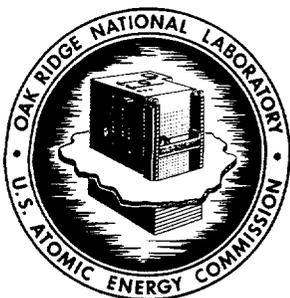


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71-10-14

DATE: October 15, 1971  
SUBJECT: Recommendations for the Disposal of Radioactive Liquids at ORNL  
TO: D. E. Ferguson  
FROM: R. E. Blanco, F. T. Binford, H. W. Godbee, and J. M. Holmes

ABSTRACT

This report presents a survey of methods now used at ORNL for disposal of radioactive liquid, along with new alternative methods. It is recommended that hydrofracturing be retained as the principal disposal method. This report serves as a supplement to a previous report, Safety Analysis of Waste Disposal by Hydraulic Fracturing at Oak Ridge, ORNL-4665. It also will appear as Chapter 6.0 in a subsequent comprehensive report that will review present and future aspects of waste disposal at ORNL.

This document has been approved for release  
to the public by:

*David C. Hamm* 3/19/96  
Technical Information Officer ORNL Site

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ChemRisk Document No. 2784



## 6.0 RECOMMENDATIONS FOR THE DISPOSAL OF RADIOACTIVE LIQUIDS AT ORNL

R. E. Blanco  
F. T. Binford

H. W. Godbee  
J. M. Holmes

### 6.1 Introduction

This section of the report reviews the methods used at ORNL for the disposal of liquid wastes and contains recommendations and a suggested schedule for improving these procedures. In-depth surveys and cost estimates of alternative procedures could not be made within the scheduled limitations of the study. However, overall advantages and disadvantages of alternative methods and preliminary cost estimates are presented along with the areas where additional detailed engineering evaluation and development work are required.

Historically, two types of liquid radioactive waste have been produced at ORNL, i.e., process (low level) waste, and intermediate-level waste (ILW). Alternatives for the disposal of ILW and the residual sludge, i.e., process waste sludge, from the low-level treatment plant are described in this section of the report. The treatment of process waste and the limitations on discharge of the treated effluent are presented in Section 8.0. Intermediate-level waste (ILW) is composed of a mixture of all of the liquid wastes, other than process waste, that are produced in hot cell, pilot plant, and reactor operations, including relatively small volumes of organic reagents and solvent. Thus, the high-level wastes, produced in pilot plants and hot cells, are diluted by the lower level research wastes to form ILW. In the 1940's, the ILW was neutralized with caustic and retained in six 170,000-gal concrete storage tanks. In the 1950's and until 1965 the supernates from these tanks, along with the process waste sludge, was discharged to surface pits in the Conasauga shale in the burial ground. In 1965, an evaporator and two 50,000-gal stainless-steel, high-level waste storage tanks were installed. Currently, these waste tanks are empty, except for a few thousand gallons of waste from the transuranium processing plant (TRU facility). The evaporator has been used to concentrate the ILW supernate from the concrete

storage tanks by a factor of 20 to 30, and the concentrate stored in tank W-8 prior to disposal by hydrofracture. The concentrates contained about 1 Ci/gal of beta-gamma activity and 0.2 mg/gal of plutonium. Currently (mid 1971), the process waste sludge is discharged to a surface waste pit that was previously used for ILW.

In December 1966, hydrofracturing was selected as the method for disposal of ILW rather than disposal in pits. In 1970, the routine disposal of ILW by hydrofracturing was stopped until a new safety analysis<sup>2</sup> was completed. The safety analysis was issued in August 1972, and it confirmed that hydrofracture is a safe, reliable disposal method. We believe that the review bodies will accept this analysis and that hydrofracturing will be resumed. This disposal method meets the requirements of an AEC policy statement which stated that it is desirable to dispose of wastes permanently "on site", if suitable methods are available.<sup>1</sup>

## 6.2 Recommendations

### 1. Disposal Method

Hydrofracture should be retained as the permanent method for disposal of all liquid radioactive wastes at ORNL based on safety and cost (Sect. 6.5). This recommendation is in agreement with AEC policy<sup>1</sup> which favors "on site" disposal, if practical.

Safety. — The safety analysis<sup>2</sup> states that "the hydraulic fracturing procedure is probably the safest and most effective method currently available for permanently removing radioactive waste from man's environment." We concur with this assessment. However, improvements to the present facility are needed to dispose of present wastes and a major permanent revision is required prior to disposal of tank sludges or pilot plant wastes.

Cost. — The estimated capital cost for hydrofracture of ORNL wastes, including intermediate-level waste (ILW), tank sludges, and diluted high-level pilot plant wastes (HLW) in a permanent facility, is \$1.6 million (1975

dollars). The estimated annual operating costs are about \$80,000 for ILW alone, and an additional \$220,000 per year (over 36 months) for tank sludges or an additional \$440,000 per year for diluted HLW (1975 dollars). Forty months are required for the diluted HLW, if the beta-gamma limit is 10 Ci/gal or 20 months if the limit is 20 Ci/gal (Sect. 6.5.1.8).

The estimated capital cost for solidification of these wastes by calcination, including a shipping cask, will be \$4.5 million in an existing building or \$7.9 million in a new building (1975 dollars). The estimated annual operating cost, including charges for final disposal in a federal repository, is about \$1.7 million for ILW, or \$1.8 million for ILW plus tank sludges, or ILW plus HLW (1975 dollars). This facility would be capable of calcining the tank sludges plus the ILW over a 10-yr period followed by calcination of ILW-HLW mixtures over a 3-yr period. Calcination of these wastes would produce large quantities of slightly contaminated nitric acid during calcination, so hydrofracturing would probably have to be continued in order to dispose of this waste stream. This additional cost has not been included in the above costs (Sects. 6.5.1.9 and 6.4.7).

These estimates for high-level wastes represent maximum costs, since it is assumed that all three pilot plants for reprocessing fuels (LMFBR, LWR, and HTGR) will be operated at ORNL.

## 2. Operating Limits

Generally, the operating limits for hydrofracture recommended in the safety analysis<sup>2</sup> should be retained for future waste disposal operations. The estimated volume of the present well and underground rock storage area is sufficient to contain all wastes foreseen for the next 20 to 30 years, with the exception of process (low-level) waste sludges. These wastes include intermediate-level waste, tank sludge, and diluted high-level pilot plant wastes. The limit of  $10 \mu\text{Ci}/\text{kg}$  of rock plus grout for long-lived alpha ( $^{239}\text{Pu}$ ) in the underground storage area will not be exceeded and the temperature rise will

not exceed 70°F. After permanent revision of the facility is completed, the limit for surface operations should be increased to 20  $\beta$ - $\gamma$  Ci/gal and to  $1 \times 10^{-2}$   $\alpha$  Ci/gal. These limits will permit (a) the injection of more concentrated wastes and thus conserve underground storage space, and (b) the handling of wastes containing higher concentrations of  $^{244}\text{Cm}$  (Sect. 6.5.1.2).

### 3. Capital Investment

Interim Facility. — The present hydrofracture plant should be upgraded in FY 1972 at a cost of \$177,000 to permit routine disposal of ILW for about 5 years, as recommended in the recent safety analysis.<sup>2</sup> (Sect. 6.5.1.6).

Permanent Facility. — An improved permanent hydrofracture facility should be constructed at the present well site in FY 1975 for approximately \$1.6 million. This improved facility is required for safe handling of the tank sludges and future pilot plant wastes (Sect. 6.5.1.7).

### 4. Operating Schedule

Intermediate-Level Waste. — One or two batches of intermediate-level waste will be injected each year. The hydrofracture of intermediate-level waste should continue in the interim facility until FY 1975 and indefinitely after FY 1976 in the permanent facility.

Tank Sludge. — Methods for removal of the tank sludges, preparation of grout, and hydrofracture of the sludge-grout mixture should be developed in FY 1972, 1973, and 1974. These wastes could then be hydrofractured in the period 1976 to 1981 (Sect. 6.5.1.4).

High-Level and TRU Wastes. — The TRU wastes should be neutralized and combined with intermediate-level waste and hydrofractured each year. With this system, greater than 99% of the  $^{244}\text{Cm}$  and  $^{239}\text{Pu}$  and a large fraction of the other radionuclides will be precipitated and held in tank W-6 for future disposal with the sludge when the permanent hydrofracture facility is ready (Sect. 6.5.1.3).

The concentrated, acidic, high-level pilot plant wastes (approximately 19,000 gal) should be stored in the 50,000-gal high-level waste tank for decay from FY 1977 to FY 1981. They should be diluted to 20  $\beta$ - $\gamma$  Ci/gal and hydrofractured during FY 1981, 1982, and 1983 (Sect. 6.5.1.3).

Process Waste Sludge. — The disposal of process waste sludge in pits in the burial ground should continue as long as the lime-soda treatment plant is operated. The volume of the sludge is very large and, if hydrofractured with the tank sludges and diluted high-level wastes, the life of the underground storage area would be reduced to 2-10 years. These sludges contain less than a total of 10 Ci of  $^{90}\text{Sr}$  per year and the alpha content is below detection. This amount of strontium is negligible compared to the thousands of curies of  $^{90}\text{Sr}$  now stored in the burial pits. The disposal procedure could be re-evaluated during the design of a new low-level waste treatment plant. The volume and weight of solids produced in the new plant is expected to be 2 to 3 times smaller and it may be feasible to dispose of these wastes by hydrofracture (Sects. 6.5.1.2 and 6.4.6).

#### 5. Organic Wastes

A relatively small volume (~ 1,000 gal) of radioactive, organic solvent waste is collected in the ILW system annually. The present practice of decontaminating this waste by distillation in the ILW evaporator and further treatment in the process waste treatment plant with final discharge to White Oak Creek should be continued, since the amount of organic waste is small and the collection system is established. The organic materials remaining in the evaporator (ILW) concentrate would be disposed of by hydrofracturing as a component of ILW (Sects. 6.3.2, 6.4.5, and 7.0).

#### 6. Chemical (Nitrate) Wastes

The nitrates in aqueous wastes are decomposed, volatilized as  $\text{NO}_x$ , and recovered as slightly contaminated nitric acid during solidification and calcination. Thus, if ORNL wastes are calcined, rather than hydrofractured, a

new acidic waste containing about 44 tons of nitrate per year will be formed. The discharge of the nitrates to White Oak Creek is not recommended. Thus, hydrofracturing should continue, in any event, for the disposal of nitrate and other chemical wastes (Sect. 6.4.7).

