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ALTERNATIVES FOR DISPOSAL OF RAFFINATE FROM THE TRUBX PROCESS

by

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Chemical Technology Division

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GLOSSARY

AECB	Atomic Energy Control Board (Canadian).
BNFL	British Nuclear Fuels Limited.
CEA	Commissariat a l'Energie Atomique (French Atomic Energy Commission).
CC	Complexant Concentrates.
CFR	Code of Federal Regulations.
CRW	Cladding Removal Waste.
DOE	United States Department of Energy.
DOT	United States Department of Transportation.
EPA	United States Environmental Protection Agency.
FP,FPs	Fission Product, Fission Products.
FR	Federal Register.
FRG	Federal Republic of Germany.
GCD	Greater Confinement Disposal.
GTCC	Greater-Than-Class-C low-level waste ; waste with concentrations of radionuclides that exceed the limits for Class C wastes defined by 10 CFR 61.
HLLW	High-Level Liquid Waste.
HLW	High-Level Waste.
HSA	High Specific Activity.
IAEA	International Atomic Energy Agency.
ILW	Intermediate-Level Waste.
INEL	Idaho National Engineering Laboratory.
LLW	Low-Level Waste.
MLW	Medium-Level Waste.
NCAW	Neutralized Current Acid Waste.
NCRW	Neutralized Cladding Removal Waste.
NEPA	National Environmental Policy Act of 1969.
NRC	United States Nuclear Regulatory Commission.
NTS	Nevada Test Site.
NWPA	Nuclear Waste Policy Act of 1983.
PFP	Plutonium Finishing Plant (located on the Hanford Reservation).
SFR	Swedish Final Repository.
SKB	Swedish Nuclear Fuel and Waste Management Company.
SLB	Shallow Land Burial.
SNM	Special Nuclear Material.
SRP	Savannah River Plant.
TRU	Material in which the concentration of nuclides of the trans-uranium elements is >100 nCi/g.
TRUEX	A process for removing actinides from high-level liquid waste by solvent extraction using the complexing agent octyl,phenyl-N,N-diisobutylcarbamoylphosphine oxide (CMPO).
WIPP	Waste Isolation Pilot Plant.

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ABSTRACT

Possible methods for disposing of the immobilized raffinate from TRUEX processing are reviewed. The purpose of the TRUEX process is to extract transuranium elements from high-level and TRU wastes into a small volume that can be managed at lower cost than that of the original wastes. The raffinate from the TRUEX process, containing negligible concentrations of transuranium elements, would be combined with salt solutions also derived from processing high-level waste, and the mixture would be converted to grout. The most important determinants of the destination of this grout are concentration of fission products in it, the interpretation of waste-classification rules, and the characteristics of the site on which the wastes were generated (whether they permit on-site disposal). Most likely, the treatment of this grout will result in its classification as LLW, and emplacement in preformed concrete vaults, in a near-surface location, on the site where the waste had been generated. Nevertheless, examinations of hypothetical cases which would result in its classification as a Greater-than-Class-C low-level waste or a high-level waste are also examined. Characteristics of the waste management systems that might be involved in handling the TRUEX raffinate for each of these waste classifications are briefly described.

I. INTRODUCTION

The TRUEX process is a solvent extraction process for separating transuranic elements from aqueous acidic nitrate solutions. The transuranic elements are extracted into an organic solvent consisting of an extractant, octyl,phenyl-N,N-diisobutylcarbamoylphosphine oxide, synthesized specifically for this application [HORWITZ], tributyl phosphate, and an organic diluent such as a paraffin hydrocarbon or tetrachloroethylene. The aqueous raffinate from this operation contains most of the other constituents of the original acidic nitrate solution.

The TRUEX process was developed [VANDEGRIFT, LEONARD-1985] to transform high-level waste (HLW) or transuranic (TRU) waste, such as those now stored at some DOE reservations, into two fractions: a small volume containing the transuranic elements and a large volume of raffinate. Depending on the value of the special nuclear material (SNM) involved, the transuranic products could be either recovered or disposed of as a waste. The raffinate, having concentrations of transuranic elements lower than the TRU limit for low-level waste (100 nCi/g), might be disposed of as a low-level waste. Thus, this transformation could permit a major fraction of the original waste volume to be managed using waste forms and disposal facilities less costly than those normally planned for HLW.

Work on the TRUEX process has been concerned mainly with its chemistry, engineering equipment for its application, and process modeling; disposal of the raffinate from the process has not been considered in detail. This waste disposal problem warrants attention because the appropriate destination of the solidified raffinate is not immediately obvious from its expected characteristics. Furthermore, the question of whether concentrations of Cs-137 and Sr-90 in solidified raffinate from some HLW streams could exceed the limits of Class C low-level waste [10 CFR 61], thereby requiring special arrangements for its disposal, needs to be considered.

In reviewing the alternatives for disposing of TRUEX raffinate, this study briefly examines the schemes for waste treatment, selection of waste form, selection of a disposal site, and designs of disposal units into which TRUEX raffinate might be emplaced. It will become evident that the possibilities for disposing of TRUEX raffinate depend on the operations used in the waste treatment scheme and the regulatory classification of the solid waste product.

II. SUMMARY

Methods for disposal of raffinates from the TRUEX process, as dictated by their waste classification, are examined. The TRUEX process separates the transuranic elements from high-level waste or TRU waste, concentrates these elements in streams of smaller volume, and leaves the remaining materials in a raffinate with relatively large volume and with negligible concentrations of transuranic elements. The aim of the process is an economically significant decrease in the volume of waste that must be managed as HLW or TRU waste.

The transuranic-containing product (TRU fraction) from the TRUEX process could--as one option--be sent to a vitrification facility for conversion to glass and disposal in a geologic repository. Alternatively, this TRU fraction could be converted to a waste form much less expensive than glass, e.g., grout, and disposed of in a facility for TRU waste of the type exemplified by the Waste Isolation Pilot Plant (WIPP). Disposal of the TRU fraction is not covered in detail in this report.

The raffinate from the TRUEX process would most likely be combined with salt wastes remaining from the treatment of HLW, and the mixture would be converted to grout. Most likely, the treatment of this grout will result in its classification as low-level waste (LLW), and emplacement in preformed concrete vaults, in a near-surface location, on the site where the waste had been generated. Wet grout would be discharged as a LLW into near-surface, preformed concrete vaults similar to those being designed for grout disposal at Hanford and saltstone disposal at the Savannah River Plant, where it would cure into large, monolithic blocks. This disposal mode would be preferred for economy. To illustrate this disposal mode, the Hanford Transportable Grout Facility is briefly described. Nevertheless, hypothetical cases which would result in its classification as a Greater-than-Class-C low-level waste (GTCC-LLW) or a high-level waste are also examined.

For some HLW streams, unless additional operations are applied to remove some fission products (mainly ^{137}Cs and ^{90}Sr), the immobilized TRUEX raffinate might have to be managed as a GTCC-LLW. The Department of Energy has outlined plans for management of GTCC-LLW generated in commercial operations but not for those generated in DOE operations. Management of GTCC-LLW might involve emplacement in an intermediate facility whose design might eventually be developed from methods for Greater Confinement Disposal in the U.S. or from methods for intermediate-level waste (ILW) disposal in foreign countries.

Since the definition of HLW by the National Waste Policy Act is based on source rather than on radioactivity concentrations, a strict interpretation of this definition could require that TRUEX raffinate be classified as HLW. If this is done, rulings that such wastes must be emplaced in a geologic repository could follow, although present laws and regulations do not specify that a geologic repository must be used for HLW.

III. CANDIDATE WASTES FOR TRUEX APPLICATION

Candidate wastes are of two main types: TRU and HLW.

The TRU wastes to which the application of TRUEX might be appropriate are stored at Hanford, WA; Rocky Flats, CO; and Los Alamos, NM. The two fractions resulting from TRUEX processing of these wastes would consist of (1) a relatively small volume in which the transuranic elements would be concentrated and (2) a relatively large volume of nonTRU raffinate containing no fission products.

Candidate high-level wastes are stored as solids and solutions at the West Valley Demonstration Project, the Savannah River Plant (SRP), the Idaho National Engineering Laboratory (INEL), and the Hanford Site. The two fractions resulting from application of TRUEX to high-level wastes would consist of (1) a small volume containing the transuranic elements and (2) a relatively large volume of raffinate containing the mixture of dissolved salts associated with the original HLW. The distribution of fission products, particularly Cs-137 and Sr-90, between the two fractions depends on the selection of operations used in the waste-processing facility, of which TRUEX would be one component.

The schedules for managing the high-level wastes at West Valley and at SRP are at such an advanced stage that the application of TRUEX to those wastes is probably not a realistic option. Because the use of the TRUEX process in schemes for the disposal of the Hanford wastes has already been considered [U.S. DEPARTMENT OF ENERGY-1987F, SCHULZ], the disposal of TRUEX raffinate is conveniently illustrated by describing its application to those wastes. Application of the TRUEX process to INEL wastes has been given less consideration, but that possibility is briefly described here also.

A. Hanford Tank Wastes

Wastes to which TRUEX would be applied are sludges, slurries, and salt solutions that are stored in tanks on the Hanford site [U.S. DEPARTMENT OF ENERGY-1987I]. Plans for the management of these wastes divide them into two groups: existing tank wastes and future tank wastes [U.S. DEPARTMENT OF ENERGY-1987D].

1. Existing Tank Wastes

Existing tank wastes consist of fuel-reprocessing wastes generated before November 1983.

a. Existing Tank Wastes in Single-Shell Tanks

The characteristics of existing tank wastes have resulted from the practice of adding caustic to high-level liquid wastes and storing the resulting mixtures in single-shell tanks. During storage in these tanks, the mixtures have separated into salt cake, sludge, and supernatant liquid.

b. Existing Tank Wastes in Double-Shell Tanks

From all single-shell tanks, the supernatant liquids are being transferred, in an ongoing program, into double-shell tanks [U.S. DEPARTMENT

OF ENERGY-1987B, U.S. DEPARTMENT OF ENERGY-1987D]. At the completion of the ongoing program, existing tank wastes will be distributed between 149 single-shell tanks containing sludge and salt cake and 14 double-shell tanks containing slurries of concentrated salt solutions, called double-shell slurry, and concentrated solutions of organic complexes, called complexant concentrates (CC). Low-level double-shell slurry will likely not require pretreatment prior to grout disposal [WODRICH]. Retrieved complexant concentrates and supernate will both be acidified for treatment using the TRUEX process [WODRICH]. In the reference plan for dealing with single-shell tanks [U.S. DEPARTMENT OF ENERGY-1987D], the wastes would remain in the tanks and be stabilized by filling the tanks with gravel [RISENMA]. Whether this plan for in-place stabilization would be permitted, however, is not certain. If it is not permitted, then an alternative plan involving mechanical retrieval and chemical processing of the salt and sludge may be carried out.

2. Future Tank Wastes

a. Storage

Future tank wastes consist of wastes generated by the PUREX plant since it resumed operation in November 1983. These wastes include three types [U.S. DEPARTMENT OF ENERGY-1987D] that would be candidates for TRUEX application: Plutonium Finishing Plant (PFP) waste, Neutralized Current Acid Waste (NCAW), and Neutralized Cladding Removal Waste (NCRW). All future tank wastes are stored in double-shell tanks.

b. Processing

Several alternative schemes have been proposed [U.S. DEPARTMENT OF ENERGY-1987D] for processing and disposal of wastes. The general scheme involving TRUEX is summarized by Fig. III-1. This scheme involves an initial retrieval of solids and slurries from the tanks by sluicing and transferring them to an on-site radionuclide-concentration facility [U.S. DEPARTMENT OF ENERGY-1987H]. Solid contents of the tanks would be separated from supernatant liquid and subjected to a washing operation.

The supernatant liquid together with liquids from washing of tank solids (salt wash) might be sent to operations for destruction of organic materials and removal of fission products. The fission products would be routed to the vitrification facility. The decontaminated salt wash would be converted to a grout for disposal [COLLINS, U.S. DEPARTMENT OF ENERGY-1987E].

The washed solids would be dissolved in an acidic solution. The resulting solution would be the feed to the TRUEX process. The transuranic elements isolated by this process would be sent to the vitrification facility, and the raffinate would be added to the decontaminated salt wash as grout feed.

Several options are considered for the scheme shown in

Fig. III-1:

- (1) The TRU isolated from the sludge could be additionally processed to recover fissionable material, or disposed of as vitrified HLW, or as a TRU waste in a low-cost form such as grout.

Fig. III-1. General Process to Remove TRU from Hanford HLW

- (2) The fission product removal step may not be required for some double-shell tank waste; it would be used as necessary to ensure that the concentration of fission products in grout would not exceed the limits defined for Class C LLW by 10 CFR 61.
- (3) Cesium would be removed from the salt solution in the FP removal step and sent to the vitrification facility. The intent of removing fission products from the salt wash is to make it possible to dispose of the decontaminated salt wash as a grout that would fall within the limits given for near-surface disposal [10 CFR 61], e.g., using the Transportable Grout Facility described in Appendix A [U.S. DEPARTMENT OF ENERGY-1987E, COLLINS].
- (4) Destruction of organics may be required for the TRUEX raffinate prior to disposal in grout.

