site-specific basis.

In addition, the interfaces of the environmental monitoring program between the plant owner, the appropriate state agencies, and the EPA should be stipulated in the application for the amended license, as may be required by 10 CFR, Part 70, Section 70.23(a), 70.32(b), and 70.34.

A suggested minimum program of environmental radiological surveillance to be conducted outside the plant during decommissioning operations for the purpose of establishing population dose is shown in Table F.5-1. This minimum program continues until all radioactive waste shipments from the site have been completed. At that time, the program is reduced in scope.

The analytical detection limits given in Table F.5-1 are based on the practicability of routine radioactivity measurement techniques, and in all cases should be sufficient to quantify radionuclide concentrations that would

TABLE F.5-1. Recommended Basic Environmental Monitoring Program for the Period of Active Decommissioning Operations

Sample Type	Frequency	Analysis	Analytical Detection Limit ^(a)	Number of Sampling Stations Onsite	Number of Stations Offsite
<u>Terrestrial Samples</u>					
Air Particulate	Weekly	Gross Alpha	0.002 pCi/m ³	2	4
	Monthly	Gross Beta	0.002 pCi/m ³	2	4
	Monthly	Gamma Scan ^(b)	0.3 pCi/m ³ /isotope	2	4
Direct Radiation	Quarterly	TLD ^(c)	1.25 mrem/quarter increase	8	10
Rainfall	Monthly	Gross Alpha	0.5 pCi/l	1	2
	Monthly	Gamma Scan ^(d)	25 pCi/l/isotope	1	2
Soil	Semiannually	Gamma Scan	0.1 pCi/g/isotope (dry)	3	3
Vegetation	Semiannually	Gamma Scan	50 pCi/kg/isotope (wet)	2	3
Animals	Semiannually	Gamma Scan	50 pCi/kg/isotope (wet)	2	3
<u>Aquatic Samples</u>					
Surface Water	Monthly	Gross Alpha	0.5 pCi/l	2	3
	Monthly	Gamma Scan ^(e)	25 pCi/l/isotope	2	3
Well Water	Quarterly	Gross Alpha	0.5 pCi/l	1	3
	Quarterly	Gamma Scan ^(d)	25 pCi/l/isotope	1	3
Bottom Sediment	Semiannually	Gamma Scan	0.1 pCi/g/isotope (dry)	1	3
Vegetation	Semiannually	Gamma Scan	100 pCi/kg/isotope (wet)	1	3
Shoreline Soil	Semiannually	Gamma Scan	0.1 pCi/kg/isotope (dry)	0	3
Fish	Semiannually	Gamma Scan	100 pCi/kg/isotope (wet)	1	3

(a) Analytical detection limit is defined here as that concentration that is three standard deviations above the average concentration in a blank sample, and assures accuracy limits of $\pm 25\%$.

(b) To be performed if gross alpha exceeds 0.1 pCi/m³.

(c) TLD: thermoluminescent dosimeter.

(d) To be performed if gross alpha exceeds 10 pCi/l.

result in conservatively estimated whole body and individual organ doses in the range of 1 to 50 mrem/year from specific nuclides. These nuclides are identified upon completion of the operational monitoring program. They are then defined in terms of concentrations in environmental media at levels which, if sustained, would result in doses in excess of acceptable limits. The derivation of these levels corresponds with the site-specific method of assessing offsite doses from radioactive materials in estimated gaseous and liquid effluents (if any), and is assumed to be consistent with the recommendations of the Federal Radiation Council⁽⁹⁾ and the International Commission on Radiological Protection.⁽¹⁰⁾ The environmental radiological monitoring program is designed to integrate fully with any ongoing programs of the state where the reference UF₆ conversion plant is located. Quality assurance is achieved, in part, by the site-specific state QA program and by participation in the EPA analytical quality assurance program. Sample collections and radiation measurements required to meet the schedule suggested in Table F.5-1 are assumed to be conducted at locations and in amounts previously established during the operational lifetime of the reference plant. The program and any changes thereto are reportable items to the NRC and to other appropriate regulatory agencies, as environmental technical specifications, in accordance with the terms of the amended license. Quality control mechanisms are exercised for all procedures involved.

Environmental monitoring during the period of decommissioning activities is carried out by the Health Physics group of the decommissioning organization as a part of their normal duties.

F.5.2 Environmental Surveillance Program for the Safe Storage Period of Passive SAFSTOR

An abbreviated version of the environmental monitoring program conducted during decommissioning is carried out during the safe storage period. During this time there is no intent to provide a surveillance program adequate for all potential nonroutine or accidental releases, although the proposed program will be useful in evaluating most lapses of control. For the emergency situation involving releases from events such as fire or malicious acts that may

require prompt emergency actions to minimize public risk, special surveillance requirements apply. The purpose of the program is to identify and quantify any releases of radioactivity to the environment.

Quality control mechanisms are exercised for all procedures involved. The surveillance system operation is assumed to be checked at least quarterly. Instrument calibrations are run at least quarterly and following maintenance. Remedial measures are taken as needed. Sampling quality control may necessitate temporary reinstatement of some aspects of the facility's former procedures, e.g., multiple sampling, sample splitting, and varying sampling locations. Similar analytical quality assurance procedures, such as treating audit samples of known concentrations, could be expected to be instituted.

Changes in background levels, in environmental accumulations (e.g., fallout from nuclear weapons testing), and especially in land usage and population distribution may, over a period of years, justify modifications in the post-operation surveillance program. Experience will indicate, and the licensee should be alert for, relationships between local environmental results that would reasonably permit reduction of the program.

The entire program should be reviewed on a periodic basis to assess its adequacy and to determine whether changes and/or reductions should be made.

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APPENDIX G

DECOMMISSIONING ACTIVITY DETAILS

The details presented in this appendix are specifically for decommissioning the uranium hexafluoride conversion (UF_6) plant by the DECON alternative or by the passive SAFSTOR alternative. Section G.1 contains information common to both decommissioning alternatives in regard to final processing inventory cleanout. Section G.2 contains information specific to the DECON alternative, and Section G.3 contains details necessary to complete passive SAFSTOR. Section G.4 summarizes details of lagoon waste onsite disposal, and Section G.5 gives details of lagoon waste stabilization. The details presented here amplify the information presented in Section 9 and provide the basic information needed to assess the safety and costs of decommissioning the reference UF_6 conversion plant.

G.1 FINAL PROCESSING INVENTORY CLEANOUT

After termination of production operations, an extensive process inventory cleanout is conducted. Final cleanout operations are similar to those operations conducted periodically to audit the uranium inventory. These cleanout operations are charged to operations and not to decommissioning. In general, the following activities are conducted during final inventory cleanout:

- All unneeded process chemicals are removed.
- All uranium products (UF_6 and U_3O_8 powder; scrap liquids and solids; contaminated liquid and solid wastes) in the building are removed and shipped offsite to appropriate dispositions.
- Vacuuming and conventional handwiping and brushing of contaminated surfaces are performed.
- All scrap materials stored on outside pads and in other buildings are shipped to appropriate disposition sites.
- The sludge in UF_6 and HF scrubbers is cleaned out.
- The wet processing equipment is flushed with water and appropriate cleaners (minimum of three flushes).

- The extraction fluoride waste systems are flushed twice with tributyl phosphate (TBP) and rinse water.
- A comprehensive radiation survey is performed to determine residual uranium inventory and to locate and identify localized contamination (hot spots that may require special handling or decontamination techniques).

All contaminated noncombustible solids generated during these cleanout operations are processed through the decontamination facility to recover their uranium content. The resulting contaminated liquid waste is processed through the radwaste recovery system before decommissioning operations begin. Combustible solid wastes are packaged in plastic and burned in an incinerator facility to recover uranium. The recovered uranium is shipped offsite for reuse. After removal of the in-process powder, scrap, and rework uranium materials, it is estimated (see Section 3) that cleanout operations recover approximately one-half of the uranium remaining in the building.

G.2 DECON ACTIVITIES

Figure G.2-1 shows the sequence and schedule for the DECON activities. The order of decommissioning for plant areas is shown in Figure A.2-2 in Appendix A.

Upon completion of final processing inventory cleanout, DECON activities commence. These are outlined as follows:

1. The areas close to uranium receiving areas are decontaminated first to provide staging areas for the storage, packaging, and shipment of equipment and materials from the plant.
2. To improve accessibility, the remaining plant-controlled areas are decontaminated, generally in the order of decreasing amount of contamination and increasing distance from the staging area.
3. The decontamination, radwaste, and incinerator facility are decontaminated near the end of the campaign, so they can be used to support the decommissioning operations.
4. The temporary laundry room and change rooms are the final rooms to be decontaminated.

FIGURE G.2-1. Schedule and Sequence for DECON Activities of the Reference UF₆ Conversion Plant

Concurrent with steps 1 through 4 above, the field facilities outside the plant are dismantled. The fluoride waste system is dismantled and decontaminated first, the extraction waste system second, and the temporary radwaste system last.

The final DECON activity is to conduct a thorough radiation survey and cleanup of all the uncontrolled areas in and around the UF₆ plant.

G.2.1 DECON Details

Following the final inventory cleanout, equipment removal and final decontamination operations are initiated. The conditions outlined below are met before commencing these operations.

1. Responsible management and safety personnel approve the following plans and procedures:
 - radiation work, industrial safety, and emergency procedures
 - equipment handling, disassembly, cleaning, and packaging and shipping procedures
 - equipment and facility decontamination plans and procedures.
2. Disposition is predetermined for all equipment. The equipment can either be decontaminated for reuse, sold as scrap, buried in the local landfill dump, partially decontaminated for use at another restricted plant, or shipped to a burial ground licensed for contaminated materials.
3. All ventilation equipment, personnel protection systems, emergency power systems, fire protection systems, and radiation monitoring equipment in the building and onsite are in service and fully functional.
4. All personnel and contractors are adequately trained to perform their jobs.
5. Appropriate occupational safety equipment and continuous air sampling equipment are available for equipment disassembly, transfers, and cleanup.
6. Temporary portable cleaning chambers for decontaminating equipment are available (e.g., greenhouse with tank for water and steam rinsing of equipment, washing tanks, degreasers, etc.).

7. Packaging materials and shipping containers are available.
8. All equipment for dismantlement and decontamination operations is available.
9. A comprehensive radiation survey is completed, with all results mapped and used as a basis for each room's work plan.
10. The system and procedures for the functions of special nuclear material accountability measurements and measurement control are established.
11. All process material supplies (e.g., hydrogen, nitric acid, ammonium hydroxide, fluoride) are disconnected from the plant and disposed of.

G.2.2 DECON Operations

This section identifies the major DECON steps for each area wherein contaminated processing equipment is disassembled and decontaminated (in its original area, using portable cleaning chambers and tanks where practicable), wrapped in plastic, removed to the staging area, and packaged for shipment to final disposition. Where practical, equipment is packaged in the original area. Some equipment requiring decontamination for unrestricted use is sent to the cylinder storage area for cleanup when in-situ cleanup is not practical.

In this study it is assumed that most equipment items are completely disassembled and decontaminated in their original location. After being released for unrestricted use, the parts are packaged and shipped to a local dump. Parts that cannot be cleaned sufficiently for release are packaged in plywood boxes and shipped to a licensed burial ground.

The pattern for final cleanup and decontamination of individual rooms and areas in the plant is dictated by doing the most contaminated areas first and improving the accessibility to staging areas. The need to use certain areas (e.g., scrap or waste handling facilities and hot maintenance) during DECON operations will delay their decommissioning.

For an individual area, accessibility to the equipment dictates the order of equipment removal. It is also desirable to remove the least contaminated equipment from the room earliest.

Figure G.2-1 illustrates the order and schedule for which DECON operations are postulated to take place. Certain operations are common for each area or room to be decontaminated. Equipment is carefully surveyed for radioactivity, before, during, and after all operations. Strict accountability records are kept for equipment and appurtences being decontaminated and shipped. Decommissioning personnel must wear appropriate protective clothing and breathing apparatus whenever contamination levels warrant it. Because external radiation doses to workers are relatively low, all decontamination operations are hands-on operations. Plastic tents are constructed when appropriate to contain contamination where equipment heavily contaminated with powders is being disassembled (e.g., grinder, electrostatic precipitator, hammermill, baghouse, etc.). Disassembly of piping, valves, pumps, ancilliary equipment, and ductwork external to equipment and hoods is conducted using strict contamination-control methods. Proper precautions and steps are taken to prevent spills or to confine contamination spills that may inadvertently occur. Health physics personnel observe and monitor operations to ensure that safe radiation and industrial practices are followed at all times.

To provide access, the equipment nearest the exit area of the room is generally removed first. Where possible, equipment external to the process equipment and hoods is removed first, followed by all external pipes, conduits, controls, etc., attached to the process equipment. The equipment is disconnected from exhaust ductwork and the open duct end is sealed or fitted with a filter to prevent spread of contamination.

The process equipment is disassembled and decontaminated in-situ as much as practicable to meet final disposition requirements. Portable greenhouses and tanks can be used to contain detergent washing and steam or water rinsing operations. Ends of piping, conduit, ductwork, etc., are always sealed or plugged with caps. Equipment is sealed with plastic, packaged, and removed to one of the staging areas for storage and shipment. Equipment that cannot be packaged in the original room is packaged in the staging areas.

Next, all remaining piping, electrical conduit, local filter systems, ductwork, etc., are dismantled and prepared for final disposition (most items

will be fully decontaminated and compacted in the original room). Ventilation exhausts and hood exhausts are plugged or fitted with filters to preclude contamination spread from back pressures. Light fixtures that cannot be sufficiently decontaminated are replaced with new units. The final step is to carefully survey for radiation and decontaminate the ceiling, walls, and floor with detergent cleanser, powered brushes, and handwipes. Floor tiles or strippable seal coat are removed. Where hot spots occur, the concrete floor is chipped and vacuumed to ensure complete decontamination. The recirculating room air ducts and fans are decontaminated and the filters are removed for incineration. A final survey is conducted prior to releasing the area for unrestricted use.

Details of decontaminating the individual work areas in their proper order are provided in the following subsections. Figures A.2-1 and A.2-2 in Appendix A provide plant layouts showing the area locations and decontamination sequences.

G.2.2.1 Drum Receiving - Area 1

The drum receiving area is used to store 0.21-m³ drums containing yellowcake. The only equipment to be cleaned and decontaminated are the storage pallets, the stacking equipment, and the fork-lift truck. The pallets are checked and sent to the incinerator if contaminated. The equipment is decontaminated and transferred for use at another facility.

G.2.2.2 ADU Slurry Receiving and Storage Area - Area 2

The first building area decontaminated is the ADU slurry receiving and storage area. Contamination is light in this area, which contains facilities and equipment for dissolving the ADU slurry with nitric acid. The sequence in which the equipment is dismantled, packaged, and removed is listed in Table G.2-1.

G.2.2.3 Sampling Station - Area 3

Sampling of dry concentrate is done by a dry-splitter sampling system. Drums are emptied through a falling-stream sampling unit. The general sequence in which the equipment is dismantled, packaged, and removed is listed in Table G.2-2.

TABLE G.2-1. General Sequence of Equipment Removal from the ADU Slurry Receiving and Storage Area

Sequence of Removal	Equipment Removal	Final Disposition (a)	Notes
1	2 ADR Storage Tanks	FDS	Decontaminate and Scrap
2	1 Receiving and Weigh Tank	FDS	Decontaminate and Scrap
3	2 Receiving Pumps	PDB	Disassemble and Package
4	2 Transfer Pumps	PDB	Disassemble and Package
5	Miscellaneous Equipment	PDB	Disassemble and Package

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.

TABLE G.2-2. General Sequence of Equipment Removal from the Sampling Station

Sequence of Removal	Equipment Removed	Final Disposition (a)	Notes
1	2 Drum Elevators	PDB	Disassemble and Package
2	2 Yellowcake Receiving Bins	FDS	Decontaminate and Scrap
3	2 Primary Samplers	PDB	Disassemble and Package
4	1 Secondary Sampler	PDB	Disassemble and Package
5	2 Yellowcake Transfer Screws	PDB	Disassemble and Package
6	2 Bucket Elevators	PDB	Disassemble and Package
7	2 Digester Feed Bins	FDS	Decontaminate and Scrap
8	Hood and Ductwork	70% FDB	Disassemble and Decontaminate
9	Stack and Forklift	FDS	Decontaminate and Transfer
10	Storage Racks	FDS	Decontaminate and Scrap

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.
% FDB - Percent Fully Decontaminated to Unrestricted Burial; Remainder to Licensed Burial Packaged in Plywood Boxes.

G.2.2.4 Laboratory - Area 4

The laboratory is located in the main plant building, as shown in Figure A.2-1. Most of the laboratory is used for testing and analysis of material that could cause contamination. The laboratory contains typical lab equipment, chemical hoods, isotopic instrumentation, physical testing, radiography scales, and other miscellaneous lab equipment. All of the equipment is decontaminated and excessed for unrestricted use. The general sequence of equipment removal is given in Table G.2-3.

TABLE G.2-3. General Sequence of Equipment Removal from the Laboratory

Sequence of Removal	Equipment Removal	Final Disposition (a)	Notes
1	3 Laboratory Hoods	FDS	Decontaminate and Release
2	Miscellaneous Equipment	FDS	Decontaminate and Release
3	Miscellaneous Work Benches, Tables, and Cabinets	FDS	Decontaminate and Release

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.

G.2.2.5 UF₄ Slurry Processing - Area 5

UF₄ slurry contained in drums is sampled and prepared for feed to the solvent extraction operation. The general sequence of dismantlement for this area is shown in Table G.2-4.

G.2.2.6 Ore Concentrate Dissolution - Area 6

Yellowcake and nitric acid are fed to digestors on a batch basis in this area. The general sequence of equipment removal for the ore concentrate dissolution area is shown in Table G.2-5.

G.2.2.7 Solvent Extraction Building - Area 7

This building contains equipment to recover uranium from the digester product. The general sequence of equipment removal is shown in Table G.2-6.

TABLE G.2-4. General Sequence of Equipment Removal from the UF₄ Slurry Processing Area

Sequence of Removal	Equipment Removal	Final Disposition ^(a)	Notes
1	Process Feed Hopper	PDB	Disassemble and Package
2	Mechanical Drum Pumper	PDB	Disassemble and Package
3	2 Digestion Tanks	FDS	Decontaminate and Scrap
4	2 Scrubbers	PDB	Disassemble and Package
5	2 Pumps	PDB	Disassemble and Package

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.

TABLE G.2-5. General Sequence of Equipment Removal from the Ore Concentrate Dissolution Area

Sequence of Removal	Equipment Removal	Final Disposition ^(a)	Notes
1	3 Digestion Tanks	FDS	Decontaminate and Scrap
2	2 Adjustment Tanks	FDS	Decontaminate and Scrap
3	Piping	FDS	Decontaminate and Scrap

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.

TABLE G.2-6. General Sequence of Equipment Removal from the Solvent Extraction Building

Sequence of Removal	Equipment Removal	Final Disposition ^(a)	Notes
1	6 Pumper Decanters	PDB	Disassemble and Package
2	2 Scrub Decanters	PDB	Disassemble and Package
3	2 Re-extraction Pulse Columns	PDB	Disassemble and Package
4	3 Extraction Waste Holding Tanks	FDS	Decontaminate and Scrap

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.

G.2.2.8 Uranyl Nitrate Concentration - Area 8

Concentration of uranyl nitrate is accomplished by the utilization of a primary and secondary evaporation system. The general sequence in which equipment is removed from the area is shown in Table G.2-7.

TABLE G.2-7. General Sequence of Equipment Removal from the Uranyl Nitrate Concentration Area

Sequence of Removal	Equipment Removal	Final Disposition (a)	Notes
1	Evaporator	PDB	Disassemble and Package
2	3 Boildown Tanks	FDS	Decontaminate and Scrap
3	3 Unit Filters	PDB	Disassemble and Package
4	Miscellaneous Equipment	PDB	Disassemble and Package
5	Hood and Ductwork	50% FDB	Decontaminate and Scrap
6	Piping	FDS	Decontaminate and Scrap

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.
% FDB - Percent Fully Decontaminated to Unrestricted Burial; Remainder to Licensed Burial Packaged in Plywood Boxes.

G.2.2.9 Denitration Area - Area 9

Uranyl nitrate is converted to UO_3 by heating in agitated trough denitrators. The general sequence in which denitrator equipment is removed for DECON is shown in Table G.2-8.

G.2.2.10 Nitric Acid Recovery - Area 10

A conventional absorption system is utilized for recovering nitrogen oxides from ore concentrate and fluorination ash dissolution and denitration of UNH to UO_3 . The general sequence of equipment removal is shown in Table G.2-9.

G.2.2.11 Reduction Systems - Area 11

Uranium trioxide is converted to UO_2 using dissociated ammonia. The general sequence of equipment removal is shown in Table G.2-10.

TABLE G.2-8. General Sequence of Equipment Removal from the Denitration Area

<u>Sequence of Removal</u>	<u>Equipment Removed</u>	<u>Final Disposition^(a)</u>	<u>Notes</u>
1	4 Denitrators	PDB	Disassemble and Package
2	4 Denitrator Scrubber Coolers	PDB	Disassemble and Package
3	4 Cooling Screws	PDB	Disassemble and Package
4	2 UO ₃ Vertical Elevators	PDB	Disassemble and Package
5	2 Hammer Mills	PDB	Disassemble and Package
6	2 Surge Bins	FDS	Decontaminate and Scrap
7	Hoods and Ductwork	FDS	Decontaminate and Scrap
8	Miscellaneous Equipment	PDB	Disassemble and Package

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.

TABLE G.2-9. General Sequence of Equipment Removal from the Nitric Acid Recovery Area

<u>Sequence of Removal</u>	<u>Equipment Removal</u>	<u>Final Disposition^(a)</u>	<u>Notes</u>
1	2 Tanks	FDS	Decontaminate and Scrap
2	2 Scrubbers	PDB	Disassemble and Package
3	2 Nitric Acid Absorbers	PDB	Disassemble and Package
4	2 Pumps	PDB	Disassemble and Package
5	Miscellaneous Equipment	PDB	Disassemble and Package
6	Piping	FDS	Decontaminate and Scrap

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.

TABLE G.2-10. General Sequence of Equipment Removal from the Reduction System Area

Sequence of Removal	Equipment Removal	Final Disposition (a)	Notes
1	2 UO ₃ Storage Bins	FDS	Decontaminate and Scrap
2	2 Screw Feeders	PDB	Disassemble and Package
3	4 Fluid Bed Reduction Reactors	PDB	Disassemble and Package
4	2 Ammonia Disassociators	PDB	Disassemble and Package
5	Miscellaneous Equipment	PDB	Disassemble and Package
6	Hood and Ductwork	FDS	Decontaminate and Scrap

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.

G.2.2.12 Hydrofluorination - Area 12

Uranium dioxide is converted to uranium tetrafluoride by reaction with anhydrous HF at high temperature. The general sequence of equipment removal is shown in Table G.2-11.

TABLE G.2-11. General Sequence of Equipment Removal from the Hydrofluorination Area

Sequence of Removal	Equipment Removed	Final Disposition (a)	Notes
1	2 Screw Feeders	PDB	Decontaminate and Package
2	2 UO ₂ Filter Bins	FDS	Decontaminate and Scrap
3	2 1st-Stage Fluid Bed Hydrofluorinators	75% FDB	Disassemble and Decontaminate
4	2 2nd-Stage Fluid Bed Hydrofluorinators	70% FDB	Disassemble and Decontaminate
5	2 UF ₄ Seal Hoppers	PDB	Decontaminate and Package
6	2 Interstage Hoppers	PDB	Decontaminate and Package
7	2 UF ₄ Purge Conveyors	PDB	Decontaminate and Package
8	2 Conveyors	PDB	Decontaminate and Package
9	2 UF ₄ Storage Hoppers	60% FDB	Disassemble and Decontaminate
10	2 Conveyors	60% FDB	Disassemble and Decontaminate
11	Piping	FDS	Decontaminate and Scrap

(a) PDB - Partially Decontaminated, Licensed Burial.
FDS - Fully Decontaminated, Unrestricted Scrap.
% FDB - Percent Fully Decontaminated to Unrestricted Burial; Remainder to Licensed Burial Packaged in Plywood Boxes.

G.2.2.13 Fluorination - Area 13

UF₄ is converted to UF₆ by reaction with elemental fluorine. The general sequence of equipment removal is shown in Table G.2-12.

TABLE G.2-12. General Sequence of Equipment Removal from the Fluorination Area

Sequence of Removal	Equipment Removal	Final Disposition (a)	Notes
1	5 Primary Fluorination Towers	60% PDB	Disassemble and Decontaminate
2	2 Primary Conveyors	PDB	Decontaminate and Package
3	2 Primary Filters	PDB	Decontaminate and Package
4	6 Primary Cold Traps	PDB	Decontaminate and Package
5	2 Product Cylinders.	PDB	Decontaminate and Package
6	2 Backup Filters	PDB	Decontaminate and Package
7	2 Feed Hoppers	PDB	Decontaminate and Package
8	2 Cleanup Reactors	PDB	Disassemble and Decontaminate
9	2 Backup Conveyors	70% FDB	Disassemble and Decontaminate
10	2 Backup Filters	60% FDB	Disassemble and Decontaminate
11	2 Secondary Cold Traps	PDB	Decontaminate and Package
12	Hood and Ductwork	FDS	Disassemble and Scrap
13	Piping	FDS	Disassemble and Scrap

(a) PDB - Partially Decontaminated, Licensed Burial.

FDS - Fully Decontaminated, Unrestricted Scrap.

% FDB- Percent fully Decontaminated to Unrestricted Burial; Remainder to Licensed Burial Packaged in Plywood Boxes.

G.2.2.14 Cylinder Loading - Area 14

Loading of cylinders used for transport of UF₆ is completed in this area. The equipment needed for cylinder loading is dismantled, packaged, and sent to licensed burial. Miscellaneous tools, test equipment, etc., are decontaminated and excessed for unrestricted use.

G.2.2.15 Fluorine Generation - Area 15

The general sequence of equipment removal in the fluorine area is shown in Table G.2-13.

TABLE G.2-13. General Sequence of Equipment Removal from the Fluorine Generation Area

Sequence of Removal	Equipment Removed	Final Disposition (a)	Notes
1	2 Waste Gas Burners	60% FDB	Disassemble and Decontaminate
2	120 Fluorine Cells	PDB	Package in Plywood Boxes
3	2 Surge Tanks	60% FDB	Disassemble and Decontaminate
4	4 Fluorine Filters	PDB	Package
5	2 HF Condensers	70% FDB	Disassemble and Decontaminate
6	2 HF Backup Filters	FDS	Disassemble and Scrap
7	2 Hydrogen Surge Tanks	FDS	Disassemble and Scrap
8	2 Fluorine Filters for Backup System	FDS	Disassemble and Scrap
9	2 HF Storage Tanks	FDS	Disassemble and Scrap
10	Miscellaneous Equipment	PDB	Package
11	Instrument Controls	PDB	Package
12	Piping	FDS	Disassemble and Scrap

(a) PDB - Partially Decontaminated, Licensed Burial.