#### 7. Disposal by Drying and Hydrofracture

If hydrofracturing is not approved as the disposal method for ORNL, some type of solidification process must be used or a combination of solidification and hydrofracturing. A simple drying of the sludge and hydrofracture of the supernate appears attractive and should be considered. All wastes would be neutralized (after appropriate cooling for HLW) to precipitate greater than 99% of the alpha radionuclides and <sup>90</sup>Sr, and a major fraction of the materials represented by the gross  $\beta$  activity. The solids would be separated, dried, and shipped to a federal repository. The supernate, containing the bulk of the solids as nitrates and other salts and a minor amount of radionuclides, would be hydrofractured. The system is expected to be considerably more expensive than hydrofracturing, but less expensive than the calcination system (Sect. 6.5.2.3).

### 6.3 Status of Present Disposal Methods

#### 6.3.1 Intermediate-Level Waste

In December 1966, hydrofracturing was selected as the routine disposal method at ORNL for ILW, and it was used successfully until mid-1970. In 1970 the AEC requested that ORNL stop the routine disposal of wastes by hydrofracturing until a new safety analysis was completed. The impetus was the "AEC Immediate Action Directive,"<sup>3</sup> which required that solid wastes containing known or detectable amounts of transuranium isotopes be retained in a retrievable form. Later, statements of AEC policy indicated that (1) waste containing significant amounts of beta and gamma nuclides<sup>4</sup> were not necessarily included in the directive and (2) that it was desirable to dispose of wastes permanently "on site" if suitable methods were

available.<sup>1</sup> A safety analysis was completed<sup>2</sup> and submitted to the AEC for approval in August 1971. The safety analysis<sup>2</sup> contains (1) a history of the development of the hydrofracture method, (2) a resume of routine disposal operations, (3) a safety analysis of surface operations and the underground storage area, and (4) a statement of operating limits for surface operations and the underground storage area. It was concluded that the method is completely safe. The safety analysis of surface operations was limited to the upgrading of the present hydrofracturing plant for continuing disposal operations with ILW similar to that handled previously. A new safety analysis will be required for surface operations, when the plant is extensively revised into a permanent facility for handling the tank sludges and other wastes containing 5 to 10 times higher concentrations of radionuclides as recommended in this report (Sect. 6.2). The other portions of the safety analysis including the underground storage apply directly to all future operations.

### 6.3.2 Organic Solvent and Reagent Wastes

Contaminated organic solvent and reagent wastes (<1000 gal annually) are discharged to the concrete ILW storage tanks. If the organic wastes contain fissionable material, nuclear poisons are added (Th for  $^{239}\text{Pu}$  and  $^{238}\text{U}$  for  $^{233}\text{U}$  and  $^{235}\text{U}$ ) which will not separate from the fissionable materials when they become alkaline and precipitate in the waste tanks (Sects. 6.4.5, 6.2, and 7.0). During subsequent evaporation, some organic components volatilize and condense with the overhead water and pass into the process waste system for eventual discharge to White Oak Creek. Non-volatile organic components remain with the ILW evaporator concentrate along with the radioactive materials originally contained in the organic waste.

### 6.3.3 Process Waste Sludge

Process waste sludge is collected in a tank truck and transferred to trenches in the Conasauga shale in the burial ground (Sect. 6.4.6).

## 6.4 Present and Potential Future Wastes

### 6.4.1 Intermediate-Level Waste Concentrate

About 80,000 to 150,000 gal of intermediate-level waste concentrates are accumulated at ORNL each year. During the hydrofracturing operation about 5.5 lb of solids are added per gallon of waste plus a maximum of 30,000 gal of wash water, so that the final volume of waste-grout mixture (for 80,000 gal of waste) that is pumped underground is about 125,000 gal. It is expected that the amount of wash water can be decreased significantly in future routine. This rate of formation of wastes is predicted for the next 5 to 10 years at ORNL. The approximate inert, chemical composition of ILW (after evaporation) is shown in Table 1. The principal constituent is sodium nitrate (0.8 M). The composition of the solids added during hydrofracturing is shown in Table 2. Thus, the final solidified grout contains about 0.15 gal of ILW concentrate per kg.

Table 3 shows the volumes and average radionuclide content of ILW that has been hydrofractured in past operations. Thus, about  $3.5 \times 10^5$  Ci of  $^{137}\text{Cs}$ ,  $2.5 \times 10^4$  Ci of  $^{90}\text{Sr}$ , 69 g of  $^{239}\text{Pu}$ , and 0.23 g of  $^{244}\text{Cm}$  have been hydrofractured for underground storage. Tables 4, 5, and 6 show the volumes and average concentration of radionuclides in tanks W-5, W-6, and W-8. Tank W-5 is used mainly to accumulate wastes from Bethel Valley and W-6 from Melton Valley. The supernates from these tanks are concentrated by a factor of about 30 in the evaporator and the concentrated ILW stored in W-8 until it is hydrofractured. The remaining tanks, W-7, W-9, and W-10 were essentially empty as of March 15, 1971, but they will be used to accumulate ILW concentrate after W-8 is filled. The supernate in W-8 is being held pending permission from the AEC for disposal by hydrofracture. As of March 1971, the supernate in W-8 contained  $6.6 \times 10^{-2}$  Ci/liter of gross  $\beta$  activity, and a total of  $6.05 \times 10^3$  Ci of  $^{90}\text{Sr}$ ,  $2.02 \times 10^2$  Ci of  $^{137}\text{Cs}$ , and 0.12 g of  $^{244}\text{Cm}$ . Plutonium-239 was not detectable by pulse height analysis.

Table 1. Approximate Composition of Intermediate-Level Waste Concentrate

NaOH	0.05 M
NaNO <sub>3</sub>	0.8 M
(NH <sub>4</sub> ) <sub>2</sub> SO <sub>4</sub>	0.15 M
Al <sub>2</sub> (SO <sub>4</sub> ) <sub>3</sub>	0.05 M
NaCl	0.05 M
NaCO <sub>3</sub>	0.05 M
<sup>137</sup> Cs	0.6 Ci/gal <sup>a</sup>
<sup>90</sup> Sr	0.04 Ci/gal <sup>a</sup>

<sup>a</sup>See Table 3.

Table 2. Solids Added to Form Hydrofracture Grout

Cement (gypsum retarded)	2 lb/gal waste
Fly ash	2 lb/gal waste
Attapulgis 150	1 lb/gal waste
Grundite	1/2 lb/gal waste
Retarder (DGL) <sup>a</sup>	0.003 lb/gal waste

<sup>a</sup>Delta gluconal lactone.

Table 3. Concentration of Predominant Radionuclides in Concentrated, Intermediate-Level Waste  
(Circa October 1970)

	Volume (gallons)	<sup>137</sup> Cs (Ci/gal)	<sup>90</sup> Sr (Ci/gal)	<sup>239</sup> Pu (mg/gal)	<sup>244</sup> Cm (mg/gal)
Average in waste injected to date (No. 1 through No. 7)	623,000	0.56	0.04	0.21 <sup>a</sup>	not analyzed
Highest concentration in waste injected to date (No. 6)	79,000	1.12	0.11	0.05	not analyzed
Injection Run No. 7	83,000	0.54	0.03	0.34	0.003
Average in storage tank sludge	305,000	0.30	1.64	15.7 <sup>b</sup>	0.33 <sup>c</sup>
Sludge diluted by five for mobility <sup>d</sup>	1,525,000	0.06	0.35	3.1	0.066

<sup>a</sup> Average concentration in last four injections ( $3.3 \times 10^5$  gal). The waste in the first three injections was not analyzed for <sup>239</sup>Pu.

<sup>b</sup> Plutonium figure taken from SS Accountability Office records; assumes <sup>239</sup>Pu blended with all sludge.

<sup>c</sup> Assumes 100 g <sup>244</sup>Cm blended with all sludge based on losses from TRU plant.

<sup>d</sup> E. J. Frederick, J. C. Suddath, and J. O. Blomeke, Proposed Program for Sludge Removal from ORNL Waste Tank Farm, ORNL CF 70-1-6.

Table 4. Estimated Radioactive Content of Tanks W-5, W-6, and W-8  
Circa March 15, 1971

Tank W-5: 514,000 liters (136,000 gallons) sludge;  
181,400 liters (48,000 gallons) supernate

Nuclide	Concentration		Total Amount		% of Total in Supernate	% of Total in Washes <sup>e</sup>
	Sludge	Supernate	Sludge	Supernate		
Gross $\beta^d$	1.0	$2.5 \times 10^{-3}$	$5.14 \times 10^5$	$4.5 \times 10^2$	$8 \times 10^{-2}$	0.12
$^{90}\text{Sr}$	0.50	$8.2 \times 10^{-4}$	$2.57 \times 10^5$	$1.49 \times 10^2$	0.57	c
$^{137}\text{Cs}$	$1.01 \times 10^{-2}$	$4.2 \times 10^{-4}$	$5.20 \times 10^3$	$0.76 \times 10^2$	1.3	36
$^{144}\text{Ce}$	$1.7 \times 10^{-3}$	b	$8.75 \times 10^2$	b	b	b
$^{106}\text{Ru}$	$8.6 \times 10^{-4}$	$1.05 \times 10^{-3}$	$4.43 \times 10^2$	$1.91 \times 10^2$	30.1	b
$^{152}\text{Eu}$	$1.5 \times 10^{-3}$	b	$7.72 \times 10^2$	b	b	b
$^{154}\text{Eu}$	$3.3 \times 10^{-3}$	b	$1.70 \times 10^3$	b	b	b
$^{155}\text{Eu}$	$1.9 \times 10^{-3}$	b	$9.78 \times 10^2$	b	b	b
$^{60}\text{Co}$	$1.04 \times 10^{-2}$	b	$5.35 \times 10^3$	b	b	b
$^3\text{H}$	b	$3.02 \times 10^{-5}$	b	5.5	b	b
$^{139}\text{Pu}^a$	3.6	b	1.85 kg	c	c	c
$^{244}\text{Cm}$	$2.6 \times 10^{-2}$	$2.7 \times 10^{-4}$	13.3 g	0.05 g	0.36	$3 \times 10^{-3}$

<sup>a</sup> Pu is  $^{239}\text{Pu}$  and  $^{240}\text{Pu}$  calculated as  $^{239}\text{Pu}$ ; mass analysis shows 80-88%  $^{239}\text{Pu}$  and 9-16%  $^{242}\text{Pu}$ .

<sup>b</sup> Not determined.

<sup>c</sup> Not detectable by  $\beta$  counting for  $^{90}\text{Sr}$  or  $\alpha$  pulse height analysis for Pu.

<sup>d</sup> Assumes a counting geometry of 10%.

<sup>e</sup> Sludge washed three times with equal volume of water.