The three types of future tank wastes are stored separately and could be processed separately if desired. Processing of combinations of the streams has also been considered [U.S. DEPARTMENT OF ENERGY-1987H]. The major differences in treatment of streams, according to current plans, are summarized as follows [WODRICH]:

- (1) Both NCRW and PFP waste would require sludge dissolution and TRUEX processing of the acidic solution. Fission-product removal operations will not be required.
- (2) For the NCAW, solids will be separated from supernate. The solids will be washed with water, and Cs-137 will be removed from the supernate by ion exchange. The water-washed solids and concentrated Cs-137 will be vitrified and the decontaminated supernate and water washes will be disposed of as an LLW grout.

3. Plans for Vitrification of Hanford Waste

Current plans call for starting vitrification operations in December 1999 [WODRICH]. According to this plan, the vitrification plant would have three main, separate feeds: the NCAW, GC, and PFP streams. Alternative plans also show neutralized cladding removal waste (NCRW) as a separate feed [WODRICH].

B. INEL High-Level Wastes

1. Composition of Wastes

Reprocessing of spent Naval fuels for DOE at the Idaho Chemical Processing Plant (ICPP) has resulted in the generation and accumulation of high-level waste on the INEL site. This reprocessing of Naval fuel is expected to continue, and the volume rate is expected to increase [BERRETH]. The entire fuel element, including hardware, is dissolved. As a result, inorganic solids derived from structural metals such as aluminum, stainless

steel, and Zircaloy accumulate in the wastes, thus increasing the volume of waste per unit of fuel processed and decreasing the concentrations of actinides and fission products in it. Because of the expected increase in the rate of reprocessing and the additional inorganic solids, the volumes of HLW generated at INEL are expected to eventually exceed those at SRP and Hanford [U.S. DEPARTMENT OF ENERGY-1988].

The acidic liquid wastes at INEL are typically stored for a short period before they are solidified by calcination. The product of calcination, a granular solid, is stored in steel bins at near-surface depths. It is expected that conversion of this calcine to a stable waste form and emplacement in a geologic repository will be required for long-term disposal. If TRUEX were applied to the wastes already in storage at INEL, it would be a part of this conversion.

2. Processing of Wastes

For this conversion, several processes have been evaluated in a selection program carried out at INEL [BERRETH, KNECHT, D.A.-1985]. The reference process involves conversion of the calcine to a glass-ceramic form to be managed as high-level waste [BAKER].

Alternatives to this plan involve separating the inert solids from the TRU elements and the fission products. The aims of such alternatives are similar to those mentioned earlier: (1) to concentrate the TRU elements into a smaller volume, thus decreasing the requirement for repository space, and (2) to isolate the relatively large volume of inert solids as a nonTRU LLW that could be disposed of by less costly techniques.

One alternative considered for separating the TRU from the inert solids involves neutralization of acidic HLW solutions [BERRETH]. Neutralization would result in concentrating the actinides into a precipitate whereas the fission products would remain in the supernate. The distribution of radionuclides between precipitate and supernate would be determined by the pH of the final solution.

Another alternative for separating the TRU from the inert solids in the INEL wastes involves the TRUEX process [BERRETH]. A general scheme for application of the TRUEX process is shown in a simplified diagram in Fig. III-2. This process involves retrieval of calcine from storage and treatment of it with nitric acid. Solids that do not dissolve in the acid would be routed to HLW immobilization, and the solution would be routed to the TRUEX process. If this alternative were applied, high-level liquid waste (HLLW) generated in the future would be processed without the intermediate conversion to calcine; thus, Fig. III-2 shows both calcine and HLLW as feeds to the process.

The actinides extracted by TRUEX from dissolved calcine and from HLLW would be routed to the immobilization process to produce solids that might be managed as either a TRU waste or a high-level waste. Application of fission-product removal operations to the TRUEX raffinate would produce (1) a high-activity fraction containing some of the fission products, notably Sr and

Fig. III-2. Alternatives for Processing INEL High-Level Waste

Cs, and (2) a low-activity fraction containing most of the nonradioactive inorganic salts. Fission-product separation operations are shown in Fig. III-2 by dashed lines to indicate that they are considered optional. The high-activity fraction would be routed to the immobilization process for management as high-level waste.

The TRUEX raffinate from processing the HLLW feed would be treated to nullify the hazard of fluoride ion by converting the raffinate to a slurry of slightly soluble calcium fluoride. The TRUEX raffinate from processing the calcine feed would be neutralized, resulting in a slurry of hydrated oxides. Both slurries would be neutralized and converted into a solid grout. The volume of liquid to be converted to grout and, therefore, the concentration of fission products in the grout would be determined by the entire suite of treatment operations, especially by the addition of reagents and by processing through an evaporator. The specific activity of this grout is further discussed in Sec. IV.B.

IV. DESTINATION OF IMMOBILIZED TRUEX RAFFINATE

A. Determination of Destination from Classification

Assigning the immobilized TRUEX raffinate to a waste class will be one of the most important determinants of which rules govern its disposal. This assignment will be based on the properties and characteristics of the immobilized waste, which are determined by processing methods. Furthermore, classification will have some bearing on which waste forms, packaging types, methods of transportation, and disposal unit designs will be acceptable.

The main theme of this section, that the classification of wastes is one of the most important determinants of their destination, is intended to be non-specific with respect to the site at which HLW is presently stored. It is convenient, however, to illustrate the discussion by using information on the properties of the wastes currently stored at Hanford and INEL. Because it is difficult to separate the general discussion from plans at a specific waste-storage location, it is important to summarize at the outset the current plan [WODRICH] for management of wastes at the Hanford Site:

- (1) The only waste classifications for disposal of defense wastes in double-shell tanks at the Hanford Site are HLW, LLW, and TRU wastes.
- (2) The LLW stream (TRUEX raffinate and salt solutions) will be equivalent to Class C (or lower) as defined in 10 CFR 61.
- (3) Generation of GTCC-LLW is not foreseen.
- (4) All HLW and TRU wastes will be disposed of in a geologic repository.
- (5) The LLW will be immobilized in grout for disposal in near-surface vaults.

B. Determination of Class from Concentrations

Based on the results of bench-scale demonstrations, the concentration of transuranic nuclides in TRUEX raffinate and in the grout prepared from it is expected to be less than the limit of 100 nCi/g for low-level wastes [10 CFR 61]. Thus, it might reasonably be expected that the low concentration of transuranic nuclides in TRUEX raffinate would permit its classification as LLW. The concentration of fission products, especially Cs-137 and Sr-90, in the TRUEX raffinate would also be important in its classification.

1. Grout from Hanford Wastes

The concentrations of fission products (notably Cs and Sr) in the final waste form can be estimated from the concentrations in stored wastes and the volume changes that occur when the waste is processed and converted to final waste form. The volume of final waste form (assumed to be grout for all non-high-level aqueous wastes) will be about three times that of the stored wastes. Thus, the concentration of a radionuclide in the grout will be about one-third of that estimated for the stored wastes.

Description of the concentrations in tank wastes has been complicated by variations in the plutonium-recovery processes in which they were generated and in the management of tank wastes. Furthermore, in some cases, the best quotable estimates of individual radionuclide quantities in the tank wastes are projections for the year 1995, assuming that the N reactor and PUREX plant would have been in routine operation up to that time. These estimates have been affected by the recent cancelling of N reactor operations and the consequent introduction of uncertainty into the schedule of PUREX Plant operations. Also, the estimated volume changes are complicated by the varying solids content of tank wastes.

The concentration of Sr-90 and Cs-137 in grout resulting from the processing of Hanford tank wastes by the scheme shown in Fig. III-1 was estimated using the estimates for their concentrations in stored wastes [SCHULZ, U.S. DEPARTMENT OF ENERGY-1987B] and the volume ratio described above. The results are given in Table IV-1 for the NCAW stream.

It is important to note that, in the general process scheme shown in Fig. III-1, the TRUEX raffinate would be added to an approximately equal volume of the decontaminated salt wash. Estimates of radionuclide concentrations in Table IV-1 have taken this dilution into account. Thus, although the main topic of interest here is the classification and destination of TRUEX raffinate, it is the destination of the combined streams that must be considered.

The characteristics of grout prepared from the Hanford NCAW waste stream can be estimated from the inventory of a canister of glass prepared from NCAW [U.S. DEPARTMENT OF ENERGY-1987B] and volume relationships between HLW and LLW streams. It is not expected that TRUEX will be applied in processing the NCAW stream, but this stream is useful in providing an upper-limit example because it is expected to have the highest concentration of radionuclides. Furthermore, these concentrations can be derived from the tabulated composition of a cylinder of glass prepared from the NCAW stream [U.S. DEPARTMENT OF ENERGY-1987B], using volume changes expected in waste treatment operations.

A limit of 4.9×10^9 Bq of Cs-137 per liter has been set for the feed to the Hanford Transportable Grout Facility [GIBSON], i.e., 0.132 Ci of Cs-137 per liter feed. Since the volume of grout is approximately 1.3 times the volume of liquid in the feed tank [U.S. DEPARTMENT OF ENERGY-1987E], the maximum concentration in grout would be 100 Ci of Cs-137 per cubic meter.

If the fission-product removal operations indicated in Fig. III-1 were used, the grout generated in this treatment scheme would be expected to have concentrations of individual radionuclides corresponding to those for a low-level waste of class A, B, or C.

If fission products are not removed, a determination must be made as to whether the concentrations exceed Class C limits. For mixtures of radionuclides, the class C limit [10 CFR 61] is obtained by a sum-of-fractions rule for long-lived and also for short-lived nuclides. This sum of fractions is calculated by dividing each nuclide's concentration by its Class C limit and adding the resulting values. If the sum exceeds 1 for either short- or long-lived radionuclides, the waste is Greater Than Class C. The fractions for the

Table IV-1. Estimated Concentrations of Radionuclides and Thermal Power for Grout That Would Be Generated if TRUEX Were Applied to Hanford NCAW Stream^a

Source of Grout	Total Grout Vol., m ³	Conc., Ci/m ³		Power	
		Cs-137	Sr-90	W/drum ^b	W/canister ^c
TRUEX Raffinate only	2x10 ⁴	2x10 ³ d	2x10 ³ d	5 ^d	20 ^{d,e}
TRUEX Raffinate plus Decontaminated Salt Wash	2x10 ⁴	1x10 ² f	1x10 ² f	4 ^{d,g}	10 ^{d,e,g}

^aIt is not expected that TRUEX will be applied in processing the NCAW stream, but this stream provides an upper-limit example because it is expected to have the highest concentration of radionuclides. Furthermore, these concentrations can be derived from the tabulated composition of a cylinder of glass prepared from the NCAW stream [U.S. DEPARTMENT OF ENERGY-1987B], using expected volume changes for waste-treatment operations.

^bAssuming grout is packaged in 55-gal drums for emplacement in a disposal facility.

^cAssuming GTCC-LLW must be emplaced in a geologic repository. Also, if this management mode were necessary, it is assumed that the grout form would be acceptable, but that it must be packaged in HLW canisters to be compatible with the transport and emplacement equipment of the repository.

^dBased on the assumption that all the radionuclides associated with HLW except the actinides are contained in the grout. These values were derived from tabulated contents for a canister of HLW glass prepared from the Hanford NCAW stream [U.S. DEPARTMENT OF ENERGY-1987B], a volume of 0.62 m³ of glass per canister [U.S. DEPARTMENT OF ENERGY-1987B], a volume reduction of 30:1 in the conversion of pretreatment plant feed to glass [U.S. DEPARTMENT OF ENERGY-1987B], and a volume increase of 53/38 in the conversion of grout plant feed to grout [U.S. DEPARTMENT OF ENERGY-1987E]. It was assumed that the volume of grout plant feed is about 70% greater than the liquid HLW that is fed to the pretreatment plant.

^eFor comparison, the maximum thermal power of HLW in glass form prepared from the Hanford NCAW stream without application of TRUEX is expected to be about 1160 W/canister, derived from data in [U.S. DEPARTMENT OF ENERGY-1987B]. The thermal power for HLW prepared for other Hanford waste streams would be much lower, about 50 W/canister [U.S. DEPARTMENT OF ENERGY-1987B].

^fBased on the limit of 4.9x10⁹ Bq of Cs-137 per liter of feed to the grout plant [GIBSON]; for this estimate, it was assumed that the concentration of Sr-90 was decreased to the same level. The estimate also used an increase of 53/38 in the conversion of grout plant feed to grout [U.S. DEPARTMENT OF ENERGY-1987E].

^gDerived from the bases described in footnote d except that the concentration of C-137 and Sr-90 is assumed to have been decreased to 1 x 10² Ci/m³ by fission-product removal operations.

concentrations of Cs-137 and Sr-90 listed in Table IV-1 are $2 \times 10^3/7000$ for Sr-90 and $2 \times 10^3/4600$ for Cs-137, giving a sum of about 0.7 or 70% of the Class C limit. Considering the uncertainty of the estimates, the possibility that the low-activity grout could exceed the Class C limit should not be discounted.

The concentrations of Sr-90 and Cs-137 in grout prepared from various Hanford waste streams other than the NCAW can be estimated from reported information on total quantities of radionuclides in streams, concentrations in streams, and volumes for stored wastes or volumes for feed to the grout plant. Final volumes of grout can be estimated from the volume changes expected to occur during processing and grout preparation. These estimates, which assume that no fission-product removal operations are used, are listed in Table IV-2.

Table IV-2. Estimated Concentrations of Radionuclides for Grout that Would Be Generated in Application of TRUEX (without fission product removal) to Hanford Waste Streams

Waste Stream	Grout Vol., ^a m ³	Conc., ^a Ci/m ³	
		Cs-137	Sr-90
CC	7.4×10^4	84	13-108
PFP	2.1×10^3	0	0
NCRW	9.8×10^3	5.3	1.5

^aEstimated volumes and concentrations from [WODRICH].