FDS - Fully Decontaminated, Unrestricted Scrap.

% FDB - Percent Fully Decontaminated to Unrestricted Burial; Remainder to Licensed Burial Packaged in Plywood Boxes.

G.2.2.16 F₂ Cell Rework - Area 16

This area has a low level of contamination. A thorough radiation survey is made and any spots of contamination found are cleaned. Equipment is excessed for unrestricted use.

G.2.2.17 Instrument Repair Shop - Area 17

In this shop, which is maintained very clean, electrical and electronic maintenance work is performed on restricted-area instrumentation equipment and components. The shop contains equipment such as scopes, testers, brazing and soldering, spare components, cleaning, and standards. All of this equipment is fully decontaminated and excessed for unrestricted use.

G.2.2.18 Ammonia Cracking - Area 18

This area has very slight contamination. Three ammonia dissociators provide hydrogen gas which is delivered to the reduction reactors. The equipment is disassembled, fully decontaminated, and excessed for unrestricted use.

G.2.2.19 Waste Gas Disposal - Area 19

Waste gases from the hydrofluorinator backup filter and the HF condenser are combined with off gases from the waste gas burner servicing the fluorine plant and the fluorinator system off gases. The gases are then scrubbed with water before being vented. This equipment is disassembled, partially decontaminated, and sent to burial or scrap.

G.2.2.20 Hydrogen Fluoride Condensing - Area 20

The hydrogen fluoride condensing system recovers HF from the fluorine production gas streams. Three ammonia dissociators are installed outdoors to provide hydrogen gas. The equipment is dismantled, decontaminated, and packaged for disposal.

G.2.2.21 Temporary Radwaste Facility - Area 21

This temporary facility is used to process contaminated liquid waste from decommissioning the plant. Liquids are collected in collection tanks and pumped through a centrifuge where suspended uranium compounds and other solids are removed. The clarified water enters a quarantine tank where it is sampled. If the liquid meets internal action guides, it is pumped on to the process liquid waste disposal system. If not, it is returned to the system for rework. Solids removed from the centrifuge are sealed in containers while they are still wet.

The equipment is dismantled and decontaminated sufficiently for shipment offsite to an LLW burial ground.

G.2.2.22 Temporary Decontamination Facility - Area 22

This facility is used to sort solid wastes into combustible and noncombustible categories. Combustible waste is packaged in plastic and sent to the incinerator facility. Noncombustible waste, equipment, etc., is either decontaminated or packaged for disposal at a licensed burial ground.

The decontamination room contains sorting tables, radiation survey equipment, U-material scanners, cleaning sinks, decontamination spray booths, and compacting and packaging equipment, some of which are fully decontaminated and sold as unrestricted excess equipment. The only exception is uranium survey and scanning equipment, which is sold for restricted use elsewhere.

G.2.2.23 Incinerator Facility - Area 23

This facility is left until the last decontamination phase of the plant, because it is used to burn combustible cleanup and decontamination items generated during dismantlement activities. Uranium is recovered from the ashes and HEPA filters.

The incinerator facility contains an input uranium scanner, a shredder, a blow tube unit, a vortex incinerator chamber, an offgas scrubber and filters, an ash recovery system, an output uranium scanner, and packaging equipment to prepare ashes and filters for offsite reprocessing to recover uranium.

The scanners and packaging equipment are decontaminated and excessed for further use. The remaining equipment is partially decontaminated, packaged, and shipped to a licensed burial ground.

G.2.2.24 Laundry and Change Rooms - Area 24

The laundry room has sorting and folding tables, washers, and dryers. The tables are decontaminated and excessed. The washers and dryers and HEPA filters are packaged and shipped to a licensed burial ground.

The last portion of the building to be dismantled and decontaminated contains the change rooms. These areas contain very little, if any, contamination.

The survey equipment is decontaminated and shipped to another restricted-use facility for reuse. The remaining equipment (toilets, showers, benches, lockers, cabinets, etc.) are fully decontaminated. These rooms are decontaminated using standard procedures and are released for unrestricted use to support new activities within the plant.

G.2.2.25 Final Cleaning of the Entire Facility

The original unrestricted building areas (offices, computer rooms, shipping, halls, etc.) are given a comprehensive final survey and thorough cleanup to ensure that no spots of contamination exist. The entire building is released for unrestricted use upon completion of this campaign. The timing should be such that decontamination of the full site around the building (Section G.2.3 items included) is also completed for unrestricted use.

G.2.3 Site Decontamination Operations (Excludes Main Building and Solvent Extraction Building)

This section describes the decontamination activities employed to prepare the entire site for unrestricted use. These activities are performed in parallel with the main building decontamination activities.

The general order of site decontamination is:

1. extraction waste treatment system
2. fluoride waste effluent treatment system
3. miscellaneous buried waste
4. drum storage area
5. cylinder storage and washing area
6. plant stack
7. overall final site survey and cleanup.

G.2.3.1 Extraction Waste Treatment System

The waste stream from the extraction system is primarily a solution of ammonium nitrate, nitric acid, metallic salts, minute quantities of uranium, and the radioactive daughter products of normal uranium decay. This stream is combined with spent sodium hydroxide from the solvent treatment and

miscellaneous digester scrubber systems, along with any recovered weak acids. This liquid is neutralized by ammonia and impounded in storage lagoons. Neutralization of the extraction waste removes residual uranium and thorium. Since the effluents come from plant operation, decontamination work can start on this system after conversion operations are stopped. A schematic of the extraction waste system is shown in Figure G.2-2.

FIGURE G.2-2. Schematic of the Extraction Waste Treatment System

The pipelines and tanks are first thoroughly flushed by chemical and water rinses appropriate for the pipe and equipment construction materials. The entire system is then drained and blown out.

The pipeline is disassembled in sections, using procedures to prevent contamination from spreading. Pipe containing contamination is bagged in plastic and taken to a portable decontamination station where powered rotary brushes and detergent are used for cleanup. Heavily scaled pipes are cleaned with rotary wire brushes or sand-blasting units. Pumps, valves, etc., are packaged for disposal at a licensed burial ground.

Tanks are cleaned manually with brushes, wipes, and detergents. Small tanks that cannot be cleaned sufficiently by flushing are sectioned, decontaminated, and scrapped.

Other pieces of equipment such as the pumps are disassembled and manually cleaned in scrub tanks or with handwipes. Such equipment is excessed or scrapped out.

The liquids in the lagoons are allowed to evaporate or are concentrated using the submerged combustion burners. After the lagoons are drained, the residual material is removed from the site for disposal. The extraction waste is loaded on 10-m³ dump trucks and covered with plastic sheeting for transport to a mill for reprocessing, or to LLW burial, or to a commercial dump. Contaminated sections of liner are cut out and packaged for shipment to a licensed burial ground. Earth contaminated below the clay liners is also packaged and shipped to licensed burial. The liner is punctured in place to allow drainage, and the cavity is filled with earth. Native vegetation is planted on the restored surfaces.

The general sequence in which the system is decontaminated is listed in Table G.2-14.

TABLE G.2-14. General Sequence of Dismantling the Extraction Waste Effluent Treatment System

<u>Sequence of Removal</u>	<u>Equipment Removed</u>	<u>Final Disposition^(a)</u>	<u>Notes</u>
1	Pipelines Between Tanks, Equipment and Lagoons	FDS	Decontaminate and Scrap
2	Pumps, Valves and Controls	PDB	Package in Plywood Boxes
3	3 Extraction Waste Storage Tanks	FDS	Disassemble and Decontaminate
4	Ammonia Reaction Tank	FDE	Decontaminate and Excess
5	4 Lagoon Heaters	PDB	Package and Ship
6	4 Settling Lagoons (clay-lined)	Remove and PDB	Remove for Recovery; Decontaminate Liner; Cut Out Porous Liner Sections and Remove Along with Contaminated Earth.

(a) FDS - Fully Decontaminate, Unrestricted Scrap.
 FDE - Fully Decontaminate, Unrestricted Excess.
 PDB - Partially Decontaminate, Licensed Burial.

G.2.3.2 Fluoride Waste Effluent Treatment System

The effluent is generated by the hydrofluoric acid scrubber. This fluoride waste stream is combined with waste sodium carbonate solutions originating in the fluorine cell rework area, with any acid spilled in the HF vaporizer room sump, and with laboratory wastes. The combined stream is treated with lime, which neutralizes acids and precipitates fluorides as calcium fluoride.

The alkaline sludge is permitted to settle in a retention basin to permit flocculation and sedimentation. The flow is treated with sulfuric acid to adjust the pH and precipitate excess calcium. It is then permitted to clarify. The clarified treated waste overflows and is combined with clean bypass water and the sewage lagoon overflow. A concrete stilling basin at the point of combination allows for mixing of the flow and controlled release through a flume so that the rate of discharge can be measured.

Other plant streams which are combined with the treated fluorine effluent before being discharged to the river include sanitary and domestic waste liquids.

The fluoride waste treatment system is shown schematically in Figure G.2-3. The effluent is generated primarily from the hydrofluorination system offgas scrubber.

FIGURE G.2-3. Schematic of the Fluoride Waste Treatment System

Dismantlement of components in the fluoride waste treatment system is handled similar to dismantlement of components in the extraction waste treatment system.

Outlined in Table G.2-15 is the sequence of dismantling the fluoride waste treatment system. The cleanout of the fluoride lagoons may also be considered as an option in the earlier final inventory closeout phase of the project.

TABLE G.2-15. General Sequence of Dismantling the Fluoride Waste Effluent Treatment System

<u>Sequence of Removal</u>	<u>Equipment Removed</u>	<u>Final Disposition^(a)</u>	<u>Notes</u>
1	Pipelines Between Tanks, Equipment and Lagoons	FDS	Decontaminate and Scrap
2	Pumps, Valves and Controls	PDB	Package in Plywood Boxes
3	250-m ³ Storage Tank	FDE	Decontaminate and Excess
4	2 45-m ³ Lime Treatment Tanks	FDE	Decontaminate and Excess
5	2 Fluoride Settling Lagoons (clay-lined)	SIP	Stabilize and Revegetate
6	Equilization Basin	SIP	Stabilize and Revegetate
7	Stilling Basin	SIP	Stabilize and Revegetate

(a) FDS - Fully Decontaminate, Unrestricted Scrap.
 FDE - Fully Decontaminate, Unrestricted Excess.
 PDB - Partially Decontaminate, Licensed Burial.
 SIP - Stabilize in Place.

G.2.3.3 Miscellaneous Buried Waste

Radioactive waste materials such as scrapped equipment, gloves, respirators, and other miscellaneous contaminated solids are buried onsite in accordance with the provisions of 10 CFR 20.304. This material is removed and packaged for burial at a licensed low-level waste burial ground.

G.2.3.4 Drum Storage Area

The drum storage pads are thoroughly surveyed. All uranium containers are removed during the inventory cleanout activities of Section G.1. Contamination is not anticipated, but some hot spots might occur. These spots are removed by scrubbing and chipping, if necessary. Any radioactive materials are packaged in steel drums and shipped to a licensed burial ground.

G.2.3.5 Cylinder Storage and Washing Area

The cylinder storage and washing area is thoroughly surveyed. All UF_6 cylinders are removed during the inventory cleanout activities of Section G.1. Any contaminated spots will be removed by scrubbing and chipping. Any radioactive materials removed are packaged in steel drums and shipped to licensed burial.

G.2.3.6 Plant Stack

The plant stack will be demolished and the material surveyed. If any material is contaminated, it will be packaged and shipped to low-level waste burial.

G.2.3.7 Overall Final Site Survey and Cleanup

A thorough general radiation and chemical survey of the entire site is made to ensure that no inadvertent contamination remains. Spots with excessive contamination are decontaminated and the contaminated materials are packaged in 19-l drums for shipment to a licensed burial ground.

At the conclusion of this step and the final fuel fabrication building survey, the entire site is available for unrestricted use.

G.3 PASSIVE SAFSTOR ACTIVITIES

After completion of final processing inventory cleanout, the remaining uranium is spread primarily throughout the processing areas in the main building and solvent extraction building, in locations relatively inaccessible without initiating dismantlement (e.g., crannies, equipment internals, between equipment buttressed to walls, exhaust ducts, etc). The total amount of

residual uranium is estimated to be about 1200 kg. This amount should not present a problem for the passive SAFSTOR alternative as long as access to the facility is restricted.

Outside the facility, the main concern of this decommissioning alternative is to take steps to secure the waste treatment lagoons until restart of the plant or deferred decontamination is undertaken.

The main activities undertaken to prepare the facility for safe storage are:

- stabilization of waste treatment facilities
- deactivation of unneeded equipment
- isolation of contaminated areas.

G.3.1 Stabilization of Waste Treatment Facilities

The extraction waste and fluoride lagoons require stabilization after UF_6 conversion is discontinued. The effluents settling out in the lagoons contain uranium that must be confined to the site during the safe storage period.

The extraction waste lagoons are drained and residual solids are packaged and shipped to a commercial uranium milling facility for recovery of the uranium.

The fluoride lagoons are drained and weighted plastic covers are placed over the remaining solids. A possible alternative is to remove residual materials from the lagoons and ship them to an offsite licensed burial ground.

The remainder of the waste treatment facilities are comprised of pipes, pumps, valves, tanks, etc., that provide a closed system to contain any residual uranium left after being flushed.

G.3.2 Deactivation of Unneeded Equipment

Essential systems (such as lighting, utilities, radiation detection alarms, security monitoring, fire detection and portable fire fighting equipment) remain in operation during the safe storage period. All other equipment that is unneeded during safe storage is deactivated for reasons of safety, energy conservation, and cost savings. Equipment is left in a condition that permits startup or salvage at a later date.

Equipment deactivation steps are:

- conducting a safety audit to ensure that all flammable and other hazardous materials are removed
- closing and securing valves
- disconnecting electrical, compressed air, and other utilities
- shutting down ventilation systems in contaminated areas, and capping exhaust stacks
- sealing process hoods
- tagging all deactivated equipment and systems for identification and status.

G.3.3 Isolation of Contaminated Areas

Portions of the main building containing uranium contamination are isolated by the installation of security locks on entryways. Indirect access routes, however unlikely, are also secured (e.g., roof access panels and pipe trenches). Signs are posted to warn against unauthorized entry. The same steps are taken for the incinerator and waste treatment buildings. Fences around the extraction waste treatment lagoons and fluoride storage lagoons are also secured and posted with warning signs.

G.3.4 Final Preparations for Safe Storage

Final preparations for safe storage are:

- installing and/or upgrading uranium sampling, monitoring systems and radiation alarms
- installing or relocating intrusion alarms
- shipping all recovered uranium materials offsite for disposition
- performing a comprehensive radiation survey of both restricted and unrestricted areas at the site. Any spots of excessive contamination in the areas to be unrestricted are decontaminated
- training of personnel and contractors
- conducting final release survey.

No additional steps are needed to meet the requirements of passive SAFSTOR. All radioactivity is confined onsite. Periodic surveillance, environmental monitoring, and security by the manager and contractor will maintain the confinement of radioactivity. Adjacent facilities and unrestricted areas in the fuel plant can be put to other uses.

G.4 DECOMMISSIONING ACTIVITIES FOR ONSITE DISPOSAL OF LAGOON WASTE

Decommissioning begins with planning and preparation activities that include a radiological survey of the site, the preparation of documentation describing anticipated environmental effects of proposed decommissioning procedures, and the formulation of written work procedures.

The initial radiological survey provides data to guide the planning of decommissioning operations. The survey includes the following measurements:

- measurements of external gamma radiation levels at 1 m above the ground surface on the lagoon and on the site
- surface soil samples at several onsite locations
- measurements of the rate of radon emanation from the waste
- measurements of the concentrations of radon and daughters in air on the site
- measurements of gamma radiation at various depths in core holes drilled into the soil near the edge of the lagoon
- subsurface soil samples obtained from these core holes.

At the reference site, cores are drilled to a depth of 8 meters and soil samples are obtained from each core. The samples are sent to a commercial laboratory for analysis of ^{226}Ra , ^{232}Th , and ^{238}U concentrations.

Lagoon waste stabilization details are described in Section G.5. To provide an example of the costs of stabilization, the following set of techniques is assumed for stabilization of the reference plant lagoon wastes. The lagoon waste is covered with a 50-mm-thick layer of asphalt. This asphalt layer is then covered with 1 m of soil. The soil is mounded slightly at the center of the lagoon to allow water to drain from the soil cover and to prevent the

accumulation of runoff from rainfall or snow melt. After compaction and contouring of the soil cover, the area is seeded with grass.

Long-term care activities following stabilization include administrative control, site maintenance, environmental monitoring, and vegetation management.

G.5 LAGOON WASTE STABILIZATION DETAILS

Various stabilization techniques can be used to reduce the potential for radionuclide migration from the waste lagoons analyzed in this study. Some stabilization techniques are described in this section. These techniques are:

- soil cover
- site topography adjustment
- revegetation
- leachability reduction.

A more complete description of stabilization techniques is given in Reference 1.

G.5.1 Soil Cover

A layer of soil (backfill) can be added to a contaminated site to provide a cover over the radioactive material. The backfill used is generally selected to provide a suitable base for subsequent revegetation of the area. The material is hauled in, dumped, and graded to form a layer of uniform specified depth. Standard earthmoving techniques are used. The soil is then compacted either by packing the area using heavy rolling equipment (e.g., a sheepsfoot roller,^(a) wobbly-wheel roller,^(b) or road roller) or by vibrating the earth with a vibrating baseplate compactor. In addition, other measures can be taken to aid in compaction, the most common being the use of water sprays to wet the soil. Little or no surface preparation is needed prior to backfilling and compaction of an area, although removal of vegetation and debris is desirable. Reference 2 provides a detailed treatment of fill compaction.

The new surface can be graded to preserve the original site contours, or new contours can be established. Site contours are chosen to control the

(a) A sheepsfoot roller is a cylindrical steel drum to which knob-headed spikes are fastened.

(b) A wobbly-wheel roller is a roller with freely suspended pneumatic tires.

drainage from the area and to reduce the potential for the accumulation of water on the surface. After completion of the contouring operation, the surface is revegetated or otherwise stabilized.

Backfilling and compaction can be used as a control measure against subsidence. Backfill returns the surface to its original level, and compaction reduces the extent of subsidence that might be caused by future settling.

The addition of a layer of soil can be used to reduce the background radiation level existing on the site from contaminated surface soil or from tailings dumped on the ground. The thickness of backfill required depends on the level of surface contamination and the radionuclides involved. In general, a soil thickness of 0.3 m to 3.0 m is required.

G.5.2 Site Topography Adjustment

Site topography adjustment is the grading, scraping, or other movement of surface soils to alter site contours. Site topography is adjusted using standard earthmoving and surface contouring techniques. After completion of the contouring operation, the surface is stabilized as desired, using one of the surface stabilization techniques described in this section.

Adjustment of site topography can be very effective in reducing radionuclide migration. Topography is a factor in both wind and water erosion, and adjustment therefore provides a method of reducing erosion damage. The alteration of site contours is a means of controlling runoff from rainfall and snow melt, and therefore is useful in adjusting the hydrological parameters of a site.

G.5.3 Revegetation

Revegetation is the establishment of a vegetative ground cover on a site whose surface has been disturbed. A variety of vegetation types and species can be used, depending on soil and climate conditions and also on the results desired. Shallow-rooted plants are preferred in order to limit plant-root penetration into buried radioactive material.

Before revegetation begins, soil tests are made at the site to determine the plant species to be used and the nutrient balance in the soil. Then, after

the surface to be revegetated is graded or leveled as desired and cleared of debris, the area is planted with selected vegetation species. Use of fertilizers and soil amendments to improve soil texture and nutrient balance is common, as is the use of mulches and/or chemical stabilizers to conserve moisture and protect the surface until the vegetation becomes established.

A vegetated surface must be managed to ensure the continued viability of the vegetative community and to provide remedial measures for incidental problems. A vegetation management program can include, but is not limited to, the following elements: herbicides, acting at the surface and/or subsurface to control undesirable plant growth and to limit plant-root penetration; use of competing plant species to control growth of undesirable species; periodic clearing of undesirable vegetation from the site; use of bacterial and/or insect controls to limit the growth of undesirable species; and replanting of areas damaged by erosion, pests, or human activities.

Revegetation can be used to control wind and water erosion of the ground surface.^(3,4) It also affects the site moisture balance by reducing runoff and increasing moisture return to the atmosphere through evapotranspiration. Revegetation may, in some cases, reduce mass wasting by anchoring the soil. One possible disadvantage of revegetation is that plant roots may penetrate areas where radioactivity is located.

The erosion protection afforded a site by a vegetation cover increases as the plant community becomes more established. This results in a gradual improvement of the site over several years, until the plant community reaches maturity.

G.5.4 Leachability Reduction

Leachability reduction involves the injection of suitable material into the buried radioactive material (e.g., a tailings pile) to chemically and/or physically bond the radionuclides into a stable mass, thus reducing leaching. The technique involves the injection of grout material into the radioactive waste and the surrounding soil. Possible grouting materials include cement, clays, asphalts, bitumens, silicates, lignochromes, lignosulfates, epoxy resins, acrylamide, polyester resins, polyphenolics, resorcinolformaldehyde, and other chemical polymers.⁽⁵⁾ These materials are injected by pumping through distribution pipes drive into the soil. Injection techniques are described in Reference 6.

By reducing waste leachability, radionuclide releases caused by hydrological action are decreased by limiting or eliminating dissolution of the radionuclides. Grouting to reduce leachability is a relatively expensive operation with costs ranging as high as \$350/m³.⁽⁵⁾ A useful life of 25 to 100 years is anticipated for the technique, assuming no regular maintenance.

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APPENDIX H

COST ASSESSMENT DETAILS

This appendix provides the details for the cost estimate information given in Section 10. Section H.1 gives the basic data used for making cost estimates. Cost details for DECDN and for preparations for the safe storage period of passive SAFSTOR are presented in Sections H.2 and H.3, respectively. In section H.4 cost estimates for onsite burial of the lagoon wastes is provided.

H.1 COMPILATION OF UNIT COST FACTORS

For ease in applying the information developed during this study to other decommissioning situations, the basic data used for making the decommissioning cost estimates are presented in this section. Categories for which basic cost data are given include salaries, waste management costs, and costs of special equipment and services. All data are based on early 1981 costs.

H.1.1 Unit Manpower Costs

Salary data for typical decommissioning staff positions are given in Table H.1-1. The original 1978 data base has been adjusted by a factor of 1.19 to account for escalation between 1978 and 1981 based on the Handy-Whitman Index.⁽¹⁾

H.1.2 Unit Waste Management Costs

Waste management costs include the costs of shipping containers, transportation costs, and waste disposal charges.

H.1.2.1 Unit Shipping Container Costs

The cost of containers for packaging radioactive materials for offsite shipment is a major cost item for decommissioning. The containers assumed to be used in this study, together with their unit costs, are listed in Table H.1-2.

H.1.2.2 Unit Transportation Costs

Transport of radioactive waste materials from the UF_6 conversion plant to a commercial low-level waste burial site is assumed to be by truck. The distance

TABLE H.1-1. Decommissioning Labor Cost Data

<u>Position</u>	<u>Base Pay (\$/yr)</u>	<u>Assume Overhead Rate (%)</u>	<u>Cost (\$/yr)</u>	<u>Reference</u>
<u>Management and Support Staff</u>				
Decommissioning Superintendent	52 000	70	89 100	a
Decommissioning Engineer	44 700	70	76 000	a
Asst. Decommissioning Engineer	30 800	70	52 400	a
Secretary	16 100	50	24 200	d
Clerk	16 100	50	24 200	d
Health and Safety Supervisor	35 200	70	59 800	d
Health Physicist	27 600	70	46 900	d
Industrial Safety Specialist	30 800	70	52 400	d
Radioactive Shipment Specialist	23 100	70	39 300	d
Procurement Specialist	23 100	70	39 300	d
Contracts and Accounting Specialist	27 700	70	47 100	a
Accounting Clerk	16 100	50	24 200	d
Security Supervisor	32 900	70	55 900	d
Security Shift Supervisor	21 400	70	36 400	d
Security Patrolman	16 900	50	25 400	b
Quality Assurance Supervisor	30 700	70	52 200	d
Quality Assurance Engineer	27 600	70	46 900	d
Quality Assurance Technician	18 500	50	27 800	d
Control Room Attendant	23 000	50	34 500	d
Tool Crib Attendant	18 500	50	27 800	c
Protective Equipment Attendant	18 500	50	27 800	d
Consultant	100 000	--	100 000	d
<u>Decommissioning Workers</u>				
Shift Engineer	30 700	70	52 200	d
Craft Supervisor	27 600	70	46 900	d
Craftsman	21 400	50	32 100	c
Crew Leader	26 100	70	44 400	c
Utility Operator	21 400	50	32 100	c
Laborer	20 600	50	30 900	b
Health Physics Technician	20 000	50	30 000	c

- (a) U.S. Department of Labor, Bureau of Labor Statistics, Bulletin March 1975.⁽²⁾
 (b) R. S. Means Company, Building Construction Cost Data - 1975, 33rd Edition.⁽³⁾
 (c) Hanford Atomic Metal Trades Council Pay Scales.⁽⁴⁾
 (d) Author's Estimate.

TABLE H.1-2. Unit Costs of Shipping Containers

Description	Burial Volume (m ³)	Estimated Unit Cost (\$)
Standard Steel Drum 0.21 m ³ , 23 kg empty	0.21	30
Small Steel Drum 0.11 m ³ , 18 kg empty	0.11	20
Polyethylene Drum Liner	(a)	1
Fiberglassed Plywood Box 1.2 m x 1.2 m x 2.4 m, 175 kg empty	3.64	100
Fiberglassed Plywood Box	Variable	40/m ³ of surface

(a) Included in outer steel drum, no added burial volume.

from the plant to the low-level waste burial site is assumed to be 800 km. Truck rate schedules that form the basis for transportation costs used in this study are shown in Table H.1-3. The table is reproduced from the published rates⁽⁵⁾ of a carrier licensed to transport radioactive materials.

The gross vehicle weight (GVW) for normal shipments by truck is assumed to be less than 20.41 Mg. The maximum allowed GVW varies from state to state, ranging from 30.21 Mg to 45.36 Mg. Overweight charges by states vary widely.⁽⁵⁾ For this study, the maximum allowed GVW and the overweight charges for the state of Washington are assumed to apply (those rates being among the highest). These overweight changes are shown in Table H.1-4. An additional surcharge of \$0.131 per km is imposed by the carrier for shipments with GVW between 21.32 Mg and 38.55 Mg. Shipments in excess of 38.55 Mg GVW require special equipment and special permission. Costs for these cases would have to be determined on a case-by-case basis.

For this study, the GVW of an unloaded exclusive-use van or tractor-trailer is assumed to be 12.70 Mg. Therefore, the payload per shipment in an exclusive-use van is 7.7 Mg, legal weight, and 20.41 Mg, overweight.