Table 5. Estimated Radioactive Content of Tanks W-5, W-6, and W-8 (Continued)  
 Circa March 15, 1971

Tank W-6:  $4.98 \times 10^5$  liters (132,000 gallons) sludge;  
 $2.15 \times 10^5$  liters (57,000 gallons) supernate

Nuclide	Concentration		Total Amount		% of Total in Supernate	% of Total in Washes <sup>e</sup>
	Sludge	Supernate	Sludge	Supernate		
Gross $\beta^d$	1.7	$5 \times 10^{-2}$	$8.5 \times 10^6$	1.07	$1.3 \times 10^4$	0.6
$^{90}\text{Sr}$	0.85	$8.2 \times 10^{-5}$	$4.25 \times 10^5$	$0.18 \times 10^2$	$4 \times 10^{-3}$	c
$^{137}\text{Cs}$	$9.3 \times 10^{-2}$	$6.5 \times 10^{-2}$	$4.65 \times 10^4$	$1.40 \times 10^4$	23.1	61
$^{144}\text{Ce}$	$7.2 \times 10^{-3}$	b	$3.60 \times 10^3$	b	b	b
$^{106}\text{Ru}$	$4.5 \times 10^{-2}$	b	$2.25 \times 10^4$	b	b	b
$^{152}\text{Eu}$	$8.0 \times 10^{-3}$	b	$4.00 \times 10^3$	b	b	b
$^{154}\text{Eu}$	$1.79 \times 10^{-2}$	b	$8.94 \times 10^3$	b	b	b
$^{155}\text{Eu}$	$9.1 \times 10^{-3}$	b	$4.55 \times 10^3$	b	b	b
$^{60}\text{Co}$	$1.61 \times 10^{-2}$	b	$8.04 \times 10^3$	b	b	b
$^3\text{H}$	b	$2.70 \times 10^{-5}$	b	5.8	b	b
$^{239}\text{Pu}^a$	9.6	b	4.80 kg	c	c	c
$^{244}\text{Cm}$	0.25	$2.7 \times 10^{-7}$	125 g	$5.8 \times 10^{-5}$ g	$4.6 \times 10^{-5}$	$1.4 \times 10^{-2}$

<sup>a</sup> Pu is  $^{239}\text{Pu}$  and  $^{240}\text{Pu}$  calculated as  $^{239}\text{Pu}$ ; mass analysis shows 80-88%  $^{239}\text{Pu}$  and 9-16%  $^{242}\text{Pu}$ .

<sup>b</sup> Not determined.

<sup>c</sup> Not detectable by  $\beta$  counting for  $^{90}\text{Sr}$  or  $\alpha$  pulse height analysis for Pu.

<sup>d</sup> Assumes a counting geometry of 10%.

<sup>e</sup> Sludge washed three times with equal volume of water.

Table 6. Estimated Radioactive Content of Tanks W-5, W-6, and W-8 (Continued)  
 Circa March 15, 1971

Tank W-8:  $1.70 \times 10^5$  liters (45,000 gallons) sludge:  
 $1.55 \times 10^5$  liters (41,000 gallons) supernate

Nuclide	Concentration		Total Amount		% of Total in Supernate	% of Total in Washes <sup>e</sup>
	Sludge	Supernate	Sludge	Supernate		
	Ci					
Gross $\beta^d$	1.6	$6.6 \times 10^{-2}$	$2.7 \times 10^5$	$1.0 \times 10^4$	3.7	0.15
$^{90}\text{Sr}$	0.82	$3.9 \times 10^{-2}$	$1.40 \times 10^5$	$6.05 \times 10^3$	4.3	c
$^{137}\text{Cs}$	$3.42 \times 10^{-2}$	$1.3 \times 10^{-3}$	$5.83 \times 10^3$	$2.02 \times 10^2$	3.3	23
$^{144}\text{Ce}$	$5.9 \times 10^{-3}$	b	$1.00 \times 10^3$	b	b	b
$^{106}\text{Ru}$	$1.7 \times 10^{-3}$	$3.1 \times 10^{-3}$	$2.90 \times 10^2$	$4.81 \times 10^2$	62.4	b
$^{152}\text{Eu}$	b	b	b	b	b	b
$^{154}\text{Eu}$	$1.2 \times 10^{-3}$	b	$2.04 \times 10^2$	b	b	b
$^{155}\text{Eu}$	$1.1 \times 10^{-3}$	b	$1.87 \times 10^2$	b	b	b
$^{60}\text{Co}$	b	b	b	b	b	b
$^3\text{H}$	b	$1.98 \times 10^{-5}$	b	3.1	b	b
	mg/liter					
$^{239}\text{Pu}^a$	1.6	b	0.27 kg	c	c	c
$^{244}\text{Cm}$	$9.2 \times 10^{-2}$	$7.6 \times 10^{-4}$	15.6 g	0.12 g	0.76	$1.6 \times 10^{-3}$

<sup>a</sup> Pu is  $^{239}\text{Pu}$  and  $^{240}\text{Pu}$  calculated as  $^{239}\text{Pu}$ ; mass analysis shows 80-88%  $^{239}\text{Pu}$  and 9-16%  $^{242}\text{Pu}$ .

<sup>b</sup> Not determined.

<sup>c</sup> Not detectable by  $\beta$  counting for  $^{90}\text{Sr}$  or  $\alpha$  pulse height analysis for Pu.

<sup>d</sup> Assumes a counting geometry of 10%.

<sup>e</sup> Sludge washed three times with equal volume of water.

#### 6.4.2 Waste Tank Sludge

The inert chemical and radiochemical composition of the sludges in tanks W-5, W-6, and W-8 have been determined. However, the inaccuracy of the sampling method limits the accuracy of the results. Analysis of samples in October 1970 showed that the sludge contains  $9.1 \times 10^4$  Ci of  $^{137}\text{Cs}$  and  $5.0 \times 10^5$  Ci of  $^{90}\text{Sr}$ . The sludge also contains 4.78 kg of  $^{239}\text{Pu}$  (based on accountability records) and 100 g of  $^{244}\text{Cm}$  (based on losses from the TRU facility). Analysis of samples in March 1971 (Tables 4-6) indicate that the sludge contains  $5.75 \times 10^4$  Ci of  $^{137}\text{Cs}$ ,  $8.22 \times 10^5$  Ci of  $^{90}\text{Sr}$ , 6.92 kg of  $^{239}\text{Pu}$ , and 153.9 g of  $^{244}\text{Cm}$ . The amount of  $^{137}\text{Cs}$  in the March analysis is 6.3 times higher than the October analysis, but the amounts of  $^{90}\text{Sr}$ ,  $^{239}\text{Pu}$ , and  $^{244}\text{Cm}$  are only about 1.5 times higher in the March analysis. The sludge samples were washed three times with approximately equal volumes of water and the solids separated by centrifugation. The volumes under the column "% in Wash" show that a significant fraction of the relatively important isotopes, with the exception of cesium, was not removed by washing (Sect. 6.5.2.3). Estimates for the inert components were obtained by spectrographic analysis of dried sludge (Table 7). The carbonate, sulfate, and nitrate contents were estimated from mass spectrographic analysis of the gases released from the dried samples during ignition to  $1650^\circ\text{C}$  (Tables 7 and 8).

#### 6.4.3 Transuranium Waste

Until July 1970, wastes from the Transuranium Processing Plant (or TRU Facility) were diluted and neutralized with caustic and transferred by pipeline to storage tank W-6 in Bethel Valley. The original operating regulations stated that the radionuclide content should not exceed 10 Ci/gal. However, it has been convenient to hold the level of  $<5$  Ci/gal. Solutions containing  $^{131}\text{I}$  are cooled 90 days prior to transfer. In 1970 and 1971, processing of some Savannah River slugs and target tubes in TRU resulted in a significant increase in the quantity of wastes from TRU. These increased wastes and troublesome leaks on the transfer line led to a decision to truck the "high level" wastes from the early Savannah River campaigns

Table 7. Approximate Chemical Compositions of Dried Sludge  
from Gunnite Tanks

Constituent	Estimated Composition Ranges - Dried Sludge Samples (Wt %)		
	Tank Number		
	W-5	W-6	W-8
Water	< 8.	< 4.	< 5.
Sodium	<sup>a</sup>	10. - 20.	10. - 30.
Calcium	1. - 8.	0.1 - 1.	0.1 - 1.
Silicon	2. - 20.	1. - 10.	3. - 20.
Aluminum	0.1 - 1.	2. - 11.	1. - 10.
Iron	0.2 - 2.	0.1 - 1.	0.1 - 1.
Potassium	0.3 - 3.	1. - 10.	1. - 10.
Magnesium	0.1 - 1.	0.1 - 1.	1. - 10.
Lead	1. - 10.	-	-
Uranium and Thorium	< 2.	.01 - .2	0.1 - 1.
Phosphate	0.4 - 4.	0.1 - 1.	0.4 - 4.
Sulphate	1. - 10.	1. - 3.	5. - 25.
Carbonate	1. - 10.	1. - 10.	1. - 10.
Nitrate	0.1 - .5	0.1 - 0.5	1. - 10.
Fluoride	0.01 - .1	0.1 - 0.8	.01 - .1

<sup>a</sup>Not available.

Table 8. Weight Losses on Drying and Gas Evolution During Ignition of Gunnite Tank Samples

	Tank Number		
	W-5	W-6	W-8
Weight loss during drying of sludge sample	< 73.	< 62.	< 66.
Gas evolution during ignition to 1650°C, cc/gm	292.	56.	136.

to the 50,000-gal storage tanks in Bethel Valley. A new pipeline was constructed in 1971 and transfer of the wastes by pipeline is expected to be resumed in late 1971. No difficulty in maintaining the concentrations below 10 Ci/gal are foreseen.

The composition and volume of TRU wastes expected in the year 1971-1975 and 1976-1981 are shown in Tables 9 and 10. These wastes probably represent the major source of fission product wastes at ORNL in this period.\* Both the amount produced each year and the cumulative amounts are listed. If the TRU wastes are neutralized and combined with ILW in W-6 and the supernate evaporated and hydrofractured each year, the indicated cumulative amounts of plutonium, curium, strontium, and a major fraction of the beta-gamma activity (rare earths) will remain with the sludge in W-6 and W-8.

#### 6.4.4 High-Level Waste

Pilot plant programs to develop fuel reprocessing methods for LMFBR, HTGR, and LWR fuels will produce high-level wastes. These programs have not received final approval as of September 1971, but consideration of methods for handling these wastes illustrate the capability of ORNL in this area.

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\*Major source if pilot plants are not operated.