It is evident from Table IV-2 that grouts prepared from the Hanford waste streams listed there are unlikely to exceed the Class C limits for Sr-90 (7000 Ci/m^3) and for Cs-137 (4600 Ci/m^3), either individually or by the sum of fractions rule. Furthermore, other isotopes in these wastes are not present in sufficiently high concentrations to contradict this conclusion through the sum of fractions rule.

2. Grout from INEL Wastes

It has been suggested that the volume of grout generated in the processing of waste by a scheme including TRUEX similar to that shown in Fig. III-2 would be about $23,000 \text{ m}^3/\text{yr}$ [KNECHT, D.A.-1988]. For simple estimates of the specific radioactivity and specific power of the grout, this volume is assumed to contain all the radionuclides, except the actinides, that would have been contained in the 680 canisters/yr [BERRETH] of glass-ceramic HLW if TRUEX had not been implemented. Tabulated estimates of radioactivity and thermal power per canister for the glass-ceramic prepared from INEL high-level waste are available [U.S. DEPARTMENT OF ENERGY-1987B]. In the present work, estimates of radionuclide concentration were also made assuming that the Cs-137, Sr-90, and Tc-99 had been removed from the low-level fraction. These estimates are listed in Table IV-3.

Table IV-3. Estimated Concentrations of Radionuclides and Thermal Power for Grout that Would be Generated in Treatment of INEL High-Level Waste

Case	Conc., Ci/m ³		Power	
	Cs-137	Sr-90	W/drum ^a	W/canister ^{b,c}
TRUEX only	500	500	2	6
TRUEX plus FP-removal operations	d	d	2 ^e	5 ^e

^aAssuming grout can be packaged in 55-gal drums for emplacement in a low-level or intermediate-level disposal facility.

^bAssuming GTCC-LLW must be emplaced in a geologic repository. Also, if this management mode were necessary, it is assumed that the grout form would be acceptable, but that it must be packaged in HLW canisters to be compatible with the transport and emplacement equipment of the repository.

^cFor comparison, the thermal power for INEL HLW in glass-ceramic form without removal of inerts by either neutralization or TRUEX, derived from data in [U.S. DEPARTMENT OF ENERGY-1987B], would be about 340 W/canister.

^dLow concentration, unestimated.

^eAssumes that contributions of Sr-90 and Cs-137 have been reduced to negligible values by fission-product removal operations.

If the fission-product removal operations indicated in Fig. III-2 were used, the low-activity grout generated in this treatment scheme would be expected to have radionuclide concentrations corresponding to those for a low-level waste of class A, B, or C. Even without fission-product removal, the sum of fractions for the concentrations of Cs-137 and Sr-90 listed in Table IV-3 would be only about 0.2, or 20% of the upper limit for Class C low-level waste. If this waste were to be classified as GTCC low-level waste, its estimated volume of 23,000 m³/yr would make the estimated volume of 60 m³/yr for commercial GTCC low-level waste [U.S. DEPARTMENT OF ENERGY-1987G] seem small.

C. Determination of Class from Definition of HLW

The expected concentrations of nuclides and the definition of LLW given in 10 CFR 61 support the conclusion that, depending on the HLW stream and the treatment conditions, a grout containing TRUEX raffinate could be classified as either a low-level waste or a GTCC low-level waste. But an entirely different conclusion might be derived from some current definitions of HLW. Current definitions of HLW are not equivalent.

According to the NWPA, HLW is defined as:

- (A) The highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid

material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the Commission, i.e., the Nuclear Regulatory Commission (NRC), consistent with existing law, determines by rule requires permanent isolation.

The definition given in NRC regulations [10 CFR 61] is worded differently:

(1) Irradiated reactor fuel, (2) liquid wastes resulting from the operation of the first cycle solvent extraction system, or equivalent, and the concentrated wastes from subsequent extraction cycles, or equivalent, in a facility for reprocessing irradiated reactor fuel, and (3) solids into which such liquid wastes have been converted.

These definitions of HLW are thus based on the source of the material, i.e., fuel reprocessing, rather than on concentrations of radionuclides. On February 27, 1987, the NRC published in the Federal Register an "Advance Notice of Proposed Rulemaking" (51 FR 5992), announcing that it would consider the possibility of redefining HLW on the basis of concentration. Also, a detailed study of the matter resulted in proposals for redefinition of HLW based on concentrations [KOCHER-1987]. More recently, however, NRC [U.S. NUCLEAR REGULATORY COMMISSION-1988] proposed that the current definition of HLW should be retained.

The primary source of TRUEX raffinate that would be derived from HLW is the processing of irradiated fuels. Therefore, a narrow interpretation of these definitions might result in classification of solids formed from TRUEX raffinate as HLW because of their source. Recent developments, however, suggest that disposal of TRUEX raffinate will not be complicated by such an interpretation of the definition of HLW. In a presentation to the State of Washington in August, 1989, NRC indicated that the source-based definitions of HLW are not necessarily applicable to the low-activity wastes generated in the treatment of HLW. Rather, NRC suggests that these low activity wastes conform to the concept of incidental wastes, which are defined in 10 CFR 50, Appendix F. For supernatant or TRUEX raffinates that may have components that could require classification as HLW by the source base definition, treatment of the waste to remove additional components will be necessary to convert them to incidental waste and, therefore, LLW. The incidental waste concept has also been applied to solidified low-activity waste (Saltstone) resulting from processing of HLW at the Savannah River Plant [WODRICH].

Alternatively, because the TRUEX raffinate is a DOE-generated waste, its classification might be obtained from the definitions given in DOE 5820.2A:

High-Level Waste. The highly radioactive waste material that results from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid waste derived from the liquid, that contains a combination of transuranic waste and fission products in concentrations requiring permanent isolation.

Low-Level Waste. Waste that contains radioactivity and is not classified as high-level waste, transuranic waste, or spent nuclear fuel as defined by this order. Test specimens of fissionable material irradiated for research and development only, and not for the production of power or plutonium, may be classified as low-level waste, provided the concentration of transuranics is less than 100 nCi/g.

The phrase used in the DOE definition of high-level waste, "...and any solid waste derived from the liquid, that contains a combination of transuranic waste and fission products in concentrations requiring permanent isolation," may provide support for classifying TRUEX raffinate as low-level waste. According to this phrase in the DOE definition, TRUEX raffinate would be classified as HLW if the concentrations of TRU and fission products in it were high enough to require permanent isolation. Projected concentrations of immobilized TRUEX raffinates (Tables IV-1, -2, and -3) would probably not be considered high enough to require permanent isolation since (1) the TRU content will be less than the upper limit for low-level waste, and (2) the expected concentrations of Sr-90 and Cs-137 are much less than the values of 210,000 Ci Sr-90/m³ and 138,000 Ci Cs-137/m³ that have been suggested [FEHRINGER] as the limits above which permanent isolation would be necessary. From these considerations, the classification of TRUEX raffinate as LLW could be supported.

The classification of immobilized TRUEX raffinate may depend on the plan selected for disposal. For example, if the immobilized raffinate is to be disposed of separately from commercial wastes, DOE definitions may determine its class, thus strengthening the possibility of its classification as LLW. On the other hand, if it is to share a disposal facility with commercial wastes, the definitions given by NWPA or by 10 CFR 60 might have more influence.

D. Conclusions on Permissible Disposal Methods

A disposal scheme will ultimately be allowed or disallowed on the basis of a performance assessment. The performance assessment is a systematic analysis of the risks associated with a waste management system and a comparison of these risks to a set of performance objectives. Which agencies determine the methods and execution of the performance assessment will depend on which disposal alternative is put into practice. Alternatives that involve on-site disposal are expected to be determined mainly by DOE and its waste-generating contractor. Performance assessment for DOE low-level radioactive waste disposal sites has been described in detail [CASE]. Alternatives that involve other contractors and also interaction with commercial wastes might involve more organizations, such as NRC and EPA in the performance assessment. In any case, defining performance objectives, e.g., the limits for dose to a receptor from wastes, is the responsibility of EPA [GRUHLKE].

According to DOE 5820.2A, performance objectives will be set by an Oversight and Peer Review Panel. It is expected that the Panel would refer to the EPA standards as the performance objectives to be met. The performance assessment methods, i.e., analyses of the migration of nuclides from emplaced waste resulting in calculations of dose to receptors in the environment, would

be defined by committees appointed by either NRC or by DOE. The Panel carrying out a performance assessment for DOE wastes might be a team of "technical professionals" [CASE] under contract to DOE.

1. Disposal as LLW

Immobilized raffinates produced by application of TRUEX to defense HLW streams could be classified as LLW on the basis of concentrations of radionuclides. In that case, the raffinate could be disposed of in a near-surface facility, as a grout poured into pre-constructed concrete vaults similar to those planned for disposal of grout at Hanford [U.S. DEPARTMENT OF ENERGY-1987E] or saltstone at SRP [WILHITE]. Disposal by such means is described in more detail in Appendix A.

2. Disposal as GTCC-LLW

Immobilized raffinates produced from application of TRUEX to some defense HLW streams are expected to have radionuclide concentrations near, but not in excess of, the Class C limit. For the hypothetical case in which solidified raffinate did exceed the limit and was classified as GTCC-LLW, several disposal possibilities exist. One is the development of a disposal facility that would provide confinement intermediate between near-surface and repository disposal. Another is that, in the absence of an intermediate facility, disposal of GTCC-LLW in a geologic repository would be required. It has been pointed out [KOCHER-1989] that present laws and regulations do not prohibit disposal of GTCC-LLW in a near-surface facility; this disposal route, however, like any other, would only be permitted by approval obtained through performance assessment. Additional considerations of disposal of immobilized TRUEX raffinate as GTCC-LLW are examined in Appendix B. Appendix D, though not addressed specifically to TRUEX waste, also discusses some aspects of disposal as GTCC-LLW for wastes generated in the partitioning of HLW.

3. Disposal as HLW

a. Disposal as HLW in a Geologic Repository

A further hypothetical case is that, regardless of radionuclide concentrations, immobilized TRUEX raffinate could be classified as HLW because of its origin and the statements in NWPA and 10 CFR 60 that define HLW on the basis of origin. In this event, rulings that such wastes must be emplaced in a geologic repository could follow.

b. Disposal as HLW in a Facility Other than a Geologic Repository

It has been noted [KOCHER-1989], however, that present laws and regulations do not specify that a geologic repository must be used for disposal of HLW. The suitability of any facility for wastes classified as HLW would be indicated by performance assessment. Disposal of HLW related to possibilities for management of TRUEX raffinate is further explored in Appendix C. Appendix D also includes some general considerations on HLW disposal.

APPENDIX A

DISPOSAL OF IMMOBILIZED TRUEX RAFFINATE AS LOW-LEVEL WASTE:
THE TRANSPORTABLE GROUT FACILITY AT HANFORD AS EXAMPLE

The purpose of the Transportable Grout Facility at Hanford is to convert into a solid grout the low-level aqueous wastes from several sources on the Hanford site, including the salt solutions from processing tank wastes in the Radionuclide Concentration Facility [U.S. DEPARTMENT OF ENERGY-1987E]. These salt solutions would contain the raffinate from TRUEX processing. For each type of waste stream, the process (Fig. A-1) would consist of obtaining the necessary grout-forming powders, blending the powders, mixing the liquid waste with the powders, and pumping the resulting slurry into a disposal unit. The grout waste form would be prepared and emplaced in a series of campaigns, each of which would involve a batch of grout of sufficient volume to fill one of the disposal units. Plans call for carrying out five campaigns per year. The Transportable Grout Facility will consist of two sub-facilities: (1) a Dry Materials Receiving and Handling Facility, where the mixture of powdered grout-forming solids would be blended, and (2) the Transportable Grout Equipment [COLLINS], in which a slurry would be formed, by mixing the aqueous wastes with these blended powders, and pumped into the disposal units. The grout-forming solids would be delivered by truck or railcar to the Dry Materials Receiving and Handling Facility, where they would be stockpiled in preparation for each campaign. The liquid wastes would be accumulated in an underground storage tank of 3800 m³ capacity.

A grout formula effective for immobilizing the constituents of the waste would be developed for each waste type [GILLIAM, RICHMOND]. A typical grout mixture might consist of Portland cement, a pozzolana, clay, the liquid waste, and one or more chemical additives to enhance such characteristics as the ability to immobilize selected radionuclides, improve strength, and decrease leachability. The powdered constituents of the grout would be weighed individually and blended in the solid state. The blended powders would be pneumatically transferred to a silo for storage until needed. From this storage silo, the powders would be transported to the grout-preparation area by covered, hopper trucks, and there transferred pneumatically from the trucks to the Transportable Grout Equipment, in which the solids are mixed with liquid wastes. Weighing devices, with redundant instrumentation, would ensure that the solids are fed to the grout-mixing equipment at the prescribed rate. Liquid waste would be continuously pumped from the feed tank through encased pipe to the mixer. Flow meters, also with redundant instrumentation, would be used to assure that the flow rate of liquid wastes to the mixer is correct.

The slurry would then be pumped through encased pipe into a disposal unit. In earlier designs [KALDOR, TREAT], the disposal unit consisted of a simple trench in soil. In more recent designs (Fig. A-2), the disposal units are preconstructed concrete vaults [U.S. DEPARTMENT OF ENERGY-1987E]. Each vault would have sufficient volume to accommodate the 5300 m³ of grout obtained by solidifying the liquid waste from one filling of the 3800 m³ feed tank. Within a vault the grout would solidify and cure into a solid monolith. The monoliths would lie under at least 5 m of cover. These near-surface disposal units are to be located in the 200 East area of the Hanford Site.