To ensure rapid turnaround on shipments, a second driver may be required, costing an additional \$0.093/km (\$0.15/mi).⁽⁵⁾

TABLE H.1-3. Transportation Rates for Legal-Size and -Weight Shipments^(a,b,c)

Kilometers One-Way (Not Over)	Rate in Cents per Kilometer			Kilometers One-Way (Not Over)	Rate in Cents per Kilometer		
	Column 1(d)	Column 2(e)	Column 3(f)		Column 1(d)	Column 2(e)	Column 3(f)
160	233	244	168	1200	86	103	71
200	214	226	155	1280	82	100	71
240	196	209	143	1360	81	99	71
280	179	192	133	1440	80	98	71
320	155	169	121	1520	79	97	71
360	147	162	115	1600	77	95	71
400	141	156	108	1760	77	94	71
440	134	150	101	1920	77	94	71
480	128	144	96	2080	77	93	71
520	125	141	91	2240	77	92	71
560	121	137	88	2400	77	92	71
600	116	132	84	2560	77	91	71
640	111	128	82	2720	77	91	71
680	108	124	80	2880	77	90	71
720	102	119	78	3040	77	89	71
760	100	117	76	3200	77	89	71
800	96	114	75	3360	77	88	71
880	94	111	73	3520	77	88	71
960	92	109	71	3680	77	87	71
1040	89	106	71	3840	77	86	71
1120	87	104	71	4000	77	86	71

and
Beyond

- (a) Reproduced from the published rates of a carrier⁽⁵⁾ licensed to transport radioactive materials.
- (b) Effective August 15, 1980.
- (c) Rates do not include a fuel surcharge, which amounted to 18% of the base rate as of February 13, 1981.
- (d) Column 1 rates applicable to one-way shipments having a destination east of the Mississippi River.
- (e) Column 2 rates applicable to one-way shipments having a destination west of the Mississippi River.
- (f) Column 3 rates apply to continuous excursion moves in which a subsequent shipment is made available to the carrier within 24 hours after arrival at the point of loading or unloading.

TABLE H.1-4. Additional Charges When Gross Vehicle Weight Exceeds 21.32 Mg^(a,b)

Gross Vehicle Weight (Mg)	State Surcharge (\$)	Carrier Surcharge (\$)	Total Overweight Surcharge (\$)
21.32 to 23.12	10 + 0.031/km	0.131/km	10 + 0.162/km
23.13 to 25.84	10 + 0.062/km	0.131/km	10 + 0.193/km
25.85 to 28.56	10 + 0.093/km	0.131/km	10 + 0.224/km
28.57 to 31.28	10 + 0.155/km	0.131/km	10 + 0.286/km
31.29 to 34.00	10 + 0.218/km	0.131/km	10 + 0.349/km
34.01 to 36.72	10 + 0.280/km	0.131/km	10 + 0.411/km
36.73 to 38.55	10 + 0.373/km	0.131/km	10 + 0.504/km

(a) State surcharge is based on rates for the state of Washington

(b) Carrier surcharge is based on the published rates⁽⁵⁾ of a carrier licensed to transport radioactive materials.

H.1.2.3 Unit Waste Disposal Costs

Radioactive wastes generated during the decommissioning of a UF₆ plant will be disposed of by burial at a licensed low-level waste burial site.

The costs for low-level waste burial used in this study are based on a November 17, 1980 price list⁽⁶⁾ from U.S. Ecology, Inc., which operates burial sites at Richland, Washington, and Beatty, Nevada. The costs in this study are given in Table H.1-5.

H.1.3 Unit Costs of Special Equipment and Supplies

The various types of special equipment needed for decommissioning activities are described in some detail in Section F.2 of Appendix F. Estimated unit costs for these equipment items and for decommissioning supplies are given in Table H.1-6. Costs of construction-type items (hoists, cranes, lifts, etc.) are based on costs shown in the 1981 catalog of building construction costs published by R. S. Means Company.⁽⁷⁾

TABLE H.1-5. Commercial Shallow-Land Burial Charges^(a,b)

I. DISPOSAL CHARGES, NON-TRU WASTE

A. Steel Drums, Wood Boxes

Container Surface Dose Rate (R/hr)(c)	Price per Unit Volume (\$/m ³)
0.00 to 0.20	307.20
0.201 to 1.00	335.45
1.01 to 2.00	376.05
2.01 to 5.00	459.05
5.01 to 10.00	542.00
10.01 to 20.00	702.65
20.01 to 40.00	870.40
40.01 to 60.00	1332.95
60.01 to 80.00	1601.30
80.01 to 100.00	1765.50
>100	by request

Minimum disposal charge: \$224.00/shipment

B. Disposable Liners

Container Surface Dose Rate (R/hr)(c)	Surcharge per Liner (\$)	Price per Unit Volume (\$/m ³)
0.00 to 0.20	None	307.20
0.201 to 1.00	119.00	307.20
1.01 to 2.00	292.00	307.20
2.01 to 5.00	411.00	307.20
5.01 to 10.00	594.00	307.20
10.01 to 20.00	758.00	307.20
20.01 to 40.00	941.00	307.20
40.01 to 60.00	1116.00	307.20
60.01 to 80.00	1288.00	307.20
80.01 to 100.00	1463.00	307.20
>100	by request	by request

II. SURCHARGES

A. State of Washington Surcharge:	\$10.60/m ³
B. Curie Surcharge (per load):	
Less than 100 curies	No charge
101 to 300 curies	\$660.00
301 to License Limits (i.e., 50,000 Ci)	\$660.00 + \$0.09/Ci
C. Handling Surcharge	
0 - 4.54	No charge
>4.54 Mg	\$87.50 + \$0.044/kg over 4.54 Mg
Special Equipment	By special quotation
D. Cask Handling Fee:	\$335.00 per cask

(a) Reproduced from the published rates⁽⁸⁾ of a licensed burial ground operator.

(b) Prices effective November 17, 1980.

(c) Maximum reading at container surface, irrespective of physical size or configuration.

TABLE H.1-6. Unit Costs for Special Tools and Equipment

Item	Estimated Unit Cost (\$ thousands)
Arc Saw	100
Portable Plasma-Arc Torch	20
Portable Oxyacetylene Torch	1
Guillotine Pipe Saw	4
Power-Operated Reciprocating Hacksaw	0.8
Nibbler	1
Submersible Pump with Disposable Filter	2
High-Pressure Water Jet	20
Electropolishing System	100 ^(a)
Mobile Chemical Decontamination Unit	20
Mobile Chemical Mixing & Heating Unit	5
Powered Floor Scrubber	0.3
Wet-Dry Vacuum Cleaner (HEPA Filtered)	1 to 5 ^(b)
Supplied-Air Plastic Suit	0.05
Respirator Facepiece	0.1
Power-Operated Mobile Manlift	40
10-Ton Mobile Hydraulic Crane	28
10-Ton Forklift	28
Concrete Drill with HEPA Filtered Dust Collection System	2
Concrete Surface Spaller	5
Front-End Loader (Light Duty)	20
Portable Filtered Ventilation Enclosure	2 to 10 ^(b)
Filtered-Exhaust Fan Unit	5
Blasting Mat	0.5
Paint Sprayer	0.5 to 1 ^(c)
HEPA Filter	0.2
Roughing Filter	0.1
Waste Compactor	12

TABLE H.1-6. (contd)

Item	Estimated Unit Cost (\$ thousands)
Incinerator	100 to 300 ^(c)
Tube Cutter	0.41
Ratcheting Pipe Cutter	0.07
High-Velocity Liquid Jet	6.90
Low-Velocity Liquid Jet	2.76
Concrete Drill	0.28
Electric/Pneumatic Hammer	0.70
Portable A-Frames	4.10
Portable Wash Sinks	2.76
Portable Spray Clean Booth	5.52
Portable Greenhouse Erection Kit	2.76
Portable Powered Brushes	0.21
Hand-Powered Brushes	0.21
Soap (per thousand kg)	0.28
Freon (per m ³)	2.76

(a) Electrolyte tank has dimensions of 1.2 m x 2.4 m.

(b) Depends on size and complexity.

(c) Depends on capacity of system.

H.2 CDST ASSESSMENT DETAILS FOR DECON

This section provides amplifying details on manpower requirements and on waste management requirements and costs for DECON.

H.2.1 Decommissioning Manpower Details for DECON

This section provides information about manpower requirements for DECON summarized in Section 10.1.1.

H.2.1.1 Staff Labor Job Descriptions

Decommissioning work force organizational charts for DECON and for passive SAFSTOR are presented respectively in Figures 10.1-1 and 10.2-1. For this study,

the decommissioning staff is categorized into two broad groups: 1) support staff and 2) decommissioning workers. Summaries of the functions performed by key individuals in the decommissioning staffs are presented below.

Project Manager. This person is responsible to corporate management for the complete planning, coordination and supervision of all activities carried out to dismantle the reference plant. He directs the project engineer and the health and safety supervisor to insure that the dismantlement plan is developed and implemented in a safe and cost-effective manner. He provides necessary liaison with regulatory agencies and implements regulatory requirements.

Project Engineer. This person is responsible for planning, coordinating, and carrying out the decommissioning activities in a safe manner. He provides engineering services and detailed procedures necessary to implement the decommissioning plan. He is responsible for the preparation of all routine and special reports and he compiles a chronological history of the entire decommissioning project.

Health and Safety Supervisor. This person is responsible for recommending policy in matters of radiological and industrial safety and for enforcing these policies when they are approved. He establishes and implements radiological safety, industrial safety, and emergency preparedness programs.

Security Force Supervisor. This person is responsible for all site security matters. He supervises the decommissioning security force.

Contracts and Accounting Specialist. This person is responsible for preparing documents to purchase needed supplies, equipment, and contractual services. He maintains a complete record of all costs incurred during the decommissioning project. He disburses funds with the approval of the project manager.

Radioactive Shipment Specialist. This person oversees all shipments of radioactive and nonradioactive materials that leave the site for disposal or transfer to insure that the shipments are properly packaged and decontaminated and conform with all applicable federal, state, and local regulations. He arranges with suppliers to provide shipping containers and contracts with carriers to transport all radioactive wastes.

Quality Assurance Engineer. This person prepares the decommissioning quality assurance plan and works with the project manager to implement the quality assurance program. He reports directly to corporate headquarters in quality assurance matters. He supervises the quality assurance technician, who monitors the performance of decommissioning activities and equipment suppliers and provides other QA functions as outlined in Section F.4 of Appendix F.

Maintenance and Crafts Supervisor. This person is responsible for maintaining plant equipment and services in the appropriate operating condition during decommissioning activities. He also schedules and supervises the activities of the craftsmen who assist the decommissioning crews in their decontamination and dismantlement tasks.

Operations Supervisor. This person develops detailed working procedures and supervises the performance of all decommissioning tasks. He writes specifications for special equipment and tools that must be procured or fabricated to carry out the decommissioning project and prepares routine and special reports as requested by the project engineer.

Safety Review Committee. This committee meets periodically to advise corporate headquarters on safety-related matters. It is composed of six voting members, two from corporate headquarters and four independent consultants. The project manager, health and safety supervisor, quality assurance engineer, and project engineer are nonvoting members.

H.2.1.2 Decommissioning Worker Manpower Requirements

Detailed estimates of the decommissioning worker manpower required to perform DECON of the reference plant are shown in Table H.2-1. These estimates for the decommissioning workers, plus those for the support staff, are summarized in Table 10.1-3.

H.2.2 Waste Management Details for DECON

This section provides data on waste management requirements and costs for DECON that are summarized in Section 10.1.3.

TABLE H.2-1. Estimated Decommissioning Worker Manpower Requirements for DECON of the Reference Plant

Decommissioning Activity	Estimated Man-Weeks ^(a)				
	Foremen	Technicians	Health Physicists	Craftsmen	Total
<u>Buildings (Plant Restricted Areas)</u>					
Drum Receiving Area	1	4	0.5	0.5	6
ADU Slurry Receiving Area	1	6	0.5	1	8.5
Sampling Station	1	7	0.5	1	9.5
Laboratory	3	15	1	5	24
UF ₄ Slurry Processing	5.5	30	2	5	42.5
Digestion	6	41	2.5	6.5	56
Solvent Extraction Building	10.5	56	4	9.5	80
Concentration of Uranyl Nitrate	9	56.5	4	7.5	77
Denitration Area	9.5	61	4	7.5	82
Nitric Acid Recovery	5	31	2	4	42
Reduction Area	9	56.5	4	7.5	77
Hydrofluorination Area	7.5	40	3	13	63.5
Fluorination Area	6.5	45	2.5	7.5	61.5
Cylinder Loading Area	5	31	2	9	47
F ₂ Production Cell Room	2.5	15	1	2.5	21
F ₂ Cell Rework Area	2	12	1	2	17
Instrument Repair Room	3.5	17	1.5	13	35
Ammonia Cracking Area	2.5	13.5	1	7.5	24.5
Waste Gas Disposal Area	2	12	1	2.5	17.5
H ₂ Condensing Area	5.5	34	2	5	46.5
Temporary Radwaste	1	7	1	1	10
Decontamination Area	1	4	0.5	1	6.5
Incinerator Facility	2	15	1	4	22
Temporary Laundry and Change Room	2	14	1	1.5	18.5
Overall Cleanup and Survey	3	22	1	2	28
<u>Site (External to Plant)</u>					
Extraction Waste Treatment System	20	148	8	13	189
Fluoride Waste Treatment System	22	160	5	15	202
Miscellaneous Buried Waste	2.5	17	1	4	24.5
Drum Storage Area	2	12	0.5	2	16.5
Cylinder Storage and Wash Area	5.5	30	1	6	42.5
Plant Stack	2	15	1	4	22
Overall Cleanup and Survey of Site	4	31	1.5	0	36.5
Total Man-Weeks	164.5	1058.5	62.5	170.5	1456.0
Total Man-Years ^(b)	3.29	21.17	1.25	3.41	29.12

(a) Number of significant figures shown is for computational completeness and does not imply accuracy to the number of figures shown.

(b) A total of 50 weeks/year is assumed.

H.2.2.1 Waste Management Requirements

Detailed estimates of volumes and weights of contaminated tanks and equipment, piping and ductwork, concrete rubble, solid waste buried onsite, and miscellaneous contaminated scrap from DECON of the reference plant are presented in this section. These estimates form the bases for determining packaging and shipping requirements and costs that are summarized in Section 10.1.3.

Equipment. Estimated shipping volumes for contaminated equipment are presented in the equipment tables in Section A.2 of Appendix A.

Table H.2-2 summarizes shipping volumes for contaminated equipment and materials in terms of the volumes of waste material shipped to sanitary landfill and to low-level waste burial.

HEPA and Roughing Filters. HEPA filters are located on the laboratory exhaust and the temporary decommissioning tents.

HEPA and roughing filters attached to hoods are 0.2 x 0.2 x 0.1 m thick. All of these filters are assumed to be incinerated and the residue is processed to recover any uranium.

HEPA filters for the decommissioning tents are 0.6 x 0.6 x 0.3 m thick. Roughing filters are 0.6 x 0.6 x 0.15 m thick. Ventilation HEPA and roughing filters that are not incinerated are assumed to be packaged in plywood boxes and shipped to a low-level waste burial site for disposal.

The average bulk density of the filters is assumed to be 320 kg/m³.

Concrete Rubble. The volume of potentially contaminated concrete assumed to be removed from floors and walls of the reference plant and shipped to low-level waste burial is shown in Table H.2-3. The estimated volumes shown in the table assume concrete would be removed to a depth of 0.05 m. A fraction of the surface area of each room (based on the type of operation in a specific room) is estimated to calculate the volume of concrete removed. The fraction ranges from 0.1 to 0.5 of the room area.

The shipping volume of the concrete rubble is assumed to be twice the volume of the concrete removed from the walls. The average bulk density of the concrete rubble as shipped is thus assumed to be 1200 kg/m³.

TABLE H.2-2. Shipping Volumes of Equipment, Pipes, Ducts, and Other Materials^(a)

Plant Location	Equipment and Components (m ³)		Pipe, Conduit, Duct, Trays, and Fixtures (m ³)	
	To Sanitary Landfill or Unrestricted Use ^(b)	To Licensed Low-Level Waste Burial ^(c)	To Sanitary Landfill or Unrestricted Use ^(b)	To Licensed Low-Level Waste Burial ^(c)
Drum Receiving and Sampling Area	44.0	40.0	15.2	4.8
ADU Slurry Receiving Area	41.0	10.0	44.6	7.6
Laboratory	62.0	0	72.1	4.1
UF ₄ Slurry Equipment Area	18.0	12.5	33.9	1.6
Digestion Area	32.0	0	27.7	8.2
Solvent Extraction Building	49.3	99.2	90.5	57.4
Uranyl Nitrate Concentration Area	57.4	19.5	12.9	7.8
Denitration Area	19.0	28.0	50.4	23.3
Nitric Acid Recovery	21.2	9.8	--	--
Reduction Area	88.4	76.4	--	70.4
Hydrofluorination Area	142.6	153.0	--	--
Fluorination Area	95.0	105.0	--	--
Cylinder Loading	15.2	23.7	--	--
Fluorine Generation	210.0	77.0	37.9	7.6
Instrument Shop	22.0	4	3.7	2.0
Temporary Radwaste Room	120	136.0	--	--
Decontamination Facility	12.0	19.0	--	--
Laundry	7.5	27	--	--
Change Rooms	17.5	7.5	--	--
Incinerator	40.5	7.5	--	--
Tank Farm	345.0	0	--	--
Extraction Waste System	60.8	671.0	--	--
Fluoride Waste System	73.2	0	--	--
Totals	1485.0	1526.0	388.9	194.8

(a) Number of significant figures shown is for computational completeness and does not imply precision to the number of figures shown.

(b) Items not compacted in volume.

(c) Items are assumed with compacted volumes for shipment.

TABLE H.2-3. Postulated Sources of Concrete Rubble Generated During DECON^(a)

Location	Floor ₂ Area (m ²)	Volume Removed (m ³)	Shipping Volume (m ³)
<u>Main Process Building</u>			
Drum Receiving and Sampling Area	165	4.2	8.4
ADU Slurry Receiving Area	50	0.6	1.2
Laboratory	175	2.2	4.4
UF ₄ Slurry Equipment Area	200	2.5	5.0
Digestion Area	200	5.0	10.0
Uranyl Nitrate Concentration Area	130	3.2	6.4
Dentitration Area	130	3.2	6.4
Reduction Area	270	6.7	13.4
Hydrofluorination	260	3.2	6.4
Fluorination Area	270	3.2	6.4
Cylinder Loading	130	1.6	3.2
<u>Solvent Extraction Building</u>	<u>377</u>	<u>14.1</u>	<u>28.2</u>
Totals	2357	49.7	99.4

(a) Number of significant figures shown is for computational completeness and does not imply precision to the number of figures shown.

H.2.2.2 Waste Management Costs

Details of waste management cost calculations for radioactive wastes from DECON are shown in Table H.2-4. These calculations are based on estimated waste quantities described in Section H.2.2.1 and on unit cost data for packaging, transportation, and waste disposal given in Section H.1.2.

For a particular waste form, the total waste management cost is the sum of the packaging cost plus the transportation cost plus the disposal cost.

TABLE H.2-4. Details of Estimated Costs for Radioactive Waste Management for DECON^(a)

Waste Category	Shipping Weight (kg)	Shipping Volume (m ³)	Type of Container	Number of Shipments	Total Packaging Cost (\$)	Basic Transportation Cost (\$)	Overweight Charge (\$)	Total Transportation Cost (\$)	Burial Cost (\$)	Total Waste Management Cost (\$)
<u>To Low-Level Waste Burial</u>										
Equipment and Components	380 000	855	Plywood Boxes	24	23 500	25 830	5 730	31 560	262 660	317 700
Pipe, Conduit, Ductwork, Trays, Fixtures, etc.	219 200	195	Plywood Boxes	12	5 400	12 910	2 870	15 780	59 900	81 100
Concrete Rubble	64 000	99	Steel Drums	7	14 160	7 530	980	8 510	30 410	53 100
Solid Waste Buried Onsite	48 000	80	Steel Drums	5	11 430			6 580	24 580	42 600
Miscellaneous	17 500	12	Steel Drums	1	1 710	1 100	220	1 320	3 690	6 700
Miscellaneous	12 000	18	Plywood Boxes	2	500	2 200	230	2 430	5 530	8 500
Totals	740 700	1 259		51	56 700			66 200	386 700	509 700

(a) Number of significant figures shown is for computational completeness and does not imply precision to the number of figures shown.

The total transportation cost is the sum of the basic transportation cost plus the overweight charge. Overweight charges are based on charges for the state of Washington. In some instances, the quantity of waste per shipment is limited by weight restrictions (i.e., the 33.1-Mg limit discussed in Section H.1.2.2) rather than by volume limits.

H.2.2.3 Lagoon Waste Management Costs

Cost data for the lagoon waste management alternatives summarized in Section 10.1.3.3 are provided in this section. The costs of the various alternatives are summarized in Table 10.1-7.

Onsite Burial. Details of the onsite burial option are shown in Table H.2-5. These costs are based on the estimated waste quantity shown, assuming site stabilization of the fluoride waste. The solvent extraction waste cannot be buried onsite and allow the site to be released for unrestricted release; thus, no costs are included for that option for the DECON alternative.

TABLE H.2-5. Details of Costs for the Onsite Burial Option

<u>Item</u>	<u>Fluoride Waste</u>
Volume (m ³)	5 720
Costs (\$)	
Manpower	15 000
Equipment	11 000
Material	22 500
Soil Analysis	2 500
Contractors Fee	<u>3 900</u>
Total	54 900

Shipment to Uranium Mill. The solvent extraction waste can be shipped to a uranium mill to remove the uranium and dispose of the remainder of the waste with the mill tailings. It is not feasible to ship the fluoride waste to a mill due to the low uranium content. Table H.2-6 summarizes the estimated costs of shipment of the extraction waste to a mill for recovery of the uranium. Included

TABLE H.2-6. Details of Lagoon Waste Management Costs for Shipment to Uranium Mill^(a)

Waste Category	Shipping Weight (Mg)	Shipping Volume (m ³)	Type of Container	Number of Shipments	Total Packaging Cost (\$)	Total Transportation Cost (\$)	Burial Cost (\$)	Total Waste Management Cost (\$)
Contaminated Liner and Soil to LLW Burial	469	670	Plywood Box	37	18 500	45 920	206 130	270 600
Extraction Waste to Mill	37 260	21 600	Plywood Box	1 840	593 400	2 511 600	-- ^(b)	3 105 000
Subtotal								3 375 600
Value of Recovered Uranium								(2 156 900) ^(c)
Total								1 218 700

(a) Number of significant figures shown is for computational completeness and does not imply precision to the number of figures shown.

(b) Not applicable.

(c) Value of uranium at \$66/kg x 34 400 kg x 0.95 = \$2,156,900.

is the shipment of about 670 m³ of contaminated liner and soil material to an LLW burial ground as part of the decommissioning cost. The cost of shipment of the waste to the uranium mill is partly offset by the value of the recovered uranium at \$66 per kilogram times 34,400 kg of uranium recovered. The efficiency of recovery of uranium at the mill is assumed to be 0.95.

Shipment to LLW Burial Ground. Costs for shipment of all lagoon waste to an LLW burial ground is covered in this section. Table H.2-7 shows the details of the cost of shipment to an LLW burial ground for both extraction and fluoride wastes. For the extraction waste, costs are shown for the liner and soil shipped to the burial ground.

Shipment to Commercial Waste Dump. It is not feasible to ship the extraction waste to a commercial dump. It would have to be mixed with soil at a 9 part soil to 1 part waste ratio, which would make it very costly. Costs for shipment of fluoride waste to a commercial dump are shown in Table H.2-8.

H.3 COST ASSESSMENT DETAILS FOR PREPARATIONS FOR THE SAFE STORAGE PERIOD OF PASSIVE SAFSTOR

This section provides details additional to those provided in Section 10 on manpower requirements and costs for preparations for safe storage.

H.3.1 Decommissioning Worker Manpower Requirements for Preparations for Safe Storage

Detailed estimates of the number of man-weeks required to perform the decommissioning tasks required to prepare the reference plant for safe storage are shown in Table H.3-1. These estimates for decommissioning workers plus those for the support staff are summarized in Table 10.2-3.

TABLE H.2-7. Details of Lagoon Waste Management Costs for Shipment to LLW Burial^(a)

Waste Category	Shipping Weight (Mg)	Shipping Volume (m ³)	Type of Container	Number of Shipments	Total Packaging Cost (\$)	Total Transporta- tion Cost (\$)	Burial Cost (\$)	Total Waste Management Cost (\$)
Extraction Waste	37 260	21 600	Plywood Box	1 840	593 400	2 511 600	6 635 500	9 740 500
Liner and Soil	469	670	Plywood Box	37	18 500	45 900	206 100	207 600
Total								10 011 100
Fluoride Waste	18 200	5 720	Plywood Box	552	817 500	726 340	1 757 200	3 301 000

(a) Number of significant figures shown is for computational completeness and does not imply precision to the number of figures shown.

TABLE H.2-8. Details of Estimated Costs for Shipment of Lagoon Wastes to a Commercial Waste Dump^(a)

Item	Fluoride Waste
Shipping Weight (Mg)	18 200
Shipping Volume (m ³)	5 720
Number of Shipments	552
Cost (\$)	
Packaging	85 000
Transportation	726 300
Burial	200 000
Total Cost	1 011 300

(a) Number of significant figures shown is for computational completeness and does not imply precision to the number of figures shown.

TABLE H.3-1. Estimated Decommissioning Worker Manpower Requirements While Preparing the Reference Plant for Safe Storage

Decommissioning Activity	Estimated Man-Weeks				
	Foremen	Technicians	Health Physicists	Craftsmen	Total
<u>Waste Treatment Facilities Stabilization</u>					
Drain Lagoons and Cover w/Plastic and Weights	2	12	2	2	18
<u>Equipment Deactivation</u>					
Audit all Pumps and Pipelines	1	2	1	2	6
Secure Valves, Hoods, and Conveyors	1	3	1	2	7
Disconnect Services No Longer Required (electrical, air, etc.)	2	2	1	4	9
Tag Equipment and Systems to Identify Status	1	2	2	2	7
Safety Audit	1	2	2	2	7
<u>Isolation of Contaminated Areas</u>					
Install Building Access Locks and Warning Signs	1	1	0	2	4
Perform Building Systems Check	1	2	1	2	6
Install Fence and Locks on Outside Facilities	1	1	0	1	3
<u>Final Preparations for Safe Storage</u>					
Complete Offsite Shipment of all Recovered Uranium	1	4	2	0.5	7.5
Complete Intrusion Alarm System Installation	1	2	1	6	10
Complete Radiation Monitoring System Installation	1	2	1	4	8
Perform Comprehensive Radiation Survey	1	4	4	0	9
Total man-weeks ^(a)	16	39	18	29	101.5

(a) A total of 50 weeks/year is assumed.

H.4 LAGOON WASTE ONSITE BURIAL

Cost data for onsite burial of lagoon waste are provided in this section.

