ENGINEERING EVALUATION OF ALTERNATIVES
FOR THE DISPOSITION OF NIAGARA FALLS
STORAGE SITE, ITS RESIDUES AND WASTES

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By

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FOREWORD

This document presents certain engineering, occupational radiation exposure, construction schedule, maintenance and surveillance, and cost details related to four basic scenarios for the long-term management of the existing radioactive wastes and residues stored at the Niagara Falls Storage Site (NFSS). This document was prepared to support the Environmental Impact Statement (EIS) currently being prepared by Argonne National Laboratory for the U.S. Department of Energy (DOE). Three general scenarios were identified in DOE's Notice of Intent to prepare an EIS, published in the U.S. Federal Register on February 1, 1983. These scenarios were:

1. No Action -- Continue interim storage of all existing wastes and residues at NFSS in the diked and capped containment area in the southwest corner in NFSS. (Interim storage construction will take place from 1983 to 1985.)
2. Long-Term Management at NFSS -- Carry out actions in addition to the interim storage actions to improve the containment of the stored wastes and residues, or modify the form of some of the residues to reduce potential migration of contaminated materials from NFSS.
3. Long-Term Management at Other Sites -- Remove all wastes and residues from NFSS and transport them to other sites for long-term management or disposal.

The public scoping procedure for the EIS resulted in a decision to consider a fourth scenario which will also receive comprehensive environmental impact analysis in the EIS:

4. Offsite Storage of Some Residues (K-65, L-30, L-50, and F-32) -- Remove some of the more radioactive residues to another site for long-term storage. Leave the remaining wastes and residues at NFSS for long-term management.

This document is not intended to stand alone; rather, it serves as a reference engineering document for use in National Environmental Policy Act (NEPA) activities and as an information source for a future decision by DOE of a final disposition alternative for the NFSS. Other reports, including the EIS, an ocean disposal study being prepared by Sandia National Laboratory, as well as studies referenced in this document, will also be used in the decision-making process.

The scope and arrangement of the information presented in this document is not intended to arbitrarily restrain the range of alternatives within each scenario that DOE may determine should be evaluated in the EIS, or ultimately considered by the decision makers in determining the long-range disposition of the NFSS or any of the materials currently stored there. Additional alternatives for the final disposition of the NFSS may be developed at a later date using portions of alternatives that are currently developed or that may be proposed in the future.

This document contains conceptual engineering information, details of which may be subject to change. No actual designs for the final disposition have been prepared. The engineering approaches described for implementing the alternatives could change as new information becomes available. Significant deviation from the interim remedial action activities planned at the site would change the baseline conditions assumed in this document.

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ACRONYMS

AEC	Atomic Energy Commission
ALARA	As Low As Reasonably Achievable
ANL	Argonne National Laboratory
DOE	Department of Energy
DOT	Department of Transportation
EIS	Environmental Impact Statement
EPA	Environmental Protection Agency
FRC	Federal Radiation Council
FUSRAP	Formerly Utilized Sites Remedial Action Program
HEPA	High Energy Particulate Arrestor
ICRP	International Commission on Radiological Protection
KAPL	Knolls Atomic Power Laboratory
LOOW	Lake Ontario Ordnance Works
LSA	Low Specific Activity
MED	Manhattan Engineer District
msl	Mean Sea Level
NCRP	National Council on Radiation Protection
NEPA	National Environmental Policy Act
NFSS	Niagara Falls Storage Site
NRC	Nuclear Regulatory Commission
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
SFMP	Surplus Facilities Management Program

ABBREVIATIONS

Ci, curie
 Ci/g, curies per gram
 cm, centimeter
 cm/s, centimeter per second
 dis/s, disintegrations per second
 dpm, disintegrations per minute
 ft, foot
 ft³, cubic foot
 ft³/s, cubic foot per second
 ft/yr, feet per year
 ft/100 yr, feet per hundred years
 gal, gallon
 gal/min, gallons per minute
 ha, hectare
 in., inch
 kg, kilogram
 kg/m², kilograms per square meter
 kg/m³, kilograms per cubic meter
 km, kilometer
 km/h, kilometers per hour
 kWh, kilowatt hour
 lb, pound
 lb/ft², pounds per square foot
 lb/ft³, pounds per cubic foot
 l/s, liters per second
 m, meter
 m³, cubic meter
 m/100 yr, meters per hundred years
 mg/g, milligrams per gram
 mg/l, milligrams per liter
 mi, mile
 uR/h, microrentgen per hour
 ml, milliliter
 mR/h, milliroentgen per hour
 mrad/h, millirad per hour
 mrem, millirem
 mrem/yr, millirem per year
 pCi/g, picocuries per gram
 pCi/l, picocuries per liter
 pCi/m²-s, picocuries per square meter per second
 pCi/ml, picocuries per milliliter
 pop., population
 ppm, parts per million
 psi, pounds per square inch
 psig, pounds per square inch-gauge
 PVC, polyvinylchloride
 t, tonne
 yd³, cubic yard

1.0 EXECUTIVE SUMMARY

The Niagara Falls Storage Site (NFSS), located in northwestern New York state, is a surplus facility owned by the U.S. Department of Energy (DOE) and currently managed by Bechtel National, Inc. An inventory of pitchblende residues and radioactive wastes is stored at the site.

Several investigations, conceptual designs, and scoping studies have been conducted regarding the site under DOE and predecessor agency sponsorship. These have focused on specific residues or wastes stored at the site. While some of these studies have focused on the conditions at the site and interim measures for waste management, no engineering evaluation has been conducted regarding final disposition alternatives for the site or its residues and wastes. DOE has assigned Bechtel the responsibility for preparing this engineering evaluation of final disposition alternatives for the site and its waste and residue inventory. Bechtel is also performing additional geologic and hydrogeologic investigations to further evaluate the suitability of the site for long-term storage of radioactive materials.

Because this document is intended to be used for environmental assessment and conceptual design considerations, it is not the intention to state conclusions regarding a preferred alternative. Rather, it compares the DOE-approved alternatives and is intended to serve as the basis of the National Environmental Policy Act (NEPA) activities and a future selection by DOE of a final disposition alternative to be designed and implemented at NFSS.

This document is not intended to stand alone; rather, it serves as a reference engineering document for use in NEPA activities and as an information source for a future selection by DOE of a final disposition alternative for the NFSS. The Environmental Impact Statement (EIS) currently being prepared for the NFSS will include environmental risk and non-occupational radiation dose evaluations. Other reports, including a study of ocean disposal by Sandia National Laboratory and those cited in the reference list in this document, will also be used in the determination of the final disposition of the NFSS.

The final disposition scenarios selected by DOE for assessment in this document are consistent with those stated in the Notice of Intent to prepare an EIS for the NFSS (DOE, 1983d) and the modifications to the alternatives resulting from the public scoping process. The scenarios are:

- o Take no action beyond interim remedial measures other than maintenance and surveillance of the NFSS;
- o Retain and manage the NFSS as a long-term waste management facility for the wastes and residues on the site;
- o Decontaminate, certify, and release the NFSS for other use, with long-term management of the wastes and residues at other DOE sites; and
- o Partially decontaminate the NFSS by removal and transport off site of only the more radioactive residues, and upgrade containment of the remaining wastes and residues on site.

The objective of this document is to present to DOE the conceptual engineering, occupational radiation exposure, construction schedule, maintenance and surveillance requirements, and cost information relevant to design and implementation of each of the four scenarios. The specific alternatives within each scenario used as the basis for discussion in this document were evaluated on the bases of engineering considerations, technical feasibility, and regulatory requirements. Selected alternatives determined to be acceptable for each of the four final disposition scenarios for the NFSS were approved by DOE to be assessed and costed in this document. These alternatives are also the subject of the EIS for the NFSS currently being prepared by Argonne National Laboratory (ANL).

1.1 STATUS OF NFSS

Interim remedial measures currently in progress and planned at the NFSS are designed to assure interim control of all radionuclides

on the site by consolidation of all of the residues and wastes within a waste containment area located in the southwest portion of the site. The containment area will be bounded by a low-permeability gray clay layer present beneath the site, a compacted clay dike and cut-off wall on the sides, and a topsoil/sand/clay/membrane cover. This engineered containment is being designed to have a minimum life of 25 years with proper maintenance, and will provide barrier isolation which will reduce radon release, erosion, and the probability of offsite migration of contained materials.

The objectives of the interim remedial action activities, defined in the DOE NFSS Project Management Plan (DOE, 1983c), are to provide for more stable and improved storage conditions; minimize occupational and public radiation exposure; and bring the site into compliance with DOE operating orders and other applicable federal and state regulations while not jeopardizing or precluding any long-term option for the site. The process that DOE is using for selecting the interim remedial actions to accomplish the stated objectives is given in the Action Description Memorandum (DOE, 1983a) for accelerated interim remedial action at the NFSS. Completion of these interim remedial actions is presently scheduled for 1985. Significant deviation from the planned interim remedial actions would change the base condition being evaluated and could require revisions to the alternatives being assessed in this document.

Within the containment area, the higher specific activity residues (the K-65, L-30, L-50, and F-32 residues) will be consolidated during interim work in existing concrete buildings to the extent practicable. A clay cap will be placed over each building after emplacement of the residues. Other waste materials include the less radioactive R-10 residues, wastes consolidated from onsite areas, materials from ditch decontamination activities, contaminated rubble, and contaminated materials from former Lake Ontario Ordnance Works (LOOW) areas. These LOOW areas or vicinity properties are currently being radiologically surveyed by Oak Ridge Associated Universities (ORAU) for DOE to identify the need for

remedial action. Contaminated soils from these areas will be placed inside the containment area or in an extension to the area depending on when the LOOW areas are designated for remedial action.

For the purpose of this document, the status of the NFSS for the assessment of final disposition alternatives is assumed to be that which is expected to exist when the interim remedial measures described above are completed.

1.2 METHOD OF ASSESSMENT

The method of assessment used in this investigation was to identify assumptions, determine applicable criteria, document the array of alternatives and variations on alternatives within the four scenarios, select acceptable alternatives, and then assess the selected alternatives. The assessments were made on the basis of:

- o Design life
- o Occupational radiation exposure (during handling and transportation)
- o Schedule for implementation
- o Manpower requirements
- o Maintenance and surveillance requirements
- o Costs.

1.3 SELECTED ALTERNATIVES

Twelve alternatives were approved by DOE for assessment in this document. For the purpose of discussing these alternatives, the more radioactive residues (the K-65, L-30, L-50, and F-32 residues) are referred to as "residues." Only these materials are considered in alternatives that include processing or immobilization because (1) they contain recoverable resources (uranium, precious metals, and base metals), and (2) they have specific activity that may warrant immobilization for risk reduction. The R-10 residues are not generally included in "residues" as used in this document and are not considered in processing or immobilization schemes because they have a lower specific activity and their concentration has been reduced as they have leached into the underlying soils. The alternatives assessed in this document are:

No Action Scenario

Alternative:

- (1) Perform maintenance and surveillance of the site in its post-interim-remedial-action condition.

Retain and Manage Scenario

Alternatives:

- (2) Upgrade containment of the stored wastes.
- (3) Immobilize the residues and upgrade containment.
- (4) Process the residues to extract precious and base metals, immobilize the residues, and upgrade containment.

Decontaminate and Release Scenario

Alternatives:

- (5) Transport containerized K-65 residues and all other residues and wastes in bulk for management at an existing DOE burial site in the eastern U.S.
- (6) Immobilize the residues, place immobilized residues in shipping containers, leave the remaining radiologically contaminated material in bulk, and transport all wastes for management at an existing DOE burial site in the eastern U.S.
- (7) Process for resource recovery and immobilize the residues, place vitrified slag in shipping containers, leave the remaining radiologically contaminated material in bulk, and transport all wastes for management at an existing DOE burial site in the eastern U.S.
- (8) Transport containerized K-65 residues and all other residues and wastes in bulk for management at an existing DOE burial site in the western U.S.
- (9) Immobilize the residues, place immobilized residues in shipping containers, leave the remaining radiologically contaminated material in bulk, and transport all wastes for management at an existing DOE burial site in the western U.S.

- (10) Process for resource recovery and immobilize the residues, place vitrified slag in shipping containers, leave the remaining radiologically contaminated material in bulk, and transport all wastes for management at an existing DOE burial site in the western U.S.

Partial Decontamination Scenario

Alternatives:

- (11) Transport containerized K-65 residues and bulk L-30, L-50, and F-32 residues for management at a DOE burial site in the eastern U.S., and upgrade containment of the remaining wastes on site at NFSS.
- (12) Transport containerized K-65 residues and bulk L-30, L-50, and F-32 residues for management at a DOE burial site in the western U.S., and upgrade containment of the remaining wastes on site at NFSS.

1.4 COMPARISON OF ALTERNATIVES

The assessment of alternatives consisted of comparing the alternatives based on conceptual engineering design information and estimates of occupational radiation exposure, cost, schedule, manpower requirements, and maintenance and surveillance requirements. A comparison of the 12 alternatives using these parameters is presented in Table 1-1. Applicable design criteria and design life are also included. It must be noted that the engineering presented in this document is strictly conceptual. Costs and schedules are, therefore, only approximate and subject to later revision.

1.4.1 Occupational Radiation Exposure

Considering occupational exposure only, Alternative 1 (maintenance and surveillance)-or Alternative 2 (upgraded containment without immobilization or processing) are the most desirable. All alternatives involving immobilization or processing and immobilization of the residues involve higher occupational exposure to workers handling the residues.

TABLE 1-1
COMPARISON OF ALTERNATIVES FOR THE FINAL
DISPOSITION OF THE NFSS RESIDUES AND WASTES

Alternative Number(a)	Minimum Design Life(b) (Yrs)	Occupational Radiation Exposure(c) (Manrem)		Construction Schedule (Yrs)	Manpower Requirements (Manhours)	Maintenance and Surveillance (Yrs)	Costs (\$ x 1000)
		On Site and Disposal (Excluding In Transit)	Transport (In Transit)				
1 Maintenance and Surveillance	25	8	NA	0	10,000/yr	Continuing	13,900 ^(d)
2 Upgrade Containment	200	2	NA	5.5	466,200	200	12,000 ^(e)
3 Immobilize, Upgrade Containment	200	211	NA	9	715,400	200	41,100 ^(e)
4 Process, Upgrade Containment	200	261	NA	9	758,300	200	38,400 ^(e)
5 Transport Materials East	NA	110	6	8	750,600	NA	168,300
6 Immobilize, Transport Materials East	NA	180	6	8	852,600	NA	185,400
7 Process, Transport Materials East	NA	240	6	8	937,600	NA	181,400
8 Transport Materials West	NA	110	20	8	1,111,500	NA	221,100
9 Immobilize, Transport Materials West	NA	180	20	8	1,190,500	NA	241,900
10 Process, Transport Materials West	NA	240	20	8	1,233,500	NA	239,600
11 Transport Residues East, Upgrade Containment	200	101	6	10	746,400	200	26,700 ^(e)
12 Transport Residues West, Upgrade Containment	200	101	19	10	901,800	200	28,900 ^(e)

(a) See Section 1.3 for description of alternatives.

(b) Design life values indicated are minimums that must be achieved.

(c) Calculated based on the estimated duration of each specific task. See Appendix F for schedules.

(d) This present value cost is needed to provide \$418,000/yr for maintenance and surveillance based on an assumed real interest rate of 3 percent.

(e) Alternatives 2, 3, 4, 11, and 12 include 5 years of full maintenance and surveillance and at least 195 years of custodial maintenance and surveillance. The present value cost of maintenance and surveillance for the 200-year period is included in the reported costs based on an average cost of \$418,000 for full maintenance and surveillance and \$20,000 for custodial maintenance and surveillance.

The occupational exposure to workers will be within regulatory exposure limits for all the alternatives.

1.4.2 Estimated Cost

Table 1-1 indicates that Alternatives 1 and 2 under the No Action and Retain and Manage scenarios are the least costly of the alternatives. The present value cost for required maintenance and surveillance associated with these alternatives is included in the costs based on an assumed real interest rate of 3 percent. Immobilization or processing and immobilization of the residues adds significantly to the cost as indicated in the table. The cost for processing could be significantly higher than reported because of uncertainties with regard to process development costs. This potential increase in costs for the alternatives involving processing is somewhat offset by the potential for increasing recovered metal values.

The costs reported for the alternatives under the Decontaminate and Release scenario (Alternatives 5 through 10) are much greater than those under the Retain and Manage scenario. Excavation, transport, and disposal costs represent the majority of the Decontaminate and Release scenario costs. In all of the alternatives involving transportation, transport to an existing DOE disposal site in the western U.S. is significantly more expensive compared with transport to an existing DOE disposal site in the eastern U.S.

The Partial Decontamination alternatives (11 and 12) are less costly than the Decontaminate and Release alternatives. Only Alternatives 1 and 2 are lower in estimated cost.

Recent revisions (September 1983) to 10 CFR 71 and 49 CFR 171-174 pertaining to shipment of radioactive materials could affect requirements for transporting the NFSS residues and wastes to a burial site. For the purpose of preparing cost estimates for this document, it was assumed that, if changes to packaging requirements for the residues and wastes result from these regulation modifications, it will be possible to obtain exemptions from these

requirements if necessary and that, consequently, the current cost estimate would not be affected. Further attention will be given to interpretation of the subject regulation revisions at a future date.

1.4.3 Preliminary Schedule and Manpower Requirements

The schedule for Alternative 2, upgrading containment, under the Retain and Manage scenario requires 5-1/2 years to implement whereas immobilization, processing, or transport alternatives (Alternatives 3 through 12) require about 8 to 10 years to implement.

Manpower requirements for Retain and Manage alternatives vary from 0.47 to 0.76 million manhours. Decontaminate and Release scenario alternatives require about 0.75 to 1.23 million manhours. Partial Decontamination scenario manpower requirements vary from 0.75 to 0.90 million manhours. In all of the alternatives involving transportation, the use of the western disposal site requires a significantly greater number of manhours than the use of the eastern disposal site.

1.4.4 Maintenance and Surveillance Requirements

The No Action scenario requires that full maintenance and surveillance continue indefinitely. The scenarios for Retain and Manage and Partial Decontamination require 5 years of full maintenance and surveillance and at least 195 years of custodial maintenance and surveillance. The Decontaminate and Release scenario does not require maintenance and surveillance of the site after the site has been decontaminated.

1.4.5 Summary of Alternative Comparisons

Table 1-1 indicates that Alternatives 1 and 2 (no action or upgrade containment) are lowest in terms of occupational radiation exposure, manpower requirements, and cost, with Alternatives 11 and 12 (partial decontamination) the next lowest rank. Alternatives 3 and 4 are the third lowest in terms of manpower requirements and costs, but the occupational radiation exposure is high for these two alternatives due to extensive handling involved in the immobilization or processing and immobilization operations. The six alternatives

involving decontamination and release of the NFSS are the highest in terms of manpower requirements and cost. Again, as with Alternatives 3 and 4, those involving immobilization or processing and immobilization are highest in occupational radiation exposure. Comparing Alternatives 5 vs. 8, 6 vs. 9, and 7 vs 10, transportation to the western disposal site is much more expensive than transportation to the eastern disposal site in terms of manpower requirements and cost and incurs greater occupational radiation exposure.

2.0 INTRODUCTION

2.1 BACKGROUND

The NFSS, located in northwestern New York state, is a surplus facility owned by the DOE and currently managed by Bechtel National, Inc. The site is the responsibility of DOE's Surplus Facilities Management Program (SFMP). The objectives of the SFMP as they relate to the NFSS are to:

- o Provide for maintenance and surveillance of surplus radioactively contaminated DOE facilities in order to meet all applicable DOE standards, to protect public and onsite personnel health and safety, and to reduce potential environmental risks to as low as reasonably achievable.
- o Place the site in an acceptable interim condition for control and management of radioactive materials pending determination and implementation of final plans.
- o Select, through the procedures that implement the NEPA, a preferred long-range disposition option for the site.
- o Implement a structured program to accomplish the final disposition of the site under the responsibility of the DOE Division of Remedial Action Projects.

The NFSS is located within what was the Department of Army's LOOW site in the townships of Porter and Lewiston. Some other portions of the former LOOW site are within the jurisdiction of the SFMP's companion DOE program, the Formerly Utilized Sites Remedial Action Program (FUSRAP) which evaluates former Manhattan Engineer District/Atomic Energy Commission (MED/AEC) sites and conducts remedial action activities where appropriate.

An inventory of radioactive residues and wastes is stored at the NFSS. Some areas of the site have also become contaminated as a result of previous burial of contaminated materials and from earlier spills.

Several investigations, conceptual designs, and scoping studies (Acres American, 1981; Battelle 1981; NLO, 1980) have been conducted regarding the NFSS. These have focused on specific wastes and/or residues stored at the site. While some of these studies have evaluated interim storage and stabilization alternatives, no study has yet addressed the disposition alternatives for the site as well as its residues and wastes. DOE has assigned Bechtel the task of preparing this document to address four disposition scenarios:

- (1) no action beyond interim storage of all existing wastes and residues at NFSS except continued site maintenance and surveillance;
- (2) retention and management as a long-term waste management site for those wastes on the site at the completion of interim actions, involving upgrade of the containment or modification of the residues to minimize potential migration of contamination from NFSS;
- (3) decontamination, certification, and release of the site for other use, including transfer of all radioactive materials exceeding release criteria to another site for long-term management or disposal; and
- (4) partial decontamination of the site, involving removal of the more radioactive residues to another site and upgrading containment of the remaining wastes.

2.2 OBJECTIVE AND SCOPE

This document presents the conceptual engineering, occupational radiation exposure, construction schedule, maintenance and surveillance, and cost information relevant to design and implementation of each final disposition scenario. Within each scenario, selected engineering alternatives are identified and discussed. Conceptual engineering designs for implementation of the alternatives are described, occupational radiation exposures evaluated, and costs estimated so that the alternatives can be compared with one another.

Factors affecting the evaluation and selection of the disposition alternatives include: the physical, chemical, and radiological characteristics of the residues and wastes at NFSS; the physical characteristics of the NFSS; design constraints for

occupational radiation exposure control; worker and public safety; technical feasibility; regulatory compliance; and schedules and costs.

The DOE is currently performing interim remedial actions as part of its ongoing maintenance and caretaker responsibilities at the NFSS. These activities include improving the storage of residues, decontaminating specific areas, and safe storage of all radioactive materials to achieve compliance with DOE Order 5480.1A.

Although the interim remedial actions will improve control and containment of radioactive substances on the site, they are designed for short-term storage (25 years) and, therefore, DOE must decide how to manage the radioactive wastes and residues over the long term. This document provides the conceptual engineering design descriptions, order-of-magnitude (25 percent contingency) cost estimates, and occupational radiation exposure control requirements that affect this decision. In support of this decision-making process, DOE has also issued a Notice of Intent to prepare an EIS which is currently being prepared by ANL. The EIS will assess and compare the potential environmental impacts of selected alternatives described in this document for the final disposition of the NFSS.

This document considers only the residues and wastes stored within the NFSS when the interim remedial actions have been completed. These include the residues and wastes currently on the site as well as those being recovered from nearby offsite ditches and other former LOOW areas that became contaminated during MED/AEC activities. The nearby vicinity properties are currently being surveyed to define the areas needing remedial action.

The scope of this document is limited to final disposition alternatives for the NFSS and its residues and wastes. It is assumed that the interim remedial actions will have achieved and will remain in compliance with DOE orders and applicable federal and state regulations for at least a 25-year period, based on SFMP direction.

2.3 CONTENT

The remainder of this document consists of four sections. Chapter 3, Description and Status of the NFSS, describes the site history, the conditions that will exist when the interim remedial actions are completed, and the site physical characteristics. It also describes the physical, chemical, and radiological characteristics of the residues at the NFSS and identifies the baseline environmental conditions affecting the engineering design. Chapter 4, Method of Assessment, describes the methods used to determine acceptable engineering concepts and to identify alternatives. Chapter 5, Identification of Alternatives for the Final Disposition of the NFSS, discusses the alternatives and their variations based on engineering considerations and technical and regulatory constraints. Chapter 6, Assessment of Alternatives, provides conceptual engineering descriptions for implementation of each of the selected alternatives as well as occupational radiation exposures, manpower requirements, maintenance and surveillance requirements, and estimated costs for each. The alternatives are compared; however, no conclusions are drawn. The appendices are:

- A Characteristics of Residues and Wastes at NFSS
- B Immobilization of Radioactive Waste
- C Resource Recovery Process
- D Hydraulic Mining and K-65 Residue Handling Sequence
- E U.S. Department of Energy Proposed Radiological Criteria for Formerly Utilized Sites Remedial Action Program and Remote Surplus Facilities Management Program Sites
- F Preliminary Schedule and Manpower Requirements.

3.0 DESCRIPTION AND STATUS OF NIAGARA FALLS STORAGE SITE

3.1 SITE HISTORY AND STATUS

3.1.1 Historical Use of Site

The current 191-acre NFSS is a small portion of the original LOOW established by the Department of Army in the early 1940's. Part of the original 3,035-ha (7,500-acre) LOOW tract was used for TNT production early in World War II. When these operations ceased, the Department of Army assigned the site to the War Assets Administration. The War Assets Administration transferred the property to the MED, which in 1947 became the AEC. Figure 3-1 shows the NFSS within the regional context of northwestern New York state; Figure 3-2 shows the structures and features of the site prior to interim remedial action activities.

During 1944, the LOOW began to be used for the storage of pitchblende (uranium ore) processing residues. The first materials to arrive at the site were low-grade residues and byproducts from the Linde Air Products Division in Tonawanda, New York (the L-30, L-50, and R-10 residues) and from the Middlesex Sampling Plant in Middlesex, New Jersey (the F-32 residues). The L-30 and L-50 residues were stored in Buildings 411, 413, and 414, while the F-32 residues were stored in a recarbonation pit directly west of Building 411. The R-10 residues, as well as associated iron cake, were stored in an open area north of Building 411 at NFSS and have been subject to environmental processes which transported contaminants into the soil and drainageways.

The Middlesex sands were obtained from decontamination activities conducted at the Middlesex Sampling Plant and thus are not a uranium processing residue. They were stored in Building 410.

Shortly after World War II, radiologically contaminated processing equipment from Linde, and contaminated metal, concrete, ceramics, and lumber from decommissioning activities conducted at other facilities used for similar operations during the war were placed on the site.

**FIGURE 3-1 THE REGIONAL CONTEXT OF
THE NIAGARA FALLS STORAGE SITE**

FIGURE 3-2 DIAGRAM OF THE NFSS SHOWING STRUCTURES AND FEATURES OF THE SITE PRIOR TO INTERIM REMEDIAL ACTIONS

In 1949, pitchblende residues (the K-65 residues) resulting from uranium extraction conducted at a St. Louis plant were transported to the LOOW in 55-gal drums. Some of these were stored outdoors along existing roads and rail lines; others were stored in Building 410. From 1950 to 1952, the K-65 residues were transferred to a renovated concrete water tower on site (Building 434).

Between 1949 and 1952, the LOOW was used as a temporary storage location for radium sources and uranium and thorium rods that were processed by various facilities located in New York State. These materials were subsequently shipped off site. In addition, radioactive materials from the Knolls Atomic Power Laboratory (KAPL), the University of Rochester, and the Electro-Metallurgical Company were received at the site. The KAPL wastes consisted of combustible materials stored in wooden crates and processing wastes stored in 55-gal drums. These materials were later transferred to an AEC burial ground at Oak Ridge, Tennessee. The University of Rochester waste included carcasses and associated laboratory animal waste and miscellaneous material containing strontium-90 and cesium-137. The Electro-Metallurgical Company material included metal casting equipment which was contaminated with recoverable quantities of uranium. This equipment was shipped to the Oak Ridge Y-12 Plant in the late 1950's.

The original steam plant on site (Building 401) was modified and used for boron-10 production between 1953 and 1959, and again between 1965 and 1971. During the first period of operation, a major site cleanup was conducted. The cleanup included consolidation and removal of surface debris, packaging of KAPL waste for shipment to Oak Ridge, and sale of bulk metallic scrap. After 1971, the boron production facility was disassembled, and the instrumentation and hardware were disposed as surplus materials. Since the boron-10 operations ceased, most of the original LOOW site property has been transferred or sold. The remaining 77.3 ha (191 acres) comprise the NFSS.