Fig. A-1. Diagram of Process for Converting Low-Level Aqueous Waste to Grout at Hanford Site

Source: [U.S. DEPARTMENT OF ENERGY-1987E]

Studies to predict the long-term performance of the emplaced grout at Hanford have been carried out in the selection of grout formulation and design of disposal units [HUIZENGA]. The long-term containment of wastes is expected to be supported by the aridity of the climate and soils at the Hanford Site, a property that should favor the integrity of the monoliths and limit the mobility of the constituents of the waste.

Fig. A-2. Concept of Vault Proposed for Grout Disposal at Hanford Site

Source: [U.S. DEPARTMENT OF ENERGY-1987E]

APPENDIX B

DISPOSAL OF IMMOBILIZED TRUEX RAFFINATE AS GTCC LOW-LEVEL WASTE

This section describes possible management systems for the case in which a solidified TRUEX raffinate is classified as GTCC-LLW. The occurrence of this case is unlikely. Even without FP removal, radionuclide concentrations in a solidified raffinate are expected to be a little below the class C limit for the waste stream of highest radionuclide concentration, e.g., if the NCAW stream were to be processed by TRUEX (Sec. IV.B.1). Moreover, it is expected that if more accurate characterization of waste streams and process operation were to indicate that the raffinate would exceed class C limits, process operators would include a fission product removal operation to assure that the raffinate could meet Class C limits. Nevertheless, the hypothetical case of disposal of raffinate as GTCC-LLW is described here. A complete disposal system would include waste treatment (waste form preparation), packaging (optional), transportation (if necessary), and emplacement in a disposal facility.

If the immobilized TRUEX raffinate can be classified through the use of DOE definitions as a low-level waste, the presence of fission products at concentrations greater than those defined for Class C [10 CFR 61] would require that the raffinate receive special attention, according to DOE 5820.2A. Disposal systems for such waste must be justified by a specific performance assessment that would meet the requirements of the National Environmental Policy Act (NEPA) and with the concurrence of the Director of Defense Waste and Transportation Management for all disposal facilities within the purview of the Assistant Secretary for Defense Programs and of the Office of Terminal Waste Disposal and Remedial Action for those disposal facilities under the cognizance of the Assistant Secretary for Nuclear Energy.

Methods for disposal of GTCC-LLW are still being debated. It is commonly assumed that disposal will require some "intermediate facility," i.e., one providing confinement intermediate between that offered by a near-surface facility and that offered by a geologic repository. On the other hand, NRC has tentatively recommended that, unless an intermediate facility becomes available, GTCC-LLW should be disposed of in a geologic repository [U.S. NUCLEAR REGULATORY COMMISSION-1988]. It has been pointed out [KOCHER-1989] that there is nothing in present regulations that prohibits GTCC-LLW from being disposed of in a near-surface facility. Such action, of course, would be permitted only on the basis of performance assessment.

The Office of Technology Assessment (OTA) has recently evaluated the management of GTCC-LLW and made some recommendations. The OTA study [OTA] notes that under provisions of the Low-Level Waste Policy Amendments Act, DOE can accept commercially generated GTCC-LLW for storage at federal facilities. It also foresees the possibility that DOE may be called upon to accept commercial GTCC-LLW by a generator or, in special cases where public health is endangered, by the NRC.

The OTA study predicts that a permanent disposal facility may not be available for 20 years. It recommends, therefore, that the Department of Energy develop an "extended storage facility" for GTCC-LLW and, until it is ready, provide storage for this waste at one of DOE's existing facilities.

One of the most important problems, in OTA's view, is the need for generators to extend the time of interim storage for such waste. For many generators, the volume of wastes stored on-site is said to be approaching permissible capacity. Furthermore, the potential for degradation of waste and failure of packages during interim storage was cited. An additional concern is the need for off-site storage of wastes from generators who have discontinued operations.

The OTA study suggested that DOE should consider expanding an existing facility or constructing a new facility at one of its national laboratories. It also recommended that Congress require licensing of an "extended-storage facility" for commercial GTCC-LLW by the Nuclear Regulatory Commission to satisfy public concerns. But OTA foresees that before an appropriate disposal facility is constructed, some generators may need access to a few thousand cubic feet of storage capacity in an existing commercial or federal storage facility, particularly in the case of several thousand sealed sources that must be disposed of. The OTA study recommends, therefore, that limited access be provided at an existing, unlicensed DOE storage facility. That waste could then be transferred to a licensed facility when one becomes operational. This recommendation, however, seems to be at odds with a pending senate bill [S2443] which contains a short paragraph requiring that any facility used for storage or disposal of GTCC-LLW must be licensed.

The OTA study recognizes that some policy makers call for disposal of GTCC-LLW in a geologic repository, but recommends that DOE investigate alternative disposal technologies.

The Department of Energy has briefly described plans for carrying out its responsibility for the management of commercial GTCC-LLW [U.S. DEPARTMENT OF ENERGY-1987G], a responsibility with which it was charged by Public Law 99-240; however, DOE has not yet indicated its plans for management of DOE-generated GTCC-LLW. In fact, DOE recently invited proposals for development of alternative concepts for managing DOE-owned high activity wastes, indicating that planning is still at an early stage. Perhaps procedures for managing DOE-owned GTCC-LLW might eventually be derived from practices and proposals for managing high-activity low-level wastes, e.g., from demonstrations of Greater Confinement Disposal by DOE or from proposals for dealing with intermediate-level waste (ILW) by foreign countries. Additional descriptions of these are given in Sec. B.4.

Three general alternatives, diagrammed in Fig. B-1, seem obvious if it is necessary to manage immobilized TRUEX raffinate as a GTCC-LLW: (1) disposal at an on-site facility, (2) disposal at an off-site facility dedicated to DOE wastes, and (3) disposal at an off-site facility shared with commercial GTCC-LLW. For these alternatives, it is assumed that an intermediate disposal facility will become available. Characteristics of each of these three alternatives are described in Sec. B.1-B.3.

Fig. B-1. Alternatives for Disposal of Immobilized TRUEX Raffinate as GTCC-LLW

1. Disposal in On-site Facility

DOE Order 5820.2A recommends on-site disposal of LLW unless it is impractical. On-site disposal involves emplacement of the wastes in a facility on the DOE reservation where it was generated. This action should be the most economic; it avoids, at least, the costs associated with transportation of the wastes to another site. This option would probably also be preferred for efficiency; only one DOE contractor would be involved, and participation by other governmental agencies would probably be less than for all the other alternatives described below.

1.1 Preparation of Waste Form for On-site Disposal

For disposal of DOE-owned wastes on the DOE reservation where they were generated, the primary rules are assumed to be DOE 5820.2A. Plans for preparation of the waste form would be determined mainly by the contractor operating the reservation, the local DOE office, and the NEPA process. In this concept, selection of the waste form and writing of waste acceptance criteria would both be under the direction of the waste-generating contractor. Although systems involving management of GTCC-LLW must be evaluated by the NEPA process and, therefore, involve reviews, minimum participation by other U.S. agencies in the evaluation of on-site disposal is expected.

Selection of the waste form would be made on the basis of

- (1) obtaining a lower preparation cost than that for a vitreous solid and
- (2) satisfying performance objectives acceptable to an Oversight and Peer Review Panel appointed by DOE. Waste acceptance criteria devised by the generating contractor would also have to be acceptable to the Oversight and Peer Review Panel. The waste form would ultimately be selected through performance assessment. The waste form performance assessment method would be approved by the Oversight and Peer Review Panel. The performance assessment method would also generally consist of a pathway analysis to determine whether the concentration of radionuclides in environmental media at the boundary of the site would be less than the limits set by performance objectives. This performance assessment would be based on characteristics of both the waste form and the disposal facility.

Among the candidate waste forms for solidification of LLW, grout would probably be the first choice. Even though grout may serve as the general waste form for both packaged and unpackaged disposal, the formula for unpackaged disposal may require additives, e.g., clays to enhance binding of some radionuclides or silicates to enhance strength and leach resistance of the grout. The grout formulation may also have to include some absorbent to control the amount of free water in packages.

1.2 Emplacement of Unpackaged Waste Form in Intermediate Disposal Facility

In this concept, the waste form without packaging would be emplaced directly in the disposal facility. Designs for the disposal of saltstone at SRP [WILHITE] and grout at Hanford involve emplacement into preformed concrete structures rather than directly into earth. Thus, for GTCC-LLW, with much

higher concentrations of radioactivity, emplacement directly in the earth would probably be acceptable only at great depths in an arid environment. Ultimately, acceptance of any disposal design will be obtained only through a favorable evaluation in a performance assessment.

Preparation of the waste form in this concept would be similar to that proposed for the Transportable Grout Equipment Facility being designed for disposal of the salt wastes at Hanford (described in Appendix A). If near-surface disposal is prohibited for GTCC-LLW, then the top surface of the emplaced grout would have to be at a depth of >30 m. Feasibility will depend on local geology. If disposal of bare grout below the water table proves unacceptable in performance assessment, then this concept would probably be practicable only in arid regions of the U.S. This method requires transporting liquid waste from the TRUEX processing plant to the disposal site. Since the concept for disposal of grout at Hanford involves transport of liquid waste by pipeline (described in Appendix A), probably any concept for preparation of grout waste form on-site should assume transportation of liquid waste to the disposal area by pipeline.

If preparation of the waste form at the disposal site proves unacceptable, the alternative is conversion to solid form at the plant and transport of solid grout blocks to the disposal unit. For economy, unpackaged solids would be desirable. The possibility of dispersing radioactivity through abrasion and chipping of the solids in handling, however, would require the use of at least minimal packaging. Some site for interim storage of final waste forms is expected to be necessary. The solid waste would be transported from the plant to the disposal unit by truck, following guidance outlined in DOE 1540.1 and DOE 1540.2.

The treatment process and facility would be chosen for simplicity and economy, recognizing that the facility need remain reliable for only a limited lifetime, sufficient to process the TRUEX wastes. The design of a facility for mixing grout should be relatively simple; the manipulation of packaging adds some complexity. Additional equipment would be used for moving, lifting, and loading of the waste blocks or packages, and the design of blocks or packages would have to include handles or hooks for transfer operations.

For this disposal alternative, the waste form would be selected by waste acceptance criteria specified by the generating contractor and its associated DOE field organization. To limit leaching of radionuclides from LLW within disposal facilities by chelating agents contained in the wastes, their concentration in LLW must be limited [10 CFR 61]. For TRUEX raffinate a step may have to be included in the process to destroy residual organics that might contribute to leaching. The choice of grout formulation would be especially important, to minimize leaching, if it is to be emplaced unpackaged into the disposal unit. Perhaps a grout formulation might be chosen from among some of those that were developed for service in contact with soil [DOLE].

1.3 Packaging of Waste Form for On-site Disposal

This concept involves converting the salt stream containing the TRUEX raffinate to a solid, sealing the solid in containers, and transporting

the package to a disposal unit on the DOE reservation where the waste was generated. Solidification and packaging would be carried out at the TRUEX processing plant. If packaged grout were to be the final waste form, it would probably be prepared by mixing the aqueous waste and cement in a steel drum and allowing it to solidify therein.

Although emplacement without packaging (Sec. B.1.2) would be preferred for economy, there are several reasons why packaging might be required:

- (1) Even if the TRUEX raffinate is cast into blocks of grout, packaging may be required to prevent the dispersal of radioactivity through chipping or abrasion of the block surfaces and edges during handling and moving.
- (2) If transport of liquid waste from the TRUEX plant to the disposal unit is not permitted, the most likely alternative is solidification at the TRUEX plant and, as a result, the use of at least minimal packaging to protect the solid in handling, as noted in item 1.
- (3) Performance assessment may show that a high-integrity container is required, as an additional migration barrier in the disposal unit, to meet performance objectives.
- (4) Use of a container may be the most convenient way for casting solid shapes and also for grappling and handling operations.

Packaging shape, size, weight, handles, and hooks would also be specified by waste acceptance criteria developed by the waste-generating contractor and its associated DOE field organization.

1.4 Emplacement of Packaged Waste Form in On-site Facility

Emplacement of packaged waste forms would involve two main considerations: (1) package handling--the equipment and procedures to be used in transfer of packages into the disposal unit, and (2) shielding of operators until the packages are emplaced in the disposal unit. Shielding may be an important consideration if the concentrations of Cs-137 and Sr-90 in the stream bearing TRUEX raffinate are high. Moreover, equipment for remote emplacement might be necessary. Such equipment was used in the demonstrations of Greater Confinement Disposal (GCD) at the Nevada Test Site for the emplacement of packages with high radiation levels [WILLIAMS]. For emplacement of packages with high radiation levels, annular shields have been used; these shields accompany the package into the disposal unit and are then lifted back out of the disposal unit for re-use.

For high-activity low-level wastes, such as GTCC low-level wastes, the waste generator may opt to exclude lead shielding from waste packages, unless unusually high radiation levels make it necessary. The purpose would be to avoid the special requirements that are associated with disposal of mixed wastes. Shipment to burial sites of waste packages containing metallic lead has been common practice according to a recent survey [BOWERMAN].