Cost details for the onsite burial of lagoon waste are shown in Table H.4-1. These costs are based on the estimated waste quantities shown assuming site stabilization of the solvent extraction waste and the fluoride waste.

TABLE H.4-1. Details of Costs for Lagoon Waste Onsite Burial

<u>Item</u>	<u>Extraction Waste</u>	<u>Fluoride Waste</u>
Volume (m ³)	21 600	5 720
Costs (\$)		
Manpower	37 200	15 000
Equipment	26 000	11 000
Material	86 800	22 500
Soil Analysis	6 200	2 500
Contractors Fee	<u>12 000</u>	<u>3 900</u>
Subtotal	168 200	54 900
25% Contingency	<u>42 000</u>	<u>13 700</u>
Totals	210 200	68 600

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APPENDIX I

SAFETY ASSESSMENT DETAILS

The purpose of this appendix is to quantify the parameters and define the methodology for estimating the public and occupational safety impacts of decommissioning the reference UF_6 conversion plant.

The impacts of decommissioning on the safety of the public are principally related to the hazards associated with the atmospheric release of radioactive materials during decommissioning, both from planned tasks and from accidents. For DECON and SAFSTOR, atmospheric releases of radioactivity are calculated for the decommissioning tasks that are assumed to be performed. The scenarios analyzed are designed to use realistically maximized parameters. These parameters are carefully chosen to cover a broad spectrum of conditions. Atmospheric release mechanisms are quantified using either measured release factors or information about known physical or chemical behaviors under the postulated conditions. The estimated routine radionuclide releases to the atmosphere are tabulated in the chronological order of the decommissioning tasks for each area of the main building, for each decommissioning alternative. A detailed probabilistic analysis of the postulated accident scenarios is not within the scope of this study. However, selected generic operational accident categories are considered.

Estimates of occupational radiation doses are based on the postulated dose rates in various areas of the reference UF_6 conversion plant and the estimated staff labor required to complete the decommissioning work.

The following sections contain detailed discussions of the technical approach and public and occupational safety details for DECON, SAFSTOR, and waste transportation. A summary of this information is given in Section 11.

I.1 TECHNICAL APPROACH

To estimate the public and occupational safety impacts of the release of radioactivity to the atmosphere from decommissioning tasks, the following basic assumptions are made:

1. The facility has fulfilled existing criteria for the containment of radio-nuclides for accidents and natural phenomena during its operational life-time.
2. All production process solutions are emptied and flushed from all piping, equipment, and sumps. All process chemicals not required for decommissioning are safely removed from the site.
3. All systems used during the various phases of decommissioning are functional and their operability is verified by appropriate tests.
4. To ensure proper air flow for the decommissioning workers, and to protect against uncontrolled atmospheric releases of radioactivity, the main process building is equipped with adequate ventilation, as defined in Appendix A. The air flow rates used in this analysis are $1.6 \times 10^5 \text{ m}^3/\text{h}$ for the main building, $6.2 \times 10^4 \text{ m}^3/\text{h}$ for the fluorine cell, and $4.1 \times 10^4 \text{ m}^3/\text{h}$ for the solvent extraction area. All of the exhaust air is assumed to pass through a bag-type dust collector that results in an overall building transmission factor of 1×10^{-1} .
5. In selected areas with potentially higher levels of radioactive contamination, a temporarily installed "greenhouse," or contamination control envelope, is used. The contamination control envelope has a HEPA filter with a transmission factor of $5 \times 10^{-4(1)}$ to reduce the airborne radio-nuclide concentrations in the UF_6 conversion plant from selected decommissioning tasks.
6. The leakage rate from a contamination control envelope is 10% for all tasks involving its use. This assumed leakage is used as a maximized value to account for routine ruptures or failures of the temporary structures.
7. The airflow rate in this study for contamination control envelopes is: $3000 \text{ m}^3/\text{hr}$.

8. Unneeded hazardous chemicals and equipment are removed after the reference UF_6 conversion plant is shut down. Decontamination agents such as phosphoric acid, ethylenediaminetetraacetic acid (EDTA), oxalic acid, and citric acid are available in the plant. Unneeded ion exchange resins and resin beds are removed.
9. The reference radionuclide inventory used in this appendix for the radiation dose calculations is defined in Appendix C and is used for all surfaces including building structural materials and process equipment.
10. All chemical or manual decontamination tasks remove 90% of the surface contamination.
11. The airborne concentrations of dust or liquid droplets are $1 \times 10^{-2} \text{ g/m}^3$, equal to the concentrations observed at the Elk River reactor decommissioning.^(2,3) For explosions, the airborne concentration is a factor of 10 higher, or $1 \times 10^{-1} \text{ g/m}^3$.⁽⁴⁾
12. All radioactive wastes shipped offsite are shipped in accordance with Department of Transportation (DOT) regulations to a low-level waste burial ground 800 km away. The cumulative dose to the public from truck shipments is based on the methods and data given in WASH-1238.⁽⁵⁾
13. Atmospheric dispersion conditions given in NRC Regulatory Guide 1.4⁽⁶⁾ for a 0- to 8-hour ground-level release are used to calculate the radiation doses resulting from accidental releases during decommissioning. The maximum-exposed individual is located 1000 meters from the point of release. For transportation accidents, the maximum-exposed individual is located 100 m from the release point.
14. Radiation doses to the maximum-exposed individual and to the population residing within 80 km of the reference site are calculated using the environmental data and assumptions discussed in Appendix E of this study. These methods are consistent with the methods outlined in Regulatory Guide 1.109.⁽⁷⁾

15. For onsite waste exhumation operations, atmospheric releases are found by using the mechanical mixing resuspension analysis defined for exhumation operations at a low-level waste burial ground.⁽⁸⁾ A constant resuspension rate of $1 \times 10^{-7} \text{ sec}^{-1}$ is assumed.⁽⁹⁾ It is further assumed that only 10% of the waste/soil mixture is of a correct physical form and particle size to permit airborne transport.^(10,11,12) Because of the wet nature of waste in the bottom of the ponds and lagoons, the airborne dust concentration is assumed to be further reduced by a factor of 10.

Other assumptions relating to specific decommissioning tasks are made and discussed where they apply to the analysis.

I.2 DECON PUBLIC SAFETY ASSESSMENT

The first option considered in the safety evaluation of decommissioning the reference UF_6 conversion plant is DECON (immediate decontamination) of the facility. This alternative results in removal of the radioactively contaminated portions of the buildings, the equipment, and the site so that unrestricted use by the public can be permitted.

The assessment of public safety during DECON includes a consideration of both routine tasks and postulated accidents. These tasks and accidents can generate considerable amounts of airborne radioactivity, primarily in the form of solid particles and/or suspended liquid droplets. Airborne contamination control measures are assumed to be used where they are considered necessary. The use of other appropriate controls is also considered.

In the following sections, the atmospheric releases from DECON are described for both routine tasks and postulated accidents. The atmospheric releases are estimated by determining the realistic maximum atmospheric release for each task or accident, and using this value whenever the task or accident is encountered, even for areas containing lower contamination levels.

I.2.1 Routine DECON Tasks

A complete discussion of the tasks for DECON at the reference UF_6 conversion plant is contained in Appendix G. The chronological sequence of DECON

tasks is listed in Figure G.2-1. The following subsections contain discussions of the methods used for calculating atmospheric releases of radioactivity and the public radiation doses from DECON of the reference UF_6 conversion plant.

I.2.1.1 Methods for Calculating Atmospheric Releases of Radioactivity

Reference DECON methods are defined and analyzed in the following paragraphs so that generic airborne release mechanisms can be determined. A summary of the reference DECON tasks considered in this study is given in Table I.2-1. A complete discussion of each task considered is given below:

Radiation Survey. Radiation surveys are performed at the start of decommissioning, prior to termination of the nuclear license, and at other times during decommissioning to determine the residual levels of radioactive contamination. Radiation surveys involve crews of health physics technicians who walk through the plant and who are assumed to disturb settled radioactivity. The general housekeeping practices followed prior to and during dismantlement determine the amount of loose surface contamination, which varies with location within the reference UF_6 conversion plant.

The relationship between the amount of material in the air above a surface and the contamination level of that surface has been studied experimentally.⁽¹³⁾ This relationship is shown by Equation I.1.

$$S_f = \frac{C_a}{C_s} \quad (I.1)$$

where:

S_f • the resuspension factor, m^{-1}

C_a • the concentration of airborne radioactivity at a given location inside a building, Ci/m^3

C_s • the concentration of radioactivity on building surfaces, Ci/m^2 .

Measured values of S_f vary between 1×10^{-5} and $2 \times 10^{-3} m^{-1}$, depending upon experimental conditions and measurement methods.⁽¹³⁻¹⁵⁾

TABLE I.2-1. DECON Safety Analysis Summary

<u>Task</u>	<u>Operation Description</u>	<u>Routine Airborne Release</u>	<u>Routine Atmospheric Release Calculation Summary</u>	<u>Postulated Hazardous Situations</u>
Radiation Survey	Crews of Health Physics Technicians Walk Through the Plant to Locate and Characterize Residual Radioactive Contamination	Airborne Dust is Removed from Floor Surfaces	Resuspension Analysis: (Resuspension Factor) x (Surface Radioactive Concentration) x (Air Volume Covering Floor) x (Building Transmission Factor) = (Atmospheric Release)	• None Analyzed
Surface Cleaning 1. Sweeping	Small Quantities of Loose Contamination on Floors are Removed by Manual Sweeping with Sweeping Compounds	Airborne Dust is Removed from Floor Surfaces During Sweeping	Resuspension Analysis: (Resuspension Factor) x (Surface Radioactivity Concentration) x (Air Volume Covering Floor) x (Building Transmission Factor) = (Atmospheric Release)	• Fire in Contaminated Sweeping Compound
2. Vacuuming	Small Quantities of Loose Contamination on Surfaces are Removed by Manual Vacuuming Procedures	Airborne Dust is Removed from Floor Surfaces During Vacuuming	Resuspension Analysis: (Resuspension Factor) x (Surface Radioactivity Concentration) x (Air Volume Through Vacuum Cleaner) x (Vacuum HEPA Transmission Factor) x (Building Transmission Factor) = (Atmospheric Release)	• Vacuum Bag Rupture
3. Water-Jet	Hand-Held High-Pressure Water Spray is Used to Remove Loose and Soluble Surface Contamination	Airborne Particles are Caused by High-Pressure Water Sprays	Entrainment of Contaminated Water: (Surface Contamination Removal Fraction) x (Surface Radioactive Concentration) x (Water-Jet Cleaning Rate) ÷ (Liquid Flow Rate) ÷ (Solution Density) x (Airborne Concentration) x (Building Air Flow Rate) x (Water-Jet Use Time) x (Building Transmission Factor) = (Atmospheric Release)	• Spilling Contaminated Liquids • Loss of Services
4. Hand Washing	Limited Areas Are Cleaned with Mops or Sponges	Liquids are Made Airborne as Sprays	Not Calculated	• Spilling Contaminated Liquids
Chemical Decontamination 1. Liquid Leaks	Loop Chemical Decontamination Is Used in Piping and Equipment	Liquids are Made Airborne from Liquid Leaks	Entrainment of Contaminated Water: (Surface Contamination Removal Fraction) (Internal Surface Radioactivity Concentration) x (Internal Surface Area) x (Leak-Weighted Air-Volume) x (Airborne Concentration) ÷ (Volume of Liquid) ÷ (Density of Liquid) x (Building Transmission Factor) = (Atmospheric Release)	• Loss of Services • Spills • Seal Failures

TABLE I.2-1. (Contd)

Task	Operation Description	Routine Airborne Release	Routine Atmospheric Release Calculation Summary	Postulated Hazardous Situations
Chemical Decontamination (Contd)				
2. Spray Leaks	Loop Chemical Decontamination is Used in Piping and Equipment	Liquids are Made Airborne from Spray Leaks	Entrainment of Spray: $(\text{Surface Contamination Removal Fraction}) \times (\text{Internal Surface Radioactivity Concentration}) \times (\text{Internal Surface Area}) \times (\text{Volume of Leak}) \times (\text{Total Liquid Volume}) \times (\text{Fraction of Leak to Reach Filter System}) \times (\text{Building Filter Transmission Factor}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none"> • Loss of Services • Material Failures • Seal Failures
Piping or Equipment Removal and Segmentation in Air				
	Piping and Equipment is Segmented for Offsite Disposal	Vaporized Metal is Made Airborne (Contamination Control Envelope)	Contaminated Piping or Equipment: $(\text{Length of Cut}) \times (\text{Width of Cut}) \times (\text{Surface Radioactivity Concentration}) \times (\text{Building Transmission Factor}) = (\text{Atmospheric Release})$ Activated Piping or Equipment: $\text{Same as Above} \times (\text{Material Thickness}) : (\text{Material Density})$	<ul style="list-style-type: none"> • Loss of Services • Oxyacetylene Explosion • Local Loss of Contamination Control
Concrete Removal				
1. Drilling	Holes are Drilled for Insertion of Concrete Spallers or Explosive Charges	Concrete Dust is Made Airborne (Contamination Control Envelope)	Concrete Dust Generation: $(\text{Number of Holes}) \times (\text{Surface Area per Hole}) \times (\text{Surface Radioactivity Concentration}) \times (\text{Airborne Concentration}) \times (\text{Control Envelope Air Flow Rate}) \times (\text{Drilling Time}) \times (\text{Control Envelope Leak Rate}) \times (\text{Building Filter Transmission Factor}) : (\text{Volume of Holes Drilled}) : (\text{Density of Concrete}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none"> • Loss of Services • Equipment Failure • Local Loss of Contamination Control
2. Spalling	Hydraulically Controlled Concrete Spallers	Concrete Dust is Made Airborne (Contamination Control Envelope)	Not Calculated	<ul style="list-style-type: none"> • Loss of Services • Loss of Contamination Control
3. Packaging	LNG-Powered Front-End Loader	Concrete Dust is Made Airborne (Contamination Control Envelope)	Not Calculated	<ul style="list-style-type: none"> • LNG Explosion
Onsite Waste Exhumation				
1. Exhumation/Packaging	Crews Use Front-end Loaders, Dozers, Loading Bins, Lift Trucks, and Other Equipment As Necessary to Exhume and Package Wastes	Dust Generated During Exhumation and Packaging Operations in Open Air	Waste Dust Generation: $(\text{Resuspension Rate}) \times (\text{Total Mixing Time}) \times (\text{Volume of Waste}) \times (\text{Specific Activity}) = (\text{Atmospheric Release})$	<ul style="list-style-type: none"> • None analyzed

The concentration of airborne radioactivity at a given location inside a building is calculated by rearranging Equation I.1 as shown in Equation I.2.

$$C_a = S_f C_s \quad (I.2)$$

where the terms S_f , C_s , and C_a are the same as defined for Equation I.1.

The total radioactivity made airborne in a building is calculated by using an assumed average resuspension factor associated with the radiation survey of $1 \times 10^{-4} \text{ m}^{-1}$, the surface radionuclide concentrations in the building, and the total volume of air contaminated during the radiation survey as shown in Equation I.3.

$$Q_s = (1 \times 10^{-4} \text{ m}^{-1}) C_s V_s \quad (I.3)$$

where:

- Q_s • the total airborne radioactivity inside a building, Ci
- C_s • the surface radioactivity concentration on building surfaces, Ci/m²
- V_s • the volume of air covering the surface involved in the radiation survey (assumed to be 1 m deep), m³.

The airborne radioactivity generation rate during the radiation survey is found by dividing the airborne radioactivity, Q_s , by the time required to perform the survey. The total radioactivity released to the atmosphere is found by multiplying the airborne radioactivity generated by the building transmission factor.

Surface Cleaning. The surface cleaning operations considered are sweeping, vacuuming, and water-jet sprays.

1. Sweeping. Loose floor contamination is assumed to be removed by manual sweeping operations with the use of sweeping compounds. The amount of airborne radioactivity generated during sweeping is determined by using a resuspension factor analysis similar to the one discussed for the radiation survey. The amount of radioactivity made airborne is highly variable and

is characterized by the properties of the loose contamination and the sweeping method. Sweeping compounds are designed to lower the resuspension by binding small particles of surface contamination to larger particles of sweeping compound.

Data from experimental studies of resuspension during sweeping are limited. One study suggests a resuspension factor of $1.9 \times 10^{-4} \text{ m}^{-1}$.⁽¹⁴⁾ For this study, a conservatively large resuspension factor of $5 \times 10^{-4} \text{ m}^{-1}$ is used. This factor is 5 times larger than the resuspension factor assumed for the radiation survey. The resuspended radioactivity is assumed to be confined in the lower meter of air in the room in which the sweeping occurs. Thus, the airborne radioactivity in a building is calculated using the volume of air disturbed by sweeping (assumed to be the volume 1 m deep above the surface area swept), the surface radioactivity concentration, and the assumed resuspension factor. The total atmospheric radioactivity released from sweeping is found by multiplying the airborne radioactivity generated in each location in the building by the building transmission factor.

2. Vacuuming. Vacuuming can be used as an alternative to manual sweeping for removal of loose surface contamination. Thus, certain areas of the building are designated for vacuuming. The vacuum exhaust is assumed to be fitted with a 99.95%-efficient HEPA filter system. The airborne radioactivity in a building resulting from vacuuming is calculated by finding the product of the surface contamination level, the total area vacuumed, the assumed surface contamination removal fraction for vacuuming (0.5), and the vacuum system HEPA filter transmission factor. The total radioactivity released to the atmosphere is found by multiplying the airborne radioactivity from vacuuming by the building transmission factor.
3. Water-Jet Sprays. The water-jet is a hand-held high-pressure water spray designed to remove loose surface contamination. The operating water pressure is about $3.1 \times 10^4 \text{ kPa}$. The principal mechanism for the generation of airborne radioactivity from the water-jet is the resuspension of liquid droplets containing contamination removed from surfaces. The water spray produces fine droplets smaller than $300 \text{ }\mu\text{m}$ in diameter (about the size of the droplets found in fogs or mists). Larger droplets are broken into

smaller droplets as they impact on the contaminated surfaces. Thus, there is the potential for a significant formation of small droplets with a considerable variation in the total quantity of radioactivity contained in the droplets.

Direct data are not available to define the quantity of droplets formed. A conservative estimate is made by assuming that a sufficient quantity of droplets are generated to maintain an airborne liquid concentration of $1 \times 10^{-2} \text{ g/m}^3$ with vigorous mixing in air. This is the maximum mass concentration that can be found in air velocities less than 0.046 m/sec.⁽³⁾ Measurements have shown that there is fairly constant distribution of 10- to 20- μm diameter particles in a water-jet spray at a concentration of about $1 \times 10^{-2} \text{ g/m}^3$, including the gross entrainment of larger particles.⁽¹⁶⁾ Since particles smaller than 10 μm are in the respirable-size range, they are potentially hazardous. The quantity of radioactivity found in these airborne droplets is influenced by factors such as the quantity of radioactivity on the surface, the solubility of the surface radioactivity in the liquid, and the contact between the liquid and the surface.

Operating parameters of the water-jet vary with the requirements of the surface decontamination effort. For this study, a water flow rate of $2.3 \times 10^{-2} \text{ m}^3/\text{min}$ and a cleaning rate of $0.77 \text{ m}^2/\text{min}$ are assumed. The water-jet is assumed to remove 90% of the surface contamination.

The concentration of radioactivity in the liquid resulting from use of the water-jet is defined by Equation I.4.

$$C_v = 0.9 C_s \left(\frac{0.77 \text{ m}^2/\text{min}}{2.3 \times 10^{-2} \text{ m}^3/\text{min}} \right) \quad (\text{I.4})$$

where:

C_v • the concentration of radioactivity in the liquid resulting from use of the water-jet, Ci/m³

0.9 • the fraction of the surface contamination removed by the water spray

C_s • the radioactivity concentration on the building surfaces, Ci/m²

0.77 • the water-jet cleaning rate, m²/min

2.3×10^{-2} • the water flow rate for the water-jet, m³/min.

The airborne radioactivity generated in a building resulting from use of the water-jet is calculated from the liquid radioactivity concentration using Equation I.5.

$$Q_h = \frac{C_v}{\rho} (1 \times 10^{-2} \text{ g/m}^3) F_v t \quad (I.5)$$

where:

Q_h • the airborne radioactivity in the building resulting from water-jet operations, Ci

C_v • liquid radioactivity concentration, Ci/m³

ρ • the density of the water-jet solution, g/m³

1×10^{-2} • the airborne concentration, g/m³

F_v • the building ventilation flow rate, m³/h

t • the time spent using the water-jet, h.

The atmospheric release of radioactivity from water-jet sprays is calculated by multiplying the airborne radioactivity in the building by the building transmission factor.

Chemical Decontamination. Liquid and spray leaks are assumed to occur during loop chemical decontamination of piping and equipment.

1. Liquid Leaks. A liquid leak results in a small airborne release in a building, since the liquid will retain most of the dissolved radioactive contamination until it reaches a floor drain. Therefore, only a small volume of air is assumed to contain 1×10^{-2} g/m³ of airborne radioactivity.

The concentration of radioactivity in the decontamination solution is a function of the radioactivity removal fraction, the surface contamination level, and the volume of decontamination solution, as shown in Equation I.6.

$$C_v = \frac{0.9 C_s}{V_c} A_s \quad (I.6)$$

where:

- C_v • the liquid radioactivity concentration, Ci/m³
- 0.9 • the surface contamination removal fraction
- C_s • surface concentration of radioactivity on internal pipe and equipment surfaces, Ci/m²
- V_c • the volume of decontamination solution in the loop, m³
- A_s • the internal surface area decontamination, m².

It is assumed that only a small volume of air above a liquid leak becomes contaminated. This volume is assumed to be the amount of air 2 m deep above the floor surface area contaminated by the liquid leak. The floor surface covered by a leak is assumed to be 1 m² for each liter of leak. Thus, the total airborne radioactivity is calculated using Equation I.7.

$$Q_1 = \frac{V_r}{\rho} (1 \times 10^{-2} \text{ g/m}^3) C_v \quad (I.7)$$

where:

- Q_1 • the airborne radioactivity in the building resulting from the liquid leak, Ci
- V_r • the volume of room air containing the airborne release (the volume of air 2 m deep over the surface area contaminated, which is assumed to be 1 m² per 1 x 10⁻³ m³ of leak), m³
- 1 x 10⁻² • the airborne concentration, g/m³
- ρ • the density of the decontamination liquid, g/m³
- C_v • the liquid radioactivity concentration, Ci/m³.

The total radioactivity released to the atmosphere from a liquid leak is calculated by multiplying the airborne radioactivity in the building by the building HEPA filter transmission factor.

2. Spray Leaks. Spray leaks of chemical decontamination liquids can lead to a major airborne release since the radioactive liquids are directly entrained in the building air. For this study, 0.3% of the airborne contamination in the building is assumed to be of the correct particle size to reach the building filter.⁽¹⁷⁾ Thus, the total release to the environment from spray leaks is 0.3% of the leaked radioactivity multiplied by the building HEPA filter transmission factor.

Piping or Equipment Removal. Contaminated piping and equipment are assumed to be removed and segmented for disposal during DECON. For this study, four cutting methods are assumed to be used: arc saw, plasma-arc torch, oxyacetylene torch, and power hacksaw. A summary of the cutting parameters for these methods is listed in Table I.2-2.

TABLE I.2-2. Cutting Parameters for Piping and Equipment Removal

Cutting Method	Cut (kerf) Width (m)	Airborne Release
Arc Saw	9.5×10^{-3}	Metal Fragments, Vapors
Plasma Torch	3.1×10^{-3} to 6.4×10^{-3}	Condensed Metals, Gases, Smoke Particles
Oxyacetylene Torch	6.4×10^{-3}	Condensed Metals, Vapors
Power Hacksaw	6.4×10^{-3}	Metal Fragments, Vapors

For piping or equipment containing surface contamination, the airborne release in the building from cutting tasks in air is calculated using Equation I.8.

$$Q_c = L K C_s \quad (1.8)$$

where:

Q_c • the airborne radioactivity from cutting contaminated pipe or equipment, Ci

L • the length of cut, m

K • the width of the cut (kerf), m

C_s • the surface radioactivity concentration, Ci/m².

Equation I.8 is based on the conservative assumption that all of the surface radioactive contamination associated with the cut width is vaporized and made airborne during cutting operations.

Contaminated Concrete Removal. Structural concrete that cannot be decontaminated to unrestricted-use levels by surface cleaning methods must be mechanically removed. For this study, the only method considered is the use of concrete spallers to remove these contaminated concrete areas. Airborne contamination control is assumed to be provided by the use of a vacuum exhaust system during this dusty task. Holes are drilled into the concrete surfaces for insertion of the concrete spaller. To calculate the airborne radioactivity in a building associated with the dust generated during drilling, the volume of concrete for each hole, the number of holes drilled, the radioactive contamination level and the effectiveness of the vacuum system must be determined. Each hole is assumed to be 0.051 m deep and to remove a contaminated surface area of 0.05 m².

For structural concrete with surface contamination only, the total radioactivity associated with drilling is found by multiplying the number of holes drilled by the surface area per hole by the surface contamination level. The surface contamination is assumed to be uniformly mixed in volume of dust associated with drilling the holes. A vacuum exhaust system is used to collect the dust generated during drilling. This system is assumed to have an airflow of 1000 m³/hr and reduce the air concentration of dust by a factor of 0.3. The quantity of airborne radioactivity generated by drilling is given by Equation I.9.

$$Q_d = \frac{N_h (7.8 \times 10^{-5} \text{ m}^2/\text{hole}) C_s}{V_h \rho} F_v t (0.3) 1 \times 10^{-1} \text{ g/m}^3 \quad (\text{I.9})$$

where:

Q_d • the quantity of radioactivity made airborne during drilling, Ci

N_h • the number of holes drilled

- 7.8×10^{-5} • the surface area removed per hole, m^2/hole
- C_s • the concentration of surface radioactivity, Ci/m^2
- V_h • the total volume of all holes drilled, m^3
- ρ • the density of concrete, $2.5 \times 10^3 \text{ kg}/\text{m}^3$
- F_v' • the vacuum system airflow rate, $1.0 \times 10^3 \text{ m}^3/\text{h}$
- t • the duration of the drilling operation, h
- 0.3 • airborne cleanup factor
- 1×10^{-1} • the airborne concentration in the vacuum system, g/m^3 .

The radioactivity released to the atmosphere is found by multiplying the quantity of radioactivity generated by the leakage factor from the control envelope (if one is used) and by the building transmission factor.

After the holes are drilled, the concrete spaller is inserted into the holes and the concrete surface is scarfed. The concrete fractures and spalls in large pieces with a relatively small amount of dust generated. Water sprays are assumed to be used to control the airborne dust during the concrete spalling. Thus, the airborne release from concrete spalling is assumed to be insignificant when compared to the drilling operations and is not considered further in this study.