Table 9.  
Projections of Radioactive Waste Generation  
Project: Transuranium Processing Plant

	1971		1972		1973		1974		1975	
	Produced	Cumulative <sup>a</sup>								
1. Total volume (gal)	2,600	2,600	1,600	4,200	400	4,600	400	5,000	400	5,400
1a. Volume after evaporation (gal)	2,600	3,400	1,600	5,000	400	5,400	400	5,800	400	6,200
2. Total $\beta$ , $\gamma$ activity (Ci)	265,000	280,000	153,000	228,960	108,000	186,960	108,000	166,760	108,000	154,070
3. Total $\alpha$ activity (Ci)	1,625	2,025	1,625	3,650	1,625	5,275	1,625	6,900	1,625	8,525
4. Heat generation (watts)	552	753	405	645	349	620	349	638	349	669
5. Radionuclides										
Pu (g)	0	0	0	0	0	0	0	0	0	0
Pu (Ci)	0	0	0	0	0	0	0	0	0	0
<sup>244</sup> Cm (g)	20	25	20	45	20	65	20	85	20	105
<sup>244</sup> Cm (Ci)	1,625	2,025	1,625	3,650	1,625	5,275	1,625	6,900	1,625	8,525
<sup>137</sup> Cs (g)	82	96	54	147	13.5	157	13.5	167	13.5	176
<sup>137</sup> Cs (Ci)	8,091	9,481	5,294	14,528	1,330	15,480	1,330	16,408	1,330	17,312
<sup>90</sup> Sr (g)	14	17	9.7	26	2.9	28	2.9	31	2.9	33
<sup>90</sup> Sr (Ci)	2,077	2,425	1,397	3,761	420	4,087	420	4,405	420	4,715
<sup>3</sup> H in total waste (Ci)	584	584	422	974	162	1,082	162	1,185	162	1,282
<sup>3</sup> H in concentrated waste (Ci)	584	584	422	974	162	1,082	162	1,185	162	1,282
6. Total solids (g)	990,000	1,120,000	555,000	1,675,000	115,000	1,790,000	115,000	1,905,000	115,000	2,020,000
7. Acid concentration (N)	0	0	0	0	0	0	0	0	0	0
8. Total FP solids (g)	1,625	1,925	1,175	3,100	450	3,550	450	4,000	450	4,450

<sup>a</sup>As of the end of each year.

Table 10.  
Projections of High-Level Radioactive Aqueous Waste Generation  
Project: TRU

	1976		1977		1978		1979		1980		1981	
	Produced	Cumulative <sup>a</sup>										
1. Total volume (gal)	4.0E2	5.8E3	4.0E2	6.2E3	4.0E2	6.6E3	4.0E2	7.0E3	4.0E2	7.4E3	4.0E2	7.8E3
1a. Volume after evaporation (gal)	4.0E2	6.6E3	4.0E2	7.0E3	4.0E2	7.4E3	4.0E2	7.8E3	4.0E2	8.2E3	4.0E2	8.6E3
2. Total $\beta, \gamma$ activity (Ci)	1.1E5	1.5E5	1.1E5	1.4E5	1.1E5	1.3E5	1.1E5	1.3E5	1.1E5	1.3E5	1.1E5	1.3E5
3. Total $\alpha$ activity (Ci)	1.6E3	1.0E4	1.6E3	1.2E4	1.6E3	1.3E4	1.6E3	1.5E4	1.6E3	1.7E4	1.6E3	1.8E4
4. Heat generation (watts)	3.5E2	7.1E2	3.5E2	7.5E2	3.5E2	7.9E2	3.5E2	8.3E2	3.5E2	8.7E2	3.5E2	9.1E2
5. Radionuclides												
Pu (g)	-	-	-	-	-	-	-	-	-	-	-	-
Pu (Ci)	-	-	-	-	-	-	-	-	-	-	-	-
<sup>244</sup> Cm (g)	2.0E1	1.3E2	2.0E1	1.5E2	2.0E1	1.7E2	2.0E1	1.9E2	2.0E1	2.1E2	2.0E1	2.3E2
<sup>244</sup> Cm (Ci)	1.6E3	1.0E4	1.6E3	1.2E4	1.6E3	1.3E4	1.6E3	1.5E4	1.6E3	1.7E4	1.6E3	1.8E4
<sup>137</sup> Cs (g)	1.4E1	1.9E2	1.4E1	1.9E2	1.4E1	2.0E2	1.4E1	2.1E2	1.4E1	2.2E2	1.4E1	2.3E2
<sup>137</sup> Cs (Ci)	1.3E3	1.8E4	1.3E3	1.9E4	1.3E3	2.0E4	1.3E3	2.1E4	1.3E3	2.1E4	1.3E3	2.2E4
<sup>90</sup> Sr (g)	2.9E0	3.5E1	2.9E0	3.7E1	2.9E0	3.9E1	2.9E0	4.1E1	2.9E0	4.3E1	2.9E0	4.5E1
<sup>90</sup> Sr (Ci)	4.2E2	5.0E3	4.2E2	5.3E3	4.2E2	5.6E3	4.2E2	5.9E3	4.2E2	6.2E3	4.2E2	6.4E3
<sup>3</sup> H in total waste (Ci)	1.6E2	1.4E3	1.6E2	1.5E3	1.6E2	1.5E3	1.6E2	1.6E3	1.6E2	1.7E3	1.6E2	1.8E3
<sup>3</sup> H in concentrated waste (Ci)	1.6E2	1.4E3	1.6E2	1.5E3	1.6E2	1.5E3	1.6E2	1.6E3	1.6E2	1.7E3	1.6E2	1.8E3
6. Total solids (g)	1.2E5	2.1E6	1.2E5	2.3E6	1.2E5	2.4E6	1.2E5	2.5E6	1.2E5	2.6E6	1.2E5	2.7E6
7. Acid concentration (N)	Neutralized wastes											
8. Total FP solids (g)	4.5E2	4.9E3	4.5E2	5.4E3	4.5E2	5.8E3	4.5E2	6.3E3	4.5E2	6.7E3	4.5E2	7.2E3

<sup>a</sup>As of the end of each year.

Tables 10, 11, 12, 13, 14, and 15 show the amount of high- and intermediate-level wastes produced by the proposed pilot plants each year and the cumulative amounts for the years 1976 to 1981 (TRU, HTGR, LMFBR, and LWBR). The pilot plants would operate in the period from 1976 to 1979 and the wastes would be evaporated and stored in one of the 50,000-gal stainless-steel storage tanks. The total volume is 19,000 gal and in the year 1981, the combined wastes will contain  $3.1 \times 10^7$  Ci of gross  $\beta$ - $\gamma$ , 2.0 kg  $^{239}\text{Pu}$ , 250 g of  $^{244}\text{Cm}$ ,  $4.0 \times 10^6$  Ci of  $^{137}\text{Cs}$ , and  $2.9 \times 10^6$  Ci of  $^{90}\text{Sr}$ . The TRU wastes (Table 10) have also been included in the listing for better comparison with present waste generation rates, although these wastes will probably be combined with other ILW and disposed of routinely.

We assumed that each pilot plant would evaporate its waste to the volumes shown and transfer it to one of the stainless steel tanks for storage. The LMFBR, LWR, and HTGR wastes would be stored as acidic solutions to maintain a minimum volume and to prevent formation of a sludge-precipitate. The waste would be neutralized prior to disposal by hydrofracture. If stored, the TRU wastes must be neutralized with caustic and held in the second stainless steel storage tank since these wastes contain high concentrations of chloride and they would be extremely corrosive during storage as acidic solutions. The intermediate-level wastes from the pilot plants will contain  $<0.1$   $\beta$ - $\gamma$  Ci/gal and would be blended with regular ILW. The combined high-level wastes would be held for decay till about 1981. Disposal operations would start in 1981 and extend over a 3 to 4-year period, thus providing an average cooling time of about 5 years (Sect. 6.5.1.8).

#### 6.4.5 Organic Solvent Reagent Wastes

The volume and composition of the contaminated organic solvent and reagent wastes are estimated as less than 1,000 gal per year (Sect. 6.3.2 and Sect. 7). Most of the solvent waste originates in the TRU plant but significant amounts of alcohols and other reagents are discharged in other buildings. Additional solvent would arise in the proposed pilot plants from solvent extraction operations.

Table 11.  
Projections of High-Level Radioactive Aqueous Waste Generation  
Project: LWBR - 1st Cycle

	1976		1977		1978		1979		1980		1981	
	Produced	Cumulative <sup>a</sup>										
1. Total volume (gal)	5.6E3	5.6E3	1.1E4	1.7E4	2.8E4	4.3E4	5.6E4	1.0E5	-	1.0E5	-	1.0E5
1a. Volume after evaporation (gal)	1.4E2	1.4E2	2.8E2	4.2E2	6.9E2	1.1E3	1.4E3	2.5E3	-	2.5E3	-	2.5E3
2. Total $\beta$ , $\gamma$ activity (Ci)	1.1E6	1.1E6	1.2E6	1.7E6	1.6E6	2.5E6	2.2E6	4.0E6	-	3.2E6	-	2.7E6
3. Total $\alpha$ activity (Ci)	4.4E3	4.4E3	8.8E3	1.3E4	5.4E3	8.6E3	1.1E4	1.9E4	-	1.3E4	-	9.7E3
5. Radionuclides												
Pu (g)	4.2E0	4.2E0	8.4E0	1.3E1	2.1E1	3.3E1	4.2E1	7.5E1	-	7.5E1	-	7.4E1
Pu (Ci)	5.0E1	5.0E1	9.7E1	1.5E2	2.3E2	3.7E2	4.4E2	7.9E2	-	7.6E2	-	7.2E2
<sup>244</sup> Cm (g)	7.6E-4	7.6E-4	9.2E-4	1.4E-3	1.3E-3	2.0E-3	2.0E-3	3.7E-3	-	3.4E-3	-	3.2E-2
<sup>244</sup> Cm (Ci)	1.84E0	1.84E0	1.7E0	2.5E0	9.6E-1	1.5E0	5.4E-1	9.8E-1	-	4.4E-1	-	3.3E-1
<sup>137</sup> Cs (g)	3.6E2	3.6E2	7.2E2	1.1E3	1.8E3	2.8E3	3.5E3	6.3E3	-	6.0E3	-	5.9E3
<sup>137</sup> Cs (Ci)	3.2E4	3.2E4	6.3E4	9.5E4	1.5E5	2.4E5	3.1E5	5.7E5	-	5.6E5	-	5.5E5
<sup>90</sup> Sr (g)	2.3E2	2.3E2	4.7E2	7.0E2	1.1E3	1.8E3	2.2E3	3.9E3	-	3.8E3	-	3.8E3
<sup>90</sup> Sr (Ci)	3.3E4	3.3E4	6.6E4	1.0E5	1.6E5	2.5E5	3.1E5	5.7E5	-	5.4E5	-	5.2E5
<sup>3</sup> H in total waste (Ci)	1.2E2	1.2E2	2.4E2	3.5E2	5.5E2	8.8E2	1.1E3	1.9E3	-	1.8E3	-	1.7E3
<sup>3</sup> H in concentrated waste <sup>b</sup> (Ci)	3.1E0	3.1E0	5.9E0	8.8E0	1.4E1	2.2E1	2.6E1	4.7E1	-	4.5E1	-	4.2E1
6. Total solids (g)	6.4E4	6.4E4	1.3E5	1.9E5	1.3E5	2.0E5	2.5E5	4.5E5	-	4.5E5	-	4.5E5
7. Acid concentration (N)	2.1	2.1	2.1	2.1	2.1	2.1	2.1	2.1	-	2.1	-	2.1
8. Total FP solids (g)	9.9E3	9.9E3	2.0E4	3.0E4	4.9E4	7.9E4	9.9E4	1.8E5	-	1.8E5	-	1.8E5

<sup>a</sup>As of the end of each year.