In 1971, the AEC Oak Ridge Operations Office, assisted by Oak Ridge National Laboratory (ORNL), performed a preliminary radiological survey of the former LOOW site and identified several

areas contaminated as a result of storage of the radioactive materials described above. The concentrations of various radionuclides in these areas exceeded then-existing environmental standards. In 1972, approximately 11,470 m³ (15,000 yd³) of contaminated soils were excavated and placed over a portion of the R-10 residue storage area, forming the R-10 "spoils pile" area.

3.1.2 Condition of the NFSS After Completion of Interim Remedial Actions

The conditions expected to exist after completion of current and planned interim remedial actions are discussed in this section. The planned interim remedial actions are consistent with the DOE NFSS Project Management Plan (DOE, 1983c) and involve gaining interim control of radionuclides on the site by consolidating the residues and wastes that exceed criteria specified in Appendix E for radium-226 in soils/sediments within a new, engineered waste containment area located in the southwest portion of the site. This includes transferring the K-65 residues from Building 434, where there is concern that the tower may be unstable if subjected to seismic loadings (DOE, 1983a; Rupley Bahler Blake, 1981), to Building 411. The containment area will be bounded by a low-permeability gray clay layer present beneath the site, a compacted clay dike and cut-off wall on the sides, and a topsoil/sand/clay cover as described in the following sections.

The purposes of the planned interim remedial actions are to provide for more stable and improved storage conditions, to minimize occupational and public radiation exposure, and to bring the site into compliance with DOE orders and all other applicable federal and state regulations while not jeopardizing or precluding any long-term option for the site. Justification for, and alternatives to, these remedial actions are discussed in the Action Description Memorandum for the planned interim remedial actions (DOE, 1983a).

If the planned interim remedial actions are significantly modified before completion, the assessment of alternatives presented in Chapter 6 may need to be revised. Minor modifications of such parameters as volumes of contaminated soils within the waste

containment area, variations in the dike configuration or cover design, or the number of uncontaminated buildings to be demolished, should not significantly affect the evaluation. Deviations such as a change in the storage location of any residues outside the planned waste containment area would be significant.

The environmental monitoring program for the NFSS focuses on the detection of radionuclides in air, surface water and groundwater as well as external gamma radiation. Emphasis is placed on areas of known contamination and the site boundary. As additional geohydrological and meteorological data are acquired or site conditions change as a result of remedial actions performed, the program is reviewed and revised. Air and groundwater monitor locations in the immediate vicinity of the R-10 area are included in Figure 3-4. Wells 1 through 8 were sampled on a quarterly basis prior to and during the installation of the north dike. On the basis of recent geohydrological investigations and geologic conditions observed during dike excavation, wells A42, A43, A50, A51, and A52 were installed for monitoring purposes. These wells are incorporated into the monitoring program and wells 1 through 8 are being phased out. As future remedial action is implemented, additional wells will be drilled and monitored to ensure that any potential migration of radionuclides via groundwater is detected. Typical locations for such wells are indicated as B-1, B-2, and B-3 on Figure 3-4.

3.1.2.1 Residue and Waste Containment

After the completion of interim remedial actions, the residues and radiologically contaminated soils, sediments, and building rubble will have been placed and compacted within the confines of the waste containment area located in the southwest portion of the NFSS (see Figures 3-3 and 3-4).

The waste containment area will be enclosed within a dike and cut-off wall, each constructed of compacted clay. The dike will have a minimum thickness of 2.4 m (8 ft) and will be constructed to a height of 1.5 m (5 ft) above grade. The dike rests upon the cut-off wall which has a minimum thickness of 3.6 m (12 ft) and extends into the gray clay layer described in Section 3.3.1.3.

FIGURE 3-3 NFSS AFTER COMPLETION OF INTERIM REMEDIAL ACTIONS

FIGURE 3-4 PLAN VIEW OF THE WASTE CONTAINMENT AREA AT THE COMPLETION OF INTERIM REMEDIAL ACTIONS

The dike and cut-off wall function as a barrier to radionuclide migration in both ground water and surface water. Both utilize compacted clay which is designed to achieve a low permeability of 10^{-7} cm/s. The cut-off wall will be keyed into the gray clay layer which has a measured permeability of 7×10^{-8} cm/s (Acres American, 1981).

The containment area will be covered with a cap designed to minimize water infiltration, radon emanation, erosion, and frost heave damage. The lower layer of the cover will consist of an impervious membrane designed to minimize radon flux, provide an identification layer between the waste and overlying cover layers, and control the moisture content of the overlying clay. The next layer will consist of 0.9 m (3 ft) of compacted clay keyed into the dike. The clay layer is also designed to minimize radon flux. The clay layer will be sloped to enhance natural drainage away from the storage area. Above the clay, a 15-cm (6-in.) sand layer is designed to promote drainage. The upper layer is a 46-cm (18-in.) topsoil vegetative cover designed to minimize erosion and frost heave damage to the underlying clay layer. The upper layer is graded, as are the previous layers, to promote runoff. Total elevation of the cover will be 104 m (342 ft) (msl). Cross-sections of the waste containment are presented in Figure 3-5. The cap will be 125 m (411 ft) wide and 287 m (941 ft) long measured between the outer edges of the dike at its intercept with ground level.

The cover may require some maintenance during the interim storage period due to small animals burrowing, frost heave, or other natural causes. An interim cover design which would further reduce the need for maintenance would require a much thicker cover over the clay, similar to the upgraded cover design presented in Chapter 6 for Alternatives 2, 3, 4, 11, and 12.

At the completion of the described interim remedial actions, the residues and radiologically contaminated materials will be located in an engineered waste containment area that provides barrier isolation to reduce radon release, erosion, and the probability of offsite migration.

**FIGURE 3-5 CROSS-SECTION OF THE WASTE CONTAINMENT AREA
AT THE COMPLETION OF INTERIM REMEDIAL ACTIONS
(NOT TO SCALE)**

3.1.2.2 Residue and Waste Disposal Within the Waste Containment Area

It is assumed that the residues and wastes that will be stored within the waste containment area (dike, cut-off wall, and cover) are:

- o The residues and wastes now on site
- o Contaminated soils, sediments, and sands from onsite and offsite vicinity properties decontamination activities
- o Contaminated building rubble.

The locations of all of these materials prior to interim remedial action and after the completion of interim remedial action are shown in Table 3-1. Where possible, residues will be placed in concrete buildings located within the waste containment area, thereby adding an additional barrier to the migration of radionuclides.

The K-65, L-30, and F-32 residues will be located in Building 411, as shown in Figure 3-6. The building was originally designed for water storage. Building 411 is a 62 x 56 x 6 m (200 x 180 x 19 ft) concrete structure which is divided into two separate bays. The roof and upper walls of the building will be removed. The K-65 residues will be placed in the west bay and the L-30 and F-32 residues will be placed in the east bay of the building and in the recarbonation pit (considered part of Building 411 for the purpose of this document). All residues will be dewatered, consolidated, and covered with an identification membrane layer, contaminated soil, and clay at the conclusion of interim remedial actions. The covering will reduce radon emanation. The building foundation, which will be inspected and patched as required before the residues are placed in it, will act as an additional barrier to radionuclide migration.

Standing water resulting from hydraulic mining transfer of the K-65 and F-32 residues into Building 411 will be removed by pump for treatment. Interstitial water, also from hydraulic mining, will be partially removed by utilizing a vertical plastic wick drain system with a vacuum-assisted drainage bed placed under the residues being

TABLE 3-1

**LOCATIONS OF RESIDUES AND CONTAMINATED MATERIALS
BEFORE AND AFTER INTERIM REMEDIAL ACTIONS ARE COMPLETED**

Material	Before Interim Remedial Action	After Interim Remedial Action
K-65 residues	Building 434	West bay of Building 411-residues dewatered and covered with clay; contaminated Building 434 rubble stored inside the waste containment area
L-30 residues	Building 411, covered with water	East bay of Building 411 and in the recarbonation pit - residues dewatered and covered with clay
L-50 residues	Buildings 413, 414	Buildings 413, 414 - residues are covered
F-32 residues	Recarbonation pit, covered with water	East bay of Building 411-residues dewatered and covered with clay
Middlesex sands	Building 410, second level and lower level	Lower level of Building 410
R-10 residue and spoils pile	Soil surface north of Building 411	Soil surface beneath clay cover, north of Building 411, inside the waste containment area
Contaminated sediments from Central and West Drainage Ditches	Onsite and offsite portions of the Central and West Drainage Ditches	Inside the waste containment area
Contaminated soils	At various locations within the NFSS	Inside the waste containment area
Contaminated demolition rubble from Buildings 410, 411, 412, 415, and 434	--	Inside the waste containment area

Note: Other materials from former LOOW areas may also be stored inside the waste containment area as part of interim remedial actions.

**FIGURE 3-6 BUILDING 411 AFTER COMPLETION
OF INTERIM REMEDIAL ACTIONS**

dewatered. The drainage bed consists of a layer of nonwoven filter fabric covering a 30-cm (12-in.) sand layer containing slotted PVC vacuum piping. The vertical wick drains will be located on 1.2- to 1.5-m (4- to 5-ft) centers and will penetrate the drainage bed and residues. Figure 3-7 presents a cross-sectional view of the residues and the planned dewatering system in Building 411. The residues, at the conclusion of interim remedial actions, will be in a well-consolidated, semi-saturated (about 30 percent water by volume) condition.

The L-50 residues are currently contained in Buildings 413 and 414, which are round concrete tanks approximately 19 m (62 ft) in diameter and 6 m (20 ft) deep. The roofs of these buildings have been removed, and the residues in each building are covered with layers of compacted clay separated by a membrane layer as shown in Figure 3-8. The area immediately surrounding Buildings 413 and 414 will be backfilled with contaminated clayey soil to form partially entombed structures.

The roof and upper walls and floors of Building 410, portions of Building 415, and all of Buildings 412 and 434 will be demolished during interim remedial actions. Contaminated rubble from the demolition of these buildings and the Middlesex sands will be placed in the lower level of Building 410 or elsewhere inside the waste containment area.

Radiologically contaminated soils and sediments from onsite and offsite remedial actions will be located throughout the waste containment area. The total volume of these soils and sediments, including contaminated underlying soil resulting from the leaching of the R-10 residues and contaminated soils, is approximately $144,500 \text{ m}^3$ ($189,000 \text{ yd}^3$) based on decontamination criteria for radium-226 in soil stated in Appendix E. Before the cover is placed over the entire waste containment area, the spoils pile will be compacted to minimize potential subsidence and water infiltration.

FIGURE 3-7 CROSS-SECTION OF BUILDING 411 AT
THE COMPLETION OF INTERIM REMEDIAL ACTIONS

FIGURE 3-8 DESIGN OF RESIDUE CONTAINMENT STRUCTURES -
BUILDINGS 413 AND 414

As indicated in the DOE NFSS Project Management Plan for the interim remedial actions (DOE, 1983c), current plans are that uncontaminated or cleaned structures (Buildings 401, 401A, 409, 430, 431, 432, and 433), located outside of the waste containment area, will also be demolished during interim remedial actions as funding permits. Demolition of these buildings could be completed during final disposition construction or site surveillance activities. The remaining structures (Buildings 402, 403, 416, 423, and 429), and possibly others presently scheduled for demolition, will be retained for use during construction activities related to the final disposition of the site.

3.1.3 Description of Residues and Wastes Stored at NFSS

A description of the residues and the wastes stored at NFSS is given in the following sections. The weight, volume, and radium inventory of the residues and sands at NFSS are given in Table 3-2. Detailed chemical, physical, and radiological data for the residues are presented in Appendix A.

3.1.3.1 K-65 Residues

The K-65 residues resulted from the processing of high-grade pitchblende ore (35 to 60 percent uranium oxide -- U_3O_8) by a St. Louis processing plant. The K-65 residues are composed of two fractions: (1) a slime fraction (passes through a 400-mesh screen) containing insoluble salt fractions including radium-contaminated barium sulfate, and (2) a sand fraction (does not pass through a 400-mesh screen) containing undissolved ore particles and somewhat insoluble silicate secondary minerals. Based on the best available data (Battelle, 1981; 1983a) the K-65 residues have a uranium concentration of 1,000 to 2,000 ppm and a radium concentration of approximately 0.22 micro-Ci/g. They also contain the decay products of uranium and radium. The K-65 residues have the highest specific activity of the residues within the NFSS. The physical form of the residues is a damp muscovite clay with an appreciable alpha quartz fraction. Chemically, the K-65 residues are mixtures of oxides,

TABLE 3-2

SUMMARY OF THE MAJOR PITCHBLEND E RESIDUES
AND MIDDLESEX SANDS STORED AT THE NFSS

Residue	Weight(a)		Volume(a)		Estimated Radium Inventory(b) Ci
	kg x10 ⁶	tons	m ³	yd ³	
K-65	3.53	3,891	3,120	4,080	760
L-30	7.46	8,227	6,080	7,960	79
L-50	1.70	1,878	1,640	2,150	20
F-32	0.13	138	340	440	0.7
R-10	7.47	8,235	7,180	9,400	24
Middlesex Sands	0.002	2	175	229	2 x 10 ⁻⁵

(a) Battelle, 1981.

(b) Preliminary estimates subject to change upon verification of radium concentration. Additional curie inventory from contaminated soils and rubble is estimated to be approximately 8 Ci.

silicates, carbonates, and sulfates with uranium primarily in the form of sodium uranyl carbonate. Lead and barium are the metals most prevalent in these residues.

3.1.3.2 L-30 Residues

The L-30 residues resulted from the processing of pitchblende ore (approximately 10 percent uranium oxide -- U_3O_8) by the Linde Ceramics Plant, Tonawanda, New York, from December 1943 through October 1944. The Linde process began with a hot sulphuric acid leach followed by addition of barium chloride. The resulting sulphuric acid leachate was neutralized with sodium carbonate. Uranium was precipitated from the pregnant solution with ammonium hydroxide and was dried to obtain pure uranium oxide (U_3O_8). Available data suggest uranium concentrations ranging from 400 to 5,000 ppm and radium concentrations from 2 to 12 ppb (Battelle, 1981; 1983a). Recent tests indicate radium concentrations could be as high as 30 ppb. The wide range of values is indicative of the heterogeneity of these residues. The metals occurring in the largest concentrations are nickel, iron, barium, and lead.

The L-30 residues are composed primarily of chamosite clay (70 percent) with smaller quantities of alpha quartz (20 percent), barite, and sodium uranyl carbonate.

3.1.3.3 L-50 Residues

The L-50 residues resulted from uranium extraction of pitchblende ores containing approximately 7 percent uranium oxide. Extraction conducted by the Linde Plant was similar to that described for the L-30 residues. The uranium and radium concentrations in the L-50 residues range from 100 to 4,000 ppm and 7 to 12 ppb, respectively (Battelle, 1981; 1983a). They are composed primarily of antigorite clay (60 percent) and alpha quartz (33 percent). No sodium uranyl carbonate was detected. Nickel, iron, barium, and lead are in the highest concentrations of the observed metals.

3.1.3.4 F-32 Residues

These residues are the result of a similar uranium extraction process to that which produced the L-30 and L-50 residues. Inventory data indicate that the uranium concentration of the pitchblende ore used by Linde ranged from 4,000 to 6,500 ppm (Battelle, 1981; 1983a). Radium concentrations are estimated to be 5 ppb.

3.1.3.5 R-10 Residues

The R-10 residues resulted from the processing of ore containing 3.5 percent uranium oxide (U_3O_8). The extraction process conducted by Linde was similar to that which produced the L-30 and L-50 residues. The R-10 residues were stored on the soil surface north of Building 411. The initial residue inventory suggested a uranium concentration of approximately 2,300 ppm. Radium concentrations are estimated to be 3 ppb.

3.1.3.6 Middlesex Sands

The Middlesex sands are the result of decontamination activities (sandblasting) conducted at the Middlesex Sampling Plant, Middlesex, New Jersey. This plant was used for sampling, weighing, assaying, and storage of uranium and thorium ores. Measurements made during the Battelle site characterization (Battelle, 1981) indicate current levels of less than 100 ppm uranium and less than 0.01 ppb radium-226. The Middlesex sands have an estimated radium inventory of 2×10^{-5} Ci.

3.1.3.7 Radiologically Contaminated Soils and Sediments

After completion of interim remedial action activities, all radiologically contaminated soils and sediments exceeding criteria established in Appendix E for radium-226 will have been placed in the waste containment area. The total volume of contaminated soils that will be contained within the waste containment area is estimated to be $144,500 \text{ m}^3$ ($189,000 \text{ yd}^3$). It was liberally estimated that the approximate concentration of radium-226 in the soils and sediments (excluding residues) is less than 50 pCi/g.

3.1.3.8 Radiologically Contaminated Building Rubble

The waste containment area, at the completion of interim remedial action activities, will contain pitchblende residues in Buildings 411, 413, and 414. The lower level of Building 410 will contain radiologically contaminated rubble from the demolition of the upper structure of Building 410, portions of Building 415 (and the interconnecting concrete pipes), and Building 434. The contaminated rubble will contain long-lived activity on the surfaces previously exposed to the residues. The non-surface portions of the rubble will contain radon daughter products. The daughter product activity (lead-210 and its daughters) is expected to be an order of magnitude greater than the long-lived radium at the time of demolition due to the high radon diffusion rates relative to radium.

3.2 SITE AREA AND PHYSICAL DESCRIPTION

The NFSS is located in Niagara County in western New York, within the township of Lewiston and adjacent to the township of Porter. It is about 30 km (19 mi) north of Buffalo, New York; 10 km (6 mi) north of the city of Niagara Falls; 6.5 km (4 mi) south of Lake Ontario; and 5 km (3 mi) east of the Province of Ontario, Canada.

The NFSS is located 6.7 km (4 mi) south of the southern shore of Lake Ontario, 3.2 km (2 mi) north of the Niagara Escarpment, on the relatively flat terrain of the Erie-Ontario Lowlands Physiographic Province. Elevations at the site range between 93 and 103 m (310 and 332 ft) mean sea level (msl); the lower elevations correspond to the man-made drainage ditches, and the higher elevations correspond to the R-10 residue and spoils pile.

Land uses immediately adjacent to the site are varied. A chemical waste disposal facility operated by SCA Chemical Waste Services is located north and east of the site. A sanitary landfill (Modern Disposal) is being constructed east of the NFSS. Land south of the site is federal government property, controlled by the Department of Labor and used for training construction equipment operators. There is also a sanitary landfill south of the site,

which is owned by the town of Lewiston. West of the facility is a Niagara Mohawk Power Corporation transmission line corridor. All of these properties are located on land that was once part of the original LOOW site. There are 12 property owners (including the U.S. Government) located along offsite portions of the West and Central Drainage Ditches north of the site (Acres American, 1981).

Land uses within the townships of Lewiston and Porter are predominantly rural and include row-crop agriculture, orchards, recreation areas, abandoned fields, and second-growth forests. Three recreational areas are located near the site: Joseph Davis State Park is 4.8 km (3 mi) west of the NFSS; Fourmile Creek State Park is located at the confluence of Fourmile Creek and Lake Ontario, about 6.4 km (4.0 mi) north of the site; and Fort Niagara State Park is located at the confluence of the Niagara River and Lake Ontario, about 8 km (5 mi) northwest of the site.

The nearest permanent residence is at a KOA campground, located 0.7 km (0.4 mi) south-southwest of the R-10 pile. A trailer park is located 2.6 km (1.6 mi) northwest of the R-10 pile on Balmer Road. Workers at SCA Chemical Waste Services work outdoors, 1.2 km (0.75 mi) north of the R-10 pile. During the summer, about 40,000 to 50,000 campers camp at a KOA campground 0.7 km (0.4 mi) south-southwest of the R-10 pile on Pletcher Road. Hunters occasionally use the area west of the Niagara Mohawk corridor.

The population of Niagara County, which has declined since 1970, was 227,101 in 1980. The nearest major population centers are the city of Niagara Falls (pop. 71,384) and the Buffalo metropolitan area (pop. 1.5 million). As of May 1982, the county had a civilian work force of 104,169, with an unemployment rate of 13.6 percent.

Major highway transportation routes in the area are State Route 93 to the north, U.S. Route 104 to the south, and the Robert Moses Parkway to the west. Local roads near the site include Lutts, Cain, Balmer, Pletcher, and Porter Center roads.

3.3 BASELINE ENVIRONMENTAL CONDITIONS OF THE NFSS

This section describes only those environmental conditions at the NFSS that are relevant to design criteria and engineering design. These conditions include site geology, hydrology, and meteorology. The conditions described in this section were obtained from field investigations conducted in 1981 (Acres American, 1981) except for the long-term meteorological data which were obtained from National Oceanic and Atmospheric Administration records (NOAA, 1977). A detailed environmental evaluation will be included in the EIS for NFSS. Additional geological and geohydrological investigations were recently completed by Bechtel and will be documented in a separate report. The findings generally support the information presented below.

3.3.1 Geologic Conditions

The NFSS is located within a broad, flat area bordering Lake Ontario. This region is the Niagara Frontier and is covered by surficial, glacially derived soils. These soils are composed of glaciofluvial sands and gravels, dense tills, and glacial lacustrine clays. Beneath the soil deposits is bedrock material consisting of shales, siltstones, and mudstones, each of which is highly susceptible to erosion (Acres American, 1981).

Six major stratigraphic units have been identified within the interval from 0 to 30 m (0 to 100 ft) below ground surface at the site (Figure 3-9). In order of increasing depth, these units are: surficial soils and fill, brown clay, gray clay and silt, brown sand and gravels, red silt, and bedrock. The surface soil, brown clay, and gray clay are of the greatest significance for design and construction at the NFSS.

3.3.1.1 Soils

The surface soil at the NFSS is generally a brown or yellowish silt with organics usually present in the upper 15.2 cm (6 in.). Gravel and sand are usually present but rarely constitute more than 40 percent of the surface soil. The thickness of these silts ranges

* SOIL CLASSIFICATION BASED ON THE UNIFIED SOIL CLASSIFICATION SYSTEM" FIGURE 3.2

Source:
Acres American (1981)

FIGURE 3-9 GENERALIZED GEOLOGIC COLUMN FOR NFSS

between 0 and 1.5 m (5 ft), with an average of 0.3 to 0.6 m (1 to 2 ft). The unit is generally dry, although frequently wetted by precipitation. Relative density is in the range of loose to medium.

3.3.1.2 Brown Clay Layer

Beneath the surficial soil at the site, the brown clay layer varies in thickness from 1.8 to 7.0 m (6 to 23 ft). The clay is generally dry, of medium relative density, and grades from brown clay to clayey silt to silty sand. Sandy gravel or gravelly sand lenses are common, especially in the lower clay levels. The mean moisture content of the brown clay is 20.7 percent with a mean dry density of $1,749 \text{ kg/m}^3$ (109 lb/ft^3). Laboratory tests indicated a shear strength of $14,184 \text{ kg/m}^2$ ($2,900 \text{ lb/ft}^2$). A mean permeability of 1.1 m/100 yr (3.5 ft/100 yr) was observed by Acres American for the brown clay unit (Acres American, 1981).

3.3.1.3 Gray Clay Layer

The gray clay layer is found beneath the brown clay unit and ranges from 3.3 to 8.8 m (11 to 29 ft) in thickness at the site. The gray clay layer is the thickest soil unit within the NFSS and is characterized by gray clay which occasionally grades to a silty clay. This is the most homogeneous unit on the site. Within the clay unit, lenses of fine-to-medium-grained sand are common, together with varying quantities of gravel. The gravel is usually fine-to-medium-sized and dispersed throughout. Sands and gravels become the predominant constituent of the unit near the base where they are sometimes clean and lighter in color. The overall consistency of this unit ranges from soft to medium. The clay is saturated, with a mean moisture content of 24.6 percent and a mean dry density of $1,576 \text{ kg/m}^3$ (98 lb/ft^3). A shear strength of $3,913 \text{ kg/m}^2$ (800 lb/ft^2) was observed for the gray clay. Laboratory tests indicated a mean permeability of $7.34/10^{-8} \text{ cm/s}$ or 2.3 m/100 yr (7.5 ft/100 yr).

3.3.2 Hydrologic Conditions -- Surface Water

In the 1940's, a system of ditches was installed during the construction of the LOOW to drain surface waters to the Central Drainage Ditch. Sixmile Creek, which originally flowed through the NFSS, was diverted just outside the southwest corner of the NFSS boundary and now flows into the Southwestern Drainage Ditch (see Figure 3-10). Surface waters from the area to the southwest of the NFSS, which originally flowed eastward into Twelvemile Creek, were diverted to the South-31 Ditch.

Currently, runoff from the NFSS is diverted into three major drainage ditches: the Central Drainage Ditch, the West Ditch, and the South-31 Ditch. The West and South-31 Ditches discharge directly into the Central Drainage Ditch which flows northward into Fourmile Creek and ultimately into Lake Ontario (Acres American, 1981). Sixmile and Twelvemile Creeks do not presently receive runoff from the NFSS.

On site, the Central Drainage Ditch is approximately 3 to 4 m (10 to 15 ft) deep, 3 to 6 m (10 to 20 ft) wide at the base, and 12 to 15 m (40 to 50 ft) wide at the top. The Central Drainage Ditch is 4.8 km (3 mi) in length from its origin at the south end of the NFSS to its confluence with Fourmile Creek. The West Ditch runs parallel to the Central Drainage Ditch and drains the western portion of the NFSS prior to joining the Central Drainage Ditch, as shown in Figure 3-10. It is approximately 1.5 km (1 mi) in length. The only major offsite contributor to flow across the site is the South-31 Ditch which flows into the Central Drainage Ditch just south of Building 411. This ditch drains an offsite drainage area of approximately 81 ha (200 acres) and is about 421 m (1,400 ft) in length (Battelle, 1981).

Development at Modern Disposal, located to the east of the NFSS, may be expected to alter the upstream hydrology of the South-31 Ditch and to change the peak storm discharge. The actual effects on storm discharge are undetermined at this time. The current 100-year flood level within the NFSS is estimated to be approximately 97 m

FIGURE 3-10 SURFACE WATER DRAINAGES AT THE NFSS

(317 ft). The NFSS lies between an elevation of 93 m (305 ft) msl at the base of the drainage ditches and 101 m (330 ft) msl at the top of the R-10 spoils pile (Battelle, 1981).

3.3.3 Hydrologic Conditions -- Groundwater

Three aquifers have been identified beneath the NFSS. These include upper and lower alluvial aquifers in the soil and the bedrock aquifer of the Queenston Formation.

3.3.3.1 Upper Soil Aquifer

The upper soil aquifer occurs in the intermittent sand lenses of the brown glacial clay. The sand lenses are discontinuous and are therefore not considered a true aquifer. Recharge of the sand lenses occurs by percolation of precipitation. Field tests indicated a permeability ranging from 2.0×10^{-5} to 2.0×10^{-6} cm/s (1.9 ft/yr to 0.2 ft/yr) (Acres American, 1981). Groundwater flow patterns in the upper alluvial aquifer beneath the NFSS appear to be localized with no apparent regional flow scheme in the upper soil aquifer, as shown in the piezometric surface map (Figure 3-11).

3.3.3.2 Lower Soil Aquifer

The lower soil aquifer is located in the sand-silt-gravel unit found beneath the gray clay and above the till, which is an unstratified glacial drift consisting of clay, sand, and gravel. Where the till is absent, the lower soil aquifer and the bedrock aquifer are in contact. Recharge of this lower unit is primarily by upward migration of connate water from the underlying Queenston Formation, especially where the red till is not present. The lower soil aquifer is reported to be continuous beneath the NFSS (Acres American, 1981). This aquifer below the NFSS is 0.9 to 2.1 m (3 to 7 ft) thick, is confined, and has a piezometric surface 7.6 to 9.1 m (25 to 30 ft) above the surface of the aquifer (Figure 3-12). This figure indicates a northwesterly groundwater flow across the NFSS. Average permeability within the unit is reported to be 1.3×10^{-3} cm/s (1,300 ft/yr) (Wehran Engineering, 1982).