Onsite Waste Exhumation. Uranium-contaminated soil is assumed to be removed from the extraction waste pond, the fluoride waste lagoons, and from miscellaneous waste burial locations onsite. A summary of options considered for the DECON alternative is given in Appendix H. Only the option that results in the maximum atmospheric release is considered in this public safety assessment. Atmospheric releases are found using the mechanical mixing resuspension analysis defined for low-level waste exhumation operations.⁽⁸⁾ The airborne radioactivity resulting from mechanical resuspension is calculated by determining the product of the mechanical mixing resuspension rate ($1 \times 10^{-7} \text{ s}^{-1}$), the volume of waste disturbed (m^3), the total disturbance time (s), and the specific activity of the waste (Ci/m^3). The time estimates used are for the exhumation, waste packaging, and offsite disposal option discussed in Appendix H. The waste is assumed to be in a wet form that reduces its atmospheric concentration from resuspension by a factor of 10. It is also assumed that only 10% of the waste is of the proper physical form and particle size to permit airborne transport.

I.2.1.2 Atmospheric Releases from DECON Tasks

Releases of radioactivity resulting from DECON tasks at the reference UF_6 conversion plant are calculated using the general methods described in Section I.2.1.1. The operational data used for each area is shown in Table I.2-3. The areas listed in this table are based on the DECON tasks discussed in Appendix G. Table I.2-3 contains the necessary safety analysis data, operational times, and contamination levels associated with DECON operations for each building area.

The calculated atmospheric releases for DECON tasks at the reference UF_6 conversion plant are shown in Table I.2-4. This table contains the calculated airborne radioactivity generation rates, the total airborne contamination generated inside the main building, and the total atmospheric release associated with each area considered. These atmospheric releases are used in the radiation dose calculations to estimate the impact of DECON tasks on the public, as discussed in the following subsection.

I.2.1.3 Public Radiation Doses from DECON Tasks

Radiation doses to the public from atmospheric releases resulting from DECON tasks are calculated using the radiation dose models discussed in Appendix E. The first-year doses and the fifty-year committed dose equivalents to both the maximum-exposed individual and to the population residing within 80 km of the site are calculated for each process area considered. Each of the atmospheric releases is assumed to be a chronic release; that is, one that occurs at a uniform rate for a period of 1 year. This is done so that direct comparisons of the impacts of DECON tasks can be made.

Radiation doses from atmospheric releases of radioactivity are calculated for direct exposure, inhalation, and ingestion radiation exposure pathways. Radiation doses from air submersion are not calculated since they have been shown to be insignificant in previous decommissioning studies.^(18,19,20) For inhalation, radionuclides are assumed to have soluble clearance times (as defined in Appendix F) for the organs of reference, except for the lung where insoluble clearance times are assumed. For this study, the organs of reference for which radiation dose is calculated include: total body, bone, lung, and GI-tract (lower large intestine).

TABLE I.2-3. Public Safety Analysis Data Summary for DECON and Preparations for Safe Storage of the UF₆ Conversion Plant

Area	General Method	Operational Data			Comments	Operational Time (h)	Average Contamination Level (a) % of Surfaces at:		
							High	Medium	Low
All Areas	Initial Comprehensive Radiation Survey (b)	Surface Area (m ²) 3.7 x 10 ³	Air Volume V (m ³) 3.4 x 10 ³		Health Physics Technicians Tour Building, 3 complete surveys	3.5 x 10	-- (c)	10	90
	Water-jet (b)	Cleaning Rate (m ² /h) 4.6 x 10 ¹	Surface Area A _s (m ²) 1.0 x 10 ³	Volume Released (m ³) 3.1 x 10 ⁰	All misc. hoods, ducts, and equipment	2.2 x 10 ¹	--	10	90
	Sweeping (b) Vacuuming (b)	Surface Area (m ²) 3.7 x 10 ³ 3.7 x 10 ³	Air Volume V _s (m ³) 3.7 x 10 ³ 3.7 x 10 ³		All floor surfaces are assumed to be swept and vacuumed	2.2 x 10 ¹ 1.0 x 10 ¹	-- --	10 10	90 90
	Oxyacetylene (d) Torch and power hacksaw	Cut Length L (m) 6.1 x 10 ³	Kerf Width K (m) 6.4 x 10 ³	Cut Rate (m/h) 2.4 x 10 ¹	All misc. piping, ducts, and electrical trays	2.5 x 10 ²	--	10	90
	Contaminated Concrete Surface Removal	Number of Holes, N _H 2.2 x 10 ⁴	Hole Surface Area (m ²) 1.7 x 10 ⁰	Hole Volume V _H (m ³) 8.7 x 10 ⁻²	4 x 10 ⁻⁶ m ³ per hole, 200 holes/day, surface area = 1.8 x 10 ³ m ²	1.2 x 10 ³	--	10	90
Area 1: Drum Receiving	None				Surface decon. included as part of all area listing	--	--	--	--
Area 2: ADU Slurry Receiving and storage area	Loop chem. decon.	Release Fraction 1 x 10 ⁻² 1 x 10 ⁻³	Surface Area A _s (m ²) 6.8 x 10 ¹ 6.8 x 10 ¹	Volume Released (m ³) 4.2 x 10 ⁻¹ 4.2 x 10 ⁻²	Assumed release fractions of 1% liquid and 0.1% spray	3.2 x 10 ¹ 3.2 x 10 ¹	50 50	35 35	15 15
	Oxyacetylene Torch and power hacksaw	Cut Length, L (m) 1.7 x 10 ²	Kerf Width, K (m) 6.4 x 10 ⁻³	Cut Rate (m/h) 1.2 x 10 ¹	Cutting includes piping, equipment, and pump removal	1.4 x 10 ¹	50	35	15

TABLE I.2-3. (Contd)

Area	General Method	Operational Data			Comments	Operational Time (h)	Average Contamination Level ^(a) % of Surfaces at:		
		Cut Length, L (m)	Kerf Width, K (m)	Cut Rate (m/hr)			High	Medium	Low
Area 3: Sampling Station	Oxyacetylene Torch and power hacksaw	2.8×10^1	6.4×10^{-3}	1.2×10^1	Cutting includes local piping, equipment, and transfer screws	2.0×10^0	50	35	15
	Water-jet	Cleaning Rate (m ² /h) 4.6×10^1	Surface Area A _s (m ²) 1.6×10^2	Liquid Volume (m ³) 5.0×10^0	Cleaning hoods, ducts, racks and bins	3.5×10^0	--(c)	100	--
Area 4: Laboratory	Water-jet	4.6×10^1	1.0×10^2	3.1×10^0	Cleaning hoods, tables, and misc. items	2.2×10^0	--	--	100
Area 5: UF ₄ Slurry Processing	Loop chem. decon.	Release Fraction	Surface Area A _s (m ²)	Volume Released (m ³)					
	- liquid leak	1×10^{-2}	3.2×10^1	2.0×10^{-1}	Assumed release fractions of 1% liquid and 0.1% spray	1.5×10^1	50	30	20
	- spray leak	1×10^{-3}	3.2×10^1	2.0×10^{-2}		1.5×10^1	50	30	20
Area 6: Ore concentrate Dissolution	Loop chem. decon.								
	- liquid leak	1×10^{-2}	1.2×10^2	7.7×10^{-1}	Assumed release fractions of 1% liquid and 0.1% spray	5.9×10^1	50	30	20
	- spray leak	1×10^{-2}	1.2×10^2	7.7×10^{-2}		5.9×10^1	50	30	20
Area 7: Solvent Extraction	Loop chem. decon.								
	- liquid leak	1×10^{-2}	7.8×10^1	5.0×10^{-1}	Assumed release fractions of 1% liquid and 0.1% spray	3.8×10^1	50	30	20
	- spray leak	1×10^{-3}	7.8×10^1	5.0×10^{-2}		3.8×10^1	50	30	20
	Oxyacetylene Torch and power hacksaw	Cut Length L (m) 4.7×10^2	Kerf Width K (m) 6.4×10^{-3}	Cut Rate (m/L) 1.2×10^1	Cutting includes local piping, equipment, and decanters	3.9×10^1	--	50	50

TABLE I.2-3. (Contd)

Area	General Method	Operational Data			Comments	Operational Time (h)	Average Contamination Level % of Surfaces at:		
		Release Fraction	Surface Area A_s (m ²)	Volume Released (m ³)			High	Medium	Low
Area 8: Uranyl Nitrate Concentration	Oxyacetylene Torch and power hacksaw	2.4×10^2	6.4×10^{-3}	1.2×10^1	Cutting includes local piping, equipment, and evaporators	2.0×10^1	--	100	--
	Load chem. decon.	1×10^{-2}	6.6×10^1	4.2×10^{-1}	Assumed release fractions of 1% liquid and 0.1% spray	3.2×10^1	50	30	20
	- liquid leak	1×10^{-3}	6.6×10^1	4.2×10^{-2}		3.2×10^1	50	30	20
	- spray leak								
Area 9: Denitration	Oxyacetylene Torch and power hacksaw	Cut Length $\frac{L(m)}{4.6 \times 10^2}$	Kerf Width $\frac{K(m)}{6.4 \times 10^{-3}}$	Cut Rate $\frac{(m/L)}{1.2 \times 10^1}$	Cutting includes local piping, equipment, denitrator, and scrubber	3.8×10^1	--	50	50
Area 10: Nitric Acid Recovery	Loop chem. decon.	Release Fraction	Surface Area A_s (m ²)	Volume Released (m ³)	Assumed release fractions of 1% liquid and 0.1% spray	7.6×10^0 7.6×10^0	--	50	50
		1×10^{-2}	1.6×10^1	1.0×10^{-2}					
		1×10^{-3}	1.6×10^1	1.0×10^{-2}					
Area 11: Reduction Systems	Water-jet	Cleaning Rate $\frac{(m^2/h)}{4.6 \times 10^1}$	Surface Area A_s (m ²) 1.7×10^2	Liquid Volume $\frac{(m^3)}{5.3 \times 10^0}$	Cleaning hoods, ducts, and bins	3.7×10^0	--	100	--
	Oxyacetylene Torch and power hacksaw	Cut Length $\frac{L(m)}{1.5 \times 10^3}$	Kerf Width $\frac{K(m)}{6.4 \times 10^{-3}}$	Cut Rate $\frac{(m/L)}{1.2 \times 10^1}$	Cutting includes local piping, equipment, and reduction reactors	1.2×10^2	--	100	--
Area 12: Hydrofluorination	Oxyacetylene Torch and power hacksaw	2.8×10^1	6.4×10^{-3}	1.2×10^1	Cutting includes feeders, hoppers, and conveyors	2.2×10^0	10	40	50

TABLE I.2-3. (Contd)

Area	General Method	Operational Data			Comments	Operational Time (h)	Average Contamination Level % of Surfaces at:		
		Release Fraction	Surface Area _s (m ²)	Volume Released (m ³)			High	Medium	Low
Area 13: Fluorination	Oxyacetylene Torch and power hacksaw	2.9×10^{-2}	6.4×10^{-3}	1.2×10^1	Cutting includes fluorinator towers, product cylinders, and cold traps	2.3×10^1	10	40	50
	Loop chem. decon.								
	- liquid leak	1×10^{-2}	2.1×10^2	1.3×10^0	Assumed release fractions of 1% liquid and 0.1% spray	9.9×10^1	10	40	50
	- spray leak	1×10^{-3}	2.1×10^2	1.3×10^{-1}		9.9×10^1	10	40	50
Area 14: Cylinder loading	None				Surface decon. included part of all area listing	--	--	--	--
Area 15: Fluorine Generation									
	Loop chem. decon.								
	- liquid leak	1×10^{-2}	7.1×10^1	4.4×10^{-1}	Assumed release fractions of 1% liquid and 0.1% spray	3.4×10^1	--	20	80
	- spray leak	1×10^{-3}	7.1×10^1	4.4×10^{-2}		3.4×10^1	--	20	80
	Oxyacetylene Torch and power hacksaw	Cut Length L (m)	Kerf Width K (m)	Cut Rate m/L	Cutting includes fluorine cells, surge tanks, and storage tanks	5.5×10^1	--	20	80
		6.7×10^{-2}	6.4×10^{-3}	1.2×10^1					
Areas 16 through 24									
	Loop chem. decon.								
	- liquid leak	1×10^{-2}	2.3×10^2	1.4×10^0	Assumed release fractions of 1% liquid and 0.1% spray	1.1×10^2	--	50	50
	- spray leak	1×10^{-3}	2.3×10^2	1.4×10^{-1}		1.1×10^2	--	50	50
	Water-jet	Cleaning Rate m^2/h	Surface Area _s (m ²)	Volume Released (m ³)	Released surfaces of all piping and equipment	6.3×10^0	--	50	50
		4.6×10^1	2.9×10^2	9.0×10^0					

TABLE I.2-3. (Contd)

Area	General Method	Operational Data	Comments	Operational Time (h)	Average Contamination Level (a)		
					% of Surfaces at:		
					High	Medium	Low
Other Onsite areas: (e)							
Extraction waste pond	Exhumation and packaging for offsite shipment	Volume of Waste (m ³) 2.2 x 10 ⁴	Airborne releases are based on methods discussed for mechanical mixing tasks in NUREG/CR-0570 (Appendix I) for low-level waste relocation tasks	5.0 x 10 ²	0.28% ^(f)	--	--
				1.3 x 10 ²	0.01% ^(f)	--	--
Fluoride waste lagoons	Exhumation and packaging for offsite shipment	5.7 x 10 ³					
Misc. buried waste	Exhumation and packaging for offsite shipment	2.7 x 10 ³		6.2 x 10 ¹	0.28% ^(f)	--	--

(a) Areas of high, medium, and low contamination are assumed to be 1000 g/m², 10 g/m², and 0.01 g/m² of the reference uranium inventory, respectively.

(b) The method shown is considered as part of preparations for safe storage as well as part of DECON.

(c) Dashes indicate no data applies for the entry shown.

(d) All miscellaneous piping, ducts, and electrical trays for all areas are included in this entry.

(e) The areas shown are onsite waste treatment or storage areas, and only the waste exhumation options discussed in Appendix H are considered.

(f) Uranium concentrations in the wastes are shown in percent.

TABLE I.2-4. Calculated Atmospheric Releases for DECON at the Reference UF₆ Conversion Plant

Area	General Method	Operational Time (h)	Airborne Generation Rate Inside Building or Control Envelope (Ci/h)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination ^(a) Control System Transmission Factor	Total ^(b) Atmospheric Release (Ci/yr)
All Areas	Initial Comprehensive ^(c) Radiation Survey	3.5 x 10 ¹	1.4 x 10 ⁻⁸	4.8 x 10 ⁻⁷	1 x 10 ⁻¹	4.8 x 10 ⁻⁸
	Water-Jet ^(c)	2.2 x 10 ¹	6.9 x 10 ⁻⁸	1.5 x 10 ⁻⁶	1 x 10 ⁻¹	1.5 x 10 ⁻⁷
	Sweeping ^(c)	2.2 x 10 ¹	1.1 x 10 ⁻⁷	2.4 x 10 ⁻⁶	1 x 10 ⁻¹	2.4 x 10 ⁻⁷
	Vacuuming ^(c)	1.0 x 10 ¹	1.2 x 10 ⁻⁷	1.2 x 10 ⁻⁶	1 x 10 ⁻¹	1.2 x 10 ⁻⁸
	Oxyacetylene Torch and Power Hacksaw ^(d)	2.5 x 10 ²	2.2 x 10 ⁻⁷	5.5 x 10 ⁻⁵	1 x 10 ⁻¹	5.6 x 10 ⁻⁶
	Contaminated Concrete Surface Removal	1.2 x 10 ³	3.4 x 10 ⁻¹⁰	4.0 x 10 ⁻⁷	1 x 10 ⁻¹	4.0 x 10 ⁻⁸
Area 1: Drum Receiving	None	--(e)	--	--	--	--
Area 2: ADU Slurry Receiving and Storage Area	Loop Chem. Decon. - Liquid Leak	3.2 x 10 ¹	2.6 x 10 ⁻¹⁰	8.5 x 10 ⁻⁹	1 x 10 ⁻¹	8.5 x 10 ⁻¹⁰
	- Spray Leak	3.2 x 10 ¹	4.1 x 10 ⁻⁹	1.3 x 10 ⁻⁷	1 x 10 ⁻¹	1.3 x 10 ⁻⁸
	Oxyacetylene Torch and Power Hacksaw	1.4 x 10 ¹	5.1 x 10 ⁻⁵	7.1 x 10 ⁻⁴	1 x 10 ⁻²	7.1 x 10 ⁻⁶
Area 3: Sampling Station	Oxyacetylene Torch and Power Hacksaw	2.0 x 10 ⁰	6.0 x 10 ⁻⁵	1.2 x 10 ⁻⁴	1 x 10 ⁻²	1.2 x 10 ⁻⁶
	Water-Jet	3.5 x 10 ⁰	6.1 x 10 ⁻⁷	2.1 x 10 ⁻⁶	1 x 10 ⁻¹	2.1 x 10 ⁻⁷
Area 4: Laboratory	Water-Jet	2.2 x 10 ⁰	6.1 x 10 ⁻⁹	1.3 x 10 ⁻⁸	1 x 10 ⁻¹	1.3 x 10 ⁻⁹
Area 5: UF ₄ Slurry Processing	Loop Chem. Decon. - Liquid Leak	1.5 x 10 ¹	5.5 x 10 ⁻¹⁰	5.2 x 10 ⁻⁹	1 x 10 ⁻¹	5.2 x 10 ⁻¹⁰
	- Spray Leak	1.5 x 10 ¹	5.2 x 10 ⁻⁹	7.8 x 10 ⁻⁸	1 x 10 ⁻¹	7.8 x 10 ⁻⁹
Area 6: Ore Concentrate Dissolution	Loop Chem. Decon. - Liquid Leak	5.9 x 10 ¹	2.0 x 10 ⁻¹⁰	1.2 x 10 ⁻⁸	1 x 10 ⁻¹	1.2 x 10 ⁻⁹
	- Spray Leak	5.9 x 10 ¹	3.3 x 10 ⁻⁹	1.9 x 10 ⁻⁷	1 x 10 ⁻¹	1.9 x 10 ⁻⁸

TABLE I.2-4. (Contd)

Area	General Method	Operational Time (h)	Airborne Generation Rate Inside Building or Control Envelope (Ci/h)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination Control System Transmission Factor (a)	Total (b) Atmospheric Release (Ci/yr)
Area 7: Solvent Extraction	Loop Chem. Decon. - Liquid Leak	3.8×10^1	2.9×10^{-10}	1.1×10^{-8}	1×10^{-1}	1.1×10^{-9}
	- Spray Leak	3.8×10^1	4.2×10^{-9}	1.6×10^{-7}	1×10^{-1}	1.6×10^{-8}
	Oxyacetylene Torch and Power Hacksaw	3.9×10^1	5.0×10^{-7}	2.0×10^{-5}	1×10^{-2}	2.0×10^{-7}
Area 8: Uranyl Nitrate Concentration	Oxyacetylene Torch and Power Hacksaw	2.0×10^1	1.0×10^{-6}	2.0×10^{-5}	1×10^{-2}	2.0×10^{-7}
	Loop Chem. Decon. - Liquid Leak	3.2×10^1	2.6×10^{-10}	8.5×10^{-9}	1×10^{-1}	8.5×10^{-10}
	- Spray Leak	3.2×10^1	4.1×10^{-9}	1.3×10^{-7}	1×10^{-1}	1.3×10^{-8}
Area 9: Denitration	Oxyacetylene Torch and Power Hacksaw	3.2×10^1	5.0×10^{-7}	1.9×10^{-5}	1×10^{-2}	1.9×10^{-7}
Area 10: Nitric Acid Recovery	Loop Chem. Decon. - Liquid Leak	7.6×10^0	5.3×10^{-12}	4.0×10^{-11}	1×10^{-1}	4.0×10^{-12}
	- Spray Leak	7.6×10^0	7.9×10^{-11}	6.0×10^{-10}	1×10^{-1}	6.0×10^{-11}
Area 11: Reduction Systems	Water-Jet	3.7×10^0	6.2×10^{-7}	2.3×10^{-6}	1×10^{-1}	2.3×10^{-7}
	Oxyacetylene Torch and Power Hacksaw	1.2×10^2	1.0×10^{-4}	1.2×10^{-4}	1×10^{-2}	1.2×10^{-6}
Area 12: Hydrofluorination	Oxyacetylene Torch and Power Hacksaw	2.2×10^0	1.1×10^{-5}	2.4×10^{-5}	1×10^{-2}	2.4×10^{-7}
Area 13: Fluorination	Oxyacetylene Torch and Power Hacksaw	2.3×10^1	1.1×10^{-11}	4.6×10^{-9}	1×10^{-2}	2.5×10^{-6}
	Loop Chem. Decon. - Liquid Leak	9.9×10^1	4.2×10^{-11}	4.6×10^{-9}	1×10^{-1}	4.6×10^{-10}
	- Spray Leak	9.9×10^1	6.3×10^{-10}	6.2×10^{-8}	1×10^{-1}	6.2×10^{-9}
Area 14: Cylinder Loading	None	--	--	--	--	--

TABLE I.2-4. (Contd)

Area	General Method	Operational Time (h)	Airborne Generation Rate Inside Building or Control Envelope (Ci/h)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination ^(a) Control System Transmission Factor	Total ^(b) Atmospheric Release (Ci/yr)
Area 15: Fluorine Generation	Loop Chem. Decon. - Liquid Leak	3.4×10^1	1.2×10^{-12}	4.0×10^{-11}	1×10^{-1}	4.0×10^{-12}
	- Spray Leak	3.4×10^1	1.8×10^{-11}	6.1×10^{-10}	1×10^{-1}	6.1×10^{-11}
	Oxyacetylene Torch and Power Hacksaw	5.5×10^1	2.1×10^{-7}	1.2×10^{-5}	1×10^{-2}	1.2×10^{-7}
Areas 16 Through 24	Loop Chem. Decon. - Liquid Leak	1.1×10^2	2.1×10^{-12}	2.3×10^{-10}	1×10^{-1}	2.3×10^{-11}
	- Spray Leak	1.1×10^2	3.2×10^{-11}	3.5×10^{-9}	1×10^{-1}	3.5×10^{-10}
	Water-Jet	6.3×10^{-7}	3.4×10^{-7}	2.2×10^{-6}	1×10^{-1}	2.2×10^{-7}
Other Onsite Areas: ^(f) Waste Extraction Pond	Exhumation and Packaging For Offsite Shipment	5.0×10^2	9.6×10^{-6}	4.8×10^{-3}	1.0	4.8×10^{-3}
Fluoride Waste Lagoons	Exhumation and Packaging For Offsite Shipment	1.3×10^2	9.2×10^{-8}	1.2×10^{-5}	1.0	1.2×10^{-5}
Misc. Buried Waste	Exhumation and Packaging For Offsite Shipment	6.2×10^1	1.2×10^{-6}	7.3×10^{-5}	1.0	7.3×10^{-5}

(a) The assumed building transmission factor for all atmospheric releases from ventilation is 1×10^{-1} . For contamination control envelopes, the transmission factor is controlled by the assumed 10% leakage, and is 1×10^{-1} . Thus, the total for a control envelope and the building is 1×10^{-2} .

(b) For comparison, dose calculations are made based on annual releases.

(c) The method shown is considered as part of preparations for as well as part of DECON.

(d) All miscellaneous piping, ducts, and electrical trays for all areas are included in this entry.

(e) Dashes indicate no data applies for the entry shown.

(f) The areas shown are onsite waste treatment or storage areas, and only the waste exhumation options discussed in Appendix H are considered.

Radiation doses to the maximum-exposed individual are shown in Table I.2-5, and doses to the population are shown in Table I.2-6. The atmospheric dispersion ($\overline{X}/Q'$) values used in the dose calculations for routine atmospheric releases are 7.5×10^{-6} sec/m³ for the maximum-exposed individual and 2.4×10^{-9} sec/m³ for the population. The total population within 80 km of the site is 3.5×10^6 , as discussed in Appendix B.

I.2.2 Postulated Accidents During DECON

During DECON, situations may arise that lead to the accidental atmospheric release of radioactivity. Accident scenarios and calculated airborne radioactivity releases considered for DECON in this study are listed in Table I.2-7. The accidents are listed by decreasing atmospheric release. Accident scenarios are based on the technologies used during DECON tasks. A general estimate of the frequency of occurrence of the level of the atmospheric release associated with each postulated accident is also given. The frequency of occurrence is arrived at by considering not only the probability of the accident but also the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar or greater magnitude per year is greater than 10^{-2} , as "medium" if between 10^{-2} and 10^{-5} , and as "low" if less than 10^{-5} .

While it is beyond the scope of this study to try to evaluate every potential accident for decommissioning a reference UF₆ conversion plant, operational, natural phenomena, and indirect man-related events are discussed in the following subsections. Multiple failure-event accidents are not considered; that is, for each accident, only one failure event is analyzed.

I.2.2.1 Postulated Operational Accidents

Accidental atmospheric releases from incidents such as equipment failure, human error, or service conditions during routine DECON tasks are discussed in this section. The accident analyses are based on the assumed technologies for DECON tasks.