<sup>b</sup>Balance of <sup>3</sup>H is in evaporator condensate.

Table 12.  
Projections of High-Level Radioactive Aqueous Waste Generation  
Project: LMFBR - 1st Cycle

	1976		1977		1978		1979		1980		1981	
	Produced	Cumulative <sup>a</sup>										
1. Total volume (gal)	2.0E4	2.0E4	2.0E4	4.0E4	2.0E4	6.0E4	2.0E4	8.0E4	2.0E4	1.0E5	2.0E4	1.2E5
1a. Volume after evaporation (gal)	2.0E3	2.0E3	2.0E3	4.0E3	2.0E3	6.0E3	2.0E3	8.0E3	2.0E3	1.0E4	2.0E3	1.2E4
2. Total $\beta$ , $\gamma$ activity (Ci)	1.5E7	1.5E7	3.0E7	4.0E7	3.0E7	4.6E7	3.0E7	4.6E7	3.0E7	4.6E7	3.0E7	4.6E7
3. Total $\alpha$ activity (Ci)	1.0E4	1.0E4	1.8E4	2.1E4	1.8E4	2.2E4	1.8E4	2.2E4	1.8E4	2.2E4	1.8E4	2.2E4
4. Heat generation (watts)	5.0E4	5.0E4	1.0E5	1.3E5	1.0E5	1.5E5	1.0E5	1.5E5	1.0E5	1.5E5	1.0E5	1.5E5
5. Radionuclides												
Pu (g)	6.0E2	6.0E2	6.0E2	1.2E3	6.0E2	1.8E3	6.0E2	1.8E3	6.0E2	1.8E3	6.0E2	1.8E3
Pu (Ci)	1.0E3	1.0E3	1.0E3	2.0E3	1.0E3	3.0E3	1.0E3	3.0E3	1.0E3	3.0E3	1.0E3	3.0E3
<sup>244</sup> Cm (g)	4.5E0	4.5E0	4.5E0	9.0E0	4.5E0	1.4E1	4.5E0	1.4E1	4.5E0	1.4E1	4.5E0	1.4E1
<sup>244</sup> Cm (Ci)	3.6E2	3.6E2	3.6E2	7.3E2	3.6E2	1.1E3	3.6E2	1.1E3	3.6E2	1.1E3	3.6E2	1.1E3
<sup>137</sup> Cs (g)	5.0E3	5.0E3	5.0E3	1.0E4	5.0E3	1.4E4	5.0E3	1.4E4	5.0E3	1.4E4	5.0E3	1.4E4
<sup>137</sup> Cs (Ci)	4.5E5	4.5E5	4.5E5	9.0E5	4.5E5	1.3E6	4.5E5	1.3E6	4.5E5	1.3E6	4.5E5	1.3E6
<sup>90</sup> Sr (g)	1.2E3	1.2E3	1.2E3	2.4E3	1.2E3	3.5E3	1.2E3	3.5E3	1.2E3	3.5E3	1.2E3	3.5E3
<sup>90</sup> Sr (Ci)	1.7E5	1.7E5	1.7E5	3.4E5	1.7E5	5.0E5	1.7E5	5.0E5	1.7E5	5.0E5	1.7E5	5.0E5
<sup>3</sup> H, total <sup>b</sup> (Ci)	3.9E1	3.9E1	3.9E1	7.7E1	3.9E1	1.0E2	3.9E1	1.0E2	3.9E1	1.0E2	3.9E1	1.0E2
6. Total solids (g)	6.0E5	6.0E5	6.0E5	1.2E6	6.0E5	1.8E6	6.0E5	1.8E6	6.0E5	1.8E6	6.0E5	1.8E6
7. Acid concentration (N)	7	7	7	7	7	7	7	7	7	7	7	7
8. Total FP solids (g)	1.4E5	1.4E5	1.4E5	2.8E5	1.4E5	4.2E5	1.4E5	4.2E5	1.4E5	4.2E5	1.4E5	4.2E5

<sup>a</sup> As of the end of each year.

<sup>b</sup> Assumes that 1% of <sup>3</sup>H produced in fuel elements reaches fuel reprocessing plant, and that this 1% is removed in a head-end operation which concentrates and keeps it separate from other wastes.

Table 13.  
Projections of High-Level Radioactive Aqueous Waste Generation  
Project: HTGR - 1st Cycle

	1976		1977		1978		1979		1980		1981	
	Produced	Cumulative <sup>a</sup>										
1. Total volume (gal)	1.5E4	1.5E4	5.0E3	2.0E4	5.0E3	2.5E4	5.0E3	3.0E4	-	3.0E4	-	3.0E4
1a. Volume after evaporation (gal)	6.8E2	6.8E2	3.8E2	1.1E3	4.8E2	1.5E3	5.1E2	2.0E3	-	2.0E3	-	2.0E3
2. Total $\beta, \gamma$ activity (Ci)	6.2E6	6.2E6	5.9E6	1.0E7	7.6E6	1.4E7	8.2E6	1.7E7	-	1.2E7	-	8.4E6
3. Total $\alpha$ activity (Ci)	1.7E4	1.7E4	1.3E4	2.3E4	2.0E4	3.2E4	2.3E4	4.2E4	-	2.5E4	-	1.5E4
4. Heat generation (watts)	2.6E4	2.6E4	2.6E4	4.4E4	3.5E4	6.2E4	3.8E4	8.1E4	-	5.7E4	-	4.2E4
5. Radionuclides												
Pu (g)	9.0E0	9.0E0	3.0E1	4.0E1	8.9E1	1.3E2	1.1E2	2.4E2	-	2.4E2	-	2.4E2
Pu (Ci)	1.6E2	1.6E2	5.7E2	7.3E2	1.6E3	8.9E3	2.0E3	1.1E4	-	1.1E4	-	1.1E4
<sup>244</sup> Cm (g)	1.2E-2	1.2E-2	7.1E-2	8.3E-2	5.9E-1	6.7E-1	7.7E-1	1.4E0	-	1.4E0	-	1.4E0
<sup>244</sup> Cm (Ci)	9.8E-1	9.8E-1	5.8E0	6.8E0	4.8E1	5.5E1	6.3E1	1.2E2	-	1.2E2	-	1.2E2
<sup>137</sup> Cs (g)	5.5E3	5.5E3	5.2E3	1.1E4	7.2E3	1.8E4	7.7E3	2.6E4	-	2.6E4	-	2.6E4
<sup>137</sup> Cs (Ci)	4.8E5	4.8E5	4.5E5	9.3E5	6.3E5	1.6E6	6.7E5	2.2E6	-	2.2E6	-	2.2E6
<sup>90</sup> Sr (g)	3.4E3	3.4E3	3.2E3	6.5E3	4.3E3	1.1E4	4.5E3	1.5E4	-	1.5E4	-	1.4E4
<sup>90</sup> Sr (Ci)	4.8E5	4.8E5	4.5E5	9.2E5	6.0E5	1.5E6	6.4E5	2.1E6	-	2.0E6	-	1.9E6
<sup>3</sup> H, total <sup>b</sup> (Ci)	1.8E3	1.8E3	1.6E3	3.3E3	2.2E3	5.3E3	2.4E3	7.4E3	-	7.0E3	-	6.7E3
6. Total solids (g)	2.8E5	2.8E5	1.7E5	4.5E5	2.2E5	6.7E5	2.3E5	9.0E5	-	9.0E5	-	9.0E5
7. Acid concentration (N)	1.5	1.5	1.5	1.5	1.5	1.5	1.5	1.5	-	1.5	-	1.5
8. Total FP solids (g)	1.3E5	1.3E5	1.2E5	2.5E5	1.7E5	4.2E5	1.8E5	6.0E5	-	6.0E5	-	6.0E5

<sup>a</sup> As of the end of each year.

<sup>b</sup> Assumes that 99% of <sup>3</sup>H in the fertile particles is found in off-gases and is recovered and stored as a separate concentrated waste (preferably solid).

Table 14.  
Projections of High-Level Radioactive Aqueous Waste Generation  
Project: LMFBR + HTGR + LWBR + TRU

	1976		1977		1978		1979		1980		1981	
	Produced	Cumulative <sup>a,b</sup>	Produced	Cumulative <sup>a</sup>								
1. Total volume (gal)	4.1E4	4.6E4	3.6E4	8.3E4	5.3E4	1.3E5	6.1E4	2.0E5	4.0E2	2.0E5	4.0E2	2.0E5
1a. Volume after evaporation (gal)	2.9E3	9.4E3	3.1E3	1.3E4	3.6E3	1.6E4	2.3E3	1.8E4	4.0E2	1.9E4	4.0E2	1.9E4
2. Total $\beta, \gamma$ activity (Ci)	2.2E7	2.2E7	3.7E7	5.2E7	3.9E7	6.3E7	1.1E7	4.5E7	1.1E5	3.7E7	1.1E5	3.1E7
3. Total $\alpha$ activity (Ci)	2.9E4	3.7E4	3.3E4	5.6E4	4.0E4	6.7E4	2.5E4	7.8E4	1.6E3	6.2E4	1.6E3	5.2E4
4. Heat generation (watts)	8.1E4	8.1E4	1.4E5	1.9E5	1.4E5	2.2E5	4.9E4	2.0E5	3.5E2	1.3E5	3.5E2	1.0E5
5. Radionuclides												
Pu (g)	6.1E2	6.1E2	6.3E2	1.2E3	6.9E2	1.9E3	1.1E2	2.0E3	-	2.0E3	-	2.0E3
Pu (Ci)	1.2E3	1.2E3	1.6E3	2.7E3	2.6E3	1.2E4	2.0E3	1.4E4	-	1.4E4	-	1.4E4
<sup>244</sup> Cm (g)	2.5E1	1.3E2	2.5E1	1.6E2	2.5E1	1.8E2	2.2E1	2.1E2	2.0E1	2.3E2	2.0E1	2.5E2
<sup>244</sup> Cm (Ci)	2.0E3	1.1E4	2.0E3	1.3E4	2.0E3	1.5E4	1.8E3	1.7E4	1.6E3	1.9E4	1.6E3	2.0E4
<sup>137</sup> Cs (g)	1.1E4	1.1E4	1.1E4	2.2E4	1.4E4	3.5E4	1.1E4	4.6E4	1.4E1	4.6E4	1.4E1	4.6E4
<sup>137</sup> Cs (Ci)	9.6E5	9.6E5	9.6E5	1.9E6	1.2E6	3.1E6	9.6E5	4.0E6	1.2E3	4.0E6	1.2E3	4.0E6
<sup>90</sup> Sr (g)	4.8E3	4.8E3	4.9E3	9.6E3	6.6E3	1.6E4	6.7E3	2.2E4	2.9E0	2.2E4	2.9E0	2.1E4
<sup>90</sup> Sr (Ci)	6.7E5	6.7E5	6.9E5	1.3E6	9.2E5	2.2E6	9.4E5	3.1E6	4.1E2	3.1E6	4.1E2	2.9E6
<sup>3</sup> H in total waste <sup>c</sup> (Ci)	2.8E2	1.5E3	4.0E2	1.8E3	7.1E2	2.4E3	1.3E3	3.5E3	1.6E2	3.5E3	1.6E2	3.5E3
<sup>3</sup> H in concentrated waste (Ci)	1.6E2	1.4E3	1.7E2	1.5E3	1.7E2	1.5E3	1.9E2	1.6E3	1.6E2	1.7E3	1.6E2	1.8E3
6. Total solids (g)	1.1E6	3.0E6	1.0E6	4.1E6	1.1E6	5.1E6	6.0E5	5.7E6	1.2E5	5.8E6	1.2E5	5.9E6
7. Acid concentration (N)	-	-	-	-	-	-	-	-	-	-	-	-
8. Total FP solids (g)	2.8E5	2.8E5	2.8E5	5.6E5	3.6E5	9.2E5	2.8E5	1.2E6	4.5E2	1.2E6	4.5E2	1.2E6

<sup>a</sup> As of the end of each year.