Whether metallic lead exhibits Extraction Procedure (EP) toxicity (determined by leach tests) as defined by the EPA classification of hazardous characteristics has been discussed recently [BOWERMAN, NISHIMOTO]. A decision that metallic lead exhibits EP toxicity would result in requiring waste packages containing lead to be disposed of according to rules for mixed waste. Preliminary studies have indicated that lead bricks exhibit the characteristics of EP toxicity and, therefore, may be radioactive mixed waste [NISHIMOTO]. Perhaps the most important effect would be in determining the final disposal location for such packages.

Another recent evaluation [SISKIND] of whether a waste package containing metallic lead should be classified hazardous waste as defined by [40 CFR 261] considered both technical and regulatory factors. Two possible situations involving emplacement of metallic lead into a disposal facility were considered: (1) as a component of the waste and (2) as a shield for controlling radiation exposure of workers handling the wastes. This evaluation pointed out that the hazard represented by metallic lead depends on conditions inside the waste package, such as the reaction of other constituents of the waste with the lead to produce soluble lead compounds, and also on conditions outside the package, such as geologic characteristics that would affect the migration of lead in the environment of the disposal facility. Considering that these influences vary with the individual waste package and the disposal site, it was concluded that an assessment of the mixed-waste hazard would best be made on a case-by-case basis.

These considerations on exclusion of lead in the emplacement of packaged waste forms apply to the discussions in Sec. B.2.4, B.3.4, and C.4.

Depending on the design of the disposal unit, packaged wastes might be placed into the disposal unit onto earth, a layer of a migration barrier (e.g., special clays), or the floor of a concrete engineered structure. Similarly, after closure of the disposal unit, the package might be surrounded by earth, a migration barrier, or a backfill of grout. The latter arrangement was used in emplacement of intermediate level wastes in the French Tumulus concept (described in Sec. B.4.4); waste-filled drums were placed onto the underground floor of a concrete structure and then the entire array was filled with grout.

2. Disposal in Off-site Facility Dedicated to DOE Wastes

This concept involves an intermediate facility in which only DOE-owned wastes are emplaced. It involves disposal at some DOE reservation other than the one at which the wastes were generated, an action that would be taken only if disposal on the generating site is impractical. For GTCC-LLW, this action might be necessary if DOE decides that economy forbids the construction of an intermediate disposal facility on every site that generates GTCC waste. In fact, it seems to be a common opinion that cost would prohibit construction of more than one central facility. For disposal of DOE-generated wastes on a DOE reservation, the primary source of general technical guidance is assumed to be DOE 5820.2A. Environmental guidance would be derived from recommendations of an Oversight and Peer Review Panel appointed by DOE and from conformance to the NEPA process, with related document preparation and review. It is inferred from DOE 5820.2A that responsibility for disposal under this alternative would be assumed by the Assistant Secretary for Defense Programs,

assisted by the Director of Defense Waste and Transportation Management. Participation in environmental evaluation by other agencies would be more extensive for this alternative than for on-site disposal.

2.1 Preparation of Waste Form for Off-site Disposal in Facility Dedicated to DOE Wastes

In this disposal alternative, although the waste form would be prepared by the generating contractor, its identity and properties would have to be selected to conform to the waste-acceptance criteria developed by the contractor that operates the disposal facility. According to DOE 5820.2A, the development of large-scale waste-treatment facilities must be supported by NEPA documentation, in addition to (1) a document describing waste streams, (2) a construction design report that includes throughputs, treatment methods, and construction and operating costs, and (3) a safety analysis report. The waste form would be subjected to a performance assessment prepared by the DOE field organization that has within its cognizance the DOE reservation containing the disposal facility. Results of the performance assessment would be compared to the performance objectives listed in DOE 5820.2A. The development and application of performance assessment models would be monitored by an Oversight and Peer Review Panel consisting of personnel representing DOE, contractors, and other specialists in performance assessment. This panel would be selected by representatives of the Director of Defense Waste and Transportation Management, the Assistant Secretary for Environment and Health, and the field office. After emplacement of the waste, monitoring measurements would be made to validate or modify the models used in performance assessments. Furthermore, the waste generator would be expected to carry on a certification program to provide assurance that the waste acceptance criteria are being met. Also, the waste generator would be subject to a periodic audit by the operator of the disposal facility to which the waste is being sent.

2.2 Packaging for Off-site Disposal in Facility Dedicated to DOE Wastes

If the treated waste is to be transported to another DOE reservation for disposal by another DOE contractor, the generating contractor will be obliged to seal the solidified waste into packages that are acceptable both for transportation and for emplacement in the disposal unit.

For transportation, the acceptability of packaging would be determined by guidelines in DOE 5480.3, which, in turn, refers to requirements stated by the Department of Transportation (DOT) [49 CFR 171-178] and by the Nuclear Regulatory Commission [10 CFR 71]. Carrying out the packaging operations in compliance with waste acceptance criteria and transportation regulations would be the responsibility of the generating contractor, but the packaging operations would be subject to periodic inspection by the receiving contractor.

For disposal, acceptable package parameters would be defined in waste acceptance criteria written by the receiving contractor. The waste acceptance criteria can be expected to present several specifications for containers, e.g., size limits, material, lifetime, strength, impact resistance, and weight.

Interim storage of packaged wastes, at the generating site before transportation and at the receiving site before emplacement in the disposal unit, would be likely. Accordingly, the packages may also have to meet some requirements related to interim storage.

2.3 Transportation for Off-site Disposal in Facility Dedicated to DOE Wastes

For transportation of DOE wastes, the primary basis for regulation would be DOE Order 5480.3. This order requires compliance with regulations of the DOT [49 CFR 171-178] and also with regulations of the NRC [10 CFR 71].

The hardware necessary to carry out the transport of these wastes includes the vehicles, packaging, shielding, overpacks (if necessary), and handling and grappling equipment to permit loading and off-loading. The type and size of packaging that would be permitted for transportation can be inferred from DOT regulations and the estimated concentrations of radio-nuclides in the wastes. The need for shielding would be indicated by dose calculations and regulations on dose to operators and the public. Overpacks may also be required, both for additional shielding and for protection of the packagings in case of accident.

Routing would be selected by a model that indicates, for transportation of radioactive materials, which roads result in minimum environmental impact [JOY].

Managing and correlating the transportation operations would be the responsibility of the waste-generating contractor, making use of either DOE vehicles or those of a private transportation contractor, as feasible.

Payloads for such operations would be limited to 22,000 kg per vehicle. If it is assumed that the density of grout is 3 g/cm^3 and the payload is about 70% grout, then about 5 m^3 of grout would be carried in a truckload. This latter volume, together with the estimated volumes of grout, can be used to estimate the numbers of truckloads required to transport the TRUEX raffinate in the form of grout. Although this is not the disposal scheme called for by any official waste-management plan, these numbers are used for various Hanford waste streams to provide examples as listed in Table B-1.

Table B-1. Numbers of Truckloads Required for Transportation if Grout Containing TRUEX Raffinate from Treatment of Hanford Wastes Not Disposed of On-site

Waste Stream	Grout Volume, m^3	Truckloads, Number
CC	7.4×10^4	1.5×10^4
PFP	2.1×10^3	4.2×10^2
NCRW	9.8×10^3	2.0×10^3

2.4 Emplacement in Off-site Intermediate Facility Dedicated to DOE Wastes

Emplacement would be limited to packages meeting the waste acceptance criteria. The criteria will have been developed by the receiving contractor in cooperation with the DOE field organization whose purview includes the reservation containing the disposal unit.

Operations performed by the receiving contractor would include off-loading of packages, perhaps interim storage of packages, movement of packages to the disposal unit, emplacement of packages into the disposal unit, and closure of the disposal unit.

All shielding would probably be removed from packages as they are lowered into the disposal unit. This practice would save costs and avoid the complications of including the hazardous material normally used for shielding, lead, in the disposal unit (see Sec. B.1.4). For immobilized solids prepared from the most radioactive streams, shielding might have to be employed, however, to protect disposal operators during emplacement operations. The largest exposures at any stage of the disposal of these wastes might be experienced by the disposal operators. Exposure would have to be limited to those specified in DOE Order 5820.2A. In fact, remote handling equipment might be needed to emplace drums. Such equipment was developed and used in the demonstration of Greater Confinement Disposal at the Nevada Test Site [WILLIAMS].

Design of the disposal unit would be determined by the receiving contractor and the associated DOE field organization. The main design parameters include shape, depth, and engineered barriers. Design parameters for both the disposal unit and the container would have been selected from proposed alternatives through performance assessments conducted by the receiving contractor and the field organization.

3. Disposal in Facility Shared with Commercial GTCC-LLW

In discussions of DOE's responsibility for receiving and disposing of commercially generated GTCC low-level wastes [SMILEY; KNECHT, M.A.-1987; U.S. DEPARTMENT OF ENERGY-1987G], there has been no indication of whether DOE-generated GTCC-LLW and commercially generated GTCC-LLW would be emplaced in the same disposal facility. There are several indications that separate plans might be made for disposal of GTCC-LLW from these two sources:

- (1) The response of DOE to PL 99-240 outlined plans for management of all GTCC-LLW except that owned and generated by DOE [U.S. DEPARTMENT OF ENERGY-1987G; SMILEY; KNECHT, M.A.-1987; KNECHT, M.A.-1988].
- (2) The responsibility of DOE for commercial GTCC-LLW might require development of a management system on a shorter schedule than DOE-owned wastes; DOE's response to PL 99-240 [U.S. DEPARTMENT OF ENERGY-1987G] declared intentions of being ready to receive commercial GTCC-LLW, at least for storage, within two years. There may be a wider interest in disposal of commercial GTCC-LLW than in disposal of DOE-generated GTCC-LLW that would result in a higher priority for resolution of the commercial problem.

- (3) The NRC proposal to amend 10 CFR 61 [U.S. NUCLEAR REGULATORY COMMISSION-1988] discusses the management of GTCC-LLW but does not specifically mention wastes generated by DOE.
- (4) Commercial GTCC-LLW may be managed by a different DOE undersecretary than DOE-generated waste.

Nevertheless, some reasons for DOE wastes sharing a facility with commercial wastes are recognized:

- (1) The cost of constructing more than one intermediate disposal facility may be too great. The Nuclear Regulatory Commission has suggested that the estimated volume ($2 \times 10^3 \text{ m}^3$) of commercial GTCC that would accumulate by the year 2020 may be too small to warrant the construction of even one central intermediate facility and that, therefore, GTCC wastes might better be disposed of in a geologic repository [U.S. NUCLEAR REGULATORY COMMISSION-1988].
- (2) The NEPA process that must accompany the development of a system for managing GTCC low-level waste might show that, to limit environmental impact to a level consistent with standards [40 CFR 193], the number of intermediate disposal facilities should be minimized or restricted to one.

Although a shared disposal facility may provide lower costs and reduced environmental impact, efficiency may be reduced because the involvement with commercial wastes would increase the number of organizations participating in regulation of the facility.

3.1 Preparation of Waste Form for Off-site Disposal in Intermediate-Level Facility Shared With Commercial GTCC-LLW

The solid waste form to be used in disposal would be prepared by the waste-generating contractor, but its formulation would probably be determined primarily by the waste acceptance criteria of the receiving contractor and its associated DOE field organization. The selection of waste form would be determined from results of a performance assessment, which would be defined and carried out by the contractor and DOE field organization associated with the reservation containing the disposal facility. Perhaps the main difference between this alternative and the one described in Sec. B.2.1 is that the association with commercial wastes would require NRC involvement in performance assessment and licensing of the facility. Selection of the waste form also would probably be influenced by NRC.

3.2 Packaging for Off-site Disposal in Intermediate-Level Facility Shared with Commercial GTCC-LLW

The selection of packaging parameters suitable for transportation and also for disposal would be guided by the rules described in DOE 5480.3. Because of the association with commercial wastes in this alternative, however, the NRC licensing process may have an influence on the package design. Development and procurement of packaging compliant with the guidelines would be the responsibility of the waste-generating contractor.

3.3 Transport for Off-site Disposal in Intermediate-Level Facility Shared with Commercial GTCC-LLW

The DOE guidelines for transportation of radioactive materials, given in DOE Order 5480.3, require compliance with regulations on transportation of radioactive materials given by NRC [10 CFR 71] and the DOT [49 CFR 100-178]. The participation of NRC in the licensing of the disposal facility, however, might be expected to result in a more active role by NRC in evaluation of transportation procedures.

3.4 Emplacement in Intermediate-Level Facility Shared with Commercial GTCC-LLW

Procedures, equipment, and guidelines for emplacement operations at a shared disposal facility should be the same as those for a facility dedicated to DOE wastes as described in Sec. B.2.4. The DOE wastes might be separated from commercial wastes, by location within the same disposal unit or by emplacement in different disposal units; waste-emplacement procedures are expected to include recording of the location of packages within a disposal unit.

4. Disposal Unit Designs for Emplacement of GTCC-LLW

If TRUEX raffinate were to be handled as GTCC-LLW and to be disposed of in an "intermediate disposal facility," the design for such a facility would have to be established. Disposal facilities that have been designed to provide lower risk than shallow land burial can be considered to indicate concepts that might be adapted for the disposal of GTCC-LLW. Such facilities include those used in demonstrations of Greater Confinement Disposal (GCD) in the U.S. and also those used in demonstrations or concepts for disposal of ILW in foreign countries. Facility designs and environmental risks associated with GCD concepts have been reviewed [GILBERT-1985, GILBERT-1986A, GILBERT-1986B]. Some of the concepts of GCD and also foreign facilities are briefly described here in Sec. B.4.1-B.4.11.