Oxyacetylene Explosion. Oxyacetylene cutting torches are assumed to be used for removing and segmenting various steel components during DECON. Acetylene gas has a flammability range of 2.5 to 80 volume % mixed in air.^(21,22) Violent

TABLE I.2-5. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During Routine DECON and Preparations for Safe Storage Tasks at the Reference UF₆ Conversion Plant

Area/General Method	Total ^(a) Atmospheric Release (Ci/yr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
All Areas Initial Comprehensive Radiation Survey ^(b)	4.8 x 10 ⁻⁸	3.3 x 10 ⁻¹¹	2.7 x 10 ⁻¹⁰	6.7 x 10 ⁻⁹	2.3 x 10 ⁻¹¹	5.8 x 10 ⁻¹¹	9.6 x 10 ⁻¹⁰	3.3 x 10 ⁻⁸	2.3 x 10 ⁻¹¹
Water-Jet ^(b)	1.5 x 10 ⁻⁷	1.0 x 10 ⁻¹⁰	8.6 x 10 ⁻¹⁰	2.1 x 10 ⁻⁸	7.2 x 10 ⁻¹¹	1.8 x 10 ⁻¹⁰	3.0 x 10 ⁻⁹	1.0 x 10 ⁻⁷	7.2 x 10 ⁻¹¹
Sweeping ^(b)	2.4 x 10 ⁻⁷	1.6 x 10 ⁻¹⁰	1.4 x 10 ⁻⁹	3.4 x 10 ⁻⁸	1.2 x 10 ⁻¹⁰	2.9 x 10 ⁻¹⁰	4.8 x 10 ⁻⁹	1.6 x 10 ⁻⁷	1.2 x 10 ⁻¹⁰
Vacuuming ^(b)	1.2 x 10 ⁻⁸	8.2 x 10 ⁻¹²	6.8 x 10 ⁻¹¹	1.7 x 10 ⁻⁹	5.8 x 10 ⁻¹²	1.4 x 10 ⁻¹¹	2.4 x 10 ⁻¹⁰	8.2 x 10 ⁻⁹	5.8 x 10 ⁻¹²
Oxyacetylene Torch and Power Hacksaw	5.6 x 10 ⁻⁶	3.8 x 10 ⁻⁹	3.2 x 10 ⁻⁸	7.8 x 10 ⁻⁷	2.7 x 10 ⁻⁹	6.7 x 10 ⁻⁹	1.1 x 10 ⁻⁷	3.8 x 10 ⁻⁶	2.7 x 10 ⁻⁹
Contaminated Concrete Surface Removal	5.6 x 10 ⁻⁶	3.8 x 10 ⁻⁹	3.2 x 10 ⁻⁸	7.8 x 10 ⁻⁷	2.7 x 10 ⁻⁹	6.7 x 10 ⁻⁹	1.1 x 10 ⁻⁷	3.8 x 10 ⁻⁶	2.7 x 10 ⁻⁹
Area 1: Drum Receiving	None	--(c)	--	--	--	--	--	--	--
Area 2: ADU Slurry Loop Chem. Decon. - Liquid Leak	8.5 x 10 ⁻¹⁰	5.8 x 10 ⁻¹³	4.8 x 10 ⁻¹²	1.2 x 10 ⁻¹⁰	4.1 x 10 ⁻¹³	1.0 x 10 ⁻¹²	1.7 x 10 ⁻¹¹	5.8 x 10 ⁻¹⁰	4.1 x 10 ⁻¹³
- Spray Leak	1.3 x 10 ⁻⁸	8.8 x 10 ⁻¹²	7.4 x 10 ⁻¹¹	1.8 x 10 ⁻⁹	6.2 x 10 ⁻¹²	1.6 x 10 ⁻¹¹	2.6 x 10 ⁻¹⁰	8.8 x 10 ⁻⁹	6.2 x 10 ⁻¹²
Oxyacetylene Torch and Power Hacksaw	7.1 x 10 ⁻⁶	4.8 x 10 ⁻⁹	4.0 x 10 ⁻⁸	9.9 x 10 ⁻⁷	3.4 x 10 ⁻⁹	8.5 x 10 ⁻⁹	1.4 x 10 ⁻⁷	4.8 x 10 ⁻⁶	3.4 x 10 ⁻⁹
Area 3: Sampling Station Oxyacetylene Torch and Power Hacksaw	1.2 x 10 ⁻⁶	8.2 x 10 ⁻¹⁰	6.8 x 10 ⁻⁹	1.7 x 10 ⁻⁷	5.8 x 10 ⁻¹⁰	1.4 x 10 ⁻⁹	2.4 x 10 ⁻⁸	8.2 x 10 ⁻⁷	5.8 x 10 ⁻¹⁰
Water-Jet	2.1 x 10 ⁻⁷	1.4 x 10 ⁻¹⁰	1.2 x 10 ⁻⁹	2.9 x 10 ⁻⁸	1.0 x 10 ⁻¹⁰	2.5 x 10 ⁻¹⁰	4.2 x 10 ⁻⁹	1.4 x 10 ⁻⁷	1.0 x 10 ⁻¹⁰
Area 4: Laboratory Water-Jet	1.3 x 10 ⁻⁹	8.8 x 10 ⁻¹³	7.4 x 10 ⁻¹²	1.8 x 10 ⁻¹⁰	6.2 x 10 ⁻¹³	1.6 x 10 ⁻¹²	2.6 x 10 ⁻¹¹	8.8 x 10 ⁻¹⁰	6.2 x 10 ⁻¹³

TABLE I.2-5. (Contd)

Area/General Method	Total ^(a) Atmospheric Release (Ci/yr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
Area 5: UF ₄ Slurry Processing									
Loop Chem. Decon.									
- Liquid Leak	5.2 x 10 ⁻¹⁰	3.5 x 10 ⁻¹³	3.0 x 10 ⁻¹²	7.3 x 10 ⁻¹¹	2.5 x 10 ⁻¹³	6.2 x 10 ⁻¹³	1.0 x 10 ⁻¹¹	3.5 x 10 ⁻¹⁰	2.5 x 10 ⁻¹³
- Spray Leak	7.8 x 10 ⁻⁹	5.3 x 10 ⁻¹²	4.4 x 10 ⁻¹¹	1.1 x 10 ⁻⁹	3.7 x 10 ⁻¹²	9.4 x 10 ⁻¹²	1.6 x 10 ⁻¹⁰	5.3 x 10 ⁻⁹	3.7 x 10 ⁻¹²
Area 6: Ore Conc. Diss.									
Loop Chem. Decon.									
- Liquid Leak	1.2 x 10 ⁻⁹	8.2 x 10 ⁻¹³	6.8 x 10 ⁻¹²	1.7 x 10 ⁻¹⁰	5.8 x 10 ⁻¹³	1.4 x 10 ⁻¹²	2.4 x 10 ⁻¹¹	8.2 x 10 ⁻¹⁰	5.8 x 10 ⁻¹³
- Spray Leak	1.9 x 10 ⁻⁸	1.3 x 10 ⁻¹¹	1.1 x 10 ⁻¹⁰	2.7 x 10 ⁻⁹	9.1 x 10 ⁻¹²	2.3 x 10 ⁻¹¹	3.8 x 10 ⁻¹⁰	1.3 x 10 ⁻⁸	9.1 x 10 ⁻¹²
Area 7: Solvent Extraction									
Loop Chem. Decon.									
- Liquid Leak	1.1 x 10 ⁻⁹	8.1 x 10 ⁻¹³	6.5 x 10 ⁻¹²	1.6 x 10 ⁻¹⁰	5.7 x 10 ⁻¹³	1.3 x 10 ⁻¹²	2.2 x 10 ⁻¹¹	8.0 x 10 ⁻¹⁰	5.7 x 10 ⁻¹³
- Spray Leak	1.6 x 10 ⁻⁸	1.1 x 10 ⁻¹¹	9.1 x 10 ⁻¹¹	2.2 x 10 ⁻⁹	7.7 x 10 ⁻¹²	1.9 x 10 ⁻¹¹	3.2 x 10 ⁻¹⁰	1.1 x 10 ⁻⁸	7.7 x 10 ⁻¹²
Oxyacetylene Torch and Power Hacksaw	2.0 x 10 ⁻⁷	1.4 x 10 ⁻¹⁰	1.1 x 10 ⁻⁹	2.7 x 10 ⁻⁸	9.6 x 10 ⁻¹¹	2.4 x 10 ⁻¹⁰	4.1 x 10 ⁻⁹	1.3 x 10 ⁻⁷	9.6 x 10 ⁻¹¹
Area 8: Uranyl Nitrate Conc.									
Oxyacetylene Torch and Power Hacksaw	2.0 x 10 ⁻⁷	1.4 x 10 ⁻¹⁰	1.1 x 10 ⁻⁹	2.7 x 10 ⁻⁸	9.6 x 10 ⁻¹¹	2.4 x 10 ⁻¹⁰	4.1 x 10 ⁻⁹	1.3 x 10 ⁻⁷	9.6 x 10 ⁻¹¹
Loop Chem. Decon.									
- Liquid Leak	8.5 x 10 ⁻¹⁰	5.8 x 10 ⁻¹³	4.8 x 10 ⁻¹²	1.2 x 10 ⁻¹⁰	4.1 x 10 ⁻¹³	1.0 x 10 ⁻¹²	1.7 x 10 ⁻¹¹	5.8 x 10 ⁻¹⁰	4.1 x 10 ⁻¹³
- Spray Leak	1.3 x 10 ⁻⁸	8.8 x 10 ⁻¹²	7.4 x 10 ⁻¹¹	1.8 x 10 ⁻⁹	6.2 x 10 ⁻¹²	1.6 x 10 ⁻¹¹	2.6 x 10 ⁻¹⁰	8.8 x 10 ⁻⁹	6.2 x 10 ⁻¹²
Area 9: Denitration									
Oxyacetylene Torch and Power Hacksaw	1.9 x 10 ⁻⁷	1.3 x 10 ⁻¹⁰	1.0 x 10 ⁻⁹	2.7 x 10 ⁻⁸	9.1 x 10 ⁻¹¹	2.3 x 10 ⁻¹⁰	3.8 x 10 ⁻⁹	1.3 x 10 ⁻⁷	9.1 x 10 ⁻¹¹

TABLE I.2-5. (Contd)

Area/General Method	Total ^(a) Atmospheric Release (Ci/yr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
Area 10: Nitric Acid Recovery Loop Chem. Decon. - Liquid Leak	4.0×10^{-12}	2.8×10^{-15}	2.2×10^{-14}	5.4×10^{-13}	1.9×10^{-15}	4.8×10^{-15}	8.2×10^{-14}	2.6×10^{-12}	1.9×10^{-15}
- Spray Leak	6.0×10^{-11}	4.1×10^{-14}	3.4×10^{-13}	8.4×10^{-12}	2.9×10^{-14}	7.2×10^{-14}	1.2×10^{-12}	4.1×10^{-11}	2.9×10^{-14}
Area 11: Reduction System Water-Jet	2.3×10^{-7}	1.6×10^{-10}	1.3×10^{-9}	3.2×10^{-8}	1.1×10^{-10}	2.8×10^{-10}	4.5×10^{-9}	1.6×10^{-7}	1.1×10^{-10}
Oxyacetylene Torch and Power Hacksaw	1.2×10^{-6}	8.2×10^{-10}	6.8×10^{-9}	1.7×10^{-7}	5.8×10^{-10}	1.4×10^{-9}	2.4×10^{-8}	8.2×10^{-7}	5.8×10^{-10}
Area 12: Hydrofluori- nation Oxyacetylene Torch and Power Hacksaw	2.4×10^{-7}	1.6×10^{-10}	1.4×10^{-9}	3.4×10^{-8}	1.2×10^{-10}	2.9×10^{-10}	4.8×10^{-9}	1.6×10^{-7}	1.2×10^{-10}
Area 13: Fluorination Oxyacetylene Torch and Power Hacksaw	2.5×10^{-6}	1.7×10^{-9}	1.4×10^{-8}	3.5×10^{-7}	1.2×10^{-9}	3.0×10^{-9}	5.0×10^{-8}	1.7×10^{-6}	1.2×10^{-9}
Loop Chem. Decon. - Liquid Leak	4.6×10^{-10}	3.1×10^{-13}	2.6×10^{-12}	6.4×10^{-11}	2.2×10^{-13}	5.5×10^{-13}	9.2×10^{-12}	3.1×10^{-10}	2.2×10^{-13}
- Spray Leak	6.2×10^{-9}	4.2×10^{-12}	3.5×10^{-11}	8.7×10^{-10}	3.0×10^{-12}	7.4×10^{-12}	1.2×10^{-10}	4.2×10^{-9}	3.0×10^{-12}
Area 14: Cylinder Loading	None	--	--	--	--	--	--	--	--
Area 15: Fluorine Generation Loop Chem. Decon. - Liquid Leak	4.0×10^{-12}	2.8×10^{-15}	2.2×10^{-14}	5.4×10^{-13}	1.9×10^{-15}	4.8×10^{-15}	8.2×10^{-14}	2.6×10^{-12}	1.9×10^{-15}
- Spray Leak	6.1×10^{-11}	4.2×10^{-14}	3.5×10^{-13}	8.7×10^{-12}	3.0×10^{-14}	7.4×10^{-14}	1.2×10^{-12}	4.2×10^{-11}	3.0×10^{-14}
Oxyacetylene Torch and Power Hacksaw	1.2×10^{-7}	8.2×10^{-11}	6.8×10^{-10}	1.7×10^{-8}	5.8×10^{-11}	1.4×10^{-10}	2.4×10^{-9}	8.2×10^{-8}	5.8×10^{-11}

TABLE I.2-5. (Contd)

Area/General Method	Total ^(a) Atmospheric Release (Ci/yr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
Areas 16 Through 24 Loop Chem. Decon.									
- Liquid Leak	2.3×10^{-11}	1.6×10^{-14}	1.3×10^{-13}	3.2×10^{-12}	1.1×10^{-14}	2.8×10^{-14}	4.5×10^{-13}	1.6×10^{-11}	1.1×10^{-14}
- Spray Leak	3.5×10^{-10}	2.4×10^{-13}	2.0×10^{-12}	4.9×10^{-11}	1.7×10^{-13}	4.2×10^{-13}	7.0×10^{-12}	2.4×10^{-10}	1.7×10^{-13}
Water-Jet	2.2×10^{-7}	1.4×10^{-10}	1.3×10^{-9}	3.0×10^{-8}	1.2×10^{-10}	2.6×10^{-10}	4.3×10^{-9}	1.6×10^{-7}	1.2×10^{-10}
Other Onsite Areas: ^(d)									
- Waste Extraction Pond Exhumation and Packaging	4.8×10^{-3}	3.3×10^{-6}	2.7×10^{-5}	6.7×10^{-4}	2.3×10^{-6}	5.8×10^{-6}	9.6×10^{-5}	3.3×10^{-3}	2.3×10^{-6}
- Fluoride Waste Lagoon Exhuma- tion and Packag- ing	1.2×10^{-5}	8.2×10^{-9}	6.8×10^{-8}	1.7×10^{-6}	5.8×10^{-9}	1.4×10^{-8}	2.4×10^{-7}	8.2×10^{-6}	5.8×10^{-9}
- Misc. Buried Waste Exhumation and Packaging	7.3×10^{-5}	5.0×10^{-8}	4.2×10^{-7}	1.0×10^{-5}	3.5×10^{-8}	8.8×10^{-8}	1.5×10^{-6}	5.0×10^{-5}	3.5×10^{-8}
Decon. Totals	4.9×10^{-3}	3.3×10^{-6}	2.8×10^{-5}	6.9×10^{-4}	2.4×10^{-6}	5.9×10^{-6}	9.8×10^{-5}	3.3×10^{-3}	2.4×10^{-6}
Preparations for Safe Storage Totals	4.5×10^{-7}	3.1×10^{-10}	2.6×10^{-9}	6.3×10^{-8}	2.2×10^{-10}	5.4×10^{-10}	9.0×10^{-9}	3.1×10^{-7}	2.2×10^{-10}

(a) For comparison, dose calculations are based on annual releases.

(b) The method shown is considered as part of preparations for safe storage as well as part of DECON.

(c) Dashes indicate that no calculations are necessary.

(d) The areas shown are for onsite waste storage areas only. The calculated releases are for the exhumation, packaging, and offsite disposal option only.

TABLE I.2-6. Radiation Doses to the Population from Atmospheric Releases During Routine DECON Tasks in the Reference UF₆ Conversion Plant(a)

Area/General Method	Total ^(b) Atmospheric Release (Ci/yr)	First-Year Dose (man-rem)				Fifty-Year Committed Dose Equivalent (man-rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
All Areas Initial Comprehensive Radiation Survey(c)	4.8 x 10 ⁻⁸	3 x 10 ⁻⁸	3 x 10 ⁻⁷	8 x 10 ⁻⁶	2 x 10 ⁻⁸	6 x 10 ⁻⁸	1 x 10 ⁻⁶	4 x 10 ⁻⁵	2 x 10 ⁻⁸
Water-Jet(c)	1.5 x 10 ⁻⁷	1 x 10 ⁻⁷	9 x 10 ⁻⁷	3 x 10 ⁻⁵	5 x 10 ⁻⁸	2 x 10 ⁻⁷	3 x 10 ⁻⁶	1 x 10 ⁻⁴	5 x 10 ⁻⁸
Sweeping(c)	2.4 x 10 ⁻⁷	2 x 10 ⁻⁷	1 x 10 ⁻⁶	4 x 10 ⁻⁵	8 x 10 ⁻⁸	3 x 10 ⁻⁷	5 x 10 ⁻⁶	2 x 10 ⁻⁴	8 x 10 ⁻⁸
Vacuuming(c)	1.2 x 10 ⁻⁸	8 x 10 ⁻⁹	7 x 10 ⁻⁸	2 x 10 ⁻⁶	4 x 10 ⁻⁹	1 x 10 ⁻⁸	2 x 10 ⁻⁷	9 x 10 ⁻⁶	4 x 10 ⁻⁹
Oxyacetylene Torch and Power Hacksaw	5.6 x 10 ⁻⁶	4 x 10 ⁻⁶	3 x 10 ⁻⁵	9 x 10 ⁻⁴	2 x 10 ⁻⁶	7 x 10 ⁻⁶	1 x 10 ⁻⁴	4 x 10 ⁻³	2 x 10 ⁻⁶
Contaminated Concrete Surface Removal	4.0 x 10 ⁻⁸	3 x 10 ⁻⁸	2 x 10 ⁻⁷	7 x 10 ⁻⁶	1 x 10 ⁻⁸	5 x 10 ⁻⁸	8 x 10 ⁻⁷	3 x 10 ⁻⁵	1 x 10 ⁻⁸
Area 1: Drum Receiving	None	--(d)	--	--	--	--	--	--	--
Area 2: ADU Slurry Loop Chem. Decon. - Liquid Leak	8.5 x 10 ⁻¹⁰	6 x 10 ⁻¹⁰	5 x 10 ⁻⁹	1 x 10 ⁻⁷	3 x 10 ⁻¹⁰	1 x 10 ⁻⁹	2 x 10 ⁻⁸	6 x 10 ⁻⁷	3 x 10 ⁻¹⁰
- Spray Leak	1.3 x 10 ⁻⁸	9 x 10 ⁻⁹	7 x 10 ⁻⁸	2 x 10 ⁻⁶	4 x 10 ⁻⁹	2 x 10 ⁻⁸	3 x 10 ⁻⁷	1 x 10 ⁻⁵	4 x 10 ⁻⁹
Oxyacetylene Torch and Power Hacksaw	7.1 x 10 ⁻⁶	5 x 10 ⁻⁶	4 x 10 ⁻⁵	1 x 10 ⁻³	2 x 10 ⁻⁶	8 x 10 ⁻⁶	1 x 10 ⁻⁴	5 x 10 ⁻³	2 x 10 ⁻⁶
Area 3: Sampling Station Oxyacetylene Torch and Power Hacksaw	1.2 x 10 ⁻⁶	8 x 10 ⁻⁷	7 x 10 ⁻⁶	2 x 10 ⁻⁴	4 x 10 ⁻⁷	1 x 10 ⁻⁶	2 x 10 ⁻⁵	9 x 10 ⁻⁴	4 x 10 ⁻⁷
Water-Jet	2.1 x 10 ⁻⁷	1 x 10 ⁻⁷	1 x 10 ⁻⁶	4 x 10 ⁻⁵	7 x 10 ⁻⁸	2 x 10 ⁻⁷	4 x 10 ⁻⁶	2 x 10 ⁻⁴	7 x 10 ⁻⁸
Area 4: Laboratory Water-Jet	1.3 x 10 ⁻⁹	9 x 10 ⁻¹⁰	7 x 10 ⁻⁹	2 x 10 ⁻⁷	4 x 10 ⁻¹⁰	2 x 10 ⁻⁹	3 x 10 ⁻⁸	1 x 10 ⁻⁶	4 x 10 ⁻¹⁰

TABLE I.2-6. (Contd)

Area/General Method	Total ^(b) Atmospheric Release (Ci/yr)	First-Year Dose (man-rem)				Fifty-Year Committed Dose Equivalent (man-rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
Area 5: UF ₄ Slurry Processing									
Loop Chem. Decon.									
- Liquid Leak	5.2 x 10 ⁻¹⁰	3 x 10 ⁻¹⁰	3 x 10 ⁻⁹	9 x 10 ⁻⁸	2 x 10 ⁻¹⁰	6 x 10 ⁻¹⁰	1 x 10 ⁻⁸	4 x 10 ⁻⁷	2 x 10 ⁻¹⁰
- Spray Leak	7.8 x 10 ⁻⁹	5 x 10 ⁻⁹	4 x 10 ⁻⁸	1 x 10 ⁻⁶	2 x 10 ⁻⁹	9 x 10 ⁻⁹	2 x 10 ⁻⁷	6 x 10 ⁻⁶	2 x 10 ⁻⁹
Area 6: Ore Conc. Diss.									
Loop Chem. Decon.									
- Liquid Leak	1.2 x 10 ⁻⁹	8 x 10 ⁻¹⁰	7 x 10 ⁻⁹	2 x 10 ⁻⁷	4 x 10 ⁻¹⁰	1 x 10 ⁻⁹	2 x 10 ⁻⁸	9 x 10 ⁻⁷	4 x 10 ⁻¹⁰
- Spray Leak	1.9 x 10 ⁻⁸	1 x 10 ⁻⁸	1 x 10 ⁻⁷	3 x 10 ⁻⁶	6 x 10 ⁻⁹	2 x 10 ⁻⁸	4 x 10 ⁻⁷	1 x 10 ⁻⁵	6 x 10 ⁻⁹
Area 7: Solvent Extraction									
Loop Chem. Decon.									
- Liquid Leak	1.1 x 10 ⁻⁹	8 x 10 ⁻¹⁰	7 x 10 ⁻⁹	2 x 10 ⁻⁷	4 x 10 ⁻¹⁰	1 x 10 ⁻⁹	2 x 10 ⁻⁸	9 x 10 ⁻⁷	4 x 10 ⁻¹⁰
- Spray Leak	1.6 x 10 ⁻⁸	1 x 10 ⁻⁸	9 x 10 ⁻⁸	3 x 10 ⁻⁶	5 x 10 ⁻⁹	2 x 10 ⁻⁸	3 x 10 ⁻⁷	1 x 10 ⁻⁵	5 x 10 ⁻⁹
Oxyacetylene Torch and Power Hacksaw	2.0 x 10 ⁻⁷	1 x 10 ⁻⁷	1 x 10 ⁻⁶	4 x 10 ⁻⁵	7 x 10 ⁻⁸	2 x 10 ⁻⁷	4 x 10 ⁻⁶	2 x 10 ⁻⁷	7 x 10 ⁻⁸
Area 8: Uranyl Nitrate Conc.									
Oxyacetylene Torch and Power Hacksaw	2.0 x 10 ⁻⁷	1 x 10 ⁻⁷	1 x 10 ⁻⁶	4 x 10 ⁻⁵	7 x 10 ⁻⁸	2 x 10 ⁻⁷	4 x 10 ⁻⁶	2 x 10 ⁻⁴	7 x 10 ⁻⁸
Loop Chem. Decon.									
- Liquid Leak	8.5 x 10 ⁻¹⁰	6 x 10 ⁻¹⁰	5 x 10 ⁻⁹	1 x 10 ⁻⁷	3 x 10 ⁻¹⁰	1 x 10 ⁻⁹	2 x 10 ⁻⁸	6 x 10 ⁻⁷	3 x 10 ⁻¹⁰
- Spray Leak	1.3 x 10 ⁻⁸	9 x 10 ⁻⁹	7 x 10 ⁻⁸	2 x 10 ⁻⁶	4 x 10 ⁻⁹	2 x 10 ⁻⁸	3 x 10 ⁻⁷	1 x 10 ⁻⁵	4 x 10 ⁻⁹
Area 9: Denitration									
Oxyacetylene Torch and Power Hacksaw	1.9 x 10 ⁻⁷	1 x 10 ⁻⁷	1 x 10 ⁻⁶	3 x 10 ⁻⁵	6 x 10 ⁻⁸	2 x 10 ⁻⁷	4 x 10 ⁻⁶	1 x 10 ⁻⁴	6 x 10 ⁻⁸

TABLE I.2-6. (Contd)

Area/General Method	Total ^(b) Atmospheric Release (Ci/yr)	First-Year Dose (man-rem)				Fifty-Year Committed Dose Equivalent (man-rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
Area 10: Nitric Acid Recovery Loop Chem. Decon. - Liquid Leak	4.0×10^{-12}	2×10^{-12}	2×10^{-11}	8×10^{-10}	1×10^{-12}	4×10^{-12}	8×10^{-11}	4×10^{-9}	1×10^{-12}
- Spray Leak	6.0×10^{-11}	4×10^{-11}	3×10^{-10}	1×10^{-8}	2×10^{-11}	7×10^{-11}	1×10^{-9}	4×10^{-8}	2×10^{-11}
Area 11: Reduction System Water-Jet	2.3×10^{-7}	2×10^{-7}	1×10^{-6}	4×10^{-5}	7×10^{-8}	3×10^{-7}	4×10^{-6}	2×10^{-4}	7×10^{-8}
Oxyacetylene Torch and Power Hacksaw	1.2×10^{-6}	8×10^{-7}	7×10^{-6}	2×10^{-4}	4×10^{-7}	1×10^{-6}	2×10^{-5}	9×10^{-4}	4×10^{-7}
Area 12: Hydrofluori- nation Oxyacetylene Torch and Power Hacksaw	2.4×10^{-7}	2×10^{-7}	1×10^{-6}	4×10^{-5}	8×10^{-8}	3×10^{-7}	5×10^{-5}	2×10^{-4}	8×10^{-8}
Area 13: Fluorination Oxyacetylene Torch and Power Hacksaw	2.5×10^{-6}	2×10^{-6}	1×10^{-5}	4×10^{-4}	8×10^{-7}	3×10^{-6}	5×10^{-5}	2×10^{-3}	8×10^{-7}
Loop Chem. Decon. - Liquid Leak	4.6×10^{-10}	3×10^{-10}	3×10^{-9}	8×10^{-8}	1×10^{-10}	6×10^{-10}	9×10^{-9}	3×10^{-7}	1×10^{-10}
- Spray Leak	6.2×10^{-9}	4×10^{-9}	4×10^{-8}	1×10^{-6}	2×10^{-9}	7×10^{-9}	1×10^{-7}	5×10^{-6}	2×10^{-9}
Area 14: Cylinder Loading	None	--	--	--	--	--	--	--	--
Area 15: Fluorine Generation Loop Chem. Decon. - Liquid Leak	4.0×10^{-12}	2×10^{-12}	2×10^{-11}	8×10^{-10}	1×10^{-12}	4×10^{-12}	8×10^{-11}	4×10^{-9}	1×10^{-12}
- Spray Leak	6.1×10^{-11}	4×10^{-11}	4×10^{-10}	1×10^{-8}	2×10^{-11}	7×10^{-11}	1×10^{-9}	5×10^{-8}	2×10^{-11}
Oxyacetylene Torch and Power Hacksaw	1.2×10^{-7}	8×10^{-8}	7×10^{-7}	2×10^{-5}	4×10^{-8}	1×10^{-7}	2×10^{-6}	9×10^{-5}	4×10^{-8}

TABLE I.2-6. (Contd)

Area/General Method	Total ^(b) Atmospheric Release (Ci/yr)	First-Year Dose (man-rem)				Fifty-Year Committed Dose Equivalent (man-rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
Areas 16 Through 24 Loop Chem. Decon.									
- Liquid Leak	2.3×10^{-11}	8×10^{-9}	7×10^{-8}	2×10^{-6}	4×10^{-9}	1×10^{-8}	2×10^{-7}	9×10^{-6}	4×10^{-9}
- Spray Leak	3.5×10^{-10}	2×10^{-10}	2×10^{-9}	6×10^{-8}	1×10^{-10}	4×10^{-10}	7×10^{-9}	3×10^{-7}	1×10^{-10}
Water-Jet	2.2×10^{-7}	1×10^{-7}	1×10^{-6}	4×10^{-5}	7×10^{-8}	3×10^{-7}	4×10^{-6}	2×10^{-4}	7×10^{-8}
Other Onsite Areas: ^(e)									
- Waste Extraction Pond Exhumation and Packaging	4.8×10^{-3}	3×10^{-3}	3×10^{-2}	8×10^{-1}	2×10^{-3}	6×10^{-3}	1×10^{-1}	4×10^0	2×10^{-3}
- Fluoride Waste Lagoon Exhuma- tion and Packag- ing	1.2×10^{-5}	8×10^{-6}	7×10^{-5}	2×10^{-3}	4×10^{-6}	1×10^{-5}	2×10^{-4}	9×10^{-3}	4×10^{-6}
- Misc. Buried Waste Exhumation and Packaging	7.3×10^{-5}	5×10^{-5}	4×10^{-4}	1×10^{-2}	2×10^{-5}	9×10^{-5}	2×10^{-3}	6×10^{-2}	2×10^{-5}
DECON Totals	4.9×10^{-3}	3×10^{-3}	3×10^{-2}	8×10^{-1}	2×10^{-3}	6×10^{-3}	1×10^{-1}	4×10^0	2×10^{-3}
Preparations for Safe Storage Totals	4.5×10^{-5}	3×10^{-7}	3×10^{-6}	8×10^{-5}	1×10^{-8}	5×10^{-7}	9×10^{-6}	3×10^{-4}	1×10^{-8}

(a) The calculated doses are shown to one significant figure and are to the 3.5×10^6 people residing within a radius of 80 km of the site.