<sup>b</sup> Includes previous five years of TRU operations.

<sup>c</sup> Does not include the <sup>3</sup>H from LMFBR (~100 curies) or HTGR (~7,400 curies) operations since the <sup>3</sup>H from these operations will be recovered and treated separately.

Table 15.  
Projections of Intermediate-Level Radioactive Aqueous Waste Generation  
Project: LMFBR + HTGR + LWBR

	1976		1977		1978		1979		1980		1981	
	Produced	Cumulative <sup>a</sup>	Produced	Cumulative <sup>a</sup>	Produced	Cumulative <sup>a</sup>	Produced	Cumulative <sup>a</sup>	Produced	Cumulative <sup>a</sup>	Produced	Cumulative <sup>a</sup>
1. Total volume (gal)	6.8E4	6.8E4	5.8E4	1.3E5	8.2E4	2.1E5	8.4E4	3.0E5	8.4E4	3.0E5 <sup>b</sup>	-	-
1a. Volume after evaporation (gal)	6.8E3	6.8E3	5.8E3	1.3E4	8.2E3	2.1E4	8.4E3	3.0E4	8.4E3	3.0E4	-	-
2. Total $\beta, \gamma$ activity (Ci)	-----Waste average ~ 0.1 $\beta$ - $\gamma$ curie per gallon at most. -----											
3. Total $\alpha$ activity (Ci)	-----Waste average ~ 0.01 $\alpha$ curie per gallon at most. -----											
4. Heat generation (watts)	Negligible											
5. Radionuclides												
Pu (g)	-	-	-	-	-	-	-	-	-	-	-	-
Pu (Ci)	-	-	-	-	-	-	-	-	-	-	-	-
<sup>244</sup> Cm (g)	-	-	-	-	-	-	-	-	-	-	-	-
<sup>244</sup> Cm (Ci)	-	-	-	-	-	-	-	-	-	-	-	-
<sup>137</sup> Cs (g)	-	-	-	-	-	-	-	-	-	-	-	-
<sup>137</sup> Cs (Ci)	-	-	-	-	-	-	-	-	-	-	-	-
<sup>90</sup> Sr (g)	-	-	-	-	-	-	-	-	-	-	-	-
<sup>90</sup> Sr (Ci)	-	-	-	-	-	-	-	-	-	-	-	-
<sup>3</sup> H in total waste (Ci)	-	-	-	-	-	-	-	-	-	-	-	-
<sup>3</sup> H in concentrated waste (Ci)	-	-	-	-	-	-	-	-	-	-	-	-
6. Total dissolved solids (g)	1.3E7	1.3E7	1.1E7	2.5E7	1.5E7	4.0E7	1.6E7	5.7E7	1.6E7	5.7E7	-	-
7. Acid concentration	Neutralized wastes											
8. Total FP solids (g)	-	-	-	-	-	-	-	-	-	-	-	-

<sup>a</sup> As of the end of each year.

<sup>b</sup> Waste are primarily sodium carbonate/nitrate; about 50 g per liter unconcentrated; and about 500 g per liter concentrated.

#### 6.4.6 Process Waste Sludge

The operation of the lime-soda process waste treatment plant is expected to continue for 4 to 6 years until a new process waste treatment plant is installed (Sect. 7). The sludge from the lime-soda plant consists of about 8,000 gal/week ( $4.16 \times 10^5$  gal/yr) of a water slurry containing 7,400 lbs of solids (Table 16). With possible revisions of the operating conditions, it is expected that the solids content can be reduced to 3,000 lbs in the same total volume (i.e., 4.5 wt % solids). A solids content of  $\sim 4.5$  wt % is assumed to be that required for preparing grout for hydrofracture (Sect. 6.5.1.2, Volume of waste). The  $^{90}\text{Sr}$  content is estimated to be less than 10 Ci/yr and the alpha content is below detection.

If the scavenging precipitation--ion exchange process (SPIX) is selected to replace the lime-soda process, the slurry volume will decrease to 2,400 to 3,600 gal/wk and would contain 900 to 1360 lbs of solids. If the clarification--ion exchange process is used, the volume of slurry would decrease to zero, if the calcium and other solids remain in solution in the ion exchange eluant concentrate after evaporation. The solids are expected to precipitate when the eluate-concentrate is made alkaline as part of the feed preparation step for hydrofracture. At this point, the weight of sludge-solids would be about the same from the SPIX and the clarification-IX process. However, the solution should be easier to handle than the slurry in operations prior to the hydrofracture step.

#### 6.4.7 Chemical (Nitrate) Wastes

The composition of normal chemical wastes is presented in Sect. 10. However, a large additional amount of nitrate-bearing waste will be generated, if the process, ILW, and pilot plant wastes are calcined rather than hydrofractured. During calcination, the nitrate salts are decomposed and recovered as nitric acid in the condensate.

The process waste contains 15 to 30 ppm nitrate or 5 to 10 tons of nitrate per year. This nitrate passes through the lime-soda plant to White Oak Creek and

Table 16. Estimated Composition of Process Waste Sludges

Plant - 100 gpm, 5-day week  
 Slurry - 4.5 wt % solids  
 $^{90}\text{Sr}$  - < 10 Ci/yr  
 Alpha - Not detectable

Process	Sludge (Solids) Weight (lb/wk)	Slurry Volume (gal/wk)
Present lime-soda (11.7 wt % solids)	7,400	8,000
Optimized Lime-Soda	3,000	8,000
Scavenging precipitation--Ion exchange Process		
Calculated	900	2,400
Pilot plant data (ORNL-3349)	1,360	3,600
Clarification Ion Exchange Process		
Calculated	290	0 <sup>a</sup>
	~ 1,100	~ 3,000

<sup>a</sup>Zero if solids remain in solution in evaporated and neutralized ion exchange eluate;  
 ~ 1,100 if solids precipitate on neutralization in preparation for hydrofracture.

would also pass through a scavenging-precipitation--ion exchange plant (SPIX), if the lime-soda plant is replaced by a SPIX plant. This nitrate increases the concentration of nitrate in the creek by 0.5 to 1.0 ppm (Table 17). The evaporation and calcination of the eluate from the SPIX process would produce 13.6 tons of nitrate annually, corresponding to about 1.4 ppm in White Oak Creek, if discharged on an average annual basis. The calcination of all of the high-level wastes would produce 2.7 tons of nitrate and the concentration in White Oak Creek would be 0.9 ppm, if discharged over a 3-month period or 0.18 ppm on an annual discharge basis. The total nitrates from the process waste system would amount to ~ 44 tons per year (4.4 ppm nitrate in White Oak Creek) or 46.5 tons for a single year in which the pilot plant wastes are also calcined.

## 6.5 Waste Disposal Methods

Hydrofracture is recommended by this subcommittee as the preferred method for disposal of wastes at ORNL based on safety, cost, environmental effects, AEC policy (favoring on-site disposal "if practical"), and impact on laboratory activities. The reasons for recommending hydrofracturing vs promising alternatives are presented in the following sections.

### 6.5.1 Hydrofracture

6.5.1.1 Safety. — A complete evaluation of the safety of the surface operations and of the underground storage area was prepared along with operating limits.<sup>2</sup> After extensive consideration of geologic and ecological factors, including heat dissipation, earthquakes, erosion, and effects of faulting, it was concluded "that the hydraulic fracturing process is probably the safest and most effective method currently for permanently removing radioactive waste from man's environment." We propose to use the same operating limits for the underground storage area (Sect. 6.5.1.2), and if the above safety assessment is accepted by the AEC, we believe it is logical to recommend that ORNL continue the use of hydrofracturing for all future aqueous wastes. This conforms to the announced AEC policy of favoring "on site" disposal.<sup>1</sup>

Table 17. Estimated Amounts of Nitrates Produced by Calcination of Wastes and Nitrate Concentrations, if Discharged to White Oak Creek

Public Health Service drinking limit: 44 ppm nitrate

Creek Flow: 12.5 cfs

Plant: 100 gpm, 5 day week

Process	Weight of Nitrate (tons)	Incremental Nitrate Conc. in White Oak Creek (ppm)
(1) Lime-soda process, not calcined (annually)	5-10	0.5-1.0
(2) Calcination of ILW (annually)	20.2	2
(3) Calcination of HTGR (one 12-month period)	0.4	0.05
(4) Calcination of combined high-level waste (one 12-month period)	2.7	0.18
(5) Calcination of eluate from scavenging precipitation-ion exchange process (annually)	13.6	1.4
(6) Total from (1), (2), and (5) (annually)	43.8	4.4
(7) Total from (1), (2), (4), and (5) (one 12-month period)	46.5	4.58

We recommend, however, that the limits for surface operations be revised to permit the handling of higher levels of waste, after a new, permanent, hydrofracturing facility has been constructed (Sect. 6.5.1.2).

6.5.1.2 Operating Limits. — The safety analysis<sup>2</sup> presents a complete set of operating limits for use with ILW. Most of these limits would be retained for future operations for disposal of tank sludges and high-level wastes. The general operating procedure would be to store and "cool" the combined high-level waste for 5 years or longer and then dilute the waste with ILW or water to meet the operating limits for ILW. Similarly, the tank sludges would be diluted with the water used to move them out of the tanks to meet the limits. Discussions of the crucial limits, as applied to future operations, are presented as follows. Other operating limits would be the same as in ref. 2.