4.1 Demonstration of Greater Confinement Disposal at Savannah River Plant

Some demonstrations of GCD have been carried out at a humid site, the Savannah River Plant (SRP) [COOK], where the thickness of the unsaturated stratum restricted the shaft depth to about 9 m. The suggested layout involved 40 shafts in a rectangular array within an area of 27 x 41 m [TOWLER]. This arrangement would provide a minimum distance between shafts of about 5 m. The cross-sectional dimensions of the waste layer, with a depth of 3 m to the top of the waste layer and a waste layer thickness of 6 m, were about the same as the traditional shallow land burial trench.

The fiberglass liner of the SRP augered shaft design has a thickness of about 1.3 cm, a length of 6 m, and an inside diameter of about 2.1 m. This liner has several functions: it provides a low-permeability secondary container, a supportive casing for the shaft walls, and structural support for the geological stratum in which the wastes are emplaced. A monitoring well is

provided by fixing a pipe in the annulus between the liner and the wall of the shaft. The cross section of the SRP shaft is diagrammed in Fig. B-2.

Fig. B-2. Cross Section of the SRP Shaft

Source: [U.S. NUCLEAR REGULATORY
COMMISSION 1984]

In the construction of the SRP shafts, a 3-m diameter hole is drilled to a depth of about 11 m. The hole is backfilled with a layer of gravel. A section of pipe, the length of the hole, is placed at the wall of the hole to provide a monitoring well. Wet grout is pumped into the hole, and a fiberglass liner is forced down into the grout. Wet grout is then poured into the space between the liner and earthen wall to form a grout annulus with a thickness of about 0.3 m outside the fiberglass liner. A steel collar is placed onto the top of the grout annulus to keep the hole open. The collar opening is closed with a steel cover when the shaft is not in use.

Bundles of waste-filled drums on a pallet are lowered into place with a crane. Alternating layers of waste-containing packages and wet grout are put down the shaft. Finally, all spaces between the packages are filled with grout. After the liner has been filled with layers of waste and grout, the steel collar and cover are removed for reuse, and the hole is backfilled with grout.

A recent status report [HARLEY] indicated that 20 boreholes have been constructed at SRP; a number have been filled and closed. Others are in the process of being filled.

The conceptual engineered trench proposed for GCD at the Savannah River Plant is illustrated in Fig. B-3. In this concept [TOWLER], an excavation would first be dug to a depth of 3 m below grade. From this level, a disposal trench would be constructed by driving sheet piling or injecting concrete walls into the ground around the perimeter and then excavating an area, within these walls, of 15 x 30 m to an additional depth of 6 m. The disposal trench might also be divided into four cells by sheet pile walls. Each cell would have a concrete bottom and an individual drainage collection system and would be provided with a removable cover to reduce the amount of precipitation reaching the cell bottom before final closure. Containers of waste would be placed in the cells and, after completion of a layer, grout would be poured into the voids between the containers. A recent status report indicated that one concrete-lined trench is in place at SRP [HARLEY].

4.2 Demonstration of Greater Confinement Disposal at Nevada Test Site

Some GCD demonstrations were carried out at an arid site, the Nevada Test Site (NTS) [BOLAND, DICKMAN], where the thickness of the unsaturated stratum permitted a shaft depth of about 37 m.

In the construction of the NTS shafts, a pilot hole for the disposal shaft is excavated. This hole, with a diameter of 4 m and depth of 1.2 m, is cased with corrugated metal pipe. Pilot holes for satellite monitoring wells are also excavated. These pilot holes, each with a diameter of 0.9 m, are also cased to a depth of 1.2 m. The concrete pad that supports the openings of all pilot holes for the disposal shaft and monitoring wells is constructed by laying forms and steel-reinforcing rods into position and by pouring a 0.2-m thick layer of concrete. When the concrete pad has cured sufficiently to bear heavy equipment, the disposal shaft of 3-m diameter is drilled within the pilot hole to a depth of 37 m. The monitoring wells are also drilled within

Fig. B-3. Steps in Construction of the GCD Trench at SRP

Source: [U.S. NUCLEAR REGULATORY COMMISSION-1984]

their pilot holes, each with a diameter of 0.6 m, to the same depth as the disposal shaft. An 18-m length of 1.8-m diameter pipe is placed into the shaft to direct the downflow of all materials that are added subsequently. The primary purpose of this pipe is to protect monitoring leads from damage by gravel and waste packages that are later placed down the shaft. Because monitoring instruments in the disposal shaft may be judged unnecessary in routine application, this pipe may not be used in routine service. The final step in construction of the NTS disposal unit is the emplacement of a layer of screened gravel of 6-m thickness on the bottom of the shaft.

Wastes are lowered into the shaft by an all-terrain crane with a capacity of 16,000 kg. This crane is equipped with a remote-controlled boom and grapple; the control panel for the remote equipment can be operated at a distance of up to 150 m from the source. The grapple is equipped with a tilt-pan-zoom video camera that permits remote viewing of the grapple, surroundings, and downshaft operation. After waste packages have been emplaced in the NTS augered shaft, they are covered with an 18-m thickness of backfill [REYNOLDS].

The relatively high ratio of waste volume to surface area committed by the NTS shaft yields a high land-use efficiency. This design (Fig. B-4)

Fig. B-4. Design Details of the Waste-Disposal Shaft Constructed at NTS

Source: [U.S. NUCLEAR REGULATORY COMMISSION 1984]

involves a square concrete pad, 19.5 m on an edge, at the surface of each shaft. This pad supports the heads of a number of monitoring wells; a smaller pad may be possible in routine application because of fewer monitoring wells. The minimum distance between disposal units (shafts) would presumably have the same dimension as the edge of the concrete pad surrounding the shaft, i.e.,

19.5 m. A liner is not used in the NTS augered shaft except for a 1.2-m length of 4-m diameter corrugated metal pipe at the top that is initially used to keep the pilot hole open. During waste-emplacement operations, the NTS demonstration shaft employs a caisson of 18-m length to protect the wiring located at the walls of the shaft leading to bottom monitors. Perhaps this may be used only in initial demonstrations. The wastes are passed down inside the caisson, whereas the wiring for monitors is located outside its walls. The floor consists of the earthen bottom of the shaft covered with 6 m of screened gravel [REYNOLDS]. For the NTS demonstration, depth to the top of the waste layer is 18 m, and the thickness of the waste layer is 12 m.

Nine satellite monitoring wells are used around each NTS shaft to characterize the migration of nuclides in the demonstration. Depending on the geology, perhaps fewer monitoring wells would be necessary in routine applications of this disposal method at some other site. Electrical leads that connect the detectors in the monitoring shafts with the readout point are located in PVC pipe laid under the concrete pad that surrounds and supports the shaft openings.

Disposal of high-activity LLW in augered shafts requires some special supporting equipment not normally employed in near-surface disposal operations. Probably the most important special piece of equipment for handling wastes of high irradiation levels is the waste-emplacement crane. Provisions for remote handling and remote viewing and the skills and equipment to maintain the mechanical, electrical, and electronic parts of this special crane and its auxiliaries would also be important. In the NTS demonstration, transfer of wastes with high radiation levels is controlled and viewed from behind shielding. Additional shielding is provided in high-radiation fields by a thick earthen berm between the shaft and the control point [REYNOLDS].

The status of the demonstration of GCD at the Nevada Test Site was summarized recently [DODGE]: DOE/NV has authorized the use of GCD methods at the NTS area 5 Radioactive Waste Management Site (RWMS) since 1984. Greater Confinement Disposal as practiced at NTS involves placing waste in 3-m-diameter, 37-m-deep boreholes. The bottom 15 m of the hole is filled with waste and the upper 22 m is backfilled with soil. The following guidelines for low-level waste to be placed in GCD were established by DOE/NV: (1) waste defined by [DOE Order 5820.2A] as not being suitable for shallow land disposal, e.g., Greater-Than-Class C waste, remote-handled high-specific-activity (HSA) waste; (2) volatile or mobile radionuclides, e.g., HSA tritium; and (3) low-level waste that would exist in a concentration greater than 0.1 mCi/g when decayed for 100 years. Greater Confinement Disposal can also be used for waste classified LLW at the request of the generator; DOE/NV is currently undertaking a performance assessment as specified by EPA [40 CFR 191] to see if GCD as practiced at the NTS is acceptable for the disposal of transuranic waste that cannot be certified for disposal at the Waste Isolation Pilot Plant.

At present, there are six GCD boreholes at the Area 5 RWMS designated for unclassified LLW. One borehole has been filled with HSA tritium-contaminated waste. Another is three-quarters full, mostly with HSA tritium waste. A third borehole is about half full with HSA fuel decladding

waste that must be remotely handled. The remainder of that borehole and one empty borehole are being reserved for more remote-handled waste. Two other empty boreholes are reserved for contact-handled waste such as the HSA tritium.

4.3 Waste Classification in Other Countries

Sections B.4.4-B.4.10 describe techniques and plans of other countries for disposing of wastes that may be comparable to GTCC-LLW. Thus, a brief preface on waste classification in other countries seems appropriate. It is assumed that the GTCC-LLW classification overlaps the waste categories that are generally labelled ILW or MLW in other countries. Definitions of waste classification in several countries are described in this section.

4.3.1 International Atomic Energy Agency

In 1970, the IAEA published a proposal for standardization of radioactive waste categories [IAEA-1970]. This classification scheme, however, was not adopted; instead, each country, and in some cases even each nuclear site, tended to develop some classification system of its own [IAEA-1987]. A subsequent definition of waste categories by the IAEA [IAEA-1981] was "based on how the wastes must be handled or disposed of in view of their radionuclide content" [IAEA-1987]. These later definitions, although introduced with the admission that they were qualitative and based on waste characteristics that may vary, have been subsequently quoted in IAEA documents over a period of several years. The waste categories were defined as follows [IAEA-1981]:

Low-Level Waste

Those wastes which, because of their low radionuclide content, do not require shielding during normal handling and transportation.

Intermediate-Level Waste (Medium-Level Waste)

Those wastes which, because of their low radionuclide content, require shielding but need little or no provision for heat dissipation during their handling and transportation.

High-Level Waste

(i) The highly radioactive liquid, containing mainly fission products, as well as some actinides, which is separated during chemical reprocessing of irradiated fuel (aqueous waste from the first solvent extraction cycle and those waste streams combined with it).

(ii) Any other waste with radioactivity levels intense enough to generate significant quantities of heat by the radioactive decay process.

(iii) Spent reactor fuel, if it is declared a waste. Intermediate-level and low-level wastes were further qualified by the terms short-lived and long-lived, according to the following definitions:

Long-Lived Waste

Those wastes that will not decay to acceptable activity levels in time periods during which administrative controls can be expected to last.

Short-Lived Waste

Those wastes that decay to acceptable activity levels in time periods during which administrative controls can be expected to last.

4.3.2 Belgium

In Belgium, "Medium-level waste (MLW) is defined by a beta-gamma activity that requires shielding protection during handling, transportation and storage, and by the same requirements on alpha activity as for LLW" [ZACCAI]. "Low-level waste (LLW) is currently defined by a beta-gamma activity that does not result in radiation levels requiring shielding and remote operation and by an alpha activity desired from the limit value for final disposal and the volume reduction factors obtained during processing" [ZACCAI].

It should be noted that "...in the absence of a well-defined final disposal route, no official radioactive waste classification has been adopted in Belgium. However, in practice, a distinction is made between 'non-geological' waste and 'geological waste'" [INTERNATIONAL ENERGY]. It seems reasonable to assume that medium-level wastes belong to the Belgian category termed "geological waste" because they would have radioactivity levels in excess of the limits for "non-geological wastes." Therefore, medium-level wastes are presumed to be those having concentrations of beta-gamma emitters and alpha emitters of more than 10 mCi/ton and having surface dose rates of more than 50 rad/h. Furthermore, medium-level wastes must have a surface dose rate less than 1000 rad/h, above which the waste is considered to be high level.

4.3.3 Canada

Although non-high-level wastes are referred to as LLW and ILW in Canadian publications, they apparently have not been quantitatively defined. "The AECB (Atomic Energy Control Board) plans to treat requests for storage or disposal on a case-by-case basis" [INTERNATIONAL ENERGY].

4.3.4 Federal Republic of Germany

In the Federal Republic of Germany, different types of waste (LLW, MLW, and HLW) are to be disposed of in separate areas of the Gorleben salt dome repository. It is reported [INTERNATIONAL ENERGY], however, that "no specific methods have been identified for distinguishing between high-level waste and non-high-level waste in the FRG."

4.3.5 France

A recent review of regulatory practices [INTERNATIONAL ENERGY] in France states that "CEA'S General Program for Radioactive Waste Management defines three categories of waste: Category A wastes, commonly

known as low-level and medium-level wastes (beta-gamma); Category B wastes containing isotopes of over 30 years half life (generally alpha emitters); and Category C wastes, containing fission products and heavy elements arising from reprocessing of spent fuel." Although medium-level wastes occupy locations separate from low-level wastes in disposal facilities [VAN KOTE], quantitative boundaries on their characteristics have apparently not been defined.

4.3.6 Sweden

It is reported [INTERNATIONAL ENERGY] that "...no specific activity levels have been defined in Sweden to distinguish between low-, intermediate-, and high-level waste. In general, IAEA definitions are used. SKB (Swedish Nuclear Fuel and Waste Management Company) distinguishes between the three categories of waste for determining handling requirements. Low-level waste can be handled and stored in simple packages, without special protective measures. Intermediate-level waste must be radiation-shielded, and high-level waste must be both radiation-shielded and cooled for a certain period to allow safe disposal."