FIGURE 3-11 PIEZOMETRIC SURFACE OF THE UPPER ALLUVIAL AQUIFER AT THE NFSS

LEGEND

- EXPLANATION**
- GROUNDWATER FLOW PATH
 - 316.0 — ELEVATION OF PIEZOMETRIC SURFACE IN LOWER ALLUVIAL AQUIFER ON 15 JULY 1981 (IN FEET)
 -?..... INDICATES ESTIMATED LOCATION OF PIEZOMETRIC SURFACE

NOTE: WATER LEVEL ELEVATIONS FROM
ACRES AMERICAN (1981)

FIGURE 3-12 PIEZOMETRIC SURFACE OF THE LOWER
ALLUVIAL AQUIFER AT THE NFSS

3.3.3.3 Bedrock Aquifer

The bedrock of the NFSS, as indicated previously, consists of the shales and siltstones of the Queenston Formation. These rock types contain an aquifer that is generally less permeable than other surrounding bedrock aquifers. Groundwater in the Queenston Formation beneath the NFSS generally moves in the weathered and fractured zone in the upper few feet of the formation. Portions of the formation that are unfractured are essentially impermeable. Well yields in the fractured portions of the formation are typically less than 0.4 l/s (7 gal/min) while yields in the unfractured areas are estimated to be less than 0.06 l/s (1 gal/min) (Johnson, 1964). Water levels in wells installed into the bedrock beneath the NFSS coincide with the piezometric contours of the lower soil aquifer (see Figure 3-12). Thus, at least locally, the lower soil aquifer and the bedrock aquifer are connected (Acres American, 1981).

3.3.3.4 Groundwater Use

Domestic groundwater from the lower soil aquifer and the bedrock aquifer is used in the NFSS area, but is generally of low yield and poor quality (Acres American, 1981). The only municipal water supply system using groundwater in Niagara County is located in the Middleport area in eastern Niagara County. Numerous residential wells use groundwater in the NFSS area.

3.3.4 Meteorology

The climate in the extreme westerly portion of New York is continental in nature with moderately cold winters and brief warm summers. The NFSS lies within the middle latitude westerlies which results in weather systems moving generally eastward and northeastward through the region.

Temperature extremes in the NFSS area are reduced slightly due to the proximity to Lake Ontario and Lake Erie. Temperature data collected for Buffalo, New York, from 1940 to 1970 indicate a mean daily maximum of 1.0°C (33°F) for the winter months (December through March) and 24.3°C (76°F) for the summer months (June through

September). Yearly mean precipitation is 92 cm (36 in.). Precipitation over the year is fairly evenly distributed with the maximum average monthly rainfall coming in late summer and early autumn (NOAA, 1977). Approximately 140 cm (56 in.) of snow falls per year, primarily between November and March (Acres American, 1981).

The prevailing wind direction (obtained from a meteorological station at the site) is from the southwest (Figure 3-13). Windspeeds are predominantly in the 12 to 18 km/h (7 to 10 knot) range.

**FIGURE 3-13 WIND FREQUENCY DISTRIBUTION
FOR THE NFSS**

4.0 METHOD OF ASSESSMENT

The purpose of this chapter is to identify the assumptions and rationale used to identify, select, and assess primary alternatives and variations of these alternatives for the final disposition of the NFSS. The alternatives provide specific means of implementing any of four disposition scenarios for NFSS, that is (1) no action beyond interim storage of all existing wastes and residues at NFSS except continued site maintenance and surveillance; (2) retention and management as a long-term waste management site for those wastes on the site at the completion of interim actions, involving upgrade of the containment or modification of the residues to minimize potential migration of contamination from NFSS; (3) decontamination, certification, and release of the site for other use, including transfer of all radioactive materials exceeding release criteria to another site for long-term management or disposal; and (4) partial decontamination of the site, involving removal of the more radioactive residues to another site and upgrading containment of the remaining wastes.

Because this document is intended to be used for environmental assessment and conceptual design considerations, it is not the intention to state conclusions regarding a preferred alternative. Rather, this document compares alternatives and is intended to serve as the basis of the NEPA activities and a future selection by DOE of a final disposition alternative to be designed and implemented at the NFSS.

4.1 ASSUMPTIONS

The following assumptions were made for the identification, selection, and assessment of alternatives:

- o The site conditions assumed as the baseline in this document are those that will exist after interim remedial actions are completed at the site (see Section 3.1.2). It is assumed that a cover has been installed over the waste and that the cover (with

adequate maintenance) will provide interim control of the waste for at least 25 years pending implementation of a final site disposition.

- o It is assumed that wastes are classified in accordance with DOE orders in effect June 1983.
- o Management or disposal criteria applicable to the residues and wastes stored at NFSS are consistent with DOE orders and applicable portions of 40 CFR 192. The criteria used in this document are given in Appendix E.
- o Occupational radiation exposure will be limited to levels stated in DOE orders. The conceptual design descriptions and cost estimates have been prepared considering this radiation exposure limit.
- o With regard to shipping, the K-65 residues, as well as other materials having lower specific activities, were assumed to be low specific activity (LSA) materials, defined in 10 CFR 71 and 49 CFR 173 as "uranium or thorium ores and physical or chemical concentrates of these ores."
- o Environmental risk and non-occupational radiation dose evaluations will be part of the EIS and will be evaluated in that document.
- o Environmental baseline conditions, described in Section 3.3, are included only as required for assessment of alternatives and development of conceptual designs.
- o The NFSS site characteristics are suitable for storage of radioactive wastes. This is being verified by ongoing geohydrological investigations by Bechtel.
- o The site will not be used for storage or disposal of any wastes except those within the waste containment area at the end of interim remedial actions.

4.2 FACTORS AFFECTING IDENTIFICATION OF ALTERNATIVES

Factors to be considered in the identification of acceptable alternatives for the final disposition of the site are:

- o Waste type
- o Waste form
- o Waste characteristics
- o Management and disposal criteria
- o Disposal site availability
- o Public radiation exposure
- o Environmental consequences
- o Public opinion
- o Cost-effectiveness
- o Occupational radiation exposure
- o Manpower requirements
- o Schedule
- o Technical feasibility.

There are three waste types at NFSS. These are:

- o Ore processing residues
- o Contaminated soil (including the R-10 residues),
sediments, and sands
- o Contaminated rubble.

Each of these waste types can be managed and disposed of in a number of ways, as discussed in Chapters 5 and 6.

4.3 RATIONALE FOR APPROACH

Figure 4-1 shows the general approach used to identify and assess acceptable alternatives and variations for the four disposition scenarios for the NFSS. Alternatives and variations to the alternatives for each scenario were identified and initially screened, based on technical and regulatory acceptability.

Alternatives and variations were considered to be technically acceptable if they were:

- o In compliance with the design criteria
- o Cost-effective
- o Able to be implemented within an acceptable schedule

**FIGURE 4-1 DIAGRAM OF APPROACH TO IDENTIFICATION OF ALTERNATIVES
LEADING TO A FUTURE SELECTION OF A FINAL NFSS DISPOSITION**

- o Able to minimize offsite migration of radionuclides
- o Able to minimize radon emanation and water infiltration
- o Able to provide a control system that ensures acceptable performance for the required design life.

Alternatives were determined to be acceptable from a regulatory standpoint if they were:

- o Consistent with DOE policy
- o Consistent with DOE orders
- o In compliance with applicable federal and state law.

The assumptions and approach given in this chapter are applied in Chapter 5 to identify and select acceptable alternatives for the final disposition of the NFSS.

5.0 IDENTIFICATION OF ALTERNATIVES FOR THE FINAL DISPOSITION OF THE NFSS

This chapter identifies acceptable engineering concepts potentially applicable for the four NFSS disposition scenarios. The first three scenarios, which were selected by DOE for study in this document and are consistent with the Notice of Intent to prepare an EIS for the NFSS (DOE, 1983d), are:

- (1) No Action -- No action beyond interim storage of all existing wastes and residues at NFSS except continued site maintenance and surveillance.
- (2) Retain and Manage -- Retention and management as a long-term waste management site for those wastes on the site at the completion of interim actions, involving upgrade of the containment or modification of the residues to minimize potential migration of contamination from NFSS.
- (3) Decontaminate and Release -- Decontamination, certification, and release of the site for other use, including transfer of all radioactive materials exceeding release criteria to another site for long term management or disposal.

The fourth scenario, included as a result of the EIS public scoping meeting, is:

- (4) Partial Decontamination -- Partial decontamination of the site, involving removal of the more radioactive residues to another site and upgrading containment of the remaining wastes.

The specific alternatives within each scenario, used as the basis for discussion in this chapter, were evaluated using engineering judgment and consideration of technical feasibility and regulatory requirements. Selected alternatives determined to be

acceptable for each of the final disposition scenarios for the NFSS were approved by DOE to be assessed and costed in this document. The specific alternatives are described in detail in Chapter 6.

5.1 CRITERIA

The radiological criteria determined by DOE to be applicable to cleanup, storage, and/or disposal of radioactive materials under the FUSRAP and SFMP programs, including the wastes at the NFSS, are summarized in Table 5-1. Appendix E contains detailed radiological criteria applicable to the NFSS. The FUSRAP and SFMP Design Criteria (DOE, 1983b) supplement the radiological criteria given in Appendix E. The design criteria present additional details regarding applicable federal regulations, including design codes, guides, and standards. The design criteria also present guidelines for maximum contaminant levels for inorganic chemicals in water systems.

5.2 NO ACTION SCENARIO

Under the No Action disposition, no substantive actions would be required except to maintain and monitor the NFSS in its post-interim-remedial-action condition (see Section 3.1.2 for description of interim remedial action). This alternative is not considered to be acceptable over the long term (at least 200 years) and is included here to be consistent with the Notice of Intent to prepare an EIS (DOE, 1983d).

5.3 RETAIN AND MANAGE SCENARIO

Alternatives considered for implementation of this scenario include:

- o Upgrade the waste containment area to minimize radionuclide migration and inadvertent intrusion
- o Upgrade the waste containment area and immobilize the residues
- o Upgrade the waste containment area and immobilize the residues after processing them for resource recovery.

TABLE 5-1

APPLICABLE CRITERIA FOR THE FINAL DISPOSITION OF THE NFSS(a)

Subject	Criteria	Regulation
<u>Control of Wastes</u>		
Longevity of waste containment area	Up to 1000 years to the extent reasonably achievable, but at least 200 years.	40 CFR 192
Radon emissions from waste containment area	20 pCi/m ² -s or 0.5 pCi/l in air outside of the waste containment area	40 CFR 192
<u>Water protection</u>		
Surface water discharge (uncontrolled areas)	Uranium-238 -- 600 pCi/l Radium-226 -- 30 pCi/l	DOE Order 5480.1 Chapter XI
In community water systems	Radium-226 and Radium-228 -- 5 pCi/l	40 CFR 141
<u>Soil Decontamination</u>		
Radium-226	5 pCi/g in the 15 cm surface layer 15 pCi/g in any 15 cm layer beneath the surface layer	40 CFR 192
<u>Building Decontamination</u>		
Indoor radon decay products	0.03 working level	40 CFR 192
<u>Transport of Wastes</u>		
Exposure rates	Not to exceed 10 mR/h at a distance of 2.0 m (6.6 ft) from the vehicle side and 2 mR/h in any normally occupied position in the vehicle	49 CFR 173
Transport containers	Design and licensing	10 CFR 71
Maintenance and surveillance(b)	Includes maintenance, surveillance, and environmental monitoring	DOE Order 5480.1 Chapter XI(c)

(a) All concentrations are incremental (i.e., in addition to background) except for the building decontamination criteria and radium 226 for water in restricted areas.

(b) Assumed to continue for the design life of the waste containment area.

(c) Although not specifically addressed, the need for maintenance and surveillance is implied.

The Retain and Manage scenario involves storing the residues and wastes on site in an engineered waste containment area. Variations of the identified alternatives include the type and degree of waste containment area improvements and waste form immobilization techniques.

5.3.1 Upgrade Waste Containment Area

The performance requirements stated in Appendix E can be met by upgrading the waste containment area to reduce long-term risks associated with radionuclide migration. This will involve the use of engineered barriers, including low permeability covers, cut-off walls, and slurry walls. A layer of rip-rap could be used to minimize plant root and animal intrusion and inadvertent human intrusion.

Upgrading the containment of the residues and wastes by using engineered barriers (described in detail in Section 6.2.1) will effectively:

- o Attenuate gamma radiation from the waste to near natural background levels
- o Divert surface water and ground water flow around the pile
- o Prevent surface water or ground water from percolating into the waste
- o Significantly reduce the release of radon from the material to within criteria levels
- o Present an aesthetically pleasing appearance.

Retaining the waste and residues in bulk form in the containment area eliminates the need for excavation after interim remedial actions are completed. This reduces cost and occupational radiation exposure.

Bulk disposal of NFSS wastes on site in newly constructed, clay-lined, below-grade pits was considered. This concept was eliminated because upgrading the waste containment area constructed during interim remedial action (see Section 3.1.2) is considered

equally effective and will be significantly less costly. In addition, bulk excavation of these materials and reburial in a new containment area on site would involve greater occupational radiation exposure.

5.3.2 Immobilization and Processing Schemes

In addition to storing the residues and wastes in bulk form and upgrading the waste containment area, consideration was given to stabilizing the more radioactive (K-65, L-30, L-50, and F-32) residues at the NFSS. Stabilization of these residues would involve altering the physical and chemical characteristics of the waste form to reduce the rate of liquid-enhanced processes such as migration, dispersion, or leaching, as well as the release of gases. Waste stability may be increased by altering the waste form into a solid matrix. The immobilization process is designed to reduce the mobility, leachability, and radon emanation of the residues.

Several immobilization techniques were considered for the residues within the NFSS. These techniques include the use of asphalt, cement, polymers, and resins. Also considered was vitrification in an electric furnace, vitrification (using electrodes) within the residue containment structures, and resource recovery (uranium, base metals, and precious metals) which produces a vitrified waste product.

The evaluation of the immobilization techniques was based on the following considerations:

- o Radon suppression
- o Leachability
- o Material availability
- o Product corrosion
- o Recoverability of the waste form
- o Flammability
- o Bacterial attack
- o Technological status
- o Process timing
- o Cost.

Using these evaluation criteria, it was determined that the vitrification techniques would produce a waste form with characteristics more amenable to management of these materials at the NFSS if it were determined that immobilization in addition to upgraded containment would be required. Detailed descriptions of these techniques may be found in Sections 6.2.2 and 6.2.3 and Appendices B and C.

In addition to increasing their stability, processing the residues for resource recovery may be justified by the value of the metals that would be recovered. A decision whether or not to immobilize the residues in addition to upgrading containment must consider the need for increased residue stabilization in relation to cost and the design life of the upgraded containment area. On the basis of considerations described in Chapter 6 and Appendices B and C, resistance furnace fusion and slag production from resource recovery were identified as the immobilization techniques to be used for the development of immobilization and processing alternatives in this document.

Consideration was given to immobilizing or processing and immobilizing the R-10 residues and contaminated soils, sands, and sediments at NFSS. Immobilizing the large volumes of these wastes with their minimal specific activity is not justifiable because of significantly increased costs with negligible reduction to health risks.

5.4 DECONTAMINATE AND RELEASE SCENARIO

Alternatives considered for implementation of this scenario include:

- o Transport containerized K-65 residues, bulk L-30, L-50, and F-32 residues, and remaining waste to a disposal facility in the eastern or western U.S.
- o Transport immobilized residues in containers and other wastes in bulk to a disposal facility in the eastern or western U.S.

- o Transport immobilized processing slag resulting from resource recovery from the residues in containers and other wastes in bulk to a disposal facility in the eastern or western U.S.

The Decontaminate and Release scenario involves transporting to a burial site the residues and wastes that exceed radiological criteria levels for unconditional release. Variations of the identified alternatives include the destination, the transportation mode, and the waste form.

5.4.1 Waste Destination

The residues and wastes could be transported to an existing eastern or western DOE burial site. The eastern and western sites are assumed to have space available, established operational practices, and acceptable geological and climatological characteristics that would allow safe long-term burial of the NFSS residues and wastes. Although burial at a western site involves much greater transportation distance and cost, it was given full consideration in this document because DOE directed that both a humid eastern site and an arid western site be evaluated.

New DOE burial sites might be available to receive NFSS residues and waste if they are developed before the final disposition of NFSS is implemented; development of a new site could take 10 years or longer. Due to the uncertainty that such sites would be identified, their use is not considered in this document. The cost of developing a new site would be borne by the users of that site. The cost of the alternatives under the Decontaminate and Release and the Partial Decontamination scenarios could consequently be expected to increase in proportion to the volume of NFSS wastes disposed of at the new burial site.

Alternative uses of the waste were also considered instead of long-term management at a DOE facility. Alternative uses could include fill for dams, roadbeds, or mine backfill. The physical and

chemical characteristics of the waste make its use in roadbed or dam construction inappropriate. Regulatory constraints (10 CFR 61) disallow the use of these materials in mine backfill, even in uranium mines, unless the mines are designated disposal sites and conform to disposal site criteria. Alternative use of the waste was not considered to be an acceptable variation of the main alternatives addressed in this document.

Ocean disposal (by dispersal) of radioactive low specific activity materials such as those stored at NFSS is not expected to be approved by DOE or the U.S. Environmental Protection Agency (EPA) in the near future. The non-residue wastes at the NFSS could possibly qualify for ocean disposal at some future date; however, because it is uncertain when such a decision may be made, ocean disposal/dispersal methods were not retained for consideration in this study. Ocean disposal will be evaluated in the EIS currently being prepared for the final disposition of the NFSS.

5.4.2 Transportation Modes

Examination of transportation modes for the residues and wastes stored at the NFSS required consideration of transport routes and transportation modes, and identification of any limitations they might have. Truck, barge, and rail transport were considered as indicated below.

5.4.2.1 Truck Transportation

Removal of the residues and wastes to a disposal site by truck would require that present access to the NFSS from public roads be upgraded to handle heavy truck traffic.

Truck transport of the bulk waste would be done in plastic-lined and covered tractor trailer dump trucks with gasketed tailgates. Weight restrictions would limit the volume of waste transported in each 20-yd³ truck to 12-13 yd³.

The truck loading operations and the facilities that would be used are similar to those that would be developed for rail transportation. Auxiliary equipment such as truck scales would be required to assure that highway weight limits were met. The round trip distance to the burial site is assumed to be approximately

2,350 km (1,460 mi) to the eastern site and 8,050 km (5,000 mi) to the western site. It is assumed that approximately 200 trucks would be required to serve the eastern site and approximately 400 for the western site. Each is assumed to have two drivers. Furthermore, it is assumed that the burial sites would accept the wastes only during normal working hours, Monday through Friday. One-way travel time is estimated to be approximately 2 days to the eastern site and 5-1/2 days to the western site. It is expected that one day would be needed at the burial site for unloading, decontamination, and the performance of normal maintenance on the trucks. Return to the NFSS would be by the same route in either case and would require equivalent travel time.

During the entire transportation operation, radiological monitoring of personnel and equipment would assure compliance with DOE, State of New York, and other affected state's guidelines (i.e., interstate transport of hazardous materials). Additional health and safety precautions would be employed during all phases for protection of personnel.

The cost of truck transportation is approximately \$188 per t (\$171 per ton) for transport to the eastern site and approximately \$363 per t (\$330 per ton) for transport to the western site. Purchase of trucks and trailers is not considered in this cost. The freight rates used in preparation of this estimate were based on Tri-State Motor Transit Co. published rates for handling radioactive waste.

5.4.2.2 Barge Transportation

Barges could be used to transport residues and wastes to an existing DOE disposal site. These materials would be loaded onto trucks and transported to a navigable waterway within approximately 320 km (200 mi) of the NFSS where barge loading facilities would be developed.

In the case of an eastern disposal site river barges would be used for the entire route. Use of a western disposal site would involve two enroute transfers, because combination river-ocean barges cannot be economically sized and are not commercially

available. Consequently, two intermediate loading/unloading facilities would be required. The wastes would be transferred, at a point on the lower Mississippi, to deep-draft, ocean-going barges or ships which could range in size approximately 13,600 to 27,300 t (15,000 to 30,000 dead weight tons).

The ocean-going barges or ships would transport the residues and wastes by way of the Panama Canal to a point on the coast of the western U.S. for transfer to river barges for the final leg of the journey. Unloading facilities would be built at the receiving port near the disposal site and the wastes would be trucked from there to the disposal site.

Ocean movement would, by law (the Merchant Marine Act of 1920 also known as the Jones Act), have to be in the U.S. vessels, which are unlikely to be available and which would be extremely costly to build and dedicate for 9 months per year for 4 to 5 years. The U.S. merchant fleet consists of only 20 bulk carriers suitable for this service, most of them of ancient vintage. There are even fewer tug/barge combinations in bulk material configuration, and these are all primarily assigned and dedicated to hauling coal and phosphate rock. In light of this new barges, costing approximately, \$35 million each, would have to be constructed.

Preliminary evaluations indicate that it would take about 190 barge trips to transport all contaminated materials from NFSS to another disposal site. This assumes that 1,816-t (2,000-ton) capacity barges would be used. A first estimate of moving the residues and wastes by barge to an eastern disposal site is approximately \$124 per t (\$113 per ton). For a western site this cost is approximately \$275 per t (\$250 per ton). Additional costs would be incurred for intermediate truck transportation, loading/unloading facility development, and barge purchase.

Consideration of barge transport assumes that the extensive permits, licenses, and clearances required for shipping contaminated wastes in barges could be obtained. For example, it is expected that an exemption for transporting the wastes would be required from the Department of Transportation (DOT).

5.4.2.3 Rail Transportation

Rail shipment is an alternate mode of transportation for bulk and containerized radioactive materials. The feasibility of utilizing various railcar designs for waste transport is governed by considerations such as material handling ease, security, decontamination, etc. Most types of rail equipment are not specifically designed to meet all expected requirements for waste hauling. Further, it would be necessary to purchase or lease private cars.

In general, two types of rail cars can be considered for various palletized or packaged commodities: bulk-handling cars and open or closed cars. Bulk handling railcars include open and covered hoppers, high- and low-side gondolas, and side dump cars. For bulk transport of the NFSS residues and wastes, gondolas would be used; containerized K-65 residues would be shipped in flatbed cars.

New loading facilities would be developed on the site. The wastes would be transported by truck to these facilities where each railcar would be loaded in a fully enclosed dust controlled environment. The capacity of each gondola is 91 t (100 tons); each flatbed car would hold two casks each containing fifteen 55-gal drums. After loading, each gondola would be decontaminated and a gasketed cover attached. Each train would comprise approximately 54 cars, totalling 4,903 t (5,400 tons).

At the disposal site each car would be unloaded in a fully enclosed dust-controlled environment by conventional earthmoving equipment. Prior to departing the disposal site, the train would be transferred to a decontamination facility for cleaning, final inspection, and release.

The round trip distance to the eastern burial site is assumed to be approximately 2,350 km (1,460 mi) and to the western site approximately 8,240 km (5,120 mi). It is assumed that a main line railroad track is available at both of the DOE disposal sites; however an allowance for a 0.8-km (0.5-mi) spur track has been included in the cost estimate. Complete cycle time required for travel to the disposal site, unloading, and return to the NFSS would

be approximately 15 days for the eastern site and 30 days for the western site. This will involve two trains serving the eastern site or three trains serving the western site.

The cost of rail transport to the eastern disposal site is approximately \$83 per t (\$75 per ton); to the western site it is approximately \$190 per t (\$172 per ton) . Included in this cost is approximately \$2.5 million for the construction of an 8 km (5 mi) spur to the NFSS.

5.4.2.4 Comparison of Transportation Modes

In evaluating these three modes of transportation the hazards to the general population and environment, feasibility of implementation, occupational radiation exposures, and costs associated with each must be considered.

General population exposure would be significantly lower in the case of barge or rail transport in light of the routes used. Frequency of barge or rail accidents would be potentially lower than that of truck accidents in view of the volume of traffic on these routes. However, the environmental impact of a barge or rail accident would be potentially greater than that of a truck accident, particularly when transporting the NFSS wastes in unprocessed form.

Implementation of truck or rail transportation would be more feasible than barge transportation since loading facilities could be built on the NFSS. Implementation of barge transportation would require the location and purchase of suitable land for barge loading/unloading facilities on two or more navigable waterways. Furthermore, barge transportation would require additional handling due to the necessity of trucking the wastes to these waterway facilities.

Total occupational radiation exposure experienced during rail transportation would be less than that for truck or barge due to the greater distance between the drivers and the wastes and the lower total number of manhours required to complete the removal of the wastes to the disposal site.

Comparison of the costs of transporting the NFSS wastes by the three modes discussed above indicates that rail is the most economical option for either disposal site. In the case of an eastern site, per-ton costs of truck, barge, and rail transportation are \$171, \$113, and \$75 respectively. In the case of a western site, these costs are \$330, \$250, and \$172 respectively.

It must be remembered, however, that these cost estimates were prepared based on the assumption that an exemption from the requirements of the September 1983 revisions to 10 CFR 71 and 49 CFR 171-174 could be obtained if necessary. Should this not be possible, the cost of transporting the NFSS residues and waste by any of the three modes would increase considerably.

Based on the above evaluation, rail transportation appeared to be the most favorable and was used in Chapter 6 to develop costs for transportation of the wastes to the eastern and western disposal sites.

5.4.3 Immobilization and Processing Schemes

The waste form alteration techniques identified in Section 5.3.2 are also applicable to the Decontaminate and Release scenario. The selected techniques include the use of resistance fusion furnaces and resource recovery which produces a final immobilized waste slag.

Immobilization of the residues at an offsite location after transport from the NFSS was considered but not retained as a variation of an alternative because it would be more costly than onsite immobilization and would involve greater occupational radiation exposure due to multiple handlings of the residues. The volume and weight reduction associated with immobilization would help reduce transportation cost. If fewer transport vehicles were used than for bulk waste shipment, risks would also be reduced. Offsite immobilization at a disposal site may be desirable if waste from several source sites was involved and the arrival of quantities of waste to be immobilized could be scheduled to make the operation cost-effective.

5.5 PARTIAL DECONTAMINATION SCENARIO

This scenario, which involves transportation of containerized K-65 residues and the other more radioactive residues in bulk for management at a DOE burial site in the eastern or western U.S., would remove 96 percent of the total radioactivity from the site. It has been added to the list of scenarios and alternatives as a result of the NFSS EIS public scoping process. The scenario and alternatives identified in that scoping process do not include immobilization or processing and immobilization schemes.

The Partial Decontamination scenario is a combination of alternatives under the Retain and Manage scenario for the waste remaining on site and the Decontaminate and Release scenario for the waste being shipped off site. A significant difference is that following partial decontamination, it will not be possible to release the site for other uses.

The discussion of variations of alternatives in Sections 5.3.1, 5.4.1, and 5.4.2 also applies to the Partial Decontamination scenario.

5.6 DETERMINATION OF ALTERNATIVES

The alternatives selected for each of the final disposition scenarios for the NFSS are:

No Action Scenario

Alternative:

- (1) Perform maintenance and surveillance at the site in its post-interim-remedial-action condition.

Retain and Manage Scenario

Alternatives:

- (2) Upgrade containment of the stored wastes.
- (3) Immobilize the residues and upgrade containment.
- (4) Process the residues to extract precious and base metals, immobilize residues, and upgrade containment.

Decontaminate and Release Scenario

Alternatives:

- (5) Transport containerized K-65 residues and all other residues and wastes in bulk for management at an existing DOE burial site in the eastern U.S.
- (6) Immobilize the residues, place them in shipping containers, leave the remaining radiologically contaminated material in bulk, and transport all wastes for management at an existing DOE burial site in the eastern U.S.
- (7) Process for resource recovery and immobilize the residues, place vitrified slag in shipping containers, leave the remaining radiologically contaminated material in bulk, and transport all wastes for management at an existing DOE burial site in the eastern U.S.
- (8) Transport containerized K-65 residues and all other residues and wastes in bulk for management at an existing DOE burial site in the western U.S.
- (9) Immobilize the residues, place them in shipping containers, leave the remaining radiologically contaminated material in bulk, and transport all wastes for management at an existing DOE burial site in the western U.S.
- (10) Process for resource recovery and immobilize the residues, place vitrified slag in shipping containers, leave the remaining radiologically contaminated material in bulk, and transport all wastes for management at an existing DOE burial site in the western U.S.

Partial Decontamination Scenario

Alternatives:

- (11) Transport containerized K-65 residues and bulk L-30, L-50, and F-32 residues for management at a DOE burial site in the eastern U.S., and upgrade containment of the remaining wastes on site at NFSS.

- (12) Transport containerized K-65 residues and bulk L-30, L-50, and F-32 residues for management at a DOE burial site in the western U.S., and upgrade containment of the remaining wastes on site at NFSS.