Transuranic Activity. — The safety analysis<sup>2</sup> states that "the transuranic activity of the waste shall be limited to an average of  $2 \times 10^{-3}$  Ci/gal" (meaning long-lived alpha, i.e.,  $^{239}\text{Pu}$ ). This specification would be retained. It assures that the concentration in the underground shale-grout mixture will be limited to 10  $\mu\text{Ci/kg}$  (250  $\mu\text{Ci/kg}$  in the grout alone). The tentative AEC limit for surface burial is 10  $\mu\text{Ci/kg}$  as proposed in a revision to the AEC Chapter Manual 0511 on April 9, 1971.<sup>4</sup> The concentration of long-lived alpha emitters in the surface of the earth is close to 10  $\mu\text{Ci/kg}$ . The term transuranic is interpreted to mean long-lived alpha emitters, namely  $^{239}\text{Pu}$ . The proposed revision exempted  $^{238}\text{Pu}$  ( $T_{1/2}$  89 yr) from consideration and indicated that other isotopes could also be exempted. ORNL requested exemption for  $^{244}\text{Cm}$  ( $T_{1/2}$  18.1 yr).

The safety analysis also limits the alpha content of surface solutions to a maximum of  $5 \times 10^{-3}$  Ci/gal. This specification is based on the containment capability of the revised interim facility for handling "present type" ILW. In the present type waste, >99% of the  $^{244}\text{Cm}$  remains in the bottom of tanks W-6 and W-8. Therefore, for future operations, in which the tank sludge or other wastes containing significant amounts of  $^{244}\text{Cm}$  is hydrofractured, the limit for alpha content (for surface containment) should be increased to at least  $1 \times 10^{-2}$  Ci/gal

and the permanent facility should have a corresponding containment capability (Sect. 6.5.1.7).

Beta-Gamma Activity. — The safety analysis<sup>2</sup> proposes a limit of 2 Ci/gal of beta-gamma activity based on the shielding capability of the interim facility. This limit should be increased to 10 to 20 Ci/gal in the permanent facility (Sect. 6.5.1.7). The use of a higher limit will decrease the volume of waste and increase the volumetric storage capacity of the underground storage area.

Temperature in Underground Storage Area. — The safety analysis<sup>2</sup> proposes a limit in temperature rise of 70°F and this limit would be retained. This limit would not be increased unless new data substantiating an increase became available. The limit of 70°F is based on calculations made on a 1-dimensional model and later confirmed on a three-dimensional model (finite limit, r - z cylindrical) using the IBM-360 heat calculation code (CTC-INS-980 Turner and Crowell). The heat capacity of the present well is limited to the amount that will raise the subsurface temperature 70°F, i.e.,  $1.26 \times 10^5$  watts (Table 18). On this basis, the well could accommodate all ORNL wastes for 39 years.

Table 18. Heat Limit of Present Well

Assumptions: 3 acres  
200 ft thick and 500 ft underground  
temperature limit =  $\Delta t$  of 70°F  
heat production limit =  $1.26 \times 10^5$  watts  
30 year half life

	Heat Production ( $10^5$ watts)
Previous wastes	0.018
Tank sludge	0.058
New pilot plants	1.0
ORNL-ILW at 465 watts/yr at 39 yrs	<u>0.18</u>
Total Watts	1.26

Present well could accommodate all wastes for 39 years.

Volume of Waste. — The safety report<sup>2</sup> indicates that the remaining capacity of the present well is  $7.5 \times 10^6$  gal as of January 1971 (Table 19). Approximately one million gallons of the underground storage capacity had been used in previous hydrofracture disposal operations.<sup>2</sup> McClain<sup>5</sup> has estimated a total capacity of 8 to 10 million gallons for this well and de Laguna<sup>6</sup> has estimated a capacity of 15 million gallons for a new well drilled closer to the TRU-HFIR operations area. Since the capacity of the present well seems adequate for 20 to 30 years, we recommend that the new permanent facility be constructed at the present well site.

Table 19 illustrates the volumetric limitations for three different modes of operation assuming a capacity of  $7.5 \times 10^6$  gal for grout in the old well and  $15 \times 10^6$  gal for a new well.

Case 1 shows that the present well will last 22 years or a new well >100 years, if the ILW is evaporated to the limit listed in the safety analysis, i.e., 2 Ci/gal, and the high level wastes diluted to 10 Ci/gal. This would also increase the concentration of  $\text{NaNO}_3$  to about 2.7 M in the ILW. However, satisfactory grout has been prepared from solutions containing 5 M  $\text{NaNO}_3$ .<sup>7</sup>

Case 2 shows that the life of the present well will be decreased to <2 years and a new well to 18 years, if the lime-soda process sludges are hydrofractured. Therefore, we recommended that the process sludges be disposed of in the burial ground pits, since they contain only about 10 Ci of  $^{90}\text{Sr}/\text{yr}$  (Sect. 6.4.6).

Case 3 shows that the life of the present well could be increased to 34 years, if the ILW is evaporated to 2 Ci/gal and  $\sim 2.7$  M  $\text{NaNO}_3$  for five years of operation of the interim plant and to 3.5 Ci/gal and 4.8 M  $\text{NaNO}_3$  for operation in the permanent plant. The high-level waste would be diluted to 20 Ci/gal for operation in the permanent plant. Under these conditions, the life of the present well would be reduced to 10 years, if the process wastes are hydrofractured.

The volumes of grout from the sludges are only best guesses, but they represent a large fraction of the total. Therefore, we recommend that research and development start immediately to determine the volume of water and grout required to accommodate the sludges.

Table 19. Volumetric Limit of Present and New Wells

Assumptions:

ILW - 80,000 gal/yr at 0.8 M NaNO<sub>3</sub>, 0.6 Ci/gal  
 Tank sludge -  $3.13 \times 10^5$  gal  
 High-level waste -  $1.9 \times 10^4$  gal  
 Process sludge -  $0.4 \times 10^6$  gal/yr

Dilution Factors for Each Batch Injection

(80,000 gal ILW)(1.56) = 125,000 gal grout  
 (25,000 gal tank sludge)(5) = 125,000 gal grout  
 (125,000 gal process sludge)(1) = 125,000 gal grout

Capacity of Old and New Wells

150 ft of shale available in old well  
 300 ft of shale available in new well  
 1 slot per 10 ft ~ 15 slots  
 Four 125,000-gal injections ~  $0.5 \times 10^6$  gal/slot; 1.3-in. surface rise/ $10^6$  gal;  
 10-in. rise maximum adjacent to well  
 Total remaining capacity of old well =  $7.5 \times 10^6$  gal of grout  
 Total capacity of new well =  $15 \times 10^6$  gal of grout

Case 1 - ILW at 2 Ci/gal and diluted HLW at 10 Ci/gal

<u>Waste</u>	<u>Gal of Grout</u>
ILW - 22 yrs at 2.7 M NaNO <sub>3</sub>	$0.81 \times 10^6$
Tank sludge - ( $3.13 \times 10^5$ gal)(5)	$1.57 \times 10^6$
High level waste - ( $1.9 \times 10^4$ gal)(270) <sup>a</sup>	$5.12 \times 10^6$
Total	$7.5 \times 10^6$

22 years of operation in old well.  
 < 100 years of operation in new well.

Case 2 - Same as Case 1 plus process sludge

ILW - 2 yrs	$0.07 \times 10^6$
Tank sludge	$1.57 \times 10^6$
High level waste	$5.12 \times 10^6$
Process sludge - 2 yrs	$0.8 \times 10^6$
Total	$7.5 \times 10^6$

< 2 years of operation in old well.  
 18 years of operation in new well.

Case 3 - ILW at 3.5 Ci/gal and diluted HLW at 20 Ci/gal

ILW - 5 yrs at 2 Ci/gal	$0.19 \times 10^6$
ILW - 29 yrs at 4.8 M NaNO <sub>3</sub>	$0.62 \times 10^6$
Tank sludge	$1.57 \times 10^6$
HLW	$2.56 \times 10^6$
Total	$7.5 \times 10^6$

34 years of operation in old well.  
 > 100 years of operation in new well.

<sup>a</sup>41 injections at 466 gal waste plus 79,530 gal dilution water to produce 10 Ci/gal; this dilution includes the additional dilution by factor of 1.56 during hydrofracture.

6.5.1.3 Hydrofracture of High-Level Wastes. – The high-level wastes from the pilot plants (Table 14) would be stored in the 50,000-gal tanks for an average of 5 years of cooling (Sect. 6.4.4). The acidic high-level waste concentrate would be neutralized and diluted with water to form 80,000-gal batches containing either 10 or 20 Ci/gal. Disposal operations would start in 1981, one 80,000-gal batch per month (466 gal high-level waste plus 79,530-gal water  $\approx$  10 Ci/gal) and would require 41 monthly injections over a period of 3.4 years (Cases 1 and 2, Table 19). This dilution includes the additional dilution by a factor of 1.57 during hydrofracture. If diluted to 20 Ci/gal, only 20 injections would be required over a period of 20 months (Case 3, Table 19). The ILW from the pilot plant (Table 15) will contain  $<0.1$   $\beta$ - $\gamma$  Ci/gal and it would be blended with the normal ILW directly. We recommend the use of 20 Ci/gal at the design limit for the permanent plant to conserve underground storage area.

The concentration of alpha in the underground storage area after hydrofracture of the combined TRU and HLW containing 20 beta-gamma curies/gal would be lower than the proposed limit (Sect. 6.5.1.2) of  $10 \mu\text{Ci}$  of long-lived alpha ( $^{239}\text{Pu}$ ) per kg of rock. The 2.0 kg of  $^{239}\text{Pu}$  in the HLW would amount to an average of  $0.096 \mu\text{Ci/kg}$  of rock in the total rock lens (storage area) or  $0.28 \mu\text{Ci/kg}$  of rock in the fraction of the rock storage area occupied by the HLW grout. The corresponding numbers for the 250 g of  $^{244}\text{Cm}$  are 15 and  $44 \mu\text{Ci/kg}$  of rock. The concentration of alpha in the surface solution before hydrofracture would be  $6.1 \times 10^{-3}$  Ci/gal, and thus lower than the proposed limit of  $1 \times 10^{-2}$  Ci/gal (Table 20).