4.3.7 Switzerland

Regarding waste categorization, the "Swiss authorities define high-level waste as that which remains from the spent fuel after plutonium and uranium have been separated out through reprocessing. This waste material initially is a liquid and is subsequently vitrified for the purpose of disposal. All other wastes are considered low- or intermediate-level waste, with low-level waste having a surface dose rate of the unshielded waste package of less than 0.2 rem per hour. The Swiss also distinguish alpha-emitting wastes in the low- and intermediate-level waste categories, but do not set quantitative limits on such" [INTERNATIONAL ENERGY].

4.3.8 United Kingdom

Specific categories of waste have been defined in the United Kingdom [INTERNATIONAL ENERGY] as follows:

Low-Level Wastes

Wastes containing radioactive materials not exceeding 4 GBq/ton alpha activity (about 100 mCi/ton) or 12 GBq/ton beta-gamma activity (about 300 mCi/ton).

Intermediate-Level Wastes

Wastes with levels of activity exceeding the limits for LLW, but not requiring account to be taken of the heat generated by radioactive decay in the design of storage or disposal facilities.

High-Level Wastes

Wastes in which the temperature may rise significantly as a result of their radioactivity, so that this factor has to be taken into account in designing storage or disposal facilities.

4.4 Disposal of ILW in France

Wastes have been disposed of in earth-covered tumuli (Fig. B-5) in France since the 1960s [VAN KOTE]. More than $1.7 \times 10^6 \text{ m}^3$ of waste materials were emplaced between 1969 and 1982 [BENNETT]. Typically, $10,000 \text{ m}^3$ of waste is disposed of in one disposal unit covering an area of 3000 m^2 . The French tumulus design includes a below-grade portion and an above-grade portion. The below-grade portion is used for disposal of ILW and is, therefore, of most interest here.

Fig. B-5. Structure of the Tumulus Disposal Unit Used in France

Source: [U.S. NUCLEAR REGULATORY COMMISSION 1984]

Wastes with intermediate levels of radioactivity and penetrating radiation are emplaced in the below-grade portion of the French tumulus system. The below-grade portion is a compartmented structure of poured reinforced concrete walls. After each compartment is completely occupied by waste, it is backfilled with grout. Some of the details of the drainage devices of the below-grade portion of this disposal system are shown in Fig. B-5, including floor drains that feed into a sump from which collected water can be monitored, pumped, and treated if necessary before discharge.

The above-grade portion of the tumulus is created at the level of the surrounding grade to form the floor of the surface structure. Walls are then built around its edge by stacking preformed concrete cylinders. Low-level wastes are also encapsulated in the concrete of these cylinders. Containers of low-level wastes are stacked between these walls, and all voids are backfilled with a free-flowing gravel. This type of backfill is intended to promote drainage if infiltration occurs. When the above-grade portion is filled, the

entire disposal unit is covered with clay and then topsoil, which is seeded to produce a vegetative cover.

4.5 Disposal of ILW in Finland

Plans for managing LLW and ILW from the two nuclear power plants in Finland, at Okiluoto and at Loviisa, involve disposal into caverns created in bedrock. Each reactor will have an on-site disposal facility or repository according to the descriptions of the Finnish plans [HARKONEN]. This type of disposal facility was necessary because the geologic conditions in Finland are considered unsuitable for shallow land burial.

At Loviisa (Fig. B-6), the depth of the repository will be about 120 m. The geologic medium in which it lies is described as a homogeneous granite. Solidified wastes will be emplaced in concrete containers within a concrete structure in a cavern with a horizontal length of 100 m and a cross section of 300 m². Waste containers will be emplaced with a remote bridge crane. After containers are in place, the spaces between them will be filled with flowing concrete. At closure, spaces between the concrete structure and the bedrock will be filled with crushed rock.

Fig. B-6. Layout of the LLW-ILW Repository at Loviisa, Finland

Source: [HARKONEN]

At Okiluoto (Fig. B-7), the geologic medium is described as an intact tonalite rock body surrounded by more fractured micaceous gneiss. The depth of the access tunnel and vault in the repository will be about 60 m. The wastes will lie at greater depths; two silos will be excavated into the floor of the vault, into which LLW and ILW will be placed separately. The silo that will hold ILW will have a diameter of 22 m and a height of 26 m. Furthermore, the silo for ILW will be cased with reinforced concrete walls with an inner diameter of 19 m and a thickness of 0.5 m. At closure, a reinforced concrete cap will be cast on the ILW silo and the cap will be covered with crushed rock and boulders. All spaces between the concrete walls and the surrounding granite will be filled with crushed rock.

Fig. B-7.

Layout of the LLW-ILW Repository
at Okiluoto, Finland

Source: [HARKONEN]

4.6 Disposal of ILW in Switzerland

In Switzerland, caverns are to be mined into rock for the disposal of LLW and ILW [ROMETSCH]. The design is still at the concept stage (Fig. B-8). An extensive evaluation of about one hundred locations including

Fig. B-8. Conceptual Layout for a LLW-ILW Repository
in Switzerland

Source: [ROMETSCH]

several types of host rock has been carried out since 1978 in the process of selecting sites for these repositories. The alternatives have been reduced to three primary candidate sites with three types of host rock: anhydrite (anhydrous calcium sulfate), marl, and alpine gneiss. Several other sites have not been eliminated from consideration. Whether or not to emplace LLW and ILW in separate repositories has been debated. These repositories will be used for wastes generated by the Swiss nuclear power plants and also for several types of wastes returning from plants in France and Britain which have contracted to reprocess Swiss fuels. Intermediate-level waste from reprocessing Swiss fuels abroad may be emplaced in a central, intermediate facility.

The caverns will have horizontal access and several hundred meters of overburden. The waste-reception area for this system will also be situated underground, but at a lower depth than the waste-emplacement areas.

Solidified waste will be received in canisters, and emplacement will involve the creation of additional solid matrices. The canisters will be placed into larger containers, presumably of concrete, and the spaces between canisters will be filled with wet cement.

4.7 Disposal of ILW in Sweden

The disposal facility near Forsmark, Sweden--referred to as the Swedish Final Repository (SFR)--is intended to accommodate wastes mainly from reactors (Fig. B-9). It is located in crystalline rock beneath the Baltic Sea, 1 km offshore, where the water depth is about 5 m [FORSSTROM]. The access tunnels and tops of the cavities containing the waste will be about 50 m underneath the sea floor. The design accommodates 100,000 m³ of waste and precludes reentry after it has been filled and sealed. The subsurface cavities will have different types of confinement barriers for high-activity and low-activity wastes. High-activity wastes will be emplaced in concrete silos 50 m high and 25 m in diameter (Fig. B-10). Each silo will be built on a bed of sand and bentonite, and the spaces between the silo walls and the rock chamber will be filled with either bentonite or a bentonite-sand mixture. The interior of a silo will be divided into rectangular shafts. Waste packages will be off-loaded from vehicles and transferred into the shafts by remotely operated cranes. After being filled with waste packages, each shaft will be injected with grout and then a concrete lid will be cast on top of the silo. The Forsmark facility will accommodate low-activity wastes in caverns without silos. In these caverns, wastes that require shielding will be emplaced by traveling overhead cranes in rooms partitioned by concrete walls. Separate caverns are assigned for low-level and medium-level wastes [HEDMAN 1986]. After being filled with waste packages, some rooms will be backfilled with grout [HEDMAN-1984, FORSSTROM]. Closure is planned for 2010.

4.8 Disposal of ILW in Canada

The design of an intermediate-level facility in Canada has apparently not been finalized. Waste disposal planning in Canada has included conceptual studies of the construction of rock cavities in which both LLW and ILW would be emplaced [CHARLESWORTH]. The cavities would be mined into crystalline rock, although shale and limestone were also considered as possible host formations. A maximum depth of 200 m was suggested [CHARLESWORTH];

Fig. B-9. Concept of Chambers in the Swedish Final Repository

Source: [NILSSON]

recent assessments have been made of the migration of radionuclides emplaced at depths of 50 and 500 m in caverns cut into limestone [RUSSELL]. Recently, the concept for an intermediate-depth facility, shown in Fig. B-11, was briefly described [ISAAC]. This concept involves a ramp access to a cavern mined into rock.

4.9 Disposal of ILW in Britain

In Britain, long-lived ILW will probably be encapsulated in grout [HOWARTH-1986, -1987] and be emplaced underground at a depth of about 300 m [FLOWERS]. The advantages of three types of site are being considered: under land, under sea bed with access by tunnel from the shore, and under seabed with access from a drilling rig. The development of a repository for long-lived ILW, however, will be postponed until a site for the trench disposal of LLW and short-lived ILW has been selected and developed. Short-lived ILW is defined as waste containing nuclides "which decay to very low levels within the period over which planning controls will prevent intrusion" [FLOWERS]. In the trench concept (Fig. B-12), packages containing short-lived ILW would be emplaced at a depth of 10-20 m, fixed in grout, and covered with a layer of concrete. Above this, drums of LLW would be set in trenches and fixed in grout. The system would then be covered with rolled clay, several meters thick.

Fig. B-10. Concept of Silo in the Swedish Final Repository

Source: [NILSSON]

4.10 Disposal of ILW in the Federal Republic of Germany

In the Federal Republic of Germany, LLW and MLW will be disposed of in underground facilities [TUSA]. The basic philosophy to be followed there involves imposing multiple barriers between the radioactive material and the biosphere [MERZ-1986A]. In general, these barriers consist of (1) the waste form, (2) the container, (3) the backfill material and confinement provided by the repository, and (4) the barrier provided by the geologic environment of the disposal site. For LLW and MLW, the suitable waste forms have been identified as bitumen, concrete, calcine, polymeric materials, and dried or compacted primary waste forms. These waste forms are to be contained in (1) 400-L steel drums, with or without concrete shielding, (2) concrete or cast-iron cylinders of 50-500 L volume, and (3) sheet-iron boxes of 4-11 m³ volume [MERZ-1986A].

Disposal planning has involved three sites: (1) the former Asse salt mine, (2) the Gorleben salt dome, and (3) the former Konrad iron mine [CLOSS].

Fig. B-11. Concept of Mined Rock Caverns for Intermediate Depth Disposal of LLW and ILW in Canada

Source: [ISAAC]

The former Asse salt mine near Remlingen, Lower Saxony, was used from 1967 to 1978 for the disposal, by various techniques, of LLW and MLW with negligible heat production. During those years, about 125,000 containers of LLW and 1300 drums of MLW were disposed of in that facility. At present, the Asse facility is being used not for disposal but as an underground laboratory to demonstrate emplacement techniques for several types of radioactive waste, including LLW, HLW, MLW, and High Temperature Gas Reactor fuel [CLOSS, BARNERT].

Fig. B-12. Conceptual Disposal Facility for LLW and Short-Lived ILW in Britain

Source: [FLOWERS]

In 1977, the Gorleben salt dome was designated as a potential site for a future nuclear waste repository. Since that time, the geologic characteristics of the site have been under investigation to determine its suitability as a host for all types of radioactive wastes, including those

that generate heat [CLOSS]. Explorations of the salt formation at 840-960 m are planned [GRUBLER]. Different waste types (LLW, MLW, and HLW) will be located in separate areas of the repository [INTERNATIONAL ENERGY]. Emplacement of wastes in the Gorleben facility is expected to begin in 1999-2000 [MERZ-1986B, GRUBLER].

The Konrad facility, near Salzgitter, Lower Saxony, will accommodate only nuclear wastes with negligible heat production. Konrad is expected to provide disposal space of about 10^6 m^3 . Emplacement operations will be carried out over a period of about 40 years [CLOSS]. The mine has six main levels at depths between 800 and 1300 m [ILLI]. Each disposal room will have a length of up to 800 m and a cross-sectional area of 40 m^2 . The waste containers will be stacked at the end of a disposal room, and the remaining space will be backfilled with crushed rock [HOLTZ]. This facility is expected to be ready for operation in 1990 [MERZ-1986B].

4.11 Conclusions on Disposal Cell Design

Any disposal cell design will be accepted only through a performance assessment involving specific values for parameters describing cell design, waste form, and geologic characteristics. Before that stage, however, some general observations on the adaptability of the designs reviewed in Sec. B.4.1-B.4.10 to disposal of GTCC-LLW can be made. In the demonstration of Greater Confinement Disposal at the Savannah River Plant (Sec. B.4.1), although the disposal units promise to provide lower risk than traditional shallow land burial, their depth (<30 m) qualifies as near-surface disposal, a disposal mode generally acceptable only for low-level waste. Permission to use near-surface disposal for GTCC-LLW might be made, however, on a special-case basis. The 37-m depth of the augered shafts used in the demonstration of Greater Confinement Disposal at the Nevada Test Site (Sec. B.4.2) would be more acceptable for disposal of GTCC-LLW. If disposal at depths >9 m is required, then location of a disposal facility for GTCC-LLW might be restricted to arid, western regions of the United States.

A brief review of the plans of foreign countries for disposal of ILW indicates that the most common design for a disposal unit involves construction of a cavern in bedrock or clay at a depth of 50-300 m. Rock cavities offer an intruder barrier, structural stability, and limited interchange with groundwater. On the other hand, the capital investment for a rock cavity constructed specifically for waste disposal probably makes it the most expensive of any facilities designed for disposal of non-high-level waste. The use of underground cavities not specially created for waste disposal but previously created in the mining of salt and iron ore, as planned by the Federal Republic of Germany, would avoid some of the capital expenses involved in disposing of wastes into rock formations.