Table 5-2 shows the alternatives for management of the residues and wastes at NFSS. Table 5-3 shows the alternatives for decontamination and release of the NFSS. Table 5-4 shows the alternatives for partial decontamination of the NFSS. Each alternative is shown by the waste type to which it is applicable. Descriptions of the four scenarios are presented in detail in Sections 6.1, 6.2, 6.3, and 6.4, respectively. Conceptual engineering designs are described, occupational radiation exposure factors considered, and costs and schedules estimated so that the alternatives may be compared.

TABLE 5-2

ALTERNATIVES FOR MANAGEMENT OF RESIDUES AND WASTES STORED
AT NFSS IF THE DISPOSITION CHOSEN IS TO RETAIN AND MANAGE THE SITE

Waste Type	Upgrade Containment	Immobilize and Upgrade Containment	Process for Resource Recovery, Immobilize, and Up-grade Containment
Ore Processing Residues	X	X	X
Soils, Sediments and Sands(a)	X		
Contaminated Rubble	X		

(a) Includes R-10 residues.

TABLE 5-3

ALTERNATIVES FOR MANAGEMENT OR DISPOSAL OF RESIDUES AND WASTES STORED AT
NFSS IF THE DISPOSITION CHOSEN TO DECONTAMINATE AND RELEASE THE SITE

Waste Type	Ship(a) and Land Dispose at Existing DOE Facility	Immobilize and Land Dispose at Existing DOE Facility	Process for Resource Recovery, Immobilize Waste Product, Land Dispose at Existing DOE Facility
Ore Processing Residues	X	X	X
Soils, Sediments, and Sands(b)	X		
Contaminated Rubble	X		

(a) The K-65 residues would be packaged for transport.

(b) Includes R-10 residue.

TABLE 5-4

**ALTERNATIVES FOR MANAGEMENT OR DISPOSAL OF RESIDUES AND WASTES STORED
AT NFSS IF THE DISPOSITION CHOSEN IS PARTIAL DECONTAMINATION OF THE SITE**

Waste Type	Ship(a) and Land Dispose at Existing DOE Facility	Upgrade Containment and Store On Site
Ore Processing Residues	X	
Soils, Sediments, and Sands(b)		X
Contaminated Rubble		X

(a) The K-65 residues would be packaged for transport.

(b) Includes R-10 residue.

6.0 ASSESSMENT OF ALTERNATIVES

The 12 selected alternatives for the four possible dispositions of the NFSS identified in Chapter 5 for assessment are described in the following sections and depicted in a logic diagram, Figure 6-1.

The following sections contain conceptual design descriptions, occupational radiation exposure estimates, estimated schedules and manpower requirements, and cost estimates for each of the alternatives. The NFSS site conditions after completion of interim remedial action activities are used as the base case from which the alternatives for the final disposition of the site are developed.

6.1 NO ACTION SCENARIO

No action, as used in this document, involves only those measures that effectively limit public exposure to radioactive sources, such as restricting public access to contaminated property. The property, in the post-interim-remedial-action condition, would need to be maintained and radiologically monitored. The cover over the waste containment area constructed during interim remedial action is assumed to have a design life of 25 years because there will be long-term deterioration due to freeze-thaw effects. In addition, root and small animal intrusion could occur because the interim cover does not provide a long-term intrusion barrier.

6.1.1 Alternative 1: Maintenance and Surveillance

Under this alternative, the site in its post-interim-remedial-action condition would be maintained indefinitely, with no significant alterations or additional construction. Implementation of Alternative 1 would require continuing maintenance, surveillance, and management support activities.

Several activities would be required for maintenance of the NFSS:

- o Inspection and maintenance of the multiple-layer cover over the waste containment area
- o Maintenance of the fence and vegetative cover
- o Upkeep of roads and access areas to sampling stations

FIGURE 6-1 LOGIC DIAGRAM INDICATING THE ALTERNATIVES FOR THE FINAL DISPOSITION OF THE NFSS

- o Support facility upkeep
- o Replacement of security fence (approximately every 25 years).

Repairs and maintenance would be conducted as required in order to meet performance standards established for the facility. For the purpose of costing, it is assumed that a half-time manager would be required for these activities and that maintenance and repair activities would be performed by local subcontractors as needed. Inspection of the multi-layer cover would be conducted each spring, following the last frost, and in the fall. The cover would be recompact or reworked if animals or freeze-thaw effects caused damage.

The surveillance program would be conducted in accordance with DOE Order 5480.1A. Surveillance would consist of: inspecting the fences and multi-layer cover, enforcing security, and performing regularly scheduled environmental monitoring.

Access to the waste containment area would be restricted by security fences. A local law enforcement agency would be retained to patrol the site. The security fences would be inspected quarterly and repairs made as required. It is anticipated that inspections would be conducted quarterly or more frequently if monitoring data suggested increased nuclide concentrations in air or water.

Environmental monitoring conducted for the waste containment area will be based on cumulative monitoring and other relevant data available at completion of the upgraded cover. A tentative monitoring plan for the containment area is shown in Figure 6-3 (page 6-8). Radon and external radiation monitors will be located around the containment area. Specific locations will include those with localized concentrations of contaminated wastes (i.e. buried Building 411) and other areas to be determined based on meteorological data. Groundwater monitoring will be conducted outside the perimeter of the slurry wall with wells emplaced in both the upper and lower alluvial aquifers. Other wells will be sampled down gradient including wells at the site boundary and the nearest private well. The frequency of sampling for all monitoring will

vary, with initial samples obtained over short periods of time (i.e., quarterly) and then extended (i.e., annually) as determined by examination of sample results.

During quarterly surveillance activities, the health physics program would be minimal, with documentation of exposure rates and occurrence of surface contamination provided by the environmental monitoring program. However, for maintenance, an increased emphasis would be placed on health physics with documentation of individual exposures and contamination control surveys.

During surveillance, the environmental monitoring program would provide data to determine site compliance with applicable regulations, exposure rates on the site and at the site boundary, and the performance of the waste containment barriers. Environmental monitoring would be performed by one individual employed by a subcontractor with offsite analytical support. Monitoring data would be filed with the responsible DOE Operations Office. Environmental monitoring and reporting of results would be as follows:

- o Exposure rates -- continually, quarterly analysis, annual reporting
- o Air -- continually for radon-222, quarterly analysis, annual reporting
- o Water -- groundwater and surface water for radium-226 and uranium-238, quarterly analysis, annual reporting.

The No Action scenario requires that maintenance and monitoring continue for an indefinite time period. Direction of site activities and reporting of site status would be the primary management functions. Each of the support activities would be designed to maintain the performance objectives for the waste containment area.

6.2 RETAIN AND MANAGE SCENARIO

There are three alternatives under this scenario. The intent of each is to upgrade the waste containment area (and possibly alter the waste form) to further reduce the possibility of offsite

contaminant migration and to minimize the need for ongoing maintenance and surveillance activities. The criteria used for waste containment activities are given in Appendix E.

6.2.1 Alternative 2: Upgrade Containment

With this alternative, the potential for migration of the wastes will be reduced by upgrading the interim-remedial-action cover and side-wall barriers of the waste containment area. Figure 6-2 illustrates sections taken through the upgraded waste containment area.

Upgrading the cover would first involve removing the top layers of sand and topsoil of the interim cover (described in Section 3.1.2.1) and stockpiling these materials for use as topsoil on the upgraded cover. If required, the upper portion of the clay layer would be recompacted to repair any damage caused by frost heave or animal intrusion. An additional 0.6 m (2 ft) of clay would then be added above the recompacted clay layer to provide a total thickness of 1.5 m (5 ft).

A 15-cm (6-in.) sand filter layer would be placed as a transition between the overlying rip-rap and the underlying clay. The rip-rap layer is designed as an engineered intrusion barrier. This layer would consist of 0.9 m (3 ft) of well-graded rip-rap and would serve as a barrier to animal, plant, and human intrusion into the waste. Above the rip-rap layer, another 15-cm (6-in.) sand filter layer would be placed to function as a transition between the rip-rap and the overlying topsoil. The rip-rap and sand layers would also serve as drainage layers to drain infiltrating surface water away from the waste. The topsoil layer would be composed of 46 cm (18 in.) of topsoil capable of supporting shallow-rooted grasses. The additional quantities of materials required to upgrade the interim cover include an estimated 40,500 m³ (53,000 yd³) of clay, 17,800 m³ (22,000 yd³) of sand, 44,400 m³ (58,000 yd³) of rip-rap, and 6,300 m³ (8,250 yd³) of topsoil. This totals about 11,000 truck loads if all of the material were obtained from

FIGURE 6-2 CROSS-SECTION OF THE UPGRADED WASTE CONTAINMENT AREA (ALTERNATIVES 2, 3, AND 4 - RETAIN AND MANAGE)

offsite locations. It is anticipated that rubble from building demolition could provide a portion of the required additional rip-rap.

All of the cover layers would be gently graded away from the waste containment area to enhance natural drainage and reduce erosion. The top of the cover would have a slope of 5 to 10 percent; the edges would have a slope of 5 to 1. An eastward diversion of the Central Drainage Ditch would be required, and Lutts Road would have to be relocated to be compatible with the upgraded cover. A plan of the relocated Central Drainage Ditch is shown in Figure 6-3. The water retention ponds are scheduled to be removed at the end of interim remedial actions and would have to be replaced for any alternative requiring these facilities. It is likely that Building 429 would be used for construction office facilities and that maintenance activities would be conducted from Building 403.

Improvement of sidewall barriers, if required, would entail the addition of a cut-off wall constructed of bentonite slurry or compacted clay. The cut-off wall would provide an additional barrier around the perimeter of the waste containment area from grade to the gray clay layer described in Section 3.3.1.3. The cut-off wall would be keyed into the upgraded clay cover to provide a continuous system. The decision as to the requirement for an additional cut-off wall and the type of cut-off wall to be constructed (bentonite slurry or compacted clay) is contingent upon an ongoing analysis being conducted by Bechtel concerning the effectiveness of the existing dike and cut-off wall to perform as a long-term barrier to ground water infiltration and radionuclide migration.

If the NFSS is upgraded and retained as a waste management facility, the following guidelines will govern maintenance and surveillance activities at the site. For a period of 5 years after completion of remedial actions, the site would be actively monitored, maintained, and surveilled as described in Section 6.1.1.

After the 5-year period following site closure, only custodial maintenance and surveillance would be conducted. These activities would include yearly inspection and maintenance of the site

**FIGURE 6-3 WASTE CONTAINMENT AREA WITH UPGRADED COVER
(ALTERNATIVES 2, 3, AND 4 - RETAIN AND MANAGE)**

perimeter and the waste containment cover. Tree growth would be removed as required to prevent cover damage. The perimeter fence would be replaced as required (expected to be every 25 years). It is expected that some time after site closure, the site, except for a small area around the waste containment area, would be released for other uses. At that time, only the waste containment area would require a security fence. Environmental monitoring, as described previously, would continue on a yearly basis for at least 195 years.

6.2.2 Alternative 3: Immobilize Residues, Upgrade Containment

Implementation of this alternative would involve the same waste containment area improvements described in the previous section. However, before the final cover was placed over the containment area, the K-65, L-30, L-50, and F-32 residues would be removed, immobilized, and replaced within the containment area. As discussed in Chapter 5, it is assumed that immobilization would be conducted using available fusion furnace technology. A detailed description of this process and a discussion of the adequacy of other immobilization technologies for NFSS residues is presented in Appendix B.

At the conclusion of interim remedial actions, the residues will have been placed in Buildings 411, 413, and 414 in bulk form. These buildings will be within the waste containment area as described in Section 3.1.2. It has been assumed for this document that the residues would be removed for immobilization using hydraulic mining. The necessary mining equipment, pumps, and piping would be available from interim remedial action operations. Hydraulic mining lends itself to remote operations and facilitates control of radon release during residue removal operations. Removal of the residues by conventional bulk excavation, however, could present significant problems due to handling and increased worker exposure and radon release which could not be satisfactorily resolved during the conceptual engineering evaluation. If a decision is made to implement this alternative or any other involving removal of the residues, bulk excavation techniques will be re-evaluated as to their applicability in removing these materials. Bulk excavation techniques could potentially be less costly than hydraulic mining.

The clay, identification membrane layers, and contaminated soil placed over Buildings 411, 413, and 414 during interim remedial actions would have to be removed before hydraulic mining could be initiated. Figure 6-4 shows the approximate extent of the required excavation prior to hydraulic mining.

Hydraulic mining uses high pressure water jets (180 gal/min at 200 psi) to form a residue slurry (30 percent residue by weight) that would be pumped to the process building where it would discharge into a thickener. Suspended residues would settle and be pumped to an indirectly heated drier. The resulting residue would be dried to approximately 15 percent water by volume to maximize the efficient drying of the materials in the furnace without creating a product that would be subject to dusting. It would then be added to the fusion furnace for immobilization. A continuous source of slurry water would be provided by pumping the thickener supernatant, and makeup water if necessary, back to the area being mined.

The residues would be transferred to the fusion furnace which would be located in a process building east of the Central Drainage Ditch (see Figure 6-5). Use of an existing site building as a process building may be possible and will be considered if a processing or immobilization alternative is selected. As conceptualized, the L-30 and F-32 residues will be hydraulically mined from the east bay of Building 411 and pumped to the processing building where they would be dried and then placed in the fusion furnace. The fusion furnace, for the purpose of this document, is assumed to have an immobilization capacity of 36 t (40 tons) per day. This processing rate was assumed because:

- o Increasing the process capacity would add significantly to the capital cost of the operation and reduce the portability of the equipment.
- o Reducing the process capacity would result in an extended period of time required to immobilize the residues.

PLAN

N.T.S.

TYPICAL N-S SECTION

N.T.S.

FIGURE 6-4 HYDRAULIC MINING OF RESIDUES (ALTERNATIVES 3 & 4 - IMMOBILIZATION OR PROCESSING, AND ALTERNATIVES 11 & 12 - PARTIAL DECONTAMINATION *)

* If the site is decontaminated, the entire cover would be removed.

**FIGURE 6-5 LOCATION OF PROCESS BUILDING (ALTERNATIVES 3, 4, 6, 7, 9, AND 10
- RETAIN AND MANAGE WITH IMMOBILIZATION OR PROCESSING)**

The molten residue would be transferred to an insulated trunnion transfer cask. The cask would be carried by a forklift that has been adapted to carry and pour the melt from the transfer cask onto a layer of sand on the ground outside Building 411. The temperature of the melt at the time of discharge from the furnace would allow the molten waste to be easily poured. The vitrified mass would flow, quickly cool, become viscous, and take on solid properties. The waste would be allowed to cool without a cover. Additional melts would be poured onto the previous layer. The waste heap would tend to retain some of its residual heat, thus preventing the upper and outer layers from cooling too quickly. This would minimize the tendency for the material to develop extensive thermal cracking. When the east bay of Building 411 was emptied, the operation would be reset to hydraulically mine the K-65 residues from the west bay of Building 411.

Hydraulic mining of the the K-65 residues, located in the west bay of Building 411, would require the special techniques described in the material that follows. These handling techniques are due to the specific activity of this material and the large concentrations of radon that may be expected to be released. Before the hydraulic mining procedure commenced, the clay cap and contaminated soil covering the west bay would be removed to a depth of 0.3 m (1 ft) above the identification layer. A 0.6-m (2-ft) layer of water would be placed over the remaining soil and material covering the K-65 residues. At this point, a temporary structure would be constructed over the bay. A fan would maintain negative pressure within the bay and allow a controlled release of radon gas during hydraulic mining through a 24-m (80-ft) stack located near the northern portion of the building (see Appendix D). The stack height was preliminarily estimated as the minimum required to prevent downwash of the gases as they leave the top of the stack and to assure that radon concentrations in the work area and at the site boundary do not exceed applicable DOE criteria. This analysis was based on radon emanation rates from the residues and meteorological data from the site (see Section 3.3.4). The stack height will be adjusted depending on the final design height of the temporary lightweight structure over Building 411.

After placing the temporary structure, the hydraulic mining equipment would be lowered through an opening in the roof, and the remaining soil layer would be slurried to a sedimentation pond for settling. The remaining identification layer would be sectioned and left within the bay. The exposed K-65 residues would be hydraulically mined to the processing building as described previously for the L-30 and F-32 residues. The east bay of Building 411 would then be prepared to receive the immobilized K-65 residues by adding a thin layer of sand over the concrete floor. The molten, vitrified K-65 residues would be poured onto the dried sand. The vitrified, solidified L-30 and F-32 residues that were poured onto the sand layer outside of Building 411 would be excavated using a front end loader. This solidified heap is expected to break up in distinct layers which will fracture along the individual pours. The fractured waste pieces would be dumped onto the K-65 residues. Additional melts of the vitrified K-65 residue would be poured onto the fractured layer, thus filling the voids and decreasing the surface area-to-volume ratio. This process would be repeated until all the K-65 residues are vitrified and placed in the east bay of Building 411.

After removing the clay and sand covers in Buildings 413 and 414, the L-50 residues would be removed by hydraulic mining techniques and transferred as a slurry to the processing building for drying and placement in the fusion furnace. The molten residue would be poured over a sand layer in the west bay of Building 411 and would be covered by contaminated soil. The immobilized radioactive waste in the bays of Building 411 would be: reduced in volume by approximately 50 percent in relation to non-immobilized residues, chemically inert, physically stable, and leach resistant.

The immobilized residues in both bays would be covered with contaminated soil and clay to bring the level of Building 411 up to grade. Figure 6-6 indicates the location of the immobilized residues in Building 411.

After completing hydraulic mining, the slurry piping would be decontaminated. The internal surfaces would be hydroblasted using a self-tracking nozzle. The pipe would be drained, flushed, and

FIGURE 6-6 LOCATION OF IMMOBILIZED RESIDUES IN BUILDING 411 (ALTERNATIVES 3 AND 4 - RETAIN AND MANAGE WITH IMMOBILIZATION OR PROCESSING)

segmented into lengths which could be easily handled for disposal. Internal cleaning of the pipes allows handling of the pipe segments with fewer radiological restrictions. Water used for hydraulic mining and decontamination activities would contain dissolved uranium, radium, and heavy metals which would require removal prior to discharge. The formation of complex metal compounds which may result from uranium, molybdenum, and vanadium reactions with carbonates during the hydraulic mining process would be reduced by the addition of lime. The lime would react with the soluble salts contained in the residues (primarily sodium carbonate) to form insoluble calcium carbonate. Removal of the carbonate from solution would minimize the formation of the complex metal compounds (Merritt, 1971). After the hydraulic mining procedure was completed, the clarified slurry water would be pumped through an activated carbon filter to remove suspended particulates prior to entering a radium complexer and anion exchange columns. This is a preventive measure used to protect the more expensive water treatment units. The filtered water would then be passed through a radium-specific complexer to reduce the radium concentration to below the DOE discharge limit of 30 pCi/l (DOE, 1982b). After passing through the radium complexer, the water would be cycled through an anion exchange to reduce the uranium concentration to below the DOE discharge limit of 600 pCi/l. The use of an anion resin to remove uranium from a carbonate leach circuit is well understood (Merritt, 1971). Treatment of the wastewater to remove heavy metals could involve either passing it through cation-anion exchange columns on site, evaporation, or chemical treatment on site with the resulting salts disposed in accordance with state and federal regulations, or transportation off site for appropriate heavy metals treatment. The carbon filter media, the radium complexer, and the anion resin containing removed uranium could be immobilized with the residues in the fusion furnace or could be disposed in appropriate locations within the waste containment area.

Decontamination of the fusion furnace would involve removal of the refractory material and disposal with the immobilized residues in Building 411. The furnace would then be decontaminated in a

manner similar to the other process equipment. The furnace could then be rebuilt and shipped to another site for use. The process building would be decontaminated by removing previously applied strippable paint.

After the residues were immobilized and placed in Building 411, Buildings 413 and 414 would be partially demolished and the waste containment area cover would be constructed (see Figure 6-2). Active management, maintenance, and surveillance would be conducted for 5 years after site closure. Custodial management and maintenance and environmental monitoring would be conducted for at least 195 years, as described in Section 6.2.1.

6.2.3 Alternative 4: Process Residues, Upgrade Containment

Implementation of this alternative would involve the same waste containment area improvements described in Section 6.2.1. However, before the final multi-layered cover was placed over the containment area, the residues would be removed and processed to isolate uranium, precious metals, and lead and base metal sulfides. In addition to resource recovery, processing would produce a final, vitrified, solid waste form that would have no significant resource value and would have properties similar to immobilized waste produced in the fusion furnace, including reduced radon emanation and leachability.

The residues located in Buildings 411, 413, and 414 would be transferred to the process building using the hydraulic mining procedures described in the previous section. After dewatering, the residues would undergo a carbonate leach process before being introduced into two furnaces. (See Appendix C for a detailed description of the proposed processing technology.) The leached residues and sufficient reducing agent to reduce the lead oxides, carbonates, and sulfates to the metal form would be placed in the first furnace. The molten lead produced would contain most of the precious metals and would be extracted before the remaining molten slag entered the second furnace. In the second furnace, a source of sulfide and carbon would be added to produce a nickel, cobalt, iron sulfide matte that would be cast into molds for shipment to markets

to be identified in the future. For the purpose of this document, the processing operation is assumed to have a residue capacity of 36 t (40 tons) per day for the reasons described in Section 6.2.2. The remaining waste slag would be handled and placed in Building 411 in a manner similar to the immobilized waste described in the previous section (see Figure 6-6). The waste slag would be approximately 50 percent of the original waste volume. The slag, after placement in Building 411, would be covered with contaminated soil and building rubble from the demolition of Buildings 413 and 414, which would be a further deterrent to intruders. A clay cap would bring the level of the materials in Building 411 up to grade.

The L-50 residues would be processed last. Because they contain the least amount of radium per gram, they will tend to partially decontaminate the equipment surfaces that have been in contact with the more radioactive residues. After completing the L-50 processing, the clarifier used to dewater the residue slurry would be dismantled. Accessible internal surfaces and contaminated external surfaces would be flushed, hydroblasted, and blasted with a wet abrasive. The heater equipment and accessories would be difficult to decontaminate, therefore they would be disposed in the waste containment area. Feed pumps, drum handling equipment (if used), exhaust fans, steam lines, wet scrubbers, and other support equipment would be decontaminated.

Water from hydraulic mining and decontamination operations would be treated as described in Section 6.2.2. After the residues were processed and the slag, soil, and clay were placed in Building 411, the multi-layer cover would be constructed. Active management, maintenance, and surveillance would be conducted for 5 years after site closure. Custodial maintenance and continuing environmental monitoring would be conducted for the following 195 years as described in Section 6.2.1.

6.3 DECONTAMINATE AND RELEASE SCENARIO

There are six alternatives considered under this scenario. The intent of each is to transport all radiologically contaminated material from the NFSS to a DOE burial site. The volumes of

contaminated materials that would require transport from the NFSS are listed in Table 6-1. The contaminated materials include the residues, soil, sediments, rubble, and portions of the interim clay cap, cover, and dike components. The volumes and associated weights of each material, shown in Table 6-1, are used in the following sections to determine the amount to be transported. Three alternatives assume disposal at an existing DOE site in the eastern U.S. while the remaining three alternatives assume disposal at an existing DOE site in the west. The eastern DOE site is assumed to be located in an area of moist climate and sandstone, limestone, and shale bedrock. The western DOE site is assumed to be located in an area of arid climate and basaltic bedrock.

6.3.1 Alternatives 5 and 8: Remove All Radioactive Materials, Transport East or West

Removal of the radioactive wastes from the NFSS would involve excavation of various quantities of waste material (including portions of the cover materials placed during interim remedial actions) from various locations within the containment area. The most radioactive materials are concentrated in Buildings 411, 413, and 414, while the remaining materials are distributed over the waste containment area.

For the purpose of this document, it was assumed that any material containing radium-226 greater than that specified in 40 CFR 192 would require removal from the NFSS (see Appendix E). Preliminary calculations using the 40 CFR 192 criterion for radium-226 indicate that approximately 179,900 m³ (238,000 yd³) of radiologically contaminated soils, sediments, building rubble, and cover and dike material would require bulk excavation and transport. In order to determine the volume of material that the railcars would be capable of hauling, it was necessary to estimate the weights of the radiologically contaminated materials as they would be transported from the NFSS. For the purpose of this document the following weights for the NFSS materials were assumed:

TABLE 6-1

ESTIMATED VOLUMES OF CONTAMINATED MATERIALS WHICH REQUIRE
TRANSPORT (ALTERNATIVES 5-10)

Material	Volume (yd ³)
<u>Residues(a)</u>	
K-65	4,080
L-30	7,960
L-50	2,150
F-32	440
Middlesex Sands	230
SUBTOTAL	14,860
<u>Contaminated Soils(b)</u>	
Northern Waste Storage Area(c)	
(1) Below original R-10 residue storage area(d)	35,000
(2) R-10 residues(a)	9,400
(3) 1972 remedial action(e)	15,000
(4) 1982 remedial action(f)	15,700
(5) 1983-1984 remedial action(g)	40,000
Southern Waste Storage Area(c)	
(1) Vicinity properties(h)	18,000
(2) Soil layer covering the residues in Building 411(i)	11,500
Additional below-grade soils in northern and southern waste storage area(j)	<u>13,000</u>
	157,600
20% CONTINGENCY	31,520
SUBTOTAL	189,120
<u>Contaminated Rubble(k)</u>	<u>11,500</u>
20% CONTINGENCY	2,300
SUBTOTAL	13,800

TABLE 6-1
(Continued)

Contaminated Portions of the Interim Cover,
Dike, and Fill

Interim Cover Over Entire Waste	
Containment Area ⁽¹⁾	14,000
Dike Material ^(m)	5,000
Clay Cover Over Bldgs. 413 and 414	1,000
Backfill S of Bldg. 411 ⁽ⁿ⁾	9,000
	<u>29,000</u>
20% CONTINGENCY	5,800
SUBTOTAL	34,800
TOTAL	252,580

-
- (a) Battelle, 1981. Volume also used for Alternatives 11 and 12.
- (b) Contaminated soils contain radium-226 concentrations greater than 5 pCi/g of soil.
- (c) The northern waste storage area lies north of Building 411 within the waste containment area; the southern waste storage area is located south of Building 411 within the waste containment area.
- (d) Assumes radium-226 concentrations exceeding 15 pCi/g of soil to a depth of up to 3.8 m (12.5 ft) beneath the original R-10 residue pile. Area to be backfilled after excavation such that the top 15 cm (6 in.) of soil will not exceed radium concentrations of 5 pCi/g of soil as required by 40 CFR 192.
- (e) Soils placed on the pile during remedial action conducted in 1972.
- (f) Soils placed on the pile during remedial action conducted in 1982.
- (g) Estimates of the contaminated soil resulting from onsite and offsite remedial action activities which will be added to the pile.
- (h) Allowance for contaminated soils from vicinity properties (other than West and Central Drainage Ditches). These properties are currently being surveyed to determine the actual volumes of contaminated soils which may require removal.

TABLE 6-1
(Continued)

- (i) Soils obtained from onsite and offsite decontamination activities during 1983-1984 and placed over the residues in Building 411.
- (j) Assumes radium-226 concentration exceeding 15 pCi/g of soil to a depth of 0.6 m (2 ft). Area to be backfilled after excavation such that the top 15 cm (6 in.) of soil will not exceed radium concentrations of 5 pCi/g as required by 40 CFR 192.
- (k) Contaminated rubble obtained from the demolition of Buildings 410, 411, 413, 414, 415, and 434 and miscellaneous foundations.
- (l) Assumes that 0.3 m (1 ft) of clay comprising the lower portion of the interim cover will be contaminated. The remaining cover material is assumed to be uncontaminated and may be used as backfill after waste material removal.
- (m) Assumes that 0.6 m (2 ft) of clay comprising the dike will be contaminated. The remaining clay will not be removed.
- (n) Uncontaminated backfill may be required for the area in the southern waste storage area to be compatible with the interim cover. A portion of this material may become contaminated between the completion of remedial action and the selection of a final disposition. For the purpose of this document, it is assumed that the total volume of backfill will be transported to a disposal facility in Alternatives 5-10.

- o Residues (dried to approximately 15 percent water by volume): 846 kg/m^3 ($2,435 \text{ lb/yd}^3$)
- o Soil and clay: $1,033 \text{ kg/m}^3$ ($2,970 \text{ lb/yd}^3$)
- o Rubble: $1,406 \text{ kg/m}^3$ ($4,050 \text{ lb/yd}^3$).

Bulk excavation of the contaminated soils and rubble would be performed with conventional earthmoving equipment and in a manner that minimizes contamination of cover materials and underlying soils. Any area excavated below existing grade would be backfilled with clean soils as required for drainage.