(b) For comparison, dose calculations are based on annual releases.

(c) The method shown is considered as part of preparations for safe storage as well as part of DECON.

(d) Dashes indicate that no calculations are necessary.

(e) The areas shown are for onsite waste storage areas only. The calculated releases are for the exhumation, packaging, and offsite disposal option only.

TABLE I.2-7. Postulated Accidental Atmospheric Releases During DECON and Preparations for Safe Storage at the Reference UF₆ Conversion Plant

Incident	Comments	Frequency of Occurrence ^(a)	Total Airborne Radioactivity Inside Building or Control Envelope (Ci)	Contamination Control System Transmission Factor ^(b)	Total ^(c) Atmospheric Release (Ci/h)
Oxyacetylene Explosion	Explosion During Area 2 Piping and Equipment Removal	Medium	1.8×10^{-2}	1×10^{-1}	1.8×10^{-3}
Contamination Control Envelope Rupture	Leakage During Area 2 Piping and Equipment Removal	High	2.9×10^{-4}	1×10^{-1}	2.9×10^{-5}
Vacuum Filter-Bag ^(d) Rupture	Rupture of a Full Filter-Bag During Surface Cleaning	Medium	2.4×10^{-4}	1×10^{-1}	2.4×10^{-5}
Contaminated Metal Cutting	Additional Cutting of Highly Contaminated Piping in Area 2	High	7.2×10^{-5}	1×10^{-1}	7.2×10^{-6}
Gross Leak During Loop Chemical Decontamination	A Leak 10 Times Larger Than the Routine Leak in Area 6	Low	1.9×10^{-6}	1×10^{-1}	1.9×10^{-7}
Spilling of Contaminated Solid Wastes	Spill of Contaminated Concrete From Surface Removal	High	1.3×10^{-6}	1×10^{-1}	1.3×10^{-7}
Explosion of LPG Leaked From-End Loader	4.5 kg of LPG Leaked During Con-From Surface Removal	Low	1.0×10^{-7}	1.0	1.0×10^{-7}
Contaminated Sweeping Compound Fire ^(d)	A Fire Consumes 0.5 m ³ of Used Sweeping Compound	Medium	2.0×10^{-7}	1×10^{-1}	2.0×10^{-8}
Combustible Waste Fire ^(d)	1 m ³ of Combustible Waste is Involved in a Fire	High	2.0×10^{-8}	1×10^{-1}	2.0×10^{-9}
Loss of Services	Loss of Water Supply, Electrical Power, or Air Flow	--(e)	--	--	--
Natural Phenomena	Flood, Earthquakes, Tornadoes or High Winds	--	--	--	--
Aircraft Crashes	Comparison to Routine Operation Probability	--	--	--	--
Man-Caused Events	Detailed Analysis is Beyond the Scope of This Study	--	--	--	--

(a) The frequency of occurrence considers not only the probability of the accident, but also the probability of an atmosphere release of the calculated magnitude. The frequency of occurrence is listed as "High" if the occurrence of a release is 10^{-2} per year, as "Medium" if between 10^{-2} and 10^{-5} , and as "Low" if $\leq 10^{-5}$.

(b) The assumed building transmission factor is 1×10^{-1} .

(c) For comparison, all accidental releases are assumed to occur in a 1-h period.

(d) The accident shown applies to both DECON and preparations for safe storage.

(e) Dashes indicate that no calculations were attempted for the incident listed.

explosions can occur when acetylene and oxygen are incorrectly mixed. The degree of explosive violence depends on how closely the gas mixture approximates the ratio for complete combustion. Some maximum pressures and pressure rise rates from three acetylene/air mixtures in the explosive range are shown in Table I.2-8.⁽²³⁾

TABLE I.2-8. Example Explosive Data for Acetylene/Air Mixtures

<u>Percent of Acetylene in Mixture</u>	<u>Maximum Explosive Pressure (kPa)</u>	<u>Maximum Rate of Explosive Pressure Rise (kPa/sec)</u>
5	540	17 000
13	1 000	8 300
20	690	9 700

Oxyacetylene explosions can occur from such causes as flow reversals, nozzle obstructions, and flashbacks (a flare going back up the gas hose).^(24,25) This accident is postulated to occur during removal of piping and equipment in Area 2, the ADU slurry receiving and storage area. This operation is conducted in a contamination control envelope with an assumed volume of 1000 m³. The airborne material generated will fill this volume with a concentration of 1×10^{-1} g/m³.⁽⁴⁾ Thus, a total of 100 g of radioactive dust with a specific activity of 1.3×10^{-6} Ci/g is made airborne during the accident. The total airborne activity is calculated to be 1.3×10^{-4} Ci.

It is reasonable to assume that the contamination control envelope HEPA filter system undergoes routine maintenance to ensure its full effectiveness. The mass of material deposited on filters without causing flow restrictions or an excessive pressure drop (>980 Pa of water gauge at rated flows) varies with filter design and the properties of the deposited material.^(26,27) A HEPA filter dust loading of 2.3 kg is assumed. Three pairs of HEPA and roughing filters in the contamination control envelope are calculated to contain about 14 kg of uranium dust. If all of this material is made airborne by the oxyacetylene explosion, and using the specific activity of 1.3×10^{-6} Ci/g for the uranium mixture, a total of about 1.8×10^{-2} Ci are made airborne from the filters.

Some radioactivity will be deposited on the inside surfaces of the contamination control envelope, and it is assumed to be made airborne in the building by the explosion. The total radioactivity generated during DECON of Area 2 is 7.1×10^{-4} Ci (see Table I.2-4). Assuming that 0.2% of this radioactivity is deposited on the interior surfaces of the control envelope, and that all of it is made airborne by the explosion, an additional 1.4×10^{-6} Ci could enter the building air.

The total radioactivity made airborne in the building by the postulated oxyacetylene explosion is found by summing the three previous airborne generation terms. They are: the radioactivity generated by the explosion inside the contamination control envelope (1.3×10^{-4} Ci), the radioactivity removed from the control envelope HEPA filter system by the explosion (1.8×10^{-2} Ci), and the radioactivity removed from interior surfaces in the control envelope (1.4×10^{-6}). The total airborne radioactivity in the building is 1.8×10^{-2} Ci. Thus, the amount of radioactivity released to the environment is found by multiplying the amount of material airborne in the building by the building transmission factor. The total atmospheric release is 1.8×10^{-2} . An oxyacetylene explosion resulting in this magnitude of airborne release is estimated to have a "medium" frequency of occurrence.

Contamination Control Envelope Rupture. Rupture of the contamination control envelope (greenhouse) during oxyacetylene cutting is assumed to have a "high" frequency of occurrence. Small globules of molten metal have been measured to be projected up to 9 m from a cutting operation.⁽²⁸⁾ It is postulated that such molten particles penetrate the sheet plastic walls and increase the leakage from the contamination control envelope from 0.1 to 0.5. This increase in leakage is assumed to occur during the removal of piping and equipment in Area 2, the ADU slurry receiving and storage area. It is assumed that this large leak goes undetected for the duration of the task, resulting in an increase in the atmospheric release of from 7.1×10^{-6} Ci to 3.6×10^{-5} Ci. The net additional release is calculated to be 2.9×10^{-5} Ci.

Vacuum Filter-Bag Rupture. Sharp objects, such as metal shards, could rupture a filter-bag during surface cleaning operations involving the use of a vacuum cleaner. When the filter-bag is ruptured, all of the collected material in the bag is assumed to become airborne in the building because of the mechanical and aerodynamic forces of the $1.4\text{-m}^3/\text{min}$ vacuum cleaner air flow. To maximize the calculation of the atmospheric release, the bag rupture is assumed to occur at the time just prior to bag change (i.e., when the filter bag is full). If vacuuming is assumed to remove 0.5 of the surface contamination from $3.7 \times 10^3 \text{ m}^2$ of area, and if the average contamination level is 1 g/m^2 , about $1.8 \times 10^3 \text{ g}$ are removed. Frequent filter-bag changes are anticipated in dusty areas. Assuming 10 bag changes during the entire task, each bag contains an average of 180 g of uranium. Assuming all 180 g becomes airborne during the accident, with a specific activity of $1.3 \times 10^{-6} \text{ Ci/g}$, the airborne radioactivity is $2.4 \times 10^{-4} \text{ Ci}$. The total release through the building ventilation system is calculated to be $2.4 \times 10^{-5} \text{ Ci}$. This accident is estimated to have a "medium" frequency of occurrence.

Accidental Cutting of Contaminated Metal. Segmentation of metal components in the reference facility after those components have been decontaminated is an integral part of DECON. Although rigorous monitoring and administrative controls are exercised to prevent the accidental cutting of components that have not been decontaminated, improper coordination of activities could lead to such an accident.

This accident is postulated to occur during cutting of highly contaminated piping in Area 2. Half of the cutting involves a contamination level of 1 kg/m^2 of internal surface. If an additional 10% of the piping is not decontaminated below this level prior to cutting, the airborne radioactivity is calculated to increase by $7.2 \times 10^{-5} \text{ Ci}$. The resulting release to the atmosphere is calculated to be $7.2 \times 10^{-6} \text{ Ci}$. This accident is estimated to have a high frequency of occurrence.

Gross Leak During Loop Chemical Decontamination. A massive failure of piping during loop chemical decontamination is estimated to have a "low" frequency of occurrence. There have been no such accidents in 30 years of experience with nuclear systems. Massive failures are unknown in high-pressure central steam boilers, for which there is considerably more operating experience than is currently available in the nuclear industry.⁽²⁹⁾

Pipe failures can result from improper system designs, the use of poor quality materials, base metal defects, fabrication defects, or inadequate service conditions.⁽³⁰⁾ The loop chemical decontamination takes place about 40 years after the piping systems are put in service. In chemical decontamination, a caustic chemical is circulated in piping and equipment to remove built-up internal surface contamination. The chemical solution could deteriorate the pipe or the welds and create localized leaks. Accidental leakage could also be introduced because of human error during the chemical decontamination.

Liquid and spray leaks are assumed to occur as part of the routine release analysis. A gross leak about 10 times larger than the largest spray leak is assumed for this accident. Thus, 1% of the liquid in the system is assumed to be made airborne. Such a leak would be large enough to be easily detected and is assumed to last only 1 hour. The largest routine spray leak is calculated to occur during DECON tasks in Area 6. Increasing the routine spray leak by a factor of 10 results in a leak of 1.9×10^{-6} Ci in the building, of which 1.9×10^{-7} Ci is released to the atmosphere. A gross leak resulting in a release of this magnitude is estimated to have a "low" frequency of occurrence.

Accidental Spill of Contaminated Solid Wastes. Another substantial operation is the handling of contaminated solid wastes. Accidental dumping or spilling of these wastes would lead to a release of airborne contamination from the plant.

Contaminated concrete dust is generated as a result of concrete drilling, and the dust is collected for disposal. It is postulated that this dust is accidentally spilled during handling. The collected dust should be damp from the water mist used during drilling, and when spilled would exhibit the behavior of damp sand. However, for this analysis, it is assumed that the concrete dust is sufficiently dry to produce a quasi-stable mass concentration of 1×10^{-2} g/m³ in a 100-m³ space surrounding the site of the spill. Thus, as much as 1 g of dust containing 1.3×10^{-6} Ci could be released to the room ventilation system, resulting in an estimated release to the atmosphere of 1.3×10^{-7} Ci of uranium. A spill of contaminated waste resulting in a release of this magnitude is estimated to have a "high" frequency of occurrence.

Explosion of LPG Leaked from a Front-End Loader. The use of a liquefied propane gas-powered front-end loader for loading concrete rubble and moving light equipment is assumed in this study, since this method is consistent with current industrial methods. It is anticipated that by the time large-scale decommissioning operations are required, an alternative method with safer fuel supply will be available and used instead of the LPG-powered front-end loader.

Flammability limits for LPG range from a mixture of 2.1 volume % to a mixture of 9.5 volume %.^(31,23) Information on the rate of pressure rise from explosions of various mixtures of propane in air is given in Table I.2-9.

TABLE I.2-9. LPG Explosion Information^(a)

<u>Concentration in Air (volume %)</u>	<u>Pre-Detonation Pressure (kPa)</u>	<u>Rate of Pres- sure Rise (kPa/sec)</u>
3	510	8 600
5	660	17 000
7	330	240

(a) Summarized from information in Reference 23.

An accidental leak of LPG is postulated to occur during the loading of concrete rubble from contaminated concrete surface spalling. A realistic leak of LPG of 4.5 kg is assumed, which would require about 110 m³ of air to produce a flammable mixture. An explosion of this mixture would produce a maximum pressure surge of less than 240 kPa, which would cause little structural damage unless the explosion occurred after the mixture is drawn into the building ventilation ductwork. This accident is assumed to occur in the ductwork where the force of the explosion is assumed to cause failure of the ventilation system. Loss of the ventilation blowers would aerodynamically isolate the building from the atmosphere. Remedial action such as sealing off vent outlets, would further limit the atmospheric release.

The force of the explosion is assumed to dislodge contaminated material located in the ductwork and filters. Most of the contaminated concrete removed

is in areas of low contamination, and the average concentration encountered in routine operations is assumed to be about 1 g of uranium per m^2 of surface. This is further mixed with the concrete dust generated, so that the average uranium concentration in the dust is about 1.0×10^{-5} $\mu\text{Ci/g}$. The amount of contaminated dust deposited in the ventilation system is estimated to be about 100 kg. If 10% of this material is released during the explosion, about 10 kg or 0.1 μCi of uranium are released to the atmosphere. An explosion of leaked LPG resulting in this magnitude of contamination is estimated to have a "low" frequency of occurrence.

Contaminated Sweeping Compound Fire. Sweeping compound is composed of sawdust treated with oil or other additives to enhance the collection of loose surface contamination. A fire is postulated to occur in used sweeping compound containing radioactivity removed from floor surfaces. The highest contamination level encountered in sweeping occurs on isolated floor surfaces with a contamination level of 10 g of uranium per m^2 . The average concentration on floor surfaces is 1 g of uranium per m^2 . It is assumed that an average of 2.5×10^{-4} m^3 of sweeping compound is used for each m^2 of surface swept. It is also assumed that the density of the sweeping compound is about the same as light wood, or about 6.0×10^5 g/m^3 .⁽³³⁾ It is reasonable to assume that the sweeping compound will be stored in small quantities; therefore a fire is postulated to occur in 0.5 m^3 (or 3×10^2 kg) of used sweeping compound. This amount of sweeping compound will cover about 2,000 m^2 of floor. Assuming that half of the floor contamination is removed during sweeping, about 1 kg of uranium is contained in the used sweeping compound.

The release fraction from the fire is assumed to be similar to that measured from burning waste, 1.5×10^{-4} .⁽³⁴⁾ Thus, about 0.15 g of uranium is released inside the building, and about 0.015 g is released to the atmosphere. With a specific activity of 1.3 $\mu\text{Ci/g}$, this equals an atmospheric release of about 2.0×10^{-8} Ci. A release of this magnitude from a fire in contaminated sweeping compound is estimated to have a "medium" frequency of occurrence.

Combustible Waste Fire. Absorbent materials, such as rags or paper wipes, are assumed to be used for a variety of purposes during DECON and are discarded after use. Materials that have come into contact with contaminated surfaces hold small quantities of radionuclides. Anticontamination clothing (coveralls,

caps, hoods, or shoe covers) become contaminated through use. Plastics are generally considered combustible, although many kinds do not burn readily or are self-extinguishing.

It is postulated that contaminated combustible waste containing low levels of radioactivity ignites and burns in a working area. Large amounts of these wastes are not likely to accumulate in one place, so it is assumed that one cubic meter containing 100 g of uranium is involved in a combustible waste fire. Using the release fraction from burning waste of 1.5×10^{-4} ,⁽³⁴⁾ the total release in the building is calculated to be 1.5×10^{-2} g, or 2.0×10^{-8} Ci. The atmospheric release resulting from this fire is estimated to be 2.0×10^{-9} Ci. A fire in combustible waste that results in a release of this magnitude is estimated to have a "high" frequency of occurrence.

Loss of Services. Loss of services, such as water supply, electrical power, or air flow, are immediately apparent to the workers performing the various DECON tasks. In the case of manual tasks, the operator would discontinue the task immediately upon the loss of services. Most remotely controlled tasks would be monitored frequently enough to almost immediately detect the loss of a service. Since the loss of services constitutes a lesser magnitude release than other postulated accidents, no further analysis is attempted in this study.

I.2.2.2 Natural Phenomena

The designs of structures, systems, and components are based on withstanding the most severe natural phenomena recorded for the site, plus appropriate margins to account for uncertainties in the recorded data. It is assumed that the structural integrity of the plant is preserved during the operational lifetime of the plant. It is further assumed that the structural integrity is preserved during decommissioning as long as required for safety. It is estimated that, while natural phenomena can cause severe damage to the reference UF_6 conversion plant resulting in severe radiological public safety impacts, they are low probability events. Also, the impact of these events will be less than the impacts calculated in the FSAR design-basis accidents for the operating plant. For this reason, no detailed analysis of the impacts of these phenomena is attempted beyond the brief discussions that follow.

Floods. The FSAR considers runoff floods, surges, seiches, wave action, tsunamis, wind-generated wave action, and combinations of these phenomena. Also, a probable maximum flood is proposed. The plant is designed to be safe from damage by such a flood or other potential flooding.

Earthquakes. It is estimated that the probability of an earthquake that would occur and cause damage to the plant is 10^{-7} per year.⁽³⁵⁾ At this probability, earthquake-induced accidents are extremely unlikely.

Tornadoes. The probability of a large tornado striking the site is on the average less than 5×10^{-6} per year.⁽³⁵⁾ At this probability, tornado-induced effects during decommissioning are extremely unlikely, and they are not considered further in this study.

High Winds. The design-basis wind velocity for all Category I and Category II structures is 168 km/hr at 9.1 m above nominal ground elevation. For this study it is assumed that DECON tasks do not reduce the ability of these structures to withstand high winds, and no releases are anticipated.

I.2.2.3 Aircraft Crashes

Each facility site has its own potential risk from aircraft crashes based on its proximity to airports and air-traffic lanes. However, current information leads to the conclusion that the risk of damage from aircraft crashes is low for most sites.⁽³⁶⁾ Moreover, the risk is not escalated by DECON operations. The aircraft crash probability for most plants is very low (10^{-6} to 10^{-8} per year) and no further analysis is attempted.

I.2.2.4 Man-Caused Events

Accidental airborne releases of radioactivity resulting from man-caused events could cover a wide spectrum of magnitudes, ranging from releases induced by casual trespassers to releases induced by armed terrorists. A detailed analysis of man-caused events during decommissioning is beyond the scope of this study. Thus, no airborne release calculations are made.

I.2.2.5 Radiation Doses from Postulated DECON Accidents

Radiation doses to the maximum-exposed individual from accidental releases of radionuclides during DECON are calculated using the radiation dose models discussed in Appendix E. The first-year doses and the fifty-year committed dose equivalents to the maximum-exposed individual from inhalation for the accidents listed in Table I.2-7 are listed in Table I.2-10. Each accidental release is assumed to occur for one hour so that a comparison of the accidents can be made. The atmospheric dispersion ($\overline{x}/Q'$) value used for the accidents is $6.5 \times 10^{-4} \text{ sec/m}^3$.

I.3 PASSIVE SAFSTOR PUBLIC SAFETY ASSESSMENT

The second decommissioning alternative considered for the reference UF_6 conversion plant is passive SAFSTOR. This alternative is divided into three potential phases: preparations, safe storage, and deferred decontamination. In this section, the atmospheric releases of radioactivity and the resulting radiation doses to the public from preparations for safe storage are discussed. No atmospheric releases are identified during safe storage, and the resulting doses from deferred decontamination are assumed to be the same as for DECON. The analysis conducted for preparations for safe storage uses the same data and methods defined in Section I.2 for DECON.

I.3.1 Routine Tasks During Preparations for Safe Storage

After completion of final process inventory cleanout, the remaining uranium is spread throughout the processing areas in relatively inaccessible locations. No major decontamination or inventory reduction tasks are required. The total amount of uranium left in the facility during safe storage is estimated to be about 1200 kg. This amount should not present a problem since access to the facility is restricted.

As described in Appendix G, the activities undertaken during preparations for safe storage are quite passive, and only limited releases of radioactivity are anticipated. The only tasks that may result in atmospheric releases are those identified for "all areas" in Table I.2-3 (shown by footnote c). The

TABLE I.2-10. Radiation Doses to the Maximum-Exposed Individual from Atmospheric Releases During DECON

Area/General Method	Total ^(a) Atmospheric Release (Ci/yr)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
Oxyacetylene Explosion	1.8×10^{-3}	1.5×10^{-4}	1.5×10^{-3}	5.1×10^{-2}	1.2×10^{-5}	8.3×10^{-4}	3.1×10^{-1}	1.3×10^{-1}	1.2×10^{-5}
Contamination Control Envelope Rupture	2.9×10^{-5}	2.5×10^{-6}	2.5×10^{-5}	8.1×10^{-4}	1.8×10^{-7}	2.9×10^{-6}	4.9×10^{-5}	2.0×10^{-3}	1.8×10^{-7}
Vacuum Filter-Bag Rupture ^(b)	2.4×10^{-5}	2.0×10^{-6}	2.0×10^{-5}	6.7×10^{-4}	1.5×10^{-7}	2.4×10^{-6}	4.1×10^{-5}	1.7×10^{-3}	1.5×10^{-7}
Contaminated Metal Cutting	7.2×10^{-6}	6.1×10^{-7}	6.1×10^{-6}	2.0×10^{-4}	4.5×10^{-8}	7.2×10^{-7}	1.2×10^{-5}	5.0×10^{-4}	4.5×10^{-8}
Gross Leak During Loop Chem. Decon.	1.9×10^{-7}	1.6×10^{-8}	1.6×10^{-7}	5.3×10^{-6}	1.2×10^{-9}	1.9×10^{-8}	3.2×10^{-7}	1.3×10^{-5}	1.2×10^{-9}
Spilling of Cont. Solid Wastes	1.3×10^{-7}	1.1×10^{-8}	1.1×10^{-7}	3.6×10^{-6}	8.2×10^{-10}	1.3×10^{-8}	2.2×10^{-7}	9.1×10^{-6}	8.2×10^{-10}
Explosion of LPG Leaked From a Front- End Loader	1.0×10^{-7}	8.5×10^{-9}	8.5×10^{-8}	2.8×10^{-6}	6.3×10^{-10}	1.0×10^{-8}	1.7×10^{-7}	7.0×10^{-6}	6.3×10^{-10}
Contaminated Sweeping Compound Fire ^(b)	2.0×10^{-8}	1.7×10^{-9}	1.7×10^{-8}	5.6×10^{-7}	1.3×10^{-10}	2.0×10^{-9}	3.4×10^{-8}	1.4×10^{-6}	1.3×10^{-10}
Combustible Waste Fire ^(b)	2.0×10^{-9}	1.7×10^{-10}	1.7×10^{-9}	5.6×10^{-8}	1.3×10^{-11}	2.0×10^{-10}	3.4×10^{-9}	1.4×10^{-7}	1.3×10^{-11}

(a) For comparison, all releases are assumed to occur in a 1-hour period.

(b) The accident shown applies to both DECON and preparations for safe storage.

public safety analysis data for these tasks are identical to the data identified for DECON, as shown in the table. The resulting atmospheric releases for these tasks are noted by footnote c in Table I.2-4.

First-year doses and fifty-year committed dose equivalents to both the maximum-exposed individual and the population residing within 80 km of the site are shown in Tables I.2-5 and I.2-6, respectively. No tasks unique to safe storage are identified that result in atmospheric releases or public radiation doses, thus no further analysis is required.

I.3.2 Postulated Accidents During Preparations for Safe Storage

No accidents are postulated for preparations for safe storage that are different from the accidents postulated for DECON. Since none of the tasks defined for preparations for safe storage involve loop chemical decontamination or the removal of contaminated concrete, the accidents defined for these DECON tasks do not apply. The only accidents that are considered are those shown in Table I.2-7 by footnote d. Again, the frequency of occurrence shown in Table I.2-7 for each accident is based on both the probability of the accident and on the probability of an atmospheric release of the calculated magnitude. The frequency of occurrence is listed as "high" if the occurrence of a release of similar or greater magnitude is greater than 10^{-2} , as "medium" if between 10^{-2} and 10^{-5} , and as "low" if less than 10^{-5} . The radiation doses calculated for the maximum-exposed individual from these accidents are shown in Table I.2-10 by footnote b.

I.4 OCCUPATIONAL DOSE ESTIMATES FOR DECON AND SAFSTOR OF THE REFERENCE UF₆ CONVERSION PLANT

Detailed estimates of the work requirements for each DECON and SAFSTOR task identified in this study and dose rates estimated for each area in the main process building are used to estimate the occupational exposures resulting from both decommissioning alternatives. It is assumed that internal exposure of decommissioning personnel is effectively prevented by the use of contamination control envelopes, breathing masks, and other protective measures. Although external dose rates are reduced substantially by the final inventory cleanout, additional cleaning and temporary shielding may reduce localized dose rates even further.

The results of these analyses are presented in Tables I.4-1 for DECON, I.4-2 for preparations for safe storage, and I.4-3 for safe storage. The estimates are based on a 6-hour effective working day in radiation zones for each worker. It is assumed that operations and craft personnel spend 50% of their time in the location of highest dose rate in each area. The balance of all work time is spent in locations where the radiation dose rate is characteristic of the most accessible locations in each area. The radiation dose rates are assumed to be similar to those measured for an operating uranium fuel fabrication facility, as reported in Reference 37.