APPENDIX C

DISPOSAL OF IMMOBILIZED TRUEX RAFFINATE AS HIGH-LEVEL WASTE

This section examines the hypothetical case in which immobilized TRUEX raffinate is classified as HLW and is required to go to the repository for disposal. The definition of HLW given by the NWPA (see Sec. IV.C of this report) could be interpreted to result in classifying immobilized TRUEX raffinate as HLW because of its source. Given that event, together with (1) judgements that such material should not be emplaced in near-surface facilities and (2) the absence of an intermediate facility, the immobilized TRUEX raffinate would probably have no alternative to emplacement in a geologic repository. Separating the HLW into two fractions and sending both fractions to the same type of disposal facility may seem absurd immediately, but its dismissal cannot be defended without at least considering what effect the use of a low-cost waste form for the non-TRU fraction might have on economic feasibility. The technology and procedures that would be involved in management of TRUEX raffinate as HLW to be emplaced in a repository are briefly described in Sec. C.1-C.4. The feasibility of this management mode for immobilized TRUEX raffinate is examined in Sec. C.5.

1. Preparation of Waste Form for Disposal in Geologic Repository

If immobilized TRUEX raffinate were classified as HLW and its disposal in a geologic repository were required, the economic consequences would depend on the waste form required. Because the toxicity and half-life of radioactive components of immobilized TRUEX raffinate would be much less than those of typical HLW, a waste form less durable than glass would probably be acceptable.

Waste forms acceptable for DOE high-level waste are indicated through performance characteristics specified in DOE 5820.2A by the regulations of NRC in 10 CFR 60.113, 10 CFR 60.131(b)(7), and 10 CFR 60.135 and also by the regulations of EPA [40 CFR 191]. Although the waste form involved in most concepts for disposal of HLW is a glass or ceramic, regulations are not prescriptive with respect to the chemical identity of the waste form; they require compliance with performance characteristics. For example, 10 CFR 60.113 specifies a limit on rate of release of radionuclides from the engineered barrier system, of which the waste form is only one element. The waste form must be incapable of forming a critical mass according to 10 CFR 60.131(b)(7); this concern should be of little significance with respect to the waste form from a process whose main purpose is to separate transuranium elements. Relevant to the waste form, 10 CFR 60.135 prohibits significant quantities of explosive, pyrophoric or chemically reactive materials, and free liquids. Furthermore, it requires solid form, consolidation of particulate matter, and noncombustibility. The regulations given by EPA [40 CFR 191] indirectly affect the selection of waste form by specifying general containment requirements, limits for doses received by individuals, and limits on radioactivity concentrations in groundwater near a disposal system.

A waste form less expensive than glass could be used if it were necessary to dispose of the immobilized TRUEX raffinate as HLW, providing the waste form

complies with the standards listed above. A grout form might be capable of meeting these standards; it has been claimed that grout formulations can be "engineered to produce solids that have leach rates comparable to borosilicate glass" [DOLE]. If a grout form were acceptable, it would probably be prepared in steel containers at the TRUEX plant and later shipped to the repository.

2. Packaging for Disposal in Geologic Repository

Disposal in a geologic repository would require NRC licensing. The NRC requirements for licensing of HLW packages [10 CFR 60], generally deal with standards for containment, addressing the performance of the package, i.e., the packaging and its contents. With regard to packaging, these regulations are not prescriptive with respect to such parameters as size, shape, and materials. Although the NRC licensing regulations are aimed at performance rather than design details, packaging must comply with the waste acceptance criteria for the disposal facility. For example, the packaging must be physically compatible with the handling equipment and the transporters used for receiving and emplacement in the repository. Since only one type of transporter will probably be used in emplacement operations, all wastes may have to be contained in the same type of package. The canisters designed to contain the HLW from the West Valley Demonstration Project, the Savannah River Plant, and the Hanford Site all have about the same dimensions [U.S. DEPARTMENT OF ENERGY-1987A]. Thus, if it were necessary to treat immobilized TRUEX raffinate as high-level waste, even though it were solidified in grout form, it might have to be packaged in a canister similar to those planned.

3. Transport for Disposal in Geologic Repository

If it were necessary to transport immobilized TRUEX wastes to a geologic repository, procedures would follow the guidelines for transport of DOE wastes given in DOE 5480.3 and in regulations [U.S. DEPARTMENT OF ENERGY-1987C]. This order requires compliance with some special DOE rules and also with applicable regulations of the DOT and the NRC for transport of radioactive materials. These rules and regulations are concerned mainly with containment, shielding, and prevention of criticality. If it were necessary to manage immobilized TRUEX raffinate as high-level waste, containment requirements would be satisfied by packaging solidified raffinate in HLW canisters for the same reasons as described in Sec. C.2. Regarding shielding, the canisters could be shipped in casks that are already being designed to transport the Hanford solidified tank wastes [U.S. DEPARTMENT OF ENERGY-1987B]. As noted in Sec. C.2, prevention of criticality should require minimal attention in the case of the raffinate from a process whose main objective is to separate the transuranic elements. The TRUEX raffinate solidified as grout and contained in canisters could be transported to a repository using the same mode and routing that had been planned originally for transportation of the borosilicate glass waste form [U.S. DEPARTMENT OF ENERGY-1987B].

4. Emplacement in Geologic Repository

If it were necessary to manage solidified TRUEX raffinate as HLW in canisters as described in Sec. C.2, the canisters, in cylindrical steel overpacks, would probably be managed on arrival at the repository just as

similar packages containing borosilicate glass waste form or spent fuel. After off-loading, the packages would be stored temporarily in a hot cell where they would be inventoried, monitored for temperature and contamination, and inspected for damage or leakage. The packages would then be lowered to the underground disposal level, where they would be transported to a room cut into the geologic formation and placed into cylindrical holes that had been excavated into the floor or wall of the room. After the packages are in place, the holes would be backfilled with granulated solid intended to act as a migration barrier.

5. Feasibility of Disposing of Immobilized TRUEX Raffinate in a Geologic Repository

Consider the following case:

- (1) Authorities decide that both the TRU fraction and the nonTRU fraction created by applying TRUEX must be classified as HLW because of their origin.
- (2) It is further decided that both fractions must go to a repository. This has been suggested as the most rigorous treatment of the case. A less rigorous alternative would involve disposal of the solidified nonTRU fraction, even if it is classified as HLW, in some facility other than a repository. Such treatment seems possible since disposal of HLW in a repository is not specifically required by the Code of Federal Regulations [KOCHER-1989] presently in effect.
- (3) The TRU fraction is converted to a glass form and loaded into canisters.
- (4) The nonTRU fraction, because of its low concentration of long-lived alpha activity, would be permitted, as the result of a performance assessment, to be disposed of as a grout.
- (5) Packaging of the grout in HLW canisters of standard design would be required in order that it could be handled by the transporters and emplacement equipment used in the repository. This latter assumption applies only to the hypothetical case being examined here; it is not considered in any of the quoted references.

The bases for judging the feasibility of this case are volumes and costs. The volumes of grout to be disposed of, i.e., the number of canisters, are compared with planned repository capacity. The estimated cost of disposing of the nonTRU grout in the repository is compared with the cost of disposing of it in concrete vaults on-site. The waste stream used for an example is the complex concentrates stored at the Hanford site. This hypothetical case, summarized in Table C-1, is derived from recent estimates, supplied by Westinghouse Hanford, of the volumes of glass and grout that would be generated in processing the wastes with and without TRUEX [WODRICH].

Estimates of unit costs for the management of HLW have been developed from a large number of cost elements [FED REG-1987], but collection of the

elements under three general headings is useful: (1) the cost for solidification, (2) the cost of transportation to the repository, and (3) the cost of handling and emplacement at the repository [SLATE-1989]. Estimates for these three general headings are adapted for application to this example.

The solidification (vitrification) cost of \$230,000 per canister is specific to the SRP site [SLATE-1989], but it is assumed to be applicable for the rough calculations we make here. For grouting, a solidification cost of \$1.23 per liter of liquid to be solidified in on-site vaults is used [SLATE-1987].

The costs for transport and disposal are to be applied to all HLW, regardless of origin. The estimated costs per canister are \$15,000 for transport [SLATE-1989] and \$335,000 for disposal [FED REG-1987]. The results of applying these costs to the example are summarized in Table C-1.

Table C-1. Comparison of Volumes and Costs for Three Alternative Treatments of a Selected Waste Stream

	Without TRUEX	With TRUEX	
		Disposal of Grout as LLW in On-site Vaults	Disposal of Grout as HLW in Repository
Number of Canisters			
Glass	870	580 ^a (150) ^b	580 ^a (150) ^b
Grout	0	0	1.2x10 ⁵
Volume Grout (L)	6.8x10 ⁷	7.4x10 ⁷	7.4x10 ⁷
Cost (10 ⁶ \$)			
Solidification			
Glass	2.0x10 ⁸	1.3x10 ⁸ (3.4x10 ⁷)	1.3x10 ⁸ (3.4x10 ⁷)
Grout	6.0x10 ⁷	6.5x10 ⁷	?
Transport			
Glass	1.3x10 ⁷	8.7x10 ⁶ (2.2x10 ⁶)	8.7x10 ⁶ (2.2x10 ⁶)
Grout	0	0	1.8x10 ⁹
Repository Disposal			
Glass	2.9x10 ⁸	1.9x10 ⁸ (5.0x10 ⁷)	1.9x10 ⁸ (5.0x10 ⁷)
Grout	0	0	4.0x10 ¹⁰
Total Costs	5.6x10 ⁸	3.9x10 ⁸ (8.6x10 ⁷)	>4.2x10 ¹⁰ (>4.2x10 ¹⁰)

^aBased on the conservative assumption that CC solids do not dissolve in HNO₃.

^bBased on assumption that 75% of the solids dissolve in HNO₃.

Although the costs given in Table C-1 are only approximations, they indicate that, even though the cost of the unrealistic scenario of solidification of grout in canisters is unknown, the costs of transport and disposal for the large number of canisters involved with the alternative described in

column 4 would far outweigh the costs of the other two alternatives (columns 2 and 3). The most economically reasonable case is described by column 3, disposal of grout as LLW in on-site vaults. For waste streams which would also require the fission-product removal step to permit disposal of the grout as LLW, the cost advantage of the alternative described in column 3 would diminish.

From another standpoint, the 1.2×10^5 canisters of grout for the alternative described in column 4 would be greatly out of proportion to the numbers planned for in current repository designs. The inventory of defense HLW for a repository is to be 8000 canisters total [SLATE-1987].

Thus, the hypothetical case in which immobilized TRUEX raffinate would be classified as HLW and disposed of in a geologic repository would be infeasible because of economics and its demands for repository capacity.

APPENDIX D

GENERAL CONCERNS IN FRACTIONATING OR
PARTITIONING OF HIGH-LEVEL WASTES

A recent study of classification and disposal of wastes that would result from treating HLW by partitioning and transmutation of the separated actinides [KOCHER-1989] expresses concerns similar to those presented in this report. Some of its conclusions could also be applied to the management of immobilized TRUEX raffinate.

Two major points are made in Kocher's study that have relevance to the disposal of TRUEX wastes.

First, in considering the type of disposal facility that might be permitted for partitioned wastes, the study points out that, although the base case for management of HLW involves its disposal in a geologic repository, current regulations and laws do not require that type of disposal facility. It can be inferred, therefore, that even if the wastes from partitioning were classified as HLW, their disposal in a geologic repository is not necessarily required, since neither 10 CFR 60 nor the NWA preclude disposal of HLW in a system other than a repository.

Second, present regulations and laws do not prohibit near-surface disposal of GTCC-LLW*. Therefore, if partitioned wastes were classified as GTCC-LLW, there is currently no basis for prohibiting their disposal in a near-surface facility; whether or not near-surface disposal would be permitted would be decided by NRC on a case-by-case basis. Although this disposal method for GTCC-LLW is described as a hypothetical possibility, it is suggested that partitioning wastes may be ineligible for near-surface disposal because the material remaining after actinides are removed from HLW may have radionuclide concentrations that exceed the limits for disposal in near-surface facilities. The study suggests, however, that ineligibility for near-surface disposal should not imply that partitioning wastes require the extreme of disposal in a geologic repository and that they might best be placed in facilities that would provide confinement intermediate between near-surface facilities and geologic repositories.

Disposal of partitioning wastes in intermediate facilities, however, would require some institutional developments. Whether the wastes from partitioning were classified as HLW or as GTCC-LLW, their disposal in intermediate facilities would require new legislation, new regulations, and establishment of limits on concentrations that would be acceptable for intermediate disposal. It is expected, however, that EPA's standards for land disposal of LLW [40 CFR 193] would apply to disposal of GTCC-LLW in an intermediate facility. Those standards are not associated with particular disposal technologies, even though near-surface facilities were the only ones analyzed by EPA in developing their standards.

*NRC has recommended, however, that if an intermediate facility does not become available, GTCC-LLW should be disposed of in a geologic repository [U.S. Nuclear Regulatory Commission, 1988].

Although the main part of the Kocher study was addressed to civilian wastes, it also suggested that intermediate disposal of defense reprocessing wastes is in need of evaluation. One of the questions to be answered by such an evaluation is whether civilian and defense wastes should be disposed of separately. The recommendation by President Reagan that, to reduce costs, defense HLW should share a repository with civilian spent fuel and HLW is noted. Analogy suggests that, for cost effectiveness, civilian wastes should share an intermediate disposal facility with defense wastes.

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