The residues located in Buildings 411, 413, and 414, which have a total volume of approximately $11,350 \text{ m}^3$ ($15,000 \text{ yd}^3$), would be removed by hydraulic mining techniques and heat dried to approximately 15 percent water by volume as described in Appendix D. Bulk transport of the L-30, L-50, and F-32 residues at a moisture content exceeding 15 percent water by volume could cause an extrusion of water as a result of vibration/compaction. This condition could cause leakage and therefore result in noncompliance with DOT regulations which do not permit any loss of material during transport. (See Section 6.2.2 for the description of the hydraulic mining and the heat drying processes). Prior to demolition the buildings would be decontaminated using high pressure water spray to remove residue fines. Radon daughter products could be expected to be present throughout the concrete; therefore care would be taken to prevent particulates from becoming airborne during demolition.

The R-10 residues and all contaminated soils, sediments, and building rubble, along with the dried L-30, L-50, and F-32 residues, would be loaded into trucks and transported to the onsite railcar loading facilities. Gondolas would be shunted from the main line to a loading spur constructed at the site. Approximately eight cars at a time would be handled with a car mover similar to a "Trackmobile" manufactured by the Whiting Corporation. The cars would be spotted at the loading facility and filled with contaminated soils and rubble from an overhead hopper into which truckloads of these materials have been dumped. For nonshielded bulk wastes the maximum allowable net weight per gondola has been established as 100 tons.

For bulk residues requiring shielding the weight of the shielding has been set at 10 tons per gondola, thereby reducing the net allowable weight of bulk residue per car to 90 tons.

Dust abatement controls would be utilized to limit airborne contamination during loading operations. Once the cars are loaded, fiberglass covers would be installed and the cars transferred to a decontamination facility for final cleaning, inspection, and release to the staging spur.

A train would consist of approximately 54 cars, totalling approximately 5,400 tons. The complete loading and train make-up operations would take about 7 working days equivalent to approximately 10 calendar days.

The K-65 residues would also be dried to a low moisture content and placed in 55-gal drums. These drums were selected for this document on the basis of availability and compatibility with commercially available flatbed railcars. Lots of fifteen drums would be placed in lead-shielded casks to be consistent with the 49 CFR 173 requirement that radiation exposure rates not exceed 10 mR/h at a distance of 2.0 m (6.6 ft) from the vehicle side. The casks would be lined with polyethylene sheeting and would meet the strong, tight container requirement for LSA material. The interior dimensions of each cask would be approximately 2.1 to 2.4 m long, 1.2 m wide, and 1.2 m high (7 to 8 ft x 4 ft x 4 ft) (Powers, 1982). Placement of the K-65 residues in these drums and casks reduces the potential for residue dispersal in the event of a transportation accident. A detailed description of the procedure used to place the K-65 residues in the drums, including transfer out of Building 411, dewatering, and drying, is given in Appendix D.

Once the casks are ready for shipment, they would be loaded onto flatbed trucks for transport to the railcar loading facility. There two casks would be loaded onto each flatbed railcar by means of a remote-controlled hoist suspended beside the hopper used for loading bulk wastes.

The round trip from the NFSS to the eastern disposal site (Alternative 5) is assumed to be approximately 2,350 km (1,460 mi) and to the western site (Alternative 8) approximately 8,240 km

(5,120 mi). For transport to the eastern site two dedicated unit trains would travel on round trips between NFSS and the disposal site. For transport to the western site three dedicated unit trains would be used.

The values generated by these assumptions were used to calculate preliminary occupational radiation exposure and transportation costs for the alternatives involved.

After removing and transporting the radiologically contaminated materials to a disposal site, the NFSS would be restored and surveyed for final release.

6.3.2 Alternatives 6 and 9: Immobilize Residues, Transport All Radioactive Materials East or West

After removing the multi-layered cover from the waste containment area and the clay covers from Buildings 413, 414, and 411, the residues would be hydraulically mined, dewatered, and heat dried as described in Section 6.2.2. They would then be immobilized as described in Appendix B. The molten residues from the furnace would be poured into molds measuring 0.9 x 0.9 x 0.15 m (3 x 3 x 0.5 ft) and would contain approximately 0.13 m³ (0.17 yd³) of immobilized residue. For the purpose of this document each cooled slab of immobilized waste is assumed to weigh approximately 433 kg (950 lb) with a density of 3,323 kg/m³ (5,590 lb/yd³).

For the K-65 residues, four piles of six slabs would be placed and supported in a lead- and steel-shielded transport container (see Figure 6-7). Preliminary calculations indicate that a lead equivalent side shielding of 5.24 cm (2.06 in.) and 3.02 cm (1.19 in.) of lead equivalent shielding on the remaining surfaces would be required to meet DOT general packaging and shipping requirements for the residues. The shielded containers would be mounted on flatbed railcars in full compliance with DOT regulations and DOE guidelines for transport to an eastern (Alternative 6) or western (Alternative 9) DOE facility. The shielded package, with hold-down devices that fasten it to the railcar, will be designed in accordance with 10 CFR 71 Subpart E and a license application would

FIGURE 6-7 CONCEPTUAL TRANSPORT CONTAINER THAT COULD BE USED TO TRANSPORT IMMOBILIZED K-65 RESIDUES (ALTERNATIVES 6, 7, 9, AND 10 - DECONTAMINATION AND TRANSPORT OF IMMOBILIZED K-65 RESIDUES)

be submitted in accordance with 10 CFR 71.12. The total loaded weight of the transport container would be approximately 18,200 kg (40,000 lb). Loading and transportation of the containers would be as described in Section 6.3.1.

The R-10 residues, contaminated soils, sediments, sands, and building rubble would be bulk-excavated, loaded, and transported as described in Section 6.3.1. Building decontamination and water treatment would also be conducted as described in Section 6.3.1.

After removing and transporting the contaminated materials to a disposal site, the NFSS would be restored and surveyed for final release.

6.3.3 Alternatives 7 and 10: Process Residues, Transport All Radioactive Materials East or West

After removing the multi-layered cover from the waste containment area, the residues would be hydraulically mined and dewatered as described in Section 6.2.2. They would then be processed to isolate precious metals and lead and base metal sulfides as described in Section 6.2.3 and Appendix C. The final waste slag would be handled and transported in the same manner as the immobilized residues described in the previous section.

The radiologically contaminated soils, sediments, sands, and building rubble would be excavated and bulk transported to an existing eastern DOE site (Alternative 7) or to an existing western DOE site (Alternative 10) as described in Section 6.3.1. Building decontamination and water treatment would be conducted as described in Section 6.3.1 and 6.3.2, respectively.

After removing and transporting the contaminated materials to a disposal site, the NFSS would be restored and surveyed for final release.

6.4 PARTIAL DECONTAMINATION SCENARIO

There are two alternatives considered under this scenario: Slurry mining the more radioactive K-65, L-30, L-50, and F-32 residues, heat drying them, and transporting them to an existing

eastern DOE burial site (Alternative 11) or to an existing western DOE burial site (Alternative 12). These alternatives would remove approximately 96 percent of the total radioactivity from the site, leaving only 32 Ci onsite as compared with 891 Ci for either the No Action or Retain and Manage scenario. Alternatives 11 and 12 also include upgrading the waste containment area, followed by 5 years of active maintenance and surveillance and at least 195 years of custodial maintenance and surveillance. This scenario was included as a result of the NEPA public scoping process; no processing or immobilization schemes are addressed.

6.4.1 Alternative 11: Transport Residues East, Upgrade Containment

This alternative involves excavation and removal of only the more radioactive residues from the NFSS. These would be hydraulically mined, dewatered, and heat dried as described in Section 6.2.2. Contaminated water from hydraulic mining activities and vehicle and equipment decontamination would be treated as described in Section 6.2.2. The residues will be loaded and transported as described in Section 6.3.1. Following removal of the residues from the waste containment area, the potential for migration of the remaining wastes will be reduced by upgrading the containment's interim-remedial-action cover and side-wall barriers as required. A detailed description of these activities is provided in Section 6.2.1 (Alternative 2).

If the NFSS is partially decontaminated by removing the more radioactive residues and then upgraded and retained as a waste management site, the following guidelines will govern maintenance and surveillance activities at the site. For a period of 5 years after site closure, the site would be actively monitored, maintained, and surveilled as described in Section 6.1.1. After the 5-year period following site closure, only custodial maintenance and environmental monitoring would be conducted for at least 195 years, as described in Section 6.2.1.

6.4.2 Alternative 12: Transport Residues West, Upgrade Containment

This alternative is identical to Alternative 11 except that the more radioactive residues would be slurry mined, heat dried, and transported to an existing DOE disposal site in the western U.S. The interim-remedial-action cover would be upgraded as described in Section 6.2.1 (Alternative 2). Maintenance and surveillance would be as described for Alternative 11.

6.5 OCCUPATIONAL RADIATION EXPOSURE

6.5.1 Onsite Exposure

The exposure of workers to radiation would occur primarily during handling, processing, and transporting the K-65 residues, and to a lesser extent from these same operations on the other residues and the R-10 spoils pile area. Other wastes on site would not contribute measurably to the overall occupational radiation exposure.

Based on measurements made during the original transfer of the K-65 residues to Building 434, exposure rates on contact with drums containing the K-65 residues may be as high as 300 mR/h with occasional hot spots exceeding 300 mR/h. Exposure rates from the F-32, L-30, and L-50 residues would be about 5 percent of that measured for the K-65 residues based on ratios of radium-226 concentration. An estimate of the upper limit of exposure (12 to 20 percent of that measured for the K-65 residues) for the F-32, L-30, and L-50 residues was obtained using the uranium concentrations in the original ores. Thus, the maximum expected exposure rate from contact with residues other than the K-65 residues is expected to be 15 to 60 mR/h.

Occupational exposure to radon is not considered a primary exposure pathway because radon would be controlled by ventilation and/or directly supplied air, thus reducing the dose to workers.

Processing and handling of residues would be conducted in facilities designed to provide reasonable assurance that radiation exposure rates in normally occupied areas of the NFSS are less than 2.5 mR/h and less than 0.5 mR/h in administrative areas. Typically, radiation doses would be limited to approximately 1 rem per calendar

quarter, with an annual goal of less than 5 rem. By designing process and handling facilities based on exposure rates for the K-65 residues, reasonable assurance is provided that the handling of other residues stored at NFSS would result in exposures of less than 1 mR/h and a dose of less than 1 rem per year.

Methodologies for reducing radiation exposure include minimizing direct contact with residues by using remote handling equipment where possible, providing shielding for operators, and placing maintenance items in remote or low exposure areas. Without these methodologies, radiation exposure rates in occupied areas would be much greater, causing occupational exposures to escalate.

The exposure rates from the K-65 residues and the associated problems in handling these materials while maintaining radiation exposures as low as reasonably achievable (ALARA) dictate that direct contact handling be minimized.

6.5.2 Exposure During Handling of Radioactive Materials and Site Maintenance and Surveillance

The occupational exposure of personnel on site was calculated based on the design exposure rates, conceptual designs, and the manpower requirements developed for cost estimating. These exposures are subject to change based on refinements to the existing conceptual designs and preliminary cost estimates.

Occupational exposures presented in Table 6-2 were calculated for the period of maintenance and surveillance after site closure (200 years) and for onsite handling of radioactive materials.

6.5.3 Exposure During Transport of Residues and Contaminated Soils

The radiation exposures resulting from transportation of residues and contaminated soils now at NFSS to either a dry western U.S. site or a wet eastern U.S. site are presented in Table 6-3. Exposures were calculated in accordance with models and generic transportation data presented in NUREG-0170.

TABLE 6-2

RADIATION EXPOSURE^(a) TO WORKERS
FROM ONSITE HANDLING OF RADIOACTIVE MATERIALS
AND SITE MAINTENANCE AND SURVEILLANCE

Alternative	Exposure in Manrem		Total
	Onsite Handling ^(b)	Maintenance and Surveillance ^(c)	
1	-	8	8
2	1	1	2
3	210	1	211
4	260	1	261
5	110	-	110
6	180	-	180
7	240	-	240
8	110	-	110
9	180	-	180
10	240	-	240
11	100	1	101
12	100	1	101

(a) Calculated based on the estimated duration of each specific task. See Appendix F for schedules.

(b) Includes health physics monitoring during construction and transportation.

(c) Conducted after site closure.

TABLE 6-3
RADIATION EXPOSURE^(a) TO CREW
FROM TRANSPORTATION OF RESIDUES AND CONTAMINATED SOILS
FROM THE NFSS

	In Transit Exposure in Manrem	
	Western U.S. Site	Eastern U.S. Site
K-65 Residues	15	4
L-30, L-50, F-32 Residues	4	1
R-10 Residues/Spoils	0.3	0.1
Contaminated Soils, Sands, Sediments, and Building Rubble	0.7	0.2

(a) Calculated based on the estimated duration of each specific task. See Appendix F for schedules.

If K-65 residues are not immobilized or processed for metal recovery, they would be transported in 55-gal drums with 1,920 drums transported each trip. The transport vehicle would be a shielded railcar designed to limit exposures to 2 mR/h to the crewmen and 10 mR/h 2.0 m (6.6 ft) from the edge of the transport vehicle, as required by 49 CFR 173. The average radiation exposure to each crewman transporting K-65 residue would be 0.3 mrem per trip to a western site 4,120 km (2,560 mi) from the NFSS, and 0.08 mrem per trip to an eastern site 1,175 km (730 mi) from the NFSS. The average radiation exposure to each crewman transporting the F-32, L-30, and L-50 residues would be 0.03 mrem per trip to a western site and 0.08 mrem per trip to an eastern site. Transportation of the R-10 residues and contaminated soils would be by bulk shipment in 91 t (100-ton) gondolas. Radiation exposures to crewmen would be consistent with 10 CFR 20 and are based on calculated exposure rates for each type of residue or contaminated soil.

The radiation exposure to crewmen from the shipment of immobilized (vitrified) residues or processed residues (vitrified slag from the residues after metal recovery) would be within the error of the estimated exposure for transportation of unprocessed residues. Therefore, the exposures presented in Table 6-3 are valid estimates for shipment of the residues in any of the three waste forms: untreated, immobilized, or processed. The basis of this analysis is the weight of material transported, volume reduction and density changes due to immobilization or processing, and regulatory exposure limits.

6.5.4 Comparison of Occupational Radiation Exposure

As summarized in Table 6-4, Alternatives 1 and 2 incur the lowest occupational radiation exposure. The next lowest exposure is incurred in the alternatives involving transportation without residue processing or immobilization (5, 8, 11, and 12), followed by those involving immobilization of the residues (3, 6, and 9). Alternatives involving processing and immobilization of the residues (4, 7, and 10) incur the highest exposure.

TABLE 6-4

COMPARISON OF OCCUPATIONAL RADIATION EXPOSURES^(a)

Alternative	Exposure in Manrem		Total
	On Site ^(b) and Disposal (Excluding In Transit)	Transportation (In Transit)	
1	8	-	8
2	2	-	2
3	211	-	211
4	261	-	261
5	110	6	116
6	180	6	186
7	240	6	246
8	110	20	130
9	180	20	200
10	240	20	260
11	101	6	107
12	101	19	120

(a) Calculated based on the estimated duration of each specific task. See Appendix F for schedules.

(b) Includes health physics monitoring during construction and maintenance and surveillance after site closure.

Transporting the residues from NFSS to another site merely changes the recipient of the long-term radiation exposure but not the magnitude of the exposure, assuming both sites are similar.

6.6 COST ESTIMATES

Tables 6-5 through 6-16 present an order-of-magnitude cost estimate for each of the 12 alternatives leading to a final disposition of the NFSS. The cost estimates have been prepared using standard Bechtel estimates methods for order-of-magnitude conceptual estimates. The estimates have been priced at third quarter 1983 pricing levels with escalation amounts added at an 7.0 percent per annum rate. The escalation rate has been applied to the estimates based on the preliminary schedules. The present value of costs associated with annual maintenance and surveillance after site closure was calculated to permit comparisons of total effective costs for each alternative. In estimating the present value of these costs, a real interest rate of 3 percent was assumed. This is consistent with historical interest and inflation rate information (Business Week, April 25, 1983). Management support has been estimated to be 15 percent of the total construction cost. The cost of management support is included in the surveillance and support costs shown on the tables; this cost should not be directly related to the manpower requirements given in Appendix F. Contingency has been applied to the estimates at approximately a 25 percent level.

6.6.1 Cost Assumptions

No Action Scenario

Alternative 1 -- Maintenance and Surveillance

- (1) The cost of interim remedial action is not included in this alternative.
- (2) Includes facility upkeep, equipment purchase and rental road and fence repairs, fuel, and water management.
- (3) Environmental monitoring includes quarterly air and water sampling for an indefinite period after alternative implementation.

TABLE 6-5

COST OF ALTERNATIVE 1 -- SITE MAINTENANCE AND SURVEILLANCE

Activity	Average Annual Cost (\$)
(1) Cover Repair	35,000
(2) Dredging Ditches	76,000
(3) Dike Repair	14,000
(4) Surveillance and Support	120,000
(5) Utilities	38,000
(6) Equipment Rental and Repair	20,000
(7) Road Repair	32,000
(8) Fence Repair	7,500
(9) Fuel	3,200
(10) Environmental Monitoring	20,000
(11) Contingency	<u>52,300</u>
TOTAL	418,000(a)

- (a) The present value needed to conduct the above listed activities indefinitely is \$13.9 million. This figure is calculated using a \$418,000 average annual cost of maintenance and surveillance and computing the present value of this average annual cost for an infinite period of time based on an assumed real interest rate of 3 percent.

TABLE 6-6

COST OF ALTERNATIVE 2 -- UPGRADE CONTAINMENT

Activity	Total Cost (\$)
(1) Remove Interim Cover	93,000
(2) Relocate Central Drainage Ditch	166,000
(3) Slurry Wall Construction (If Required)	520,000
(4) Final Multi-Layered Cover	2,204,000
(5) Surveillance, Construction, and Management	<u>3,217,000(a)</u>
SUBTOTAL	6,200,000
ESCALATION (7%/yr)	1,395,000
CONTINGENCY (25%)	<u>1,899,000</u>
SUBTOTAL	9,494,000
- 5 YRS MAINTENANCE AND SURVEILLANCE AFTER SITE CLOSURE	1,915,000(b)
- 195 YRS CUSTODIAL MAINTENANCE AND SURVEILLANCE	<u>573,000(c)</u>
TOTAL	11,982,000

(a) Includes engineering design, construction and management support, and radiological surveillance during construction.

(b) This figure is calculated using a \$418,000 annual cost of maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.

(c) This figure is calculated using a \$20,000 annual cost of custodial maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.

TABLE 6-7
COST OF ALTERNATIVE 3 -- IMMOBILIZE RESIDUES,
UPGRADE CONTAINMENT

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	294,000
(2) Hydraulic Mining	771,000
(3) Immobilize Residues	9,971,000
(4) Move and Store Immobilized Residues	2,326,000
(5) Demolish Buildings 413, 414 and Store in Building 411	534,000
(6) Relocate Central Drainage Ditch	166,000
(7) Slurry Wall Construction (If Required)	520,000
(8) Place Contaminated Soils and Clay Cap Cover over Building 411 and Construct Final Multi-Layered Cover	2,672,000
(9) Surveillance, Construction, and Management Support	<u>3,360,000</u> ^(a)
SUBTOTAL	20,614,000
ESCALATION (7%/yr)	10,307,000
CONTINGENCY (25%)	<u>7,730,000</u>
SUBTOTAL	38,651,000
- 5 YRS MAINTENANCE AND SURVEILLANCE AFTER SITE CLOSURE	1,915,000 ^(b)
- 195 YRS CUSTODIAL MAINTENANCE AND SURVEILLANCE	<u>573,000</u> ^(c)
TOTAL	41,139,000

- (a) Includes engineering design, construction and management support, and radiological surveillance during construction.
- (b) This figure is calculated using a \$418,000 annual cost of maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.
- (c) This figure is calculated using a \$20,000 annual cost of custodial maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.

TABLE 6-8

COST OF ALTERNATIVE 4 -- PROCESS RESIDUES,
UPGRADE CONTAINMENT

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	294,000
(2) Hydraulic Mining	771,000
(3) Process and Immobilize Residues	10,745,000
(4) Move and Store Immobilized Residues	2,326,000
(5) Demolish Buildings 413, 414 and Store in Building 411	534,000
(6) Relocate Central Drainage Ditch	166,000
(7) Slurry Wall Construction (If Required)	520,000
(8) Place Contaminated Soils and Clay Over Building 411 and Construct Final Multi-Layered Cover	2,672,000
(9) Surveillance, Construction, and Management Support	<u>3,360,000(a)</u>
SUBTOTAL	21,388,000
ESCALATION (7%/yr)	10,694,000
CONTINGENCY (25%)	<u>8,021,000</u>
SUBTOTAL	40,103,000
- 5 YRS MAINTENANCE AND SURVEILLANCE AFTER SITE CLOSURE	1,915,000(b)
- 195 YRS CUSTODIAL MAINTENANCE AND SURVEILLANCE	<u>573,000(c)</u>
SUBTOTAL	42,591,000
CREDIT FOR METAL RECOVERY	<u>-4,153,000(d)</u>
TOTAL	38,438,000

- (a) Includes engineering design, construction and management support, and radiological surveillance during construction.
- (b) This figure is calculated using a \$418,000 annual cost of maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.
- (c) This figure is calculated using a \$20,000 annual cost of custodial maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.
- (d) The metal values are not escalated. Actual values would depend upon market rates at the time the materials were sold.

TABLE 6-9

COST OF ALTERNATIVE 5 -- REMOVE ALL RADIOACTIVE MATERIALS,
TRANSPORT EAST

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	423,000
(2) Hydraulic Mining	771,000
(3) Demolish Buildings	690,000
(4) Load and Transport (a)	
A. Contaminated Soils and R-10 Residues	23,555,000
B. K-65, L-30, L-50, F-32 Residues	2,554,000
C. Rubble	2,484,000
D. Contaminated Dike Material	3,828,000
(5) Disposal	32,365,000 (b)
(6) Site Restoration	545,000
(7) Surveillance, Construction, and Management Support	22,070,000 (c)
SUBTOTAL	39,785,000
ESCALATION (7%/yr)	44,893,000
CONTINGENCY (25%)	33,670,000
TOTAL	168,348,000

- (a) Includes waste removal, temporary stockpiling where required, and vehicle loading and transport to a burial site in the eastern U.S.
- (b) Disposal fee incurred at the burial site (\$4.50/ft³ at an eastern site).
- (c) Includes engineering design, construction and management support, and radiological surveillance during construction and transport.

TABLE 6-10

**COST OF ALTERNATIVE 6 -- IMMOBILIZE RESIDUES,
TRANSPORT ALL RADIOACTIVE MATERIALS EAST**

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	423,000
(2) Hydraulic Mining	771,000
(3) Immobilization of K-65, L-30, L-50 and F-32 Residues	9,971,000(a)
(4) Demolish Buildings	690,000
(5) Load and Transport(b)	
A. Contaminated Soils and R-10 Residues	23,555,000
B. Immobilized K-65, L-30, L-50 and F-32 Residues	1,635,000
C. Rubble	2,984,000
D. Contaminated Dike Material	3,828,000
(6) Disposal	31,463,000(c)
(7) Site Restoration	545,000
(8) Surveillance, Construction, and Management Support	22,990,000(d)
SUBTOTAL	98,855,000
ESCALATION (7%/yr)	49,428,000
CONTINGENCY (25%)	37,071,000
TOTAL	185,354,000

- (a) Residue immobilization utilizes electric resistance furnaces as described in Appendix B.
- (b) Includes waste removal, temporary stockpiling where required, and vehicle loading and transport to a burial sit in the eastern U.S.
- (c) Disposal fee incurred at the burial site (\$4.50/ft³ at an eastern site).
- (d) Includes engineering design, construction, and management support, and radiological surveillance during construction and transport.

TABLE 6-11

COST OF ALTERNATIVE 7 -- PROCESS RESIDUES,
TRANSPORT ALL RADIOACTIVE MATERIALS EAST

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	423,000
(2) Hydraulic Mining	771,000
(3) Processing and Immobilization of K-65, L-30, L-50 and F-32 Residues	10,745,000(a)
(4) Demolish Buildings	690,000
(5) Load and Transport(b)	
A. Contaminated Soils and R-10 Residues	23,555,000
B. Immobilized K-65, L-30, L-50 and F-32 Residues	1,635,000
C. Rubble	2,984,000
D. Contaminated Dike Material	3,828,000
(6) Disposal	31,463,000(c)
(7) Site Restoration	545,000
(8) Surveillance, Construction, and Management Support	22,990,000(d)
SUBTOTAL	98,939,000
ESCALATION (7%/yr)	49,470,000
CONTINGENCY (25%)	37,102,000
SUBTOTAL	185,511,000
CREDIT FOR METAL RECOVERY	-4,153,000(e)
TOTAL	181,358,000

- (a) Residue processing for resource recovery is described in Appendix C. The immobilized waste product from this operation requires disposal.
- (b) Includes waste removal, temporary stockpiling where required, and vehicle loading and transport to a burial site in the eastern U.S.
- (c) Disposal fee incurred at the burial site (\$4.50/ft³ at an eastern site).
- (d) Includes engineering design, construction and management support, and radiological surveillance during construction and transport.
- (e) The metal values are not escalated. Actual values would depend upon market rates at the time the materials were sold.

TABLE 6-12

COST OF ALTERNATIVE 8 -- REMOVE ALL RADIOACTIVE MATERIALS,
TRANSPORT WEST

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	423,000
(2) Hydraulic Mining	771,000
(3) Demolish Buildings	690,000
(4) Load and Transport(a)	
A. Contaminated Soils and R-10 Residues	45,389,000
B. K-65, L-30, L-50, F-32 Residues	3,959,000
C. Rubble	5,750,000
D. Contaminated Dike Material	8,004,000
(5) Disposal	25,173,000(b)
(6) Site Restoration	545,000
(7) Surveillance, Construction, and Management Support	27,211,000(c)
SUBTOTAL	117,915,000
ESCALATION (7%/yr)	58,958,000
CONTINGENCY (25%)	44,218,000
TOTAL	221,091,000

- (a) Includes wastes removal, temporary stockpiling where required, and vehicle loading and transport to a burial site in the western U.S.
- (b) Disposal fee incurred at the burial site (\$3.50/ft³ at an western site).
- (c) Includes engineering design, construction and management support, and radiological surveillance during construction and transport.

TABLE 6-13

**COST OF ALTERNATIVE 9 -- IMMOBILIZE RESIDUES,
TRANSPORT ALL RADIOACTIVE MATERIALS WEST**

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	423,000
(2) Hydraulic Mining	771,000
(3) Immobilization of K-65, L-30, L-50 and F-32 Residues	9,971,000(a)
(4) Demolish Buildings	690,000
(5) Load and Transport(b)	
A. Contaminated Soils and R-10 Residues	45,389,000
B. Immobilized K-65, L-30, L-50 and F-32 Residues	3,356,000
C. Rubble	5,750,000
D. Contaminated Dike Material	7,866,000
(6) Disposal	24,471,000(c)
(7) Site Restoration	545,000
(8) Surveillance, Construction, and Management Support	29,770,000(d)
SUBTOTAL	129,002,000
ESCALATION (7%/yr)	64,501,000
CONTINGENCY (25%)	48,376,000
TOTAL	241,879,000

- (a) Residue immobilization utilizes electric resistance furnaces as described in Appendix B.
- (b) Includes waste removal, temporary stockpiling where required, and vehicle loading and transport to a burial site in the western U.S.
- (c) Disposal fee incurred at the burial site (\$3.50/ft³ at a western site).
- (d) Includes engineering design, construction and management support, and surveillance during construction and transport.

TABLE 6-14

COST OF ALTERNATIVE 10 -- PROCESS RESIDUES,
TRANSPORT ALL RADIOACTIVE MATERIALS WEST

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	423,000
(2) Hydraulic Mining	771,000
(3) Processing and Immobilization of K-65, L-30, L-50 and F-32 Residues	10,745,000 ^(a)
(4) Demolish Buildings	690,000
(5) Load and Transport ^(b)	
A. Contaminated Soils and R-10 Residues	45,389,000
B. Immobilized K-65, L-30, L-50 and F-32 Residues	3,356,000
C. Rubble	5,750,000
D. Contaminated Dike Material	7,866,000
(6) Disposal	24,471,000 ^(c)
(7) Site Restoration	545,000
(8) Surveillance, Construction, and Management Support	30,002,000 ^(d)
SUBTOTAL	130,008,000
ESCALATION (7%/yr)	65,004,000
CONTINGENCY (25%)	48,753,000
SUBTOTAL	243,765,000
CREDIT FOR METAL RECOVERY	-4,153,000 ^(e)
TOTAL	239,612,000

(a) Residue processing for resource recovery is described in Appendix C. The immobilized waste product from this operation requires disposal.