I.5 RADIATION DOSES FROM WASTE TRANSPORT FOR DECOMMISSIONING THE REFERENCE UF₆ CONVERSION PLANT

During decommissioning of the reference UF₆ conversion plant, radioactive waste materials are packaged and shipped offsite for burial. These wastes are assumed to be shipped to a commercial low-level waste burial facility located about 800 km from the site. All wastes are assumed to be shipped by truck. The methods used to estimate radiation doses to transportation workers and to members of the public from waste shipments are given in Reference 5. Radiation doses received by workers unloading the radioactive materials at the repository or disposal site are not considered in this study since they are assumed to occur at separate licensed facilities.

I.5.1 Technical Approach

The following assumptions are made about truck shipments of radioactive materials:

1. Each shipment of waste from the main process building contains enough material to result in the maximum exposure rates allowed by regulations. Department of Transportation (DOT) regulations set the following exposure limits:
 - 1000 mR/hr at 1 m from the external surface of any package transported in a closed vehicle
 - 200 mR/hr at the external surface of the vehicle

TABLE I.4-1. Estimated Occupational Radiation Doses for DECON

Area	Foremen			Technicians			Health Physicists			Craftsmen			Area Total Dose (man-rem)
	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	
Area 1: Drum Receiving	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	120	4.4×10^{-3}	3.7×10^{-5}	15	5.6×10^{-4}	3.7×10^{-5}	15	5.6×10^{-4}	6.6×10^{-3}
Area 2: ADU Slurry	1.5×10^{-4}	30	4.5×10^{-3}	3.7×10^{-3}	180	6.7×10^{-1}	3.7×10^{-3}	15	5.6×10^{-2}	3.7×10^{-3}	30	1.1×10^{-1}	8.4×10^{-1}
Area 3: Sampling Station	1.5×10^{-4}	30	4.5×10^{-3}	3.7×10^{-3}	210	7.8×10^{-1}	3.7×10^{-3}	15	5.6×10^{-2}	3.7×10^{-3}	30	1.1×10^{-1}	9.5×10^{-1}
Area 4: Laboratory	3.7×10^{-5}	90	3.3×10^{-3}	3.7×10^{-5}	450	1.7×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	150	5.6×10^{-3}	2.7×10^{-2}
Area 5: UF_4 Slurry	1.5×10^{-4}	165	2.5×10^{-2}	3.7×10^{-3}	900	3.3×10^0	3.7×10^{-3}	60	2.2×10^{-1}	3.7×10^{-3}	150	5.6×10^{-1}	4.1×10^0
Area 6: Ore Concentrate Dissolution	1.5×10^{-4}	180	2.7×10^{-2}	3.7×10^{-3}	1230	4.6×10^0	3.7×10^{-3}	135	5.0×10^{-1}	3.7×10^{-3}	195	7.2×10^{-1}	5.8×10^0
Area 7: Solvent-Extraction	1.5×10^{-4}	315	4.7×10^{-2}	3.7×10^{-3}	1680	6.2×10^0	3.7×10^{-3}	120	4.4×10^{-1}	3.7×10^{-3}	285	1.0×10^0	7.7×10^0
Area 8: Uranyl Nitrate	3.7×10^{-5}	270	1.0×10^{-3}	1.2×10^{-4}	1700	2.0×10^{-1}	1.2×10^{-4}	120	1.4×10^{-2}	1.2×10^{-4}	225	2.7×10^{-2}	2.4×10^{-1}
Area 9: Denitration	3.7×10^{-5}	285	1.0×10^{-2}	1.2×10^{-4}	1830	2.2×10^{-1}	1.2×10^{-4}	120	1.4×10^{-2}	1.2×10^{-4}	225	2.7×10^{-2}	2.7×10^{-1}
Area 10: Nitric Acid	3.7×10^{-5}	150	5.6×10^{-3}	1.2×10^{-4}	930	1.1×10^{-1}	1.2×10^{-4}	60	7.2×10^{-3}	1.2×10^{-4}	120	1.4×10^{-2}	1.4×10^{-1}
Area 11: Reduction Systems	3.7×10^{-5}	270	1.0×10^{-3}	1.2×10^{-4}	1700	2.0×10^{-1}	1.2×10^{-4}	120	1.4×10^{-2}	1.2×10^{-4}	225	8.3×10^{-3}	2.2×10^{-1}
Area 12: Hydrofluorination	3.7×10^{-5}	225	8.3×10^{-3}	1.2×10^{-4}	1200	1.4×10^{-1}	1.2×10^{-4}	90	1.1×10^{-2}	1.2×10^{-4}	390	4.7×10^{-2}	2.1×10^{-1}
Area 13: Fluorination	3.7×10^{-5}	195	7.2×10^{-3}	1.2×10^{-4}	1350	1.6×10^{-1}	1.2×10^{-4}	493	5.9×10^{-2}	1.2×10^{-4}	225	2.7×10^{-2}	2.5×10^{-1}
Area 14: Cylinder Loading	3.7×10^{-5}	150	5.5×10^{-3}	3.7×10^{-5}	930	3.4×10^{-2}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	270	1.0×10^{-2}	2.1×10^{-2}
Area 15: F_2 Production	3.7×10^{-5}	75	2.8×10^{-3}	3.7×10^{-5}	450	1.7×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	75	2.8×10^{-3}	2.4×10^{-2}
Area 16: F_2 Cell	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	360	1.3×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	1.8×10^{-2}
Area 17: Instrument Repair	3.7×10^{-5}	105	3.9×10^{-3}	3.7×10^{-5}	510	1.9×10^{-2}	3.7×10^{-5}	45	1.7×10^{-3}	3.7×10^{-5}	390	1.4×10^{-2}	3.9×10^{-2}
Area 18: Ammonia Cracking	3.7×10^{-5}	75	2.8×10^{-3}	3.7×10^{-5}	405	1.5×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	225	8.3×10^{-3}	2.7×10^{-2}
Area 19: Waste Gas Disposal	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	360	1.3×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	75	2.8×10^{-3}	1.9×10^{-2}

TABLE I.4-1. (Contd)

Area	Foremen			Technicians			Health Physicists			Craftsmen			Area Total Dose (man-rem)
	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	
Area 20: H ₂ Condensing	3.7×10^{-5}	165	6.1×10^{-3}	3.7×10^{-5}	102	3.8×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	150	5.6×10^{-3}	1.8×10^{-2}
Area 21: Temp. Radwaste	3.7×10^{-5}	30	1.1×10^{-3}	1.2×10^{-4}	210	2.5×10^{-2}	1.2×10^{-4}	30	1.1×10^{-3}	1.2×10^{-4}	30	3.6×10^{-3}	3.1×10^{-2}
Area 22: Decon. Area	3.7×10^{-5}	30	1.1×10^{-3}	1.2×10^{-4}	120	1.4×10^{-2}	1.2×10^{-4}	15	1.8×10^{-3}	1.2×10^{-4}	30	3.6×10^{-3}	2.1×10^{-2}
Area 23: Incinerator	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	450	1.7×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	120	4.4×10^{-3}	2.5×10^{-2}
Area 24: Laundry	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	420	1.6×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	45	1.7×10^{-3}	2.1×10^{-2}
Area 25: Overall Cleanup	3.7×10^{-5}	90	3.3×10^{-3}	3.7×10^{-5}	660	2.4×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.1×10^{-2}
Building Area Totals		3 200	1.8×10^{-1}		18 500	1.7×10^{-1}		1 810	1.4×10^{-1}		3 770	2.7×10^{-1}	2.1×10^{-1}
Site Areas: (a)													
Extraction Waste Pond	1.5×10^{-4}	600	9.0×10^{-2}	3.7×10^{-3}	4 440	1.6×10^{-1}	3.7×10^{-3}	240	8.9×10^{-1}	3.7×10^{-3}	390	1.4×10^0	1.8×10^1
Fluoride Waste Lagoons	1.5×10^{-4}	660	9.9×10^{-2}	3.7×10^{-3}	4 800	1.8×10^{-1}	3.7×10^{-3}	150	5.6×10^{-1}	3.7×10^{-3}	450	1.7×10^0	2.0×10^1
Misc. Buried Waste	1.5×10^{-4}	75	1.1×10^{-2}	3.7×10^{-3}	510	1.9×10^0	3.7×10^{-3}	30	1.1×10^{-1}	3.7×10^{-3}	120	4.4×10^{-1}	2.4×10^0
Drum Storage	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	360	1.3×10^{-2}	3.7×10^{-5}	15	5.6×10^{-4}	3.7×10^{-5}	60	2.2×10^{-3}	1.8×10^{-2}
Cylinder Storage	3.7×10^{-5}	165	6.1×10^{-3}	3.7×10^{-5}	900	3.3×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	180	6.7×10^{-3}	4.7×10^{-2}
Plant Stack	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	450	1.7×10^{-2}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	120	4.4×10^{-3}	2.5×10^{-2}
Overall Cleanup	3.7×10^{-5}	120	4.4×10^{-3}	3.7×10^{-5}	930	3.4×10^{-2}	3.7×10^{-5}	45	1.7×10^{-3}	3.7×10^{-5}	0	0	4.0×10^{-2}
Site Area Totals		1 740	2.0×10^{-1}		12 400	3.6×10^{-1}		540	1.6×10^{-1}		1 320	3.6×10^0	4.1×10^1
DECON Totals		4 940	3.8×10^{-1}		30 900	5.3×10^{-1}		2 350	3.0×10^0		5 090	6.2×10^0	6.2×10^1

(a) Only the waste exhumation, packaging, and offsite disposal option is considered.

TABLE I.4-2. Estimated Occupational Doses for Preparations for Safe Storage

Task	Foremen			Technicians			Health Physicists			Craftsmen			
	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Average Exposure Rate (R/h)	Exposure Time (man-hr)	Total Dose (man-rem)	Task Total Dose (man-rem)
Equipment Deactivation													
Audit of pumps and pipelines	3.7×10^{-5}	30	1.1×10^{-3}	1.2×10^{-4}	60	7.2×10^{-3}	1.2×10^{-4}	30	3.6×10^{-3}	1.2×10^{-4}	60	7.2×10^{-3}	1.9×10^{-2}
Secure valves, hoops, conveyors	3.7×10^{-5}	30	1.1×10^{-3}	1.2×10^{-4}	90	1.1×10^{-2}	1.2×10^{-4}	30	3.6×10^{-3}	1.2×10^{-4}	60	7.2×10^{-3}	2.3×10^{-2}
Disconnect unneeded services	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	120	4.4×10^{-3}	9.9×10^{-3}
Tag equipment to identify status	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	7.7×10^{-3}
Safety audit	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	7.7×10^{-3}
Isolation of areas													
Install building locks and signs	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	0	0	3.7×10^{-5}	60	2.2×10^{-3}	4.4×10^{-3}
Building systems check	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	6.6×10^{-3}
Install fence and outside locks	7.4×10^{-6}	30	2.2×10^{-4}	7.4×10^{-6}	30	2.2×10^{-4}	7.4×10^{-6}	0	0	7.4×10^{-6}	30	2.2×10^{-4}	6.6×10^{-4}
Final Preparations													
Complete Offsite shipment of uranium	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	120	4.4×10^{-3}	3.7×10^{-6}	60	2.2×10^{-3}	3.7×10^{-5}	15	5.6×10^{-4}	6.1×10^{-3}
Install intrusion alarm system	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	180	6.7×10^{-3}	1.1×10^{-2}
Install radiation monitoring system	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	60	2.2×10^{-3}	3.7×10^{-5}	30	1.1×10^{-3}	3.7×10^{-5}	30	4.4×10^{-3}	8.8×10^{-3}
Comprehensive radiation survey	1.2×10^{-4}	30	3.6×10^{-3}	1.2×10^{-4}	120	1.4×10^{-2}	1.2×10^{-5}	120	1.4×10^{-4}	1.2×10^{-4}	0	0	1.8×10^{-2}
Building Area Totals		390	1.6×10^{-2}		810	5.1×10^{-2}		480	1.8×10^{-2}		825	4.0×10^{-2}	1.2×10^{-1}
Onsite Waste Stabilization													
Drain ponds and lagoons and cover with weighted plastic	1.2×10^{-4}	60	7.2×10^{-3}	1.2×10^{-4}	360	4.3×10^{-2}	1.2×10^{-4}	60	7.2×10^{-3}	1.2×10^{-4}	60	7.2×10^{-3}	6.5×10^{-2}
Preparations for Safe Storage Totals		450	2.3×10^{-2}		1170	9.3×10^{-2}		540	2.5×10^{-2}		885	4.7×10^{-2}	1.8×10^{-1}

TABLE I.4-3. Estimated Occupational Radiation Doses During Safe Storage

Employment Category	Average Exposure Rate (R/h)	Total Exposure (man-hours/yr)	Total Dose (man-rem/yr)	Total Dose For 10-Years of Safe Storage (man-rem)
Facility Manager	3.7×10^{-5}	375	1.4×10^{-2}	1.4×10^{-1}
Maintenance and Operations Man	9.4×10^{-5}	750	7.0×10^{-2}	7.0×10^{-1}
Secretary ^(a)	0	0	0	0
Third-Party Inspectors	6.6×10^{-5}	45	3.0×10^{-3}	3.0×10^{-2}
TOTALS		1170	8.7×10^{-2}	8.7×10^{-1}

(a) Not expected to work in a radiation zone, but included in the analysis.

- 10 mR/hr at any point 2 m from the vehicle
- 2 mR/hr at any normally occupied position in the vehicle.⁽³⁸⁾

Shipments of wastes from the extraction pond and fluoride lagoons are assumed to have dose rates of 10% of these limits.

2. For each truck shipment of radioactive waste, two truck drivers spend 12 hours inside the cab and 1 hour outside of the cab at a distance of 2 m from the cargo.
3. For each truck shipment of radioactive waste, two garagemen each spend 10 minutes at an average distance of 2 m from the payload.
4. For each truck shipment of radioactive waste, the maximum-exposed individual is located 30 m from the route. His dose rate is calculated to be 1.2×10^{-7} rem per shipment for wastes from the main process building, and 1.2×10^{-8} rem per shipment for wastes from the extraction pond or fluoride lagoon.
5. The population density along the transport corridors is 120 persons/km².

6. All shipments maintain an average speed of 65 km/hr; thus, the cumulative dose to the public is 2.3×10^{-6} man-rem/km for wastes from the main process building, and 2.3×10^{-7} man-rem/km for wastes from the extraction pond or fluoride lagoon.⁽⁵⁾
7. The $\bar{X}/Q'$ value used for calculating the radiation dose to the maximum-exposed individual from transportation accidents is 3×10^{-2} sec/m³, as discussed in detail in Appendix N of Reference 39. This individual is assumed to be located 100 m downwind from the accident.

Using these assumptions, the following sections contain discussions of the radiation doses during DECON. Both routine and accident conditions are discussed.

1.5.2 Doses from Routine Offsite Transport of Wastes

The number of radioactive waste shipments required during DECON for wastes from the main process building is given in Table 10.1-5 as 51. In addition, 1877 shipments of waste recovered from the extraction ponds and fluoride lagoons are required. Using these numbers of shipments and the assumptions discussed in the technical approach, radiation doses to transport workers and to the public are calculated and are listed in I.5-1.

1.5.3 Transportation Accidents

Transportation accidents have a wide range of severities. Most accidents occur at low vehicle speeds and have relatively minor consequences. In general, as speed increases, accident severity also increases. However, accident severity is not a function of vehicle speed only. Other factors, (e.g., the type of accident, the kind of equipment involved, and the location of the accident) can have an important bearing on accident severity.

Furthermore, damage to a package in a transportation accident is not directly related to accident severity. In a series of accidents of the same severity, or in a single accident involving a number of packages, damage to packages may vary from none to extensive. In relatively minor accidents, serious damage to packages can occur from impacts on sharp objects or from being struck by other cargo. Conversely, even in very severe accidents, damage to packages may be minimal.

TABLE I.5-1. Calculated Radiation Doses from Routine Radioactive Waste Transportation

Alternative/Group	Radiation Dose per Shipment (man-rem) ^(a)	Number of Shipments	Total Population Dose per Group (man-rem) ^(b)
<u>DECON-Building Wastes</u>			
Truck Drivers	6.7×10^{-2}	51	3.4
Garagemen	3.3×10^{-3}	51	<u>0.17</u>
Total Transport Worker Dose			3.6
Onlookers	5.0×10^{-3}	51	0.26
General Public	1.8×10^{-3}	51	<u>0.092</u>
Total Public Dose			0.35
<u>DECON-Site Wastes</u>			
Truck Drivers	6.7×10^{-3}	1877	13
Garagemen	3.3×10^{-4}	1877	<u>0.62</u>
Total Transport Worker Dose			14
Onlookers	5.0×10^{-4}	1877	0.94
General Public	1.8×10^{-4}	1877	<u>0.38</u>
Total Public Dose			1.3
<u>Totals for DECON</u>			
Truck Drivers	-- ^(c)	1928 ^(d)	16
Garagemen	--	1928 ^(d)	<u>0.79</u>
Total Transport Worker Dose	--		17
Onlookers	--	1928 ^(d)	1.2
General Public	--	1928 ^(d)	<u>0.47</u>
Total Public Dose			1.7

(a) Based on one-way trips of 800 km.

(b) All doses are rounded to two significant figures.

(c) Dashes indicate that weighted average dose rates were not separately calculated.

(d) Total shipments are found by adding building waste plus site waste shipments.

The probabilities of truck accidents in this study are based on accident data supplied in Reference 5. Accidents are classified into five severity categories as functions of vehicle speed and fire duration. The five categories and their associated probabilities for truck accidents are shown in Table I.5-2.

TABLE I.5-2. Transportation Accident Severity Categories^(a)

Severity	Vehicle Speed (km/h)	Fire Duration (h)	Probability per Vehicle-km Truck
Minor	0-50	0	2×10^{-7}
	0-50	<0.5	4×10^{-9}
	50-80	0	6×10^{-7}
Moderate	0-50	0.5-1.0	3×10^{-11}
	50-80	<0.5	6×10^{-9}
	80-110	0	2×10^{-7}
	80-110	<0.5	3×10^{-9}
Severe	0-50	>1.0	3×10^{-12}
	50-80	>1.0	6×10^{-12}
	50-80	0.5-1.0	4×10^{-12}
	80-110	0.5-1.0	4×10^{-12}
	>110	0	6×10^{-11}
	>110	<0.5	6×10^{-11}

(a) Summarized from material in Reference 5.

The wastes shipped from the UF₆ conversion plant during decommissioning will all require Type A containers. Some of the wastes will consist of compacted combustible material contaminated with uranium. Accident descriptions involving Type A containers are given in Table I.5-3 for minor accidents and in Table I.5-4 for severe accidents. Type A waste packages containing natural uranium (transport Group III) are limited to contain 3 Ci per container.⁽⁴⁰⁾ Since natural uranium is estimated to have a low specific activity (1.3 μ Ci/g from Table C.2-1), and since most combustible wastes come from manual operations conducted in areas of low contamination level, the quantity of material per container is estimated to be about 5.5×10^{-4} Ci. This quantity is based on the concentration of contaminated sweeping compound discussed in the

TABLE I.5-3. Minor Accidents for Type A Packages During Truck Transportation

Accident Description	Sequence of Events	Atmospheric Release
Truck Collision or Overturn Involves Waste Containers	<ol style="list-style-type: none"> 1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. Type A packages may rupture, but no fire occurs. 4. Accident is reported to local and federal officials. 5. Packages recovered. 	None
Truck Collision or Overturn and a Minor Fire (1/2 Hour or Less) Involves 1 Type A Waste Container	<ol style="list-style-type: none"> 1. Collision or overturn accident occurs. 2. Truck leaves roadway and may overturn. 3. One Type A waste container is involved in a fire with 1/2 hour or less duration. 4. Accident is reported to local and federal officials. 5. Packages recovered. 	Release fraction of 5×10^{-4} of the contents of one waste package is assumed. (34)

TABLE I.5-4. Severe Accidents for Type A Packages During Truck Transportation

Accident Description	Sequence of Events	Atmospheric Release
Truck Collision or Overturn and a Major Fire (1 Hour or Longer) Involving 40 Type A Waste Containers.	<ol style="list-style-type: none"> 1. Collision or overturn accident occurs. 2. Truck leaves roadway at high speed and overturns 3. Type A packages rupture and 40 waste containers are involved in a fire. 4. Accident is reported to local and federal officials. 5. Packages recovered. 	Release fraction of 5×10^{-4} of the contents of 40 waste packages is assumed. (34)

fire scenario in Section I.2.2.1. For the minor accident, one package is assumed to burn and release $5 \times 10^{-4(34)}$ of the radioactivity in the container, or about 2.8×10^{-7} Ci. The resulting 50-year committed dose equivalent to the maximum-exposed individual is 0.9 mrem to the lung. The minor transportation accident is estimated to have a low frequency of occurrence.

For the severe accident, the contents of 40 waste containers (each containing 5.5×10^{-4} Ci of uranium) are assumed to burn, releasing a total of 1.1×10^{-5} Ci. The resulting 50-year committed dose equivalent to the maximum-exposed individual is 35 mrem to the lung. The severe accident is also estimated to have a low frequency of occurrence. A complete listing of the doses to the organs of the maximum-exposed individual from these accidents is given in Table I.5-5.

TABLE I.5-5. Radiation Doses to the Maximum-Exposed Individual for Releases from Selected Transportation Accidents

Incident	Total Atmospheric Release (ci/h) ^(a)	First-Year Dose (rem)				Fifty-Year Committed Dose Equivalent (rem)			
		Total Body	Bone	Lung	GI-LLI	Total Body	Bone	Lung	GI-LLI
Minor Transportation Accident	2.8×10^{-7}	1.1×10^{-5}	1.1×10^{-5}	3.6×10^{-4}	-- ^(b)	1.3×10^{-5}	2.2×10^{-5}	9.0×10^{-4}	--
Severe Transportation Accident	1.1×10^{-5}	4.3×10^{-4}	4.3×10^{-4}	1.4×10^{-2}	--	5.3×10^{-5}	8.4×10^{-4}	3.5×10^{-2}	--

(a) All releases are assumed to occur in a 1-hour period, for comparison purposes.

(b) For inhalation, no dose result to the GI-LLI from uranium.

I.6 HAZARDOUS CHEMICAL RELEASES DURING DECOMMISSIONING OPERATIONS

The two primary sources of chemical hazards are residual chemicals from plant operation and chemical agents used during decontamination.

I.6.1 Chemical Residuals from Plant Operations

Several potentially hazardous chemicals are used or produced in operations conducted at the reference UF₆ conversion plant. Residues from these operations may be present during the decommissioning operation, and therefore must be dealt with prior to unconditional release of the site. Table I.6-1 lists the chemicals which are of concern, either because of their relative toxicity, or because of the total amount which may be present on the site at the time of decommissioning.

TABLE I.6-1. Potential Chemical Residuals from Plant Operations

Material	Form	Source
Acetone	Liquid	Process Use
Anhydrous Ammonia	Liquid	Process Use
Aqueous Ammonia	Liquid	Process Use
Hydrochloric Acid	Liquid	Process Use
Hydrofluoric Acid	Liquid	Process Use
Nitric Acid	Liquid	Process Use
Sodium Hydroxide	Liquid	Process Use
Sulfuric Acid	Liquid	Process Use
Degreaser (solvent)	Liquid	Process Use
Helium	Gas	Process Use
Hydrogen	Gas	Process Use
Nitrogen	Gas	Process Use
Oxygen	Gas	Process Use
Propane	Gas	Process Use
Uranium Hexafluoride	Gas/ Solid	Product
Uranyl Nitrate	Solid	Feed Material
Uranium Dioxide	Solid	Feed Material
Calcium Oxide	Solid	Process Use
Calcium Fluoride	Solid	Waste Material
Ammonium Nitrate	Solid	Waste Material

Most of the chemical species listed will pose no residual toxicity problem. Those in liquid form, primarily strong acids and bases, are stored in tanks and closed systems. The inventory (if any) remaining in the storage and piping systems after plant shutdown will be treated in one of the following ways: solidification and transfer to a chemical waste disposal area; neutralization and disposal in a landfill, and where economically justified, salvage or resale to another user.

The process gases used in the plant also pose little hazard. The inventories in the respective systems will be low at the time of the final plant shutdown. With the exception of hydrogen and propane which are flammable, the gases used

present almost no biological hazard. The small amounts of propane and hydrogen remaining in those systems will be easily diluted and flushed with an inert gas such as nitrogen before dismantlement of the piping systems begins.

The uranium compounds used as feed material present both radiological and chemical toxicity hazards. The chemical toxicity of uranium compounds is largely a function of their solubility in body fluids; the more soluble compounds such as nitrates and fluorides tend to translocate to the kidney, the site of major biological damage. The threshold limit value (TLV) for all uranium compounds has been set at 0.2 mg U/m^3 .⁽⁴¹⁾ The concentration to which the maximum-exposed individual is exposed as a result of the chronic releases from routine decommissioning activities averaged over a 1-year period is $8.9 \times 10^{-7} \text{ mg U/m}^3$. Even if the largest single release (3.7 kg U from onsite waste exhumation) is assumed to occur in a 4-month period, the maximum-exposed individual would be exposed to an air concentration of $2.7 \times 10^{-6} \text{ mg U/m}^3$, or about 1×10^{-5} times the TLV for 1 week.

The waste material collected over the plant operating life, principally calcium fluoride, is potentially hazardous. Calcium fluoride has a TLV of 2.5 mg/m^3 . Using the chronic release $\bar{X}/Q$ for the maximum-exposed individual (7.6×10^{-6}) and a permissible exposure of 1/100 TLV, the maximum average CaF_2 release rate which can be tolerated is 3.3 g/sec. Accordingly, transportation and packaging methods must be selected with consideration given to the toxic nature of the fluoride. Occupational exposure to CaF_2 will be limited by conventional industrial contamination control techniques.

I.6.2 Chemical Residuals from Decontamination Agents

Chemical decontamination agents such as detergents, acid solutions, degreasers, and organic solvents may be used for various tasks at the UF_6 conversion plant for decontamination. Most such decontamination agents are of low chemical toxicity and are widely used in non-nuclear industry applications.

Nitric acid, which may be used to flush chemical process systems and to wipe down contaminated surfaces, is neutralized with caustic before disposal. The resultant nitrate salts are relatively innocuous. Detergent residues on any cleaned surface should be negligible so long as the decontamination process is carried to completion with a final rinse of the cleaned surfaces.

Degreasers and solvents such as Freon may be vaporized and released to the atmosphere during cleaning operations. These materials are relatively non-toxic, but they may pose local hazards to decommissioning personnel if allowed to accumulate in unventilated areas.

Under normal conditions, the proper selection and use of chemical decontamination agents should preclude any significant chemical toxicity hazard to workers or to the general public.

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