The TRU wastes could not be hydrofractured as a separate single batch each year under the limits proposed for the interim plant or the permanent plant, since they would exceed the alpha limits specified for surface operations, i.e.,  $5 \times 10^{-3} \alpha$  Ci/gal and  $1 \times 10^{-2} \alpha$  Ci/gal, respectively (Sect. 6.5.1). The TRU wastes would contain  $\sim 0.25 \text{ mg } ^{244}\text{Cm/gal}$  or  $2 \times 10^{-2} \alpha$  Ci/gal, after dilution to 80,000 gal. In 1971 the TRU wastes contained  $\sim 150,000$  gross  $\beta$ - $\gamma$  curies, 8,000 Ci of  $^{137}\text{Cs}$ , 1500 Ci of  $^{90}\text{Sr}$ , and  $\sim 20$  g of  $^{244}\text{Cm}$ . These amounts will decrease to an annual rate of  $\sim 110,000$  gross  $\beta$ - $\gamma$  curies, 1300 Ci of  $^{137}\text{Cs}$ , 400 Ci of  $^{90}\text{Sr}$ , and 20 g of

Table 20. Alpha Concentrations in Rock Storage Area for Combined HTGR, LMFBR, LWBR, and TRU Wastes

	Average Concentration in Total Rock Lens ( $\mu\text{Ci } \alpha/\text{kg rock}$ )	Concentration in HL Fraction of Rock Lens ( $\mu\text{Ci } \alpha/\text{kg rock}$ )
<u>Waste at 10 <math>\beta, \gamma</math> Ci/gal</u>		
$^{239}\text{Pu}$	0.096	0.14
$^{244}\text{Cm}$	15	22
<u>Waste at 20 <math>\beta, \gamma</math> Ci/gal</u>		
$^{239}\text{Pu}$	0.096	0.28
$^{244}\text{Cm}$	15	44

$^{244}\text{Cm}$  in 1973 and succeeding years (Tables 9 and 10). We recommend that the TRU wastes be processed each year in the ILW system (Sect. 6.4.3). Thus, the plutonium, curium, and a large fraction of the other radionuclides will be precipitated and held in tank W-6. They would be removed later with the sludge and hydrofractured.

6.5.1.4 Hydrofracture of Tank Sludges. — In previous budget submissions, ORNL and AEC have recognized the need for emptying the Gunnite tanks. This should be postponed until a permanent facility with better containment and shielding is available in about 1976. A development program, probably in the range of \$250,000, is required to study problems such as formulation of grout, hydraulic sluicing, slurry suspension, and test equipment items for the hydrofracture plant.

A complete plan for the development of procedures for removal of the sludge from the tanks, for grout formulation, and for hydrofracture of the sludge was prepared by Frederick et al.<sup>8</sup> The program plan suggested dilution of the sludge

by a factor of five to achieve mobility and the hydrofracture of twenty 80,000-gal batches. We recommend the activation of this plan on a schedule that would complete the sludge disposal operations prior to 1981, when the disposal of high-level wastes would start (Sect. 6.5.1.8).

The concentration of alpha (Pu) in the underground storage area would be lower than the proposed limit of 10  $\mu\text{Ci}/\text{kg}$  of rock (Sect. 6.5.1.2). The 6.92 kg of  $^{239}\text{Pu}$  and the 154 g of  $^{244}\text{Cm}$  would amount to 0.33 and 9.6  $\mu\text{Ci}/\text{kg}$  of rock, respectively, in the total storage area or 1.6 and 46  $\mu\text{Ci}/\text{kg}$  of rock, respectively, in the fraction of the rock storage area occupied by the sludge grout.

6.5.1.5 Disposal of Organic Wastes. — See Sect. 6.2 for recommendations.

6.5.1.6 Revision of Hydrofracture Facility for Interim Operations. — We recommend that the hydrofracture facility be revised immediately to permit operation for an interim period of  $\sim 5$  years until a permanent facility is available. ORNL sent AEC-ORO a letter requesting this revision in FY 1972 for an estimated cost of \$172,000 (Table 21). The facility will be capable of handling intermediate-level wastes similar to those hydrofractured in the last 5 years. The pipeline to the hydrofracture facility should also be replaced in FY 1972.

6.5.1.7 Permanent Revision of Hydrofracture Facility. — An extensive revision of the hydrofracture facility will be required to convert it to a permanent facility which can handle all ORNL wastes, including the sludges from the present tanks and wastes from future pilot plant operations. This revision is required to safely process the sludges which contain significant concentrations of  $^{239}\text{Pu}$ ,  $^{90}\text{Sr}$ , and  $^{244}\text{Cm}$ . A tentative cost estimate of  $\$1.19 \times 10^6$  (escalated to  $\$1.6 \times 10^6$  in the year 1975) for this revision is presented in Table 22. However, at the present time, we do not know how to select certain equipment items (such as backflow preventers) or to design the slurry moving system. Therefore, the cost estimate represents a "best guess." Engineering development is required to obtain the necessary information.

6.5.1.8 Schedule and Costs for Disposal by Hydrofracture. — A plan for liquid waste disposal by hydrofracture is presented in Table 23. Upgrading of the present

Table 21. Revision of Hydrofracture Facility for Interim Operations

1. Fabricate and install new mixing tub		\$ 18,807
2. New densometers and circulating pump		9,972
3. Rework piping in mixing cell		11,168
4. Shielding on injection pump and roof of mixing cell		4,665
5. Improve ventilation on mixing cell and tub		3,909
6. Double filter waste tank exhaust and add ventilation to pumphouse		4,575
7. Replace high pressure screwed piping and valves with flanged piping and remote valves		<u>48,474</u>
	Sub-Total	\$101,570
Inspection and Engineering		<u>17,550</u>
	Sub-Total	\$119,120
Contingency		<u>23,880</u>
	Sub-Total	\$143,000
Escalation after 7-1-71		<u>9,000</u>
	Sub-Total	\$152,000
Two new additional logging wells		<u>20,000</u>
	TOTAL	<u><u>\$172,000</u></u>

Table 22. Final Improvements to Hydrofracture  
(Very Preliminary)

1.	Demolish and enlarge mixing and injection cell: provide more shielding and better containment		\$ 203,000
2.	Demolish and enlarge pump and valve pit: provide more shielding and better containment		127,700
3.	Relocate and elevate cement storage tanks to provide better solids metering		70,000
4.	Purchase and install new equipment and reinstall all systems		
	a. Air dryer	\$12,400	
	b. New mixing tub viewing equipment	6,000	
	c. High pressure valves and piping	6,700	
	d. Slotting and handling equipment	<u>36,900</u>	
			62,000
5.	Provide wellhead safety valves.		28,600
6.	Equipment for slurry suspension and handling		25,500
7.	Re-do instrumentation		85,800
8.	Injection pump modifications, Halliburton		50,000
9.	Rework waste pits		22,000
10.	New waste storage tanks		<u>155,000</u>
		Sub-Total	829,600
	Engineering and Inspection		<u>167,900</u>
		Sub-Total	997,500
	Contingency		<u>199,500</u>
		Sub-Total	1,197,000
	7% 5 Years to 1975 Escalator*		<u>418,950</u>
		TOTAL	<u><u>\$1,615,950</u></u>

\*This assumes submittal of preliminary Schedule 44 by April 1972 and mid-point of construction in December 1975.

Table 23. Plan for Liquid Waste Disposal via Hydrofracture

	FY-72	FY-73	FY-74	FY-75	FY-76, 77, 78	FY-79, 80	FY-81, 82, 83
I. Present Hydrofracture Plant							
a. Minimum repairs and upgrading	\$172,000 GPP						
b. Install new transfer line	485,000 GPP						
c. Routine operation for normal wastes							
II. Development Work for Sludge Removal							
a. Mix development at ORNL	50,000 Op	\$25,000 Op					
b. Subcontract to Halliburton		50,000 Op					
c. Slurry handling problems	25,000 Op	50,000 Op					
d. Equipment for slurry handling tests	50,000 GPE						
III. Permanent Hydrofracture Facility							
a. Conceptual design	25,000 Op	25,000 Op					
b. Request for Authorization submitted	(~ \$1.6 M)						
c. Title I design							
d. Money authorized by Congress							
e. Title II design							
f. Construction							
g. Injection of normal wastes (to test new system)							
h. Slurry tests							
i. Empty sludge from Gunnite tanks (2-3 yrs at ~ \$250,000 per year operating costs)							
j. Normal waste disposal							
k. Disposal of high-level wastes							

hydrofracture facility and the construction of a new pipeline from tank W-8 to the hydrofracture site should be carried out in FY 1972 at costs of \$172,000 and \$485,000, respectively. The development of methods for mining the tank sludge, preparing a suitable grout mixture, and conceptual design of the permanent facility should start in FY 1972 at a cost of \$150,000 and should continue at the same level in FY 1973. Routine operation of the hydrofracture plant for disposal of ILW would continue into FY 1975 with an annual operating cost of \$80,000 (1975 dollars). Operations would cease during construction of the permanent hydrofracture plant at a capital cost of \$1.6 million (1975 dollars) and would resume in FY 1976 with the disposal of the tank sludge with an annual operating cost of \$220,000 (1975 dollars). The disposal of tank sludges would be completed in 2 or 3 years (FY 1976, 1977, 1978) and disposal of ILW would continue indefinitely. Total cost for processing the sludge was estimated<sup>8</sup> as roughly \$1.17 million, exclusive of costs for the continued routine hydrofracturing of ILW. About \$480,000 was for operating and amortization costs over 3 years (\$160,000 per year), \$300,000 for sludge removal and revisions to the hydrofracture facility, and \$390,000 for contingency. The report states that these costs were based on unproven technical assumptions and that a more detailed analysis is required. The disposal of high-level wastes from the anticipated pilot plants would start in FY 1981 and extend for 20 to 41 months with an annual operating cost of \$440,000 (1975 dollars). A comparison of the costs for the disposal of high-level wastes by hydrofracture and by solidification is given in Sect. 6.5.1.

## 6.5.2 Solidification

Conceptual studies and cost estimates of methods for solidifying the wastes were prepared for comparison with disposal by hydrofracture. The solidified wastes would be shipped to a national repository for final disposal.

6.5.2.1 Comparison of Costs for Calcination and Hydrofracture. — Preliminary cost estimates have been prepared by Holmes<sup>9</sup> for the disposal of HTGR or combined HTGR-LMFBR-LWBR (HLW) pilot plant, reprocessing wastes blended with ORNL

intermediate-level wastes (ILW). Disposal methods considered included hydrofracturing or solidification by pot or fluid bed calcination. The costs for solidifying the combined HTGR-LMFBR-LWBR wastes and the HTGR wastes alone were also estimated for comparison purposes. Results indicate that hydrofracturing is the lowest cost method for disposal of the high-level wastes. The incremental costs for solidification of either HTGR waste or HLW will be 0.7 to \$1 million above the cost for solidifying the ILW alone. The following cases were considered:

Case No.	Waste	Disposal Method	Location
I (A and B)	ILW <sup>a</sup>	Hydrofracture	ORNL
II	ILW + HTGR	Hydrofracture	ORNL
III	ILW + HLW <sup>c</sup>	Hydrofracture	ORNL
IV (A and B)	ILW <sup>a</sup>	Pot Calcination	FPPP <sup>b</sup>
V	ILW + HTGR	Pot Calcination	FPPP
VI	ILW + HLW <sup>c</sup>	Pot Calcination	FPPP
VII (A and B)	ILW <sup>a</sup>	Fluid Bed Calcination	FPPP
VIII	ILW + HTGR	Fluid Bed Calcination	FPPP
IX	ILW + HLW <sup>c</sup>	Fluid Bed Calcination