(b) Includes waste removal, temporary stockpiling where required, and vehicle loading and transport to a burial sit in the eastern U.S.

(c) Disposal fee incurred at the burial site (\$3.50/ft³ at a western site).

(d) Includes engineering design, construction and management support, and radiological surveillance during construction and transport.

(e) Metal values are not escalated. Actual values would depend upon market rates at the time the materials were sold.

TABLE 6-15

COST OF ALTERNATIVE 11 -- TRANSPORT
RESIDUES EAST, UPGRADE CONTAINMENT

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	294,000
(2) Hydraulic Mining	771,000
(3) Load and Transport K-65, L-30, L-50, F-32 Residues	2,554,000(a)
(4) Backfill Buildings 411, 413, 414 and Construct Final Multi-Layered Cover	2,711,000
(5) Disposal	2,564,000(b)
(6) Relocate Central Drainage Ditch	166,000
(7) Slurry Wall Construction (If Required)	520,000
(8) Surveillance, Construction, and Management Support	3,353,000(c)
SUBTOTAL	<u>12,933,000</u>
ESCALATION (7%/yr)	6,467,000
CONTINGENCY (25%)	4,850,000
SUBTOTAL	<u>24,250,000</u>
- 5 YRS MAINTENANCE AND SURVEILLANCE AFTER SITE CLOSURE	1,915,000(d)
- 195 YRS CUSTODIAL MAINTENANCE AND SURVEILLANCE	<u>573,000(e)</u>
TOTAL	<u>26,738,000</u>

- (a) Includes waste removal, temporary stockpiling where required, and vehicle loading and transport to a burial site in the eastern U.S.
- (b) Disposal fee incurred at the burial site (\$4.50/ft³ for an eastern site).
- (c) Includes engineering, design, construction and management support, and radiological support during construction and transport.
- (d) This figure is calculated using a \$418,000 annual cost of maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.
- (e) This figure is calculated using a \$20,000 annual cost of custodial maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.

TABLE 6-16

COST OF ALTERNATIVE 12 -- TRANSPORT RESIDUES
WEST, UPGRADE CONTAINMENT

Activity	Total Cost (\$)
(1) Remove Interim Cover and Overburden Over Residues	294,000
(2) Hydraulic Mining	771,000
(3) Load and Transport K-65, L-30, L-50, F-32 Residues	3,959,000(a)
(4) Backfill Buildings 411,413, 414 and Construct Final Multi-Layered Cover	2,711,000
(5) Disposal	1,994,000(b)
(6) Relocate Central Drainage Ditch	166,000
(7) Slurry Wall Construction (If Required)	520,000
(8) Surveillance, Construction, and Management Support	3,645,000(c)
SUBTOTAL	14,060,000
ESCALATION (7%/yr)	7,030,000
CONTINGENCY (25%)	5,273,000
SUBTOTAL	26,363,000
- 5 YRS MAINTENANCE AND SURVEILLANCE AFTER SITE CLOSURE	1,915,000(d)
- 195 YRS CUSTODIAL MAINTENANCE AND SURVEILLANCE	573,000(e)
TOTAL	28,851,000

(a) Includes waste removal, temporary stockpiling where required, and vehicle loading and transport to a burial site in the western U.S.

(b) Disposal fee incurred at the burial site (\$3.50/ft³ for a western site).

(c) Includes engineering, design, construction and management support, and radiological support during construction and transport.

(d) This figure is calculated using a \$418,000 annual cost of maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.

(e) This figure is calculated using a \$20,000 annual cost of custodial maintenance and surveillance and computing the present value at an assumed real interest rate of 3 percent.

Retain and Manage Scenario

Alternative 2 -- Upgrade Containment

- (1) Same as assumption 1 in Alternative 1.
- (2) Table 6-6 presents the present value of full maintenance and surveillance and custodial maintenance and surveillance. Calculation assumptions are stated in the table notes.

Alternative 3 -- Immobilize, Upgrade Containment

- (1) Same as assumption 1 in Alternative 1.
- (2) Pumps and related equipment are available from hydraulic mining operations during interim remedial action activities.
- (3) Fusion furnace is assumed to have a capacity of 36 t (40 tons) per day, based on 24-hour operation with 8 hours required for charging the furnace and 16 hours required to melt the charged material.
- (4) Hydraulic mining, drying, and immobilization are estimated to require 36 months.
- (5) Same as assumption 2 in Alternative 2.

Alternative 4 -- Process, Upgrade Containment

- (1) Same as assumption 1 in Alternative 1.
- (2) Same as assumption 2 in Alternative 2.
- (3) Same as assumption 2 in Alternative 3.
- (4) Same as assumption 3 in Alternative 3.
- (5) Previous metal value from processing is estimated using values presented in Engineering and Mining Journal (May, 1982). These values were not escalated because the fluctuation in the price of metals cannot be accurately estimated; the values would be expected to escalate.
- (6) Development problems that cannot be fully anticipated at the conceptual design level for the described processing scheme could significantly increase the cost of this alternative.

- (7) Uncertainties regarding recovery efficiency and the lack of a de minimus standard for metals that could contain a small amount of residual radioactivity could affect the costs estimated.

Decontaminate and Release Scenario

Alternatives 5 and 8 -- Transport Materials East or West

- (1) Same as assumption 1 in Alternative 1.
- (2) Same as assumption 2 in Alternative 3.
- (3) K-65 residues are placed in 55-gal drums, loaded into shielded containers at 15 drums per container, and transported on flatcars carrying 2 containers each.
- (4) Remaining residues will be transported in shielded, covered gondolas with a net weight of 82 t (90 tons) each.
- (5) R-10 residues and rubble will be transported in unshielded, covered gondolas with a net weight of 91 t (100 tons) each.
- (6) Transport values assume that 2 dedicated unit trains serving the eastern site and 3 dedicated unit trains serving the western site are available for a period of 4 years with no interruptions (strikes, etc.).

Alternatives 6 and 9 -- Immobilize, Transport Materials East or West

- (1) Same as assumption 1 in Alternative 1.
- (2) Same as assumption 2 in Alternative 3.
- (3) Same as assumption 3 in Alternative 3.
- (4) Molten K-65, F-32, L-30, and L-50 residues are poured into molds, cooled, and transported.
- (5) Immobilized waste is assumed to weigh 431 kg (950 lb) per slab.
- (6) Twenty-four slabs of K-65 residue will be transported per shielded container carrying 2 containers per flatcar; L-50 and L-30 and F-32 slabs will be transported in shielded, covered gondolas with a net weight of 82 t (90 tons) each.

- (7) Remaining soil and rubble will be transported in unshielded, covered gondolas with a net weight of 91 t (100 tons) each.
- (8) Same as assumption 6 in Alternatives 5 and 8.

Alternatives 7 and 10 -- Process, Transport Materials East or West

- (1) Same as assumption 1 in Alternative 1.
- (2) Same as assumption 2 in Alternative 3.
- (3) Same as assumption 3 in Alternative 3.
- (4) Same as assumption 5 in Alternative 4.
- (5) Same as assumption 6 in Alternative 4.
- (6) Same as assumption 7 in Alternative 4.
- (7) Slag will be transported in a shielded container.
- (8) Same as assumption 7 in Alternatives 6 and 9.
- (9) Same as assumption 6 in Alternatives 5 and 8.

Partial Decontamination Scenario

Alternatives 11 and 12 -- Transport Residues East or West, Upgrade Containment

- (1) Same as assumption 1 in Alternative 1.
- (2) Same as assumption 2 in Alternative 2.
- (3) Same as assumption 3 in Alternatives 5 and 8.
- (4) Same as assumption 4 in Alternatives 5 and 8.
- (5) Same as assumption 6 in Alternatives 5 and 8, except that the shipping period is 3 years instead of 4 years.

6.6.2 Comparison of Costs

Alternatives 1 through 4 under the No Action scenario and the Retain and Manage scenario are considerably less expensive than the Decontaminate and Release scenario which involves removal of all of the materials contaminated above criteria levels from the NFSS (Alternatives 5 through 10). Alternative 1 would require full maintenance and surveillance for an indefinite period of time

because the minimum design life of the waste containment area constructed during interim remedial measures is 25 years. Alternatives 2, 3, and 4 would require 5 years of full maintenance and surveillance during the post-closure observation phase and at least 195 additional years of custodial maintenance and surveillance. The present value cost for required maintenance and surveillance is included in the costs based on an assumed real interest rate of 3 percent. Alternatives 1 through 4 are less than about 15 percent of the cost of Alternatives 5 through 10.

Transporting the contaminated material from the NFSS to a DOE facility under the Decontaminate and Release scenario (Alternatives 5 through 10) is considerably more expensive than managing this same material at the site. Total miles and total number of shipments to the eastern and western sites are indicated in Table 6-17. Transporting the material to a DOE facility in the western U.S. (Alternatives 8, 9, and 10) is approximately one and a half times as costly as transporting the material to a DOE facility in the eastern U.S. (Alternatives 5, 6, and 7). This cost difference is due to the fact that the western facility is three times farther than the distance to the eastern facility.

Processing the residues for precious and base metal recovery with subsequent immobilization (Alternative 7) results in a 2 percent cost reduction as compared with immobilizing the residues in a fusion furnace (Alternative 6) before transport to an eastern disposal facility. Processing the residues (Alternative 10) results in a 1 percent cost reduction as compared with immobilizing the same residues in a fusion furnace (Alternative 9) before transport to a western DOE facility. The cost reduction could be greater depending on the future market value of the metals and the actual metal content in the residue. However, processing costs reported in this document do not include potential process development costs, which could significantly increase the cost of alternatives involving processing. In addition, because of the absence of a de minimus standard for radioactivity in the materials, the marketability of the metals is uncertain.

TABLE 6-17

**TRANSPORTATION REQUIREMENTS FOR THE
NFSS DISPOSITION ALTERNATIVES**

Alternative No. (a)	Transport Miles Required (x 1000)	No. of Train Loads (54 Cars)	Transport Costs (\$ x 1000) (b)
1	None	None	None
2	None	None	None
3	None	None	None
4	None	None	None
5	184,248	72	72,838
6	179,130	70	70,804
7	179,130	70	70,804
8	52,776	72	118,316
9	51,310	70	116,927
10	51,310	70	116,927
11	10,263	14	4,789
12	35,826	14	7,423

(a) See Section 1.3 for description of alternatives.

(b) These values include an escalation rate of 7.0 percent during transport and a 25 percent contingency factor. These costs, therefore, cannot be directly compared with the individual transport costs in Tables 6-9 through 6-16, where the escalation and contingency are applied after totaling the cost for all activities.

Alternatives 11 and 12 for partial decontamination are twice as costly as no action (Alternative 1), but much less costly than the alternatives under the Decontaminate and Release scenario (Alternatives 5-10). The waste containment area would be upgraded under Alternatives 11 and 12, but only the cost for excavating, shipping, and disposing of the residues would be added.

6.7 SCHEDULES AND MANPOWER REQUIREMENTS

A summary of schedule and manpower requirements for each of the 12 alternatives is given in Figure 6-8. It is expected that Alternative 1 would require maintenance and surveillance for an indefinite period of time. Detailed schedules and manpower requirements are given in Appendix F. Alternatives 3 and 4 show construction being completed about 9 years after the start of design engineering. Five additional years of full surveillance and maintenance are assumed for these two alternatives although custodial maintenance of the site would continue beyond that time period. Alternatives 5 through 10 require transport to a DOE disposal site and considerably more onsite construction activities than Alternatives 1 through 4. Construction and transport for Alternatives 5 through 10 are complete about 8 years after the start of design engineering. Total manhour requirements for these six alternatives range from approximately 0.75 million to 1.23 million.

Alternatives 11 and 12 require about 9 years for transport, disposal of the residues, and upgrading the waste containment area. Man-hour requirements are slightly less than for total decontamination of the site.

6.8 SUMMARY OF ALTERNATIVE COMPARISONS

The assessment of the 12 alternatives consisted of comparing the alternatives based on engineering design criteria and design life, occupational radiation exposure, scheduling, manpower requirements, maintenance and surveillance requirements, and cost. A comparison of the 12 alternatives using these parameters is presented in Sections 6.5.4, 6.6.2, and 6.7 and is summarized in Table 6-18.

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a		YEAR															TOTAL M'HRS.	
ALTERNATIVE		1	2	3	4	5	6	7	8	9	10	11	12	13	14	15		
1	▼	b															10,000/YR	
2	▼				c	▼			d			▼	d			f	466,200	
3	▼				c	▼									▼	f	715,400	
4	▼				c	▼						d			▼	f	758,300	
5	▼				e	▼												750,600
6	▼				e	▼												852,600
7	▼				e	▼												937,600
8	▼				e	▼												1,111,500
9	▼				e	▼												1,190,500
10	▼				e	▼												1,233,500
11	▼				g	▼			h	▼	d			▼			f	746,400
12	▼				g	▼			h	▼	d			▼			f	901,800

Notes: a. Refer to Section 5.6 for description of alternative.
 b. Maintenance and surveillance for an indefinite period of time.
 c. Includes engineering design, engineering support, and construction.
 d. Maintenance and surveillance will continue for a 5-year period following construction.

e. Includes items in c above, plus transportation and disposal.
 f. Custodial maintenance and surveillance for a period of 195 years.
 g. Includes items in c above, plus excavation and shipping of K-65, L-30, L-50, and F-32 residues.
 h. Includes completion of upgraded containment and completion of construction.

FIGURE 6-8 SCHEDULE AND MANPOWER REQUIREMENTS SUMMARY

TABLE 6-18

COMPARISON OF ALTERNATIVES FOR THE FINAL
DISPOSITION OF NFSS RESIDUES AND WASTES

Alternative Number ^(a)	Minimum Design Life ^(b) (Yrs)	Occupational Radiation Exposure ^(c) (Manrem)		Construction Schedule (Yrs)	Manpower Requirements (Manhours)	Maintenance and Surveillance (Yrs)	Costs (\$ x 1000)
		On Site or Disposal (Excluding In Transit)	Transport (In Transit)				
1	25	8	NA	0	10,000/yr	Continuing	13,900 ^(d)
2	200	2	NA	5.5	466,200	200	12,000 ^(e)
3	200	211	NA	9	715,400	200	41,100 ^(e)
4	200	261	NA	9	758,300	200	38,400 ^(e)
5	NA	110	6	8	750,600	NA	168,300
6	NA	180	6	8	852,600	NA	185,400
7	NA	240	6	8	937,600	NA	181,400
8	NA	110	20	8	1,111,500	NA	221,100
9	NA	180	20	8	1,190,500	NA	241,900
10	NA	240	20	8	1,233,500	NA	239,600
11	200	101	6	10	746,400	200	26,700 ^(e)
12	200	101	19	10	901,800	200	28,900 ^(e)

(a) See Section 1.3 for description of alternatives.

(b) Design life values indicated are minimums that must be achieved.

(c) Calculated based on the estimated duration of each specific task. See Appendix F for schedules.

(d) This present value cost is needed to provide \$418,000/yr for surveillance and maintenance based on an assumed real interest rate of 3 percent.

(e) Alternatives 2, 3, 4, 11, and 12 include 5 years of full maintenance and surveillance and at least 195 years of custodial maintenance and surveillance. The present value cost of maintenance and surveillance for the 200-year period is included in the reported costs based on an average cost of \$418,000 for full maintenance and surveillance and \$20,000 for custodial maintenance and surveillance.

6.8.1 Cost Comparison

When compared on the basis of cost, the least expensive is Alternative 2, with Alternative 1 only marginally higher. Next lowest in cost are Alternatives 11 and 12, costing roughly twice as much as Alternatives 1 and 2. Alternatives 3 and 4 are more expensive than 11 and 12 by an average of approximately 43 percent. When compared with Alternatives 1 and 2, Alternatives 3 and 4 cost roughly three times as much. All of the alternatives (5-10) that allow for total decommissioning of the NFSS are very expensive compared with the six alternatives (1, 2, 3, 4, 11, and 12) that require continued maintenance of NFSS as a storage site. When compared with the two lowest cost alternatives (1 and 2), the decommissioning alternatives cost approximately 15 times as much.

Immobilization of the residues (K-65, F-32, L-30, and L-50) is a relatively expensive process, adding an average of approximately \$22 million as compared with similar alternatives from which immobilization is omitted (i.e. 3 vs. 2, 6 vs. 5, or 9 vs. 8). Recovery of valuable materials during the immobilization process can reduce the cost of immobilization by an average of \$3.0 million as observed by comparisons between alternatives incorporating immobilization with and without material recovery (i.e. 3 vs. 4, 6 vs. 7, and 9 vs. 10).

Transportation of materials to a disposal site is much more expensive for the western disposal site than for the eastern disposal site. This is due to the greater distance to the western site. When relatively small volumes of material are being transported (Alternatives 11 and 12), the cost difference is only about \$2 million. When large volumes of material are being transported (Alternatives 5 vs. 8, 6 vs. 9, and 7 vs. 10) the cost difference is an average of approximately \$56 million.

6.8.2 Occupational Radiation Exposure Comparison

Alternative 2 has the lowest total manrem exposure with a total calculated dose of only 2 manrem. Alternative 1 has the next lowest calculated exposure at 8 manrem. All other alternatives (3-12)

involve excavation and handling or processing of the radioactive materials that have been buried as a result of the interim remedial measures previously undertaken. Because of these activities, the calculated exposure of onsite operating personnel is much larger than the exposures calculated for either Alternatives 1 or 2. Onsite exposures calculated for Alternatives 3-12 vary from 101 to 261 manrem, with the highest exposures associated with those alternatives requiring more extensive handling and processing.

For those alternatives that require transportation of residues and/or other wastes off site, radiation exposures to the crewmen were calculated. In transit exposures associated with the transportation of the radioactive material to a western site were calculated to be approximately three times as high as those associated with transportation to an eastern site. In either case, the calculated exposures directly associated with transportation off site are relatively small compared with the exposures calculated for onsite excavation, handling, and processing.

Intransit radiation exposures to crewmen were calculated in the expectation that radiation levels will not exceed the limits specified by 49 CFR 173. None of the calculated intransit exposures exceed the dose standards for individuals specified in 10 CFR 20.

6.8.3 Construction Schedule Comparison

Alternative 1 has no construction schedule associated with it since no action beyond the interim remedial action is planned other than maintenance and surveillance. Alternative 2 takes only 5-1/2 years to complete since the amount of work is relatively small. All of the other alternatives take 8 to 10 years to complete.

6.8.4 Selection of Final Disposition

Selection of a final disposition for the NFSS would involve an evaluation of the material presented in this document and other reports cited in the references. The relationship of this information to potential environmental impacts at the NFSS will be addressed in the EIS currently being prepared by ANL.

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APPENDIX A

**CHARACTERISTICS OF
RESIDUES AND WASTES
AT NFSS**

CHARACTERISTICS OF RESIDUES AND WASTES AT NFSS

Table A-1 gives the characteristics of the K-65 residues. The characteristics of the L-30, L-50, and R-10 residues are given in Table A-2, A-3, and A-4 respectively. A comparison of the mean stable element concentrations in soils from the R-10 spoils pile area to mean naturally occurring levels is given in Table A-5.

The results of carbonate leach tests recently performed (Battelle, 1983b) are presented in Tables A-6 through A-10. Information on the chemistry of the F-32 residues is included in these tables.

TABLE A-1
GENERAL CHARACTERISTICS OF THE K-65 RESIDUES

Characteristic	BCL(a)	NLO(b)
Dry wt, kg (tons)	-	3.5×10^6 (3,850)
Estimated volume, m ³ (ft ³)	-	3,080 (108,750)
Density, kg/m ³ (lb/ft ³)	-	1,179 (73)
Uranium, ppm	500(c) 18,240(d) 30,000(e) 1,410-1,961(f)	500

(a) Reported by Battelle Columbus Laboratories, 1981.

(b) Reported by National Lead of Ohio, 1980.

(c) Direct gamma spectroscopy of the residue.

(d) X-ray diffraction of the residue.

(e) Spark source mass spectroscopy.

(f) Calculated uranium from radium measurements.

Source: Battelle, 1981.

TABLE A-2

CHARACTERISTICS OF THE L-30 RESIDUES

Characteristic	BCL (a)	NLO (a)	Vitro (b)	Litz (c)	BNI (e)
Dry wt, kg (tons)	-	7.5×10^6 (8,227)	7.9×10^6 (8,740)	-	
Estimated Volume, m ³ (ft ³)	-	6,020 (215,000)	-	-	
Density, kg/m ³ (lb/ft ³)	-	842.6 (52.6)	-	-	
Uranium, ppm	5,000	1,800	1,950	830	
Lead, ppm	7,500 - 23,500	-	-	7,600	
Radium, ppb	12 ^(d)	8	-	2	30
Barium, ppm	10,000 - 20,000	-	-	1,900	
Iron, ppm	10,000 - 20,000	-	-	66,000	
Gold, ppm	0.2	-	-	0.7	
Platinum, ppm	0.5	-	-	0.2	
Palladium, ppm	2	-	-	6.2	
Silver, ppm	2	-	-	6.2	
Copper, ppm	1,500 - 5,000	2,000	-	1,100	
Cobalt, ppm	5,000 - 10,000	6,200	-	2,600	
Nickel, ppm	30,000 - 50,000	20,000	-	6,200	

(a) NLO, Inc., and Battelle Columbus Laboratories, 1980.

(b) Vitro Corp., 1952.

(c) Litz, 1974.

(d) Sample of residues stored in west bay only.

(e) Bechtel National, Inc. sample of residues stored in east bay only.

Source: Battelle, 1981.

TABLE A-3
CHARACTERISTICS OF THE L-50 RESIDUES

Characteristic	BCL(a)	NLO(b)
Dry wt, kg (tons)	-	1.7×10^6 (1,878)
Volume	-	1,624 (58,000)
Density, kg/m ³ (lb/ft ³)		1,265 (79)
Uranium, ppm	1,000 - 2,100	1,200 - 1,300
Lead, ppm	7,000	7,600
Radium, ppb	8 - 12	7.8 - 9.3
Barium, ppm	20,000	-
Iron, ppm	20,000	-
Gold, ppm	0.2	-
Platinum, ppm	0.5	-
Palladium, ppm	2 - 3	-
Silver, ppm	0.5	-
Copper, ppm	2,000 - 3,000	2,400
Cobalt, ppm	10,000	5,900
Nickel, ppm	20,000 - 30,000	19,100

(a) Battelle Columbus Laboratories, 1980.

(b) NLO, Inc., 1980.

Source: Battelle, 1981.

TABLE A-4

SUMMARY OF RADIOLOGICAL AND NONRADIOLOGICAL
CHARACTERISTICS OF THE R-10 RESIDUE STORAGE
AND SPOIL PILE AREA^(a)

Characteristic	Range of Values
Radiological	
Gamma (1 m), $\mu\text{R/h}$	13-7000
Beta-Gamma (1 cm), mrad/h	0.06-4
Surface Radium-226, pCi/g	4-9400
Total Uranium, mg/g	1-145
Radon	
Concentration 1.5 m Above Soil Surface, pCi/l	2-440 (b)
At Soil Surface, pCi/l	15-12 (b)
Nonradiological	
Lead, ppm	3-650
Arsenic, ppm	0.5-5
Chromium, ppm	20-30
Cerium, ppm	5-100
Fluorine, ppm	3-100
Strontium, ppm	50-200
Barium, ppm	100-500
Zirconium, ppm	10-1000
Copper, ppm	20-3000
Nickel, ppm	20-5000
Cobalt, ppm	50-5000
Vanadium, ppm	30-1000
Titanium, ppm	1000-3000
Bioconcentrations	
Radium-226 in Vegetation, pCi/g	5.4

(a) Includes contaminated subsurface soil and spoil pile.

(b) Measurements were taken by Battelle (1981) before additional spoils from decontamination activities were placed on the pile in 1982.

TABLE A-5

COMPARISON OF MEAN STABLE ELEMENT CONCENTRATIONS IN SOILS FROM
THE R-10 SPOIL PILE AREA TO MEAN NATURALLY OCCURRING LEVELS

Element	ppm	
	Naturally Occurring(a)	On Site (Standard Error)
Sodium	6,300	3,786 (432)
Titanium	5,000	1,484 (217)
Vanadium	100	196 (70)
Cobalt	8	1,030 (400)
Nickel	0	1,634 (538)
Copper	20	449 (210)
Manganese	850	636 (133)
Zirconium	300	154 (71)
Barium	500	315 (48)
Lithium	30	196 (26)
Strontium	300	95 (13)
Flourine	200	26 (9)
Lanthanum	30	24 (9)
Cerium	50	35 (10)
Chromium	100	26 (3)
Arsenic	6	2 (0.4)
Lead	10	321 (74)

(a) Bowen, 1966

Source: Battelle, 1981.

TABLE A-6

pH, FILTRATE VOLUME, WASH WATER VOLUME, SAMPLE WEIGHT,
AND LEACHED RESIDUE WET AND DRY WEIGHT DATA ON 12
LEACH EXPERIMENTS

Sample	Initial pH	Final pH	Filtrate Volume (ml)	Wash Water Volume (ml)	Sample Weight (g)	Leached Residue	
						Wet Weight (g)	Dry Weight (g)
L-30-1	10.16	10.36	950	1850	1001.0	1019.8	809.2
L-30-2	10.20	10.37	995	1860	1004.8	972.0	819.2
L-30-3	10.23	10.47	1005	2365	999.5	936.6	768.3
F-32-1	10.08	10.08	1030	1995	1020.8	653.5	351.0
F-32-2	10.03	10.00	1135	1865	1001.9	553.8	200.8
F-32-3	9.93	10.12	1290	1765	1012.9	641.0	288.1
L-50-1	10.5	10.5	915	1590	1000.0	777.6	647.9
L-50-2	10.5	10.5	850	2365	1000.0	943.8	625.0
L-50-3	10.5	10.5	865	2000	1000.0	886.4	622.8
K-65-1	9.63	10.1	1085	1410	1000.5	1100.0	668.4
K-65-2	9.7	10.1	1065	1355	1000.5	1105.7	645.2
K-65-3	9.6	10.1	1240	1480	1001.5	1123.4	647.3

Source: Battelle, 1983b.

TABLE A-7

BICARBONATE AND CARBONATE CONCENTRATIONS IN FILTRATES
AT CONCLUSION OF LEACH TEST COMPARED TO CONCENTRATIONS
ADDED AS SODIUM SALTS AT THE OUTSET OF THE LEACH TESTS

Sample	CO ₃ ²⁻ Added as Na ₂ CO ₃ (mg/l)	CO ₃ ²⁻ Present at end of leach (mg/l)	HCO ₃ ⁻ Added as NaHCO ₃ (mg/l)	HCO ₃ ⁻ Present at end of leach (mg/l)
L-30-1	22,620	21,240	7,259	4,760
L-30-2	22,620	19,440	7,259	5,120
L-30-3	22,620	7,800	7,259	1,830
F-32-1	22,620	8,580	7,259	1,160
F-32-2	22,620	9,600	7,259	3,600
F-32-3	22,620	8,400	7,259	1,710
L-50-1	22,620	28,620	7,259	5,500
L-50-2	22,620	33,000	7,259	-0-
L-50-3	22,620	33,000	7,259	5,610
K-65-1	22,620	13,320	7,259	2,740
K-65-2	22,620	14,040	7,259	1,520
K-65-3	22,620	11,100	7,259	1,950

Source: Battelle, 1983b.

TABLE A-8

Na⁺ CONCENTRATIONS IN FILTRATES AT CONCLUSION OF LEACH
TEST COMPARED TO Na⁺ CONCENTRATION ADDED AS Na₂CO₃ AND
NaHCO₃ AT OUTSET ALONG WITH SO₄²⁻ AND TOTAL U
CONCENTRATIONS IN FILTRATES

Sample	Na ⁺ (mg/l)	Na ⁺ Added as Na ₂ CO ₃ and NaHCO ₃ (mg/l)	SO ₄ ²⁻ (mg/l)	Total Uranium (mg/l)
L-30-1	15,600	20,095	1,330	8,277
L-30-2	14,400	20,095	1,250	7,977
L-30-3	5,700	20,095	440	3,678
F-32-1	5,600	20,095	480	1,669
F-32-2	7,300	20,095	940	2,112
F-32-3	5,100	20,095	830	2,333
L-50-1	21,300	20,095	2,830	6,763
L-50-2	23,100	20,095	3,170	6,368
L-50-3	25,500	20,095	3,260	7,505
K-65-1	10,800	20,095	2,200	3,111
K-65-2	10,700	20,095	2,260	2,474
K-65-3	8,800	20,095	1,790	2,457

Source: Battelle, 1983b.

TABLE A-9

RESULTS OF INDUCTIVELY COUPLED PLASMA-ATOMIC EMISSION
SPECTROSCOPY ANALYSES ON FILTRATES
(mg/l)

1 of 2

Sample	Ag	Al	As	B	Ba	Be	Ca	Cd	Co	Cr	Cu	Fe	Mg
L-30-1	<0.05	3.81	2.28	2.47	1.69	<0.005	2.30	<0.025	<0.025	0.44	<0.05	0.30	<0.1
L-30-2	0.068	9.10	4.29	4.67	0.95	<0.005	4.76	<0.025	<0.025	0.69	0.57	1.13	<0.1
L-30-3	<0.05	5.88	2.20	2.95	1.54	<0.005	4.66	<0.025	<0.025	0.52	0.24	0.96	<0.1
F-32-1	<0.05	3.65	3.87	2.11	2.34	<0.005	7.46	<0.025	0.97	0.13	4.48	8.36	<0.1
F-32-2	0.055	8.44	6.01	2.86	4.85	<0.005	14.80	<0.025	2.38	0.17	4.55	37.0	<0.1
F-32-3	0.070	5.26	4.52	1.97	2.55	<0.005	7.28	<0.025	1.41	0.15	5.40	11.0	<0.1
L-50-1	<0.050	7.34	2.30	7.93	1.08	<0.005	4.78	<0.025	<0.025	2.00	0.42	0.99	<0.1
L-50-2	<0.050	8.02	2.68	3.87	1.11	<0.005	5.12	<0.025	<0.025	2.08	0.42	0.99	<0.1
L-50-3	<0.050	7.39	2.74	1.66	1.10	<0.005	5.30	<0.025	<0.025	2.14	0.38	1.03	<0.1
K-65-1	<0.050	8.86	2.18	1.02	1.60	0.053	3.16	<0.025	<0.025	<0.05	0.68	0.62	<0.1
K-65-2	<0.050	8.60	2.46	0.91	1.62	0.053	2.95	<0.025	<0.025	<0.05	0.70	0.60	<0.1
K-65-3	<0.050	6.98	1.99	0.99	1.89	0.044	3.29	<0.025	<0.025	<0.05	0.57	0.74	<0.1

TABLE A-9
(Continued)

2 of 2

Sample	Mn	Mo	Ni	P	Pb	Sb	Se	Sn	Sr	Tl	V	Y	Zn
L-30-1	<0.05	85.0	<0.25	11.3	15.0	<0.5	7.30	<0.3	0.32	<0.5	<0.025	<0.025	0.28
L-30-2	0.15	120.0	<0.25	16.0	22.9	<0.5	10.3	<0.3	0.053	<0.5	<0.025	0.042	0.12
L-30-3	0.09	40.0	<0.25	7.68	2.78	<0.5	5.82	<0.3	0.076	<0.5	0.24	<0.025	0.13
F-32-1	0.35	85.0	4.13	49.1	112.0	<0.5	1.24	<0.3	0.078	<0.5	2.40	0.028	2.78
F-32-2	1.40	60.0	6.94	71.7	157.0	<0.5	1.98	<0.3	0.160	<0.5	1.69	0.060	12.7
F-32-3	0.50	60.0	4.90	66.1	146.0	<0.5	1.63	<0.3	0.082	<0.5	2.38	0.038	5.25
L-50-1	0.09	187.5	<0.25	32.0	43.3	<0.5	12.2	<0.3	0.036	<0.5	<0.025	0.032	<0.1
L-50-2	0.09	200.0	<0.25	35.8	48.6	<0.5	13.1	<0.3	0.042	<0.5	<0.025	0.036	<0.1
L-50-3	0.09	212.5	<0.25	33.8	53.2	<0.5	11.8	<0.3	0.040	<0.5	<0.025	0.037	0.11
K-65-1	<0.05	1,850	<0.25	7.28	26.6	1.63	30.6	<0.3	0.220	<0.5	<0.025	<0.025	2.08
K-65-2	<0.05	1,875	<0.25	20.6	32.9	1.70	30.8	<0.3	0.190	<0.5	<0.025	<0.025	0.64
K-65-3	<0.05	1,525	<0.25	5.40	37.7	1.29	26.1	<0.3	0.270	<0.5	<0.025	<0.025	0.63

Source: Battelle, 1983b.

TABLE A-10

SPARK-SOURCE MASS SPECTROMETRY ANALYSES
OF LEACHED RESIDUE SAMPLES (PPM OR
PERCENT WHERE NOTED)

1 of 4

Element	L30-1	L30-2	L30-3	F32-1	F32-2	F32-3	L50-1	L50-2	L50-3	K65-1	K65-2	K65-3
Lithium	40	100	40	60	100	40	60	200	40	200	100	100
Beryllium	0.7	0.3	0.3	0.3	0.2	0.2	0.7	0.7	0.7	0.3	0.5	0.3
Boron	100	30	30	20	20	20	100	30	30	200	600	600
Flourine	100	100	100	20	20	20	200	200	100	70	70	70
Sodium	6000	6000	3000	3%	3%	3%	6%	1.5	6%	3%	6%	3%
Magnesium	5%	5%	3%	2%	1.5%	1.5%	5%	2%	5%	2%	4%	2%
Aluminum	10%	6%	6%	5%	5%	3%	10%	6%	10%	10%	10%	10%
Silicon	30%	30%	30%	30%	30%	30%	30%	30%	30%	30%	30%	30%
Phosphorus	400	200	400	600	600	600	400	200	400	1000	4000	2000
Sulphur	400	60	150	1%	1%	3000	600	150	600	6000	1%	6000
Chlorine	50	20	20	70	50	70	20	20	20	30	7	5
Potassium	3000	1%	2%	2000	3000	1500	6000	1%	-	1.5%	1.5%	2%
Calcium	2000	1500	2000	6000	6000	3000	1500	3000	6000	1500	2000	2000
Scandium	30	15	30	30	15	30	50	50	30	30	30	30
Titanium	7000	2000	4000	2000	3000	2000	7000	6000	6000	3%	4%	4%
Vanadium	1000	600	600	600	600	200	2000	1000	2000	1000	2000	2000

TABLE A-10
(Continued)

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Element	L30-1	L30-2	L30-3	F32-1	F32-2	F32-3	L50-1	L50-2	L50-3	K65-1	K65-2	K65-3
Chromium	60	60	40	60	40	100	100	100	100	100	200	60
Manganese	6%	4%	10%	6000	6000	2000	4%	6%	6%	200	200	100
Iron	3%	1%	3%	2%	2%	1%	1%	2%	2%	1%	2%	2%
Cobalt	2000	2000	2000	2000	600	4000	4000	4000	6000	1000	4000	1000
Nickel	1%	1%	2%	4000	5000	4000	2%	2%	4%	2000	4000	5000
Copper	1500	1500	1500	3000	3000	1500	600	2000	1500	400	400	600
Zinc	200	70	200	2000	2000	2000	400	400	400	70	40	40
Gallium	6	4	6	4	4	4	10	6	4	40	20	6
Germanium	<10	<10	<10	<10	<10	<10	<10	<10	<10	<10	<10	<10
Arsenic	10	6	40	60	60	60	10	20	20	15	6	2
Selenium	20	4	10	70	70	70	30	20	20	100	100	70
Bromine	<1	<1	<1	<1	<4	<1	<1	<1	<1	<1	<1	4
Rubidium	10	30	30	10	10	5	10	20	20	70	50	30
Strontium	30	50	200	70	70	50	70	70	70	500	700	500
Yttrium	10	15	15	40	40	20	15	15	40	60	100	60
Zirconium	150	60	200	100	100	100	150	200	200	1000	1500	2000
Niobium	10	6	10	4	4	10	10	15	10	60	60	40
Molybdenum	40	20	60	200	200	200	150	60	150	1%	2%	1%
Ruthenium	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6
Rhodium	<6	<6	<6	<6	<6	<6	<6	<6	<6	<10	<6	<6

TABLE A-10
(Continued)

3 of 4

Element	L30-1	L30-2	L30-3	F32-1	F32-2	F32-3	L50-1	L50-2	L50-3	K65-1	K65-2	K65-3
Palladium	0.4	0.2	2	6	6	6	1.5	4	2	20	40	20
Silver	<0.2	<0.4	<0.4	6	6	6	<0.2	<0.4	<0.4	<0.4	<0.2	<0.4
Cadmium	<3	<3	<3	<3	<4	<4	<3	<3	<3	<4	<20	<3
Indium	<0.2	<0.4	<0.4	<0.2	<0.2	<0.2	<0.2	<0.4	<0.2	<0.2	<0.2	<0.2
Tin	150	400	200	100	100	100	600	200	40	20	15	10
Antimony	2	4	10	6	10	6	1	4	0.6	10	10	10
Tellurium	6	6	6	4	4	4	20	15	20	20	20	20
Iodine	0.6	0.3	0.6	1	1.5	1	0.3	0.3	0.2	1	1	0.6
Cesium	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1	<0.1
Barium	4000	1500	2000	1500	1500	1500	6000	1%	1.5%	4%	10%	6%
Lanthanum	150	150	200	200	100	150	150	150	150	4000	5000	2000
Cerium	150	150	150	200	100	150	150	200	150	5000	5000	4000
Praseodymium	15	60	150	40	15	40	20	100	40	400	1500	1000
Neodymium	100	200	400	100	60	200	100	400	60	1000	2000	2000
Samarium	10	15	40	20	20	60	20	20	10	200	400	200
Europium	6	10	6	10	10	10	10	10	10	100	200	60
Gadolinium	6	10	20	10	10	20	10	20	6	100	200	60
Terbium	2	1	2	2	2	2	6	2	1.5	20	60	10
Dysprosium	6	10	10	20	20	20	20	20	6	100	200	60
Holmium	0.6	0.2	0.6	1	2	1	2	1	0.2	6	10	2
Erbium	0.6	1	0.6	4	4	6	0.6	0.6	1	10	6	4

TABLE A-10
(Continued)

4 of 4

Element	L30-1	L30-2	L30-3	F32-1	F32-2	F32-3	L50-1	L50-2	L50-3	K65-1	K65-2	K65-3
Thulium	<0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.6	1	0.4
Ytterbium	<0.6	<0.6	0.6	0.6	2	0.6	1	0.6	<0.6	4	10	4
Lutetium	<0.2	<0.2	0.2	<0.2	0.2	0.2	0.2	0.2	<0.2	1	0.6	1
Hafnium	<3	<3	<3	<3	<3	<3	<3	<3	<3	<10	<20	<10
Tantalum	1	1	1	1	4	1	1	1.5	1	4	<3	3
Tungsten	60	20	60	0.6	0.6	1	10	1	0.6	10	10	10
Rhenium	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4
Osmium	<0.6	<0.4	<0.4	<0.6	<0.6	<0.6	<0.6	<0.6	<0.4	<0.6	<0.6	<0.4
Iridium	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4	<0.4
Platinum	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6	<0.6
Gold	<0.2	<0.2	<0.2	0.2	0.4	<0.2	<0.2	0.4	0.2	15	6	6
Mercury	<6	<6	<6	<6	<6	<6	<6	<6	<6	<6	<6	<6
Thallium	<0.4	<0.4	<0.4	<1	<1	<1	<0.4	<0.4	<0.4	<1	<0.4	<1.5
Lead(a)	1800	3000	4000	7%	7%	4%	7300	3000	4000	11%	71%	30%
Bismuth	0.6	0.2	0.6	15	15	15	0.2	0.2	0.2	0.2	0.2	0.2
Thorium	6	1	2	4	10	4	4	1	0.6	40	60	40
Uranium	100	200	200	2000	3000	2000	400	200	300	1000	2000	600
Lead-204	10	7	10	7	3	4	0.4	0.4	0.4	4	7	7
Lead-206	1000	3000	4000	7%	7%	4%	7000	3000	4000	10%	70%	30%
Lead-207	400	100	300	3000	1000	1000	300	400	300	7000	1.5%	4000
Lead-208	400	400	400	300	300	200	7	10	4	200	300	100

(a) Isotope distribution not normal.

Source: Battelle, 1981.

APPENDIX B

IMMOBILIZATION OF RADIOACTIVE WASTE

IMMOBILIZATION OF RADIOACTIVE WASTE

The purpose of waste immobilization is to alter the physical and chemical properties of the residues in a manner that will reduce the potential for the dispersal of the radiological or chemical constituents of the waste.

A number of techniques have been evaluated for the immobilization of the K-65, F-32, L-30, and L-50 residues. All have been assessed in the context of site remedial engineering designs that are strictly conceptual. Detailed engineering of the preferred NFSS disposition alternative may necessitate modifications to the immobilization process proposed, or re-evaluation of its overall practicability.

Two of the techniques evaluated include binding processes: One using a cationic asphaltic emulsion, and another using concrete. In both processes, the residues would be blended with a medium that coats the fines and fills the voids. Another process, electric furnace fusion, would fuse the residues into a molten mass that would be poured into molds, disposal containers, or other appropriate locations and allowed to solidify.

Using the emulsion process, the residues would be blended with an asphaltic medium similar to that used in highway construction. The emulsion would adhere to the residues and, due to its hydrophobic properties, expel a large portion of the moisture contained within the residue. Preliminary estimates indicate that the immobilized waste product would contain 40 percent asphalt, 56.5 percent residue, and 3.5 percent water by weight. Use of the emulsion process, however, would not retain the waste over the long term in its altered form due to the breakdown of the asphalt caused by long-term exposure to radiation. For this reason, this technology was considered inadequate for the pitchblende residues stored at the NFSS.

Using the cementation process, the residues would be mixed with a masonry cement containing equal volumes of portland cement and calcium hydroxide. The resulting waste product would be formed into a free-standing, nonfriable form free of water. This solid form would decrease the potential for physical dispersal, but the uranium and radium leachability would be only slightly improved over the unsolidified residue. This technique would increase the waste bulk, would involve high processing radiation risks at a high cost, and would not produce a waste form with a significant reduction in migration potential. For these reasons, this process was eliminated from further consideration.

Immobilizing radioactive wastes by fusion would be accomplished by heating the material to its melting point and pouring the viscous liquid into molds or trenches for cooling. The immobilized waste form would have a number of advantages over non-immobilized waste forms including:

- o Volume reduction
- o Weight reduction
- o Reduced rates of migration, dispersion, and leaching
- o Reduced radon emanation to within regulatory limits
- o Inert waste form with respect to chemical and biological activity
- o Cooled liquid waste form which, like glass, would not fracture as a crystalline matrix.

A full-scale waste immobilization system has not been developed, although electric furnace fusion technology has been successfully used for the vitrification of high-level radioactive waste and for numerous nonradiological processes. The equipment needed for electric fusion furnace operation is available commercially, and the process has been shown to be technically feasible. For these reasons, the electric furnace fusion technique is used in this document as an example of an appropriate NFSS waste immobilization technique. The process is described below.

Heat would be generated in a fusion furnace by passing an electric current between two molybdenum electrodes. The waste material, located between the electrodes, would be resistant to the electric current flow. When an electric current is impeded, a portion of the electrical energy, related to the magnitude of impedance, is transformed into heat energy. By manipulating the amount of electric current, voltage, and waste quantity between the electrodes, sufficient heat would be generated to melt the waste [1300 to 1500°C (2,372 to 2,732°F)]. At these temperatures, all free moisture, waters of crystallization, and gases would be driven off. The molten waste could then be poured into molds or directly into Building 411 for cooling. Slow cooling for molded waste would be required because differential cooling in a solid block or slab would result in a large stress between the surface and the interior. Significant thermal cracks could be avoided if the immobilized waste was insulated and slowly cooled.

During the fusion process, carbonate, nitrate, and sulfate salts would be decomposed to form metallic silicates or oxides. The solubility of most metallic silicates in water is much less than metallic salts. Therefore, the leachability of the immobilized waste would be reduced. The physical stability of the immobilized waste would result from its glass-like properties, limited porosity, and minimal surface area available for chemical reaction. Silicates in the waste would buffer most chemical reactions and thus retard the release of metallic ions (unless the solvent is highly acidic).

Figure B-1 shows a preliminary process flow diagram for immobilizing the residues at NFSS. Major features of the process include a residue drier, the fusion furnace, various filters for removing process particulates and radon daughter products, a scrubbing system for reducing the concentration of sulphur oxides before atmospheric release, and a release stack.

The process would begin by placing waste and fluxing and reducing agents in a carbon steel hopper with a capacity of approximately 1.5 m³ (2 yd³) and equipped with a vibrator/shaker. Waste within the hopper would be shaken onto a screw

1 WASTE HOPPER	11 EVAPORATIVE COOLER
2 SCREW CONVEYOR	12 LIME ($\text{Ca}(\text{OH})_2$) SPRAY SYSTEM
3 WASTE DRIER	13 AIR PREFILTER
4 CONTINUOUS MELTER
ELECTRIC FURNACE	14 HEATER
5 PARTIALLY FILLED MOLD	15 HEPA FILTER BANK
6 GLASS BLOCK	16 EXHAUST FAN
7 MECHANICAL ROLLER/CONVEYOR	17 STACK
8 COOLING STATION	18 ISOKINETIC SAMPLER
9 WRAPPING AND SHIPPING STATION	19 WATER COOLER
10 CERAMIC FILTER	20 AIR PREFILTER

**FIGURE B-1 PROCESS FLOW DIAGRAM
FOR ELECTRIC FURNACE FUSION**

conveyor system which would transfer the material to a drier. The waste would be dried by heat from:

- o The furnace below
- o Furnace offgas through the outer wall of the drier
- o Conveyed air and combustion products from the furnace skirt.

During heating, the moisture in the waste would be driven off and all interstitial radon would be released. A screw conveyor and metering devices would be used to continuously feed and control the waste as it entered the electric furnace.

It is assumed that the furnace will be operated continuously Monday through Friday at a capacity of 36 t (40 tons) per day. During the last shift on Friday, the furnace would be cooled, with the solidified product remaining in the outflow portion. During the weekend, sufficient power would be drawn to maintain the furnace temperature in its cooled condition. On the first shift on Monday, maintenance would be performed during the period that the unit is brought up to operating temperatures. Furnace electrodes would be located so as to simplify maintenance and repair. Energy requirements per 40-ton batch are estimated to be 10,000 kWh.

The furnace operation would be a continuous-flow process. Dried waste would be added continuously to the furnace, and the molten waste would flow continuously from bottom vents to molds for transport or to insulated trunnion casks for transfer to Building 411.

For transport away from the NFSS, the immobilized waste would be cast in rectangular molds with dimensions of 0.9 x 0.9 x 0.15 m (3 x 3 x 0.5 ft). The design allows removal of the blocks after proper cooling and reuse of the molds. The slabs and molds would be transferred to a cooling station by a gravity-assisted roller conveyor. After cooling below 82°C (180°F), the blocks would be wrapped with reinforced PVC sheeting held in place with banding straps.

Offgas from the fusion furnace would be directed through a unit consisting of hollow ceramic filters. The filter would remove particulate carryover from the furnace. When the buildup of particulate material on the filter became excessive, it could be removed by backflushing the filter with a burst of nitrogen gas into the furnace. When the spent ceramic filters could no longer be backflushed with nitrogen, they would be removed and added directly to the waste feed where they would be vitrified along with the residues.

Exhaust gas from the annulus of the drier would be piped to an evaporative cooler/scrubber unit. The purpose of the cooler is to reduce the temperature to less than 66°C (150°F) and to scrub sulphur oxides from the exhaust system. Water saturated with lime would be sprayed into the exhaust gas and would react with the sulphur oxides to form calcium sulphate. Precipitates would be removed and disposed as radioactive waste. Although no uranium and radium would be expected to carry over into the exhaust gas, the radium-226 daughter isotopes polonium-218, polonium-214, polonium-210, lead-214, lead-210, bismuth-214, and bismuth-210 are volatile and would be expected to occur in the offgas. The scrubber would be expected to remove lead-210 and its daughters, bismuth-210 and polonium-210, as sulphates. The residue from the scrubber would be dried and put in small steel containers for placement in a mold for transport or placed directly in Building 411 where it would be encapsulated with molten vitreous waste.

The exhaust gas from the evaporative cooler would be prefiltered to prevent filter loading. The duct work from the prefilters would be equipped with heaters which could be energized to keep the saturated air above the dew point. Gas from the heating unit would enter an array of high efficiency particulate arrestor (HEPA) filters designed to remove 99.95 percent of all particulates greater than 3 micrometers in size. The remaining low particulate gas would be released via a stack.

The fusion furnace unit and other major pieces of peripheral equipment such as ceramic filters, wet scrubbers, and exhaust fans described here could be adapted from existing process equipment, minimizing the design engineering effort required.

Decontamination of the fusion furnace would involve removal of the refractory material and disposal in Building 411 or at an existing DOE facility. The internal surfaces of the furnace would be decontaminated after which it would be dismantled and sold or transported to another site for use.

APPENDIX C

RESOURCE RECOVERY PROCESS

RESOURCE RECOVERY PROCESS

The K-65, F-32, L-30, and L-50 residues are the result of processing of high-grade pitchblende ores. Although these ores were processed for uranium and metals, the residues still contain appreciable quantities of these and other materials.

The physical form of the residue is a wet clay (70% solids by weight) with an appreciable alpha quartz fraction. The clay is a sheet-like muscovite (mica) present as $\text{KAl}_2\text{Si}_3\text{AlO}_{10}(\text{OH})_2$. The muscovite clay constitutes approximately 60 volume percent; the alpha quartz, approximately 25 volume percent (Battelle, 1981).

Chemically, the K-65 residues are mixtures of oxides, carbonates, and sulfates. Oxides are approximately 40 percent of the residues; carbonates and sulfates, 20 percent. Based on the best available data, the K-65 residues have a uranium concentration of 1,000 to 2,000 ppm and radium concentration of approximately 220 ppb. The primary form of the uranium is sodium uranyl carbonate. The majority of the radium is in the form $\text{Ba}(\text{Ra})\text{SO}_4$ (Battelle, 1981).

The K-65 residues have the highest concentrations of precious metals and uranium; the L-30 and L-50 residues also contain recoverable precious metals (see Tables C-1 and C-2). The major barrier to recovery of these components is the high concentration of radium that must be separated from the metals. The F-32 residues have unknown metal concentrations, are of low grade compared to the other ores, and have small volume (see Appendix A). Therefore, these were deleted from the market evaluation. These may be considered further if analysis of these residues warrants.

Potential Resource Values

Tables C-1 and C-2 show the potential recoverable resource values for metals and uranium in the K-65, L-30, and L-50 residues, assuming 90 percent recovery efficiency, acceptable decontamination of the products is achieved, and May 1982 prices. The metal

TABLE C-1
VALUE OF K-65 RESIDUES STORED AT NFSS

Resource	\$/kg ^(a)	ppm ^(b)	kg recovered ^(c)	\$
Uranium (U)	44.71	1,500	4,725	211,300
Lead (Pb)	.50	370,000	1,165,500	582,800
Copper (Cu)	1.74	470	1,480	2,600
Cobalt (Co)	22.91	2,000	6,300	144,300
Nickel (Ni)	5.79	3,700	11,655	67,500
Gold (Au)	13,306	9 ^(d)	28	377,200
Platinum (Pl)	12,540	0.6	1.89	23,700
Palladium (Pd)	3,406	27	85	289,700
Silver (Ag)	377	0.3	0.945	400
Value				\$1,699,500

- (a) Prices based on: U \$20.3/lb; Pb \$0.225/lb; Cu \$0.79/lb; Co \$10.4/lb; Ni \$2.63/lb; Au \$504/troy oz; Pd \$179/troy oz; Pt \$475/troy oz; Ag \$14.28/troy oz (Engineering and Mining Journal, May 1982). Co and Ni Sulfide matte is based on 80 percent of metal prices.
- (b) Uranium concentration obtained from Battelle (1983a). Remaining metal concentrations obtained from leached residue data presented in Table A-10 of this report (Battelle, 1983b).
- (c) Assumes 90 percent recovery efficiency.
- (d) Results of a fire assay analysis for gold indicated that the K-65 residues may contain 1 troy oz gold/ton of residue, or 42.67 ppm, about 4.7 times the reported value (Battelle, 1983a). The lower value is used here for conservatism.

TABLE C-2
VALUE OF L-30 AND L-50 RESIDUES STORED AT NFSS

Resource	\$/kg ^(a)	L-30 Residues			L-50 Residues		
		(ppm)	(kg Recovered) ^(b)	(\$)	(ppm)	(kg Recovered) ^(b)	(\$)
Uranium (U)	44.71	500	3,357	150,100	2,500	3,825	171,000
Lead (Pb)	0.50	2,900	19,471	9,800	4,800	7,344	3,700
Copper (Cu)	1.74	1,500	10,000	17,500	400	612	1,100
Cobalt (Co)	22.91	2,000	13,428	307,600	4,700	7,191	164,800
Nickel (Ni)	5.79	33,000	221,562	1,282,800	26,700	40,851	236,600
Gold (Au)	13,306	0.2	1.34	17,900	0.3	0.46	6,100
Platinum (Pt)	12,540	0.6	4.02	50,500	0.6	0.91	11,500
Palladium (Pd)	3,406	0.3	2.01	6,900	2.5	3.82	13,000
Silver (Ag)	377	0.3	6.7	2,500	0.3	0.46	200
Value				1,845,600			608,000

(a) Prices Troy based on: U \$20.3/lb; Pb \$0.225/lb; Cu \$0.79/lb; Co \$10.4/lb; Ni \$2.63/lb; Au \$504/oz; Pd \$179/troy oz; Pt \$475/troy oz; Ag \$14.28/troy oz (Engineering and Mining Journal, May 1982). Co and Ni sulfide matte is based on 80 percent of metal prices.

(b) Assumes 90 percent recovery efficiency.

concentrations of the residues used for this analysis were obtained from recent tests conducted by Battelle (1983a, 1983b). Considered in total, the value of potentially recoverable precious metal is less than that which would be calculated from information developed during previous investigations (See Appendix A). Table C-3 presents a summary of the recoverable resource values.

Even though the African ores were processed for precious metal recovery prior to their shipment to the U.S., significant recoverable concentrations of precious metals remain in the residues. The value of precious metals in the residues stored at NFSS is estimated to be approximately \$4,153,000 based on the assumed recovery efficiency, the value of the metals, and considering analytical uncertainty (Tables C-1, C-2, and C-3).

Proposed Processing Scheme

The processing scheme for the extraction of uranium and precious and base metals is proposed based on site remedial engineering designs that are strictly conceptual. Detailed engineering of the preferred NFSS disposition alternative may necessitate modifications to the resource recovery process as proposed. Further, the ability to decontaminate the recovered metals to the degree necessary for subsequent marketing has not been verified. At present it is assumed that further smelting will be necessary following initial recovery.

As presently proposed, the process would involve three stages. The first stage would be a carbonate leach process that would separate uranium, molybdenum, and vanadium from the residues by dissolving the uranium in a carbonate solution (see Figure C-1). The yellowcake formation process (Stage II -- Figure C-2) would purify the uranium constituent and produce ammonium diuranate (yellowcake) which could be used as a feedstock or sold. The leached residue would be transported to a smelting process (Stage III) where the precious metals would be separated from the radium and other impurities (see Figure C-3). Lead in the residues tends to collect the precious metals out of the slag while the base metals

TABLE C-3

SUMMARY OF VALUE OF URANIUM AND METALS
RECOVERABLE FROM K-65, L-30, AND L-50 RESIDUES

Resource	Value of Recoverable Metals(a) (\$)
Uranium	532,400
Lead	596,200
Copper	21,200
Cobalt	616,700
Nickel	1,586,900
Gold	401,200
Platinum	85,700
Palladium	309,600
Silver	<u>3,100</u>
TOTAL	4,153,000

(a) The values of these metals fluctuate. The values reported were calculated in 1982. See notes on Table C-1.

FIGURE C-1 CARBONATE LEACH STAGE FOR RESIDUE PROCESSING

FIGURE C-2 YELLOWCAKE FORMATION STAGE FOR RESIDUE PROCESSING

FIGURE C-3 ELECTRIC FURNACE SMELTING STAGE FOR RESIDUE PROCESSING

collect in the sulfide matte. An ingot would be formed and would contain the recovered metals. A leach-resistant stable slag would also be formed and would contain the radium, barium, and impurities. This slag could be transferred to a permanent disposal site. Design capacity for the process would be 40 tons residue per 24-hour day.

Several aspects of the proposed resource recovery process have been utilized in the industrial setting. The uranium recovery operation (Stages I and II) is similar to the carbonate leach processes used in industrial yellowcake production. The metals recovery and slag vitrification process (Stage III) has been used in smelting operations.

The final metals separation and vitrification step will require further development for these particular residues. Preliminary tests have shown that the expectations concerning metals separation were correct. The entire processing scheme will require further development considering that the process involves many steps, some at high temperatures, and also involves remote handling. During start-up, the recovery efficiency will be less than 90 percent. Development of the processing scheme could greatly exceed the currently estimated cost.

Carbonate Leach (Stage I)

The residue would be hydraulically mined from its storage area and pumped to a thickener as a slurry with 25 percent solids. Liquid would be removed from the slurry to 50 percent solids. Uranium would be leached from the residues in the form of sodium uranyl carbonate. The liquid containing the sodium uranyl carbonate would be passed through the thickener and used for the hydraulic mining of more residue. This liquid recycle would reduce the volume of water requiring treatment at the completion of processing and increase the concentration of sodium uranyl carbonate in solution. Bicarbonate must be present in the leach solution to prevent the precipitation of dissolved uranium by hydroxyl ions formed during leaching. When the concentration was sufficiently high, the liquid would be processed in Stage II (yellowcake formation).

Yellowcake Formation (Stage II)

When sodium uranyl carbonate was sufficiently concentrated in the Stage I water, the solution would be diverted into a tank and mixed with sodium hydroxide (NaOH). The sodium uranyl carbonate would be converted to sodium diuranate $[\text{Na}_2\text{U}_2\text{O}_7]$ by the following reaction:

Sodium diuranate would drop out of solution as a precipitate. The residence time required for this reaction to take place at 95 percent efficiency is estimated to be 6 to 12 hours.

The precipitate would be sent to a resolution tank where H_2SO_4 would be added. This would dissolve the diuranate and allow any excess CO_2 to be bubbled off by the following reaction:

The CO_2 would be sent to the wet scrubber and the solution passed to the NH_3 precipitation tank.

Ammonium diuranate (ADU) would be formed in the ammonia tank as a precipitate by the following reaction:

The addition of ammonia would neutralize the solution pH to about 7.2.

The precipitated solution would be passed to a filter where the ADU yellowcake would be filtered out of solution and washed with water. The yellowcake, now free of sulfates, would be dried in a screwdryer and packaged for shipment. The sulfate solution would be pumped to the wet scrubber to remove SO_2 from process gases.

The solution which would be left after the NaOH precipitation of sodium diuranate would be composed primarily of Na_2CO_3 , Na_2SO_4 , and NaHCO_3 . This solution would be reacted with $\text{Ca}(\text{OH})_2$ by the following reactions:

This step would serve to remove carbonates and sulfates from the process system by precipitating out CaCO_3 and CaSO_4 .

The solution would be filtered and sent to the wet scrubber to aid in the removal of SO_2 from process gases. The filter cake, made up primarily of CaSO_4 and CaCO_3 , would be stored temporarily in a surge pile and ultimately buried.

Electric Furnace Smelting (Stage III)

Leached and dried residue from the Stage I carbonate leach process would be fed in 2- to 3-ton batches to an electric furnace. Enough activated carbon would be added to the residue to reduce the lead oxides, carbonates, and sulfates to their elemental metals. The temperature maintained inside the furnace would be $1,300^\circ$ to $1,500^\circ\text{C}$ ($2,372$ to $2,732^\circ\text{F}$). A molten lead ingot containing Au, Pd, Pt, and Ag would be formed at the bottom of the furnace. The offgas, containing mainly CO_2 and SO_2 , would be vented to the baghouse.

After about 1 hour, the slag containing radium, barium, and all other constituents not held in the lead ingot would be poured into another electric furnace. This furnace would also be maintained at $1,300^\circ$ to $1,500^\circ\text{C}$ ($2,372$ to $2,732^\circ\text{F}$). More activated carbon would be added to reduce the remaining metal compounds (Ni, Fe, Co, and Cu). Either lime or soda ash would be added to reduce the viscosity of the slag. Iron sulfide would be added to collect the Ni, Fe, Co,

and Cu into a sulfide matte at the bottom of the furnace. Offgas would again be sent to the baghouse. After about an hour, the vitrified slag containing barium, radium, and unreacted uranium could be removed from the furnace. This slag would be poured into molds for transport offsite or would be poured directly into Building 411. The leach-resistant slag waste product would be approximately 50 percent of the original residue volume.

Offgases from both furnaces would be directed to a baghouse. Dust would be removed from the gas and returned to a surge bin to be recycled to the furnace. Gas containing primarily SO_2 would be directed to the wet scrubber where it would be mixed with gas emissions from the screwdryer. These gases would be scrubbed with a caustic solution to remove SO_2 before they were vented to the atmosphere.

Materials of Construction

All equipment would be designed to withstand caustic solutions and temperatures of 100°C (212°F), except the resolution tank which would be constructed of fiberglass. Furnaces would be designed to withstand temperatures of $1,500^\circ\text{C}$ ($2,732^\circ\text{F}$).

Energy Requirements

Energy requirements are expected to be similar to those required for immobilization, i.e., 10,000 kWh per 40-ton batch.

Safety Precautions

The residues contain sufficient radioactivity to warrant special safety precautions in the design and operation of the processing system. The K-65 residues are the most hazardous, with radium concentrations of about 220,000 pCi/g. The L-30 and L-50 residues have radium concentrations of 12,000 and 10,000 pCi/g, respectively (Battelle, 1981). The same safety precautions would be taken for the processing of all the residues. All tanks and process equipment would be designed to prevent the accumulation of toxic chemicals or radioactive gases.

To minimize operator exposure to radioactive streams, all equipment would be remotely operated from a central control room located such that visual contact with the equipment could be maintained. Manual valves and standby pumps would be provided as necessary so that equipment could be emptied prior to major maintenance.

APPENDIX D

HYDRAULIC MINING AND K-65 RESIDUE HANDLING SEQUENCE

HYDRAULIC MINING AND K-65 RESIDUE HANDLING SEQUENCE

This appendix discusses the remedial action for removal of the K-65 residues from Building 411 for alternatives involving immobilization, processing, and/or transporting the residues. The hydraulic mining and handling sequence for other residues would be similar, but would not require the same radon release controls. The actions are based strictly on conceptual engineering. Detailed engineering of the preferred NFSS disposition alternative (if it includes removal of the residues), may necessitate modification of the conceptual plans discussed below.

The clay and soil cover over the K-65 residues in Building 411 would be excavated by backhoe to a depth of 0.3 m (1 ft) above the identification layer which was placed during interim remedial action to reduce radon emanation and to identify the location of the residue. This waste would be properly stored or loaded onto trucks and removed from the site as bulk low-level radioactive waste. Water would then be added until it is 0.9 m (3 ft) in depth over the remaining soil covering the identification layer and the K-65 residues when the identification layer is removed. The water will minimize radon release from the residues. A lightweight structure would be constructed over the west bay of Building 411, and an exhaust stack would be built. The building would be kept under negative pressure; air would be exhausted via the stack.

Health physics coverage would be provided during the loading operation and during placement of the water layer. The health physics coverage would include air sample monitoring, particularly during the placement of water on top of the soil over the K-65 residues.

A crane would be used to position and lower the hydraulic mining equipment through an opening in the roof. The remaining soil would be hydraulically mined to the identification layer. The soil would be slurried to a sedimentation pond for removal. If the

identification layer was broken, the remaining material in the west bay would be considered to be contaminated with some K-65 material and would be handled accordingly. If the identification layer remained intact after soil removal, it would be sectioned and left within the west bay for later removal. Makeup water would be added to Building 411 as required. The slurry piping would be 10-cm (4-in.) diameter carbon steel with welded joints and seams. High pressure water would slurry the residue into a 30 percent by weight solution that would be pumped directly to the process building.

The process building would be erected east of the Central Drainage Ditch and just north of the South-16 ditch. The layout is shown in Figure D-1. The building would have a 10.6 x 30.4 m (35 x 100 ft) reinforced concrete slab with a 30-cm (12-in.) curb upon which a steel siding structure would be erected. A duct would connect this structure to the lightweight structure covering Building 411. A 9-m (30-ft) diameter thickener would be located in the western end of the process building.

The hydraulic mining pump would discharge the slurry through the pipe system into the thickener. Motorized rakes would move the residues into the conical bottom of the thickener where hydraulic pressure would force the material out into a sump. This sump would be located below grade east of the 1.5-m (5-ft) solid concrete block wall which isolates the thickener. A Moyno position displacement pump with variable speed drive would be used to transfer the thickened sludge from the sump by the thickener to the indirectly heated drier.

In the drier, hot oil is used to heat the jacket and hollow paddles (or screws) of the drier. The drier agitates and mixes the incoming sludge with recirculated dry solids while driving off the moisture from the sludge in the form of vapor (steam). The steam is directed to an air-cooled condenser where it is condensed. The resultant water is returned to the thickener where it blends with the supernatant liquid from the thickener for return to Building 411 in the hydraulic mining operation.

The hot oil used to heat the indirectly heater drier is heated in a direct-fired heating system with expansion tank, circulation

FIGURE D-1 PROCESS BUILDING LAYOUT FOR LOADING K-65 RESIDUES INTO DRUMS FOLLOWING HYDRAULIC MINING

pumps, and heating coils. The heated oil -- about 316°C (600°F) -- is circulated to the indirectly heated drier to furnish the heat for evaporation of the moisture from the sludge. The oil exits from the drier at about 290°C (550°F) and returns to the heater system for reheating. The overflow from the thickener would drain by gravity to a second sump located adjacent to the underdrain sump. This sump would be emptied by a high pressure (200 psig) horizontal centrifugal pump.

The pump would supply water at high pressure to the section of pipe near the thickener underdrain isolation valves to clear the pipe of settled residue slurry. The pump would also be used to supply about 0.4 ft³/s (180 gal/min) to the nozzles of the hydraulic mining pump. During the hydraulic mining operation, inspection of the pipeline and equipment would be conducted a minimum of twice per day for early detection of leaks. Handling the residues wet would minimize radon release; therefore, this would not be a concern during hydraulic mining and transfer of the residues.

The dried slurry solids are discharged from the indirectly heated drier through a rotary valve into a drum fill station weigh hopper. Preset weight limits on the weigh hopper would shut off the rotary valve and initiate the drum fill sequence when the correct fill weight is loaded into the hopper. A set of powered roller conveyors would position an empty 55-gal drum beneath the hopper. A collar would be lowered on the drum which would allow the hopper to meter about 249 kg (550 lb) of residue into the drum. The drum would be vibrated during the filling stage. The collar would be raised, and the drums indexed to a station where a gasketed drum lid would be crimped over the opening. The drum would then be advanced to the last station where the drum would be washed down with clean high pressure water. The drum would then be transferred out of the building on a powered roller conveyor. When the weigh hopper is emptied into the drum, the drum fill gate would close and the rotary valve under the drier would be turned on to load the next batch of dried solids into the weigh hopper.

In order to perform maintenance in the process building, all sumps, thickeners, and hoppers would be routinely emptied of residue so that the source inventory would be at a low level. The eastern end of the process building would be equipped with a water spray system to be used for washdown. The interior of the structure would be painted with a 20-mil coating of strippable paint prior to the start of the work. The flooring would have a smooth, sealed concrete surface which would be sloped to the drum washdown sump. During normal shutdowns (i.e., over the weekend) the residual load of dried solids would be left in the indirectly heated drier. This residual load of dried solids is needed to facilitate restart of the drying operation. The load of dried solids in the drier is, of course, radioactive and would make the vicinity of the drier a moderately radioactive (300-500 mR) area requiring normal precautions if any entry or maintenance work is needed near this equipment.

If direct maintenance on the drier or equipment in its immediate vicinity is required, the dry solids in the drier would be emptied into drums and then sprays in the domed cover of the drier would be used to flush residual material from the drier into the thickener. The dumping and flushing operations would remove most of the radioactive contamination from the drier, thereby permitting work on the drier or equipment in its immediate vicinity.

The process structure would be well-lighted and monitored with a TV camera. The equipment would be monitored remotely from a control room located in Building 403.

All drums to be filled would be premarked and labeled in accordance with DOE regulations. The west end of the process building and an area extending beyond the 1.5 m (5 ft) concrete block wall to a distance of 15 m (50 ft) from the outer western wall would be enclosed in a personnel exclusion fence. The exposure at the fence would be limited to 2.5 mR/h.

The 55-gal drums filled with K-65 residues would have a dose rate at the surface of about 300 mR/h. These drums would be picked up from the roller assembly with a three-prong drum lift using a

portable crane. The crane would be positioned on the far side of the shielded transport vehicle. The crane would remotely lift the drum off the roller, place it in the transport vehicle, and disengage.

If the K-65 residues are to be vitrified with no metal recovery, there would be no roller for the drum filling. The waste hopper would instead contain a screw feeder to be used for filling a furnace smelt pot. Adjacent to the waste hopper, a door panel would provide access for the smelt pot carried by a vehicle. The eastern wall exterior would be shielded with concrete solid blocks. A lime and ferric oxide hopper and augered screw feeder would be located adjacent to the wall. This feeder would be interlocked to meter the proper fluxing agent into the smelting pot. The pot would be fed with the correct waste quantity by controlling the number of turns of the screw feeder and by sonar level devices. When filled, it would be taken to the smelt building.

If the metals are to be recovered, a reducing agent would also be metered in. This would be done through a separate hopper with a separate screw feeder.

Figure D-2 shows the site plan with the hydraulic mining piping and process building locations.

APPENDIX E

**U. S. DEPARTMENT OF ENERGY
PROPOSED RADIOLOGICAL CRITERIA FOR
FORMERLY UTILIZED SITES REMEDIAL ACTION PROGRAM
AND
REMOTE SURPLUS FACILITIES MANAGEMENT PROGRAM SITES**

U. S. DEPARTMENT OF ENERGY
Proposed Radiological Criteria
for
Formerly Utilized Sites Remedial Action Program
and Remote
Surplus Facilities Management Program Sites

Presented here are the radiological cleanup and waste control criteria of general applicability to the FUSRAP project and remote SFMP sites.

With the exception of limits for radium-226, the soil cleanup criteria were developed on the basis of limiting maximum individual radiation exposure to DOE limits specified in DOE Order 5480.1A exclusive of exposure from natural background radiation or medical procedures. The aggregate of the contribution from all major pathways, based on scenarios for permanent intrusion, e.g., establishing residences on the site, has been assumed. In most circumstances, the probability is low that such an intrusion will occur. Also, conservative assumptions were used in deriving these criteria to ensure that a particular dose limit would not be exceeded. Use of these criteria is additionally conservative because the pathways considered in the derivation of the criteria assume all water intake and most food intake is from the site. Also, the sites often have limited agricultural capability and the contamination is generally not homogeneous. The combined effect of these factors is such that the probable radiation exposure to the average population on, or in the vicinity of, FUSRAP sites decontaminated to these criteria limits will not be appreciably different from that normally received from natural background radiation.

The cleanup criteria for surface contamination of structures were developed from a proposed ANSI standard modified as appropriate to be consistent with DOE Order 5480.1A and the specific needs of FUSRAP for cost-effective, workable guidelines which provide an adequate safety margin. The waste control criteria are based on applicable DOE Orders and EPA's regulations for inactive uranium milling sites, 40 CFR 192.

The reader should note that Sections A.1., A.2., and D. hereof have separate sets of footnotes.

August 5, 1983

PROJECT RADIOLOGICAL CRITERIA

A. Cleanup Criteria

1. Cleanup of Land (Maximum limits for unrestricted use)

Soil Remedial Action Criteria^{1/}

<u>Radionuclide</u>	<u>Soil Criteria</u> (pCi/g above background) Avg. over 100 m ²	
Ra-226	5 pCi/g, averaged over the first 15 cm of soil below the surface; 15 pCi/g when averaged over 15 cm thick soil layers more than 15 cm below the surface and less than 1.5 m below the surface.	
U-Natural ^{2/}	75	
U-238 ^{3/}	75	
U-234 ^{4/}	150	
U-235 ^{4/}	150	
Am-241	20	
Pu-241 ^{5/}	800	
Pu-239, -240	100	
Pu-238	100	
Th-230 ^{6/}	300	
Cs-137	80	
Sr-90	100	
H-3 (pCi/ml soil moisture)	5,200	
Th-232 ^{7/}	15	

^{1/} Except for radium-226, these criteria represent unrestricted-use residual concentrations above background averaged across any 15 cm thickness layer to any depth and over any contiguous 100 m² surface area. The same conditions prevail for Ra-226 except for soil layers beneath 1.5 m; beneath 1.5 m, the allowable Ra-226 concentration may be affected by site-specific conditions and must be evaluated accordingly.

^{2/} A curie of natural uranium means the sum of 3.7×10^{10} disintegrations per second from U-238 plus 3.7×10^{10} dis/sec from U-234 plus 1.7×10^9 dis/sec from U-235. One curie of natural uranium is equivalent to 3,000 kilograms or 6,615 pounds of natural uranium.

3/ This criterion is for the activity concentration of U-238 alone, but has been derived on the basis of the assumption that U-234 is also present in the soil at the same activity concentration and that the contribution from U-235 is small.

4/ Assumes no other uranium isotopes are present.

5/ The Pu-241 criterion was derived from the Am-241 concentration.

6/ Applicable only for relatively short periods of unrestricted use (up to approximately 100 years if no Ra-226 is present initially). For most applications (permanently unrestricted use), the Th-230 concentration may not exceed the Ra-226 guideline.

7/ Assumes all decay chain products are in equilibrium concentrations (e.g., Ra-228 criteria would be the same as that for Th-232).

2. Cleanup of Buildings

a. Indoor Radon Decay Products

A structure located on private property and intended for unrestricted use shall be subject to remedial action as necessary to ensure the annual average radon daughter concentration (RDC) is less than 0.03 WL within the structure.

b. Indoor Gamma Radiation

The indoor gamma radiation after cleanup shall not exceed 20 microrentgen per hour (20 microR/hr) above background.

c. Indoor/Outdoor Building Surface Contamination

Building Surface Contamination Remedial Action Criteria:

Radionuclides	Allowable Surface Contamination (dpm/100 cm ²)	
	Total	Removable
Group 1: Radionuclides for which the area concentration guide in air above background ^{2/} is 2×10^{-13} Ci/m ³ or less or for which the uncontrolled area concentration guide in water above background ^{2/} is 2×10^{-7} Ci/m ³ or less; includes Pa-231, Th-228, Th-230, Ac-227, Ra-226, Ra-228, and Pb-210.	100	20
Group 2: Radionuclides not in Group 1 for which the uncontrolled area concentration guide in air above background ^{2/} is 1×10^{-12} Ci/m ³ or less or for which an uncontrolled area concentration guide in water above background ^{2/} is 1×10^{-6} Ci/m ³ or less; includes U-232, U-238, Th-232, Ra-223, and Po-210.	1,000	200
Group 3: Those radionuclides not in Group 1 or Group 2; includes U-234, U-235, and Ra-224.	5,000	1,000

^{1/} The levels may be averaged over 1 m² provided the maximum activity in any area of 100 cm² is less than 3 times the limit value; dpm = disintegrations per minute.

^{2/} Given in Attachment I to Chapter XI, Table II, DOE Order 5480.1A.

Source: Adapted from proposed ANSI N13.12.

B. Control of Radioactive Wastes and Residues

1. Interim Storage

All operational and control requirements specified in the following DOE Orders shall apply:

- a. 5480.1A, Environmental Protection, Safety, and Health Protection Program for DOE Operations.
- b. 5480.2, Hazardous and Radioactive Mixed Waste Management.
- c. 5481.1A, Safety Analysis and Review System.
- d. 5483.1, Occupational Safety and Health Program for Government-Owned Contractor-Operated Facilities.
- e. 5484.1, Environmental Protection, Safety, and Health Protection Information Reporting Requirements.
- f. 5484.2, Unusual Occurrence Reporting System.
- g. Control and stabilization features will be designed to ensure, to the extent reasonably achievable, an effective life of 50 years, and in any case, at least 25 years.
- h. Radon concentrations in the atmosphere above facility surfaces or openings shall not (1) exceed 100 pCi/l at any given point, or an average concentration of 30 pCi/l for the facility site, or (2) exceed an average radon concentration at or above any location outside the facility site of 3.0 pCi/l (above background).

- i. For water protection, use existing State and Federal Standards; apply site-specific measures where needed.

2. Long-Term Management

- a. All operational requirements specified for Interim Storage Facilities (B.1) will apply.
- b. Control and stabilization features will be designed to ensure, to the extent reasonably achievable, an effective life of 1,000 years and, in any case, at least 200 years. Other disposal site design features shall conform with 40 CFR 192 performance guidelines/requirements.
- c. Radon emanation to the atmosphere from facility surfaces or openings shall not (1) exceed an average release rate of $20 \text{ pCi/m}^2 \text{ sec}$, or (2) increase the annual average radon concentration at or above any location outside the facility site by more than 0.5 pCi/l .
- d. For water protection, use existing State and Federal Standards; apply site-specific measures where needed.
- e. Prior to placement of any potentially biodegradable contaminated wastes in a Long-Term Management facility, such wastes will be properly conditioned to (1) insure the generation and escape of biogenic gases will not cause the criteria in paragraph 2.c. to be exceeded, and (2) insure biodegradation within the facility will not result in premature structural failure not in accordance with the criteria in paragraph 2.b. If biodegradable wastes are conditioned by incineration, incineration operations will be carried out in compliance with all applicable federal, state, and

local air emission standards and requirements, including any standards for radionuclides established pursuant to 40 CFR 61, National Emission Standards for Hazardous Air Pollutants (NESHAPS).

C. Exceptions

1. Procedure -- Analysis of site-specific conditions.
2. Applicability -- Where health and safety would be endangered, or where cost clearly outweighs benefits.

D. Criteria Sources

<u>Criteria</u>	<u>Source</u>
1. <u>Cleanup Criteria</u>	
a. Cleanup of Land ^{1/}	DOE Order 5480.1A, 40 CFR 192 ^{2/}
b. Cleanup of Buildings	40 CFR 192, proposed ANSI N13.12
2. <u>Control of Radioactive Wastes and Residues</u>	
a. Interim Storage	DOE Order 5480.1A
b. Disposal	40 CFR 192
3. <u>Exceptions</u>	
a. Procedure	40 CFR 192
b. Applicability	40 CFR 192

^{1/} The bases of the cleanup criteria are developed in ORO-831 and ORO-832.

^{2/} Based on limiting the radon daughter concentration to 0.03 WL within structures.

APPENDIX F

**PRELIMINARY SCHEDULE
AND MANPOWER REQUIREMENTS**

PRELIMINARY SCHEDULE AND MANPOWER REQUIREMENTS

Preliminary schedules and manpower requirements for each of the 12 disposition alternatives are given in Figures F-1 through F-12. The activities considered for each alternative are given in the left-hand column of each figure, and the years in which the activities occur are indicated across the top of each figure.

For each activity identified, the number of people required and the duration of their involvement is presented. The manpower requirements identified on the figures for each line item are preliminary estimates subject to change during future design phases. These manpower estimates are presented for information purposes only, and are not intended to be compared in any manner with the cost estimates presented earlier in this document.

The schedule and manpower requirements for Alternative 1, which is maintaining the site in its post-interim-remedial-action condition, are given in Figure F-1 (page F-3). This alternative requires maintenance and surveillance, in addition to continuing management support, for an indefinite period of time.

Figure F-2 shows the schedule and manpower requirements for Alternative 2, which is upgrading the waste containment area. From the start of engineering design to the completion of construction will require about 5 years. Following site closure, there will be 5 years of full maintenance and surveillance. Custodial maintenance and surveillance will continue for the balance of 200 years. The estimated manhours total is 466,200.

Alternative 3, which is to immobilize the residues and upgrade the waste containment area, is addressed in Figure F-3. Hydraulic mining and immobilization will require about 6 years, and upgrading containment will take about 3 years, for a total of 9 years. Following this, there will be 5 years of full maintenance and surveillance, and the balance of 200 years of custodial maintenance and surveillance.

Alternative 4 involves processing of the residues; schedule and manpower requirements are shown in Figure F-4. This alternative will be completed in 9 years, followed by 5 years of full maintenance and surveillance and 195 years of custodial maintenance and surveillance. Manhour requirements for Alternatives 3 and 4 are 715,400 and 758,300, respectively.

Alternatives 5 through 10 (see Figures F-5 through F-10) have similar schedules; they each require 8 years for implementation. Manhours are much greater for these alternatives because decontamination of the site requires removal and shipment of the residues, contaminated soils, and rubble. The difference in manhours between Alternatives 5, 6, and 7 and Alternatives 8, 9, and 10 is due to additional time required to ship the materials to a western DOE site. Alternatives 5, 6, and 7 require an average of 846,000 manhours, and Alternatives 8, 9, and 10 require an average of 1,178,500 manhours.

Alternatives 11 and 12 involve excavation and removal of the more radioactive residues and transport to an existing DOE disposal site in the eastern and western U.S. The waste containment area is then upgraded. From the start of engineering design to the completion of waste disposal will require about 6 years. Upgrading the waste containment area will require about 4 years. Following site closure, there will be 5 years of full maintenance and surveillance, and 195 years of custodial maintenance and surveillance. Manpower requirements for Alternatives 11 and 12 are 746,400 and 901,800, respectively.

FIGURE F-1
SCHEDULE AND MANPOWER REQUIREMENTS • ALTERNATIVE 1

ACTIVITY	YEARS										
	1	2	3	4	5	6	7	8	9	10	11
MANAGEMENT SUPPORT						1 MAN					
MAINTENANCE & SURVEILLANCE						4 MEN					

Note: Maintenance and surveillance
 continues indefinitely

TOTAL MANHOURS: 10,000 /YEAR

FIGURE F-2
SCHEDULE AND MANPOWER REQUIREMENTS • ALTERNATIVE 2

NOTES: a* Includes health physics monitoring during construction
b* Five years of maintenance and surveillance followed by
195 years of custodial maintenance and surveillance

TOTAL MANHOURS: 466,200

FIGURE F-3
SCHEDULE AND MANPOWER REQUIREMENTS • ALTERNATIVE 3

NOTES: a* Includes health physics monitoring during construction
 b* Five years of maintenance and surveillance, followed by
 195 years of custodial maintenance and surveillance

TOTAL MANHOURS: 715,400

FIGURE F-4
SCHEDULE AND MANPOWER REQUIREMENTS • ALTERNATIVE 4

NOTES: a* Includes health physics monitoring during construction
b* Five years of maintenance and surveillance, followed by
195 years of custodial maintenance and surveillance

TOTAL MANHOURS: 758,300

FIGURE F-5

FIGURE F-6
SCHEDULE AND MANPOWER REQUIREMENTS • ALTERNATIVE 6

FIGURE F-7
SCHEDULE AND MANPOWER REQUIREMENTS • ALTERNATIVE 7

FIGURE F-8
SCHEDULE AND MANPOWER REQUIREMENTS • ALTERNATIVE 8

NOTE: a* Includes health physics monitoring during construction

TOTAL MANHOURS: 1,111,500

FIGURE F-9
SCHEDULE AND MANPOWER REQUIREMENTS • ALTERNATIVE 9

NOTE: a* Includes health physics monitoring during construction.

TOTAL MANHOURS: 1,190,500

FIGURE F-10
SCHEDULE AND MANPOWER REQUIREMENTS • ALTERNATIVE 10

NOTE: a* - Includes health physics monitoring during construction

TOTAL MANHOURS: 1,233,500

FIGURE F-12
SCHEDULE AND MANPOWER REQUIREMENTS ● ALTERNATIVE 12

NOTES: a Includes health physics monitoring during construction

b* Manpower is included for 200 years of maintenance and surveillance (Alternatives 2-4).

TOTAL MANHOURS: 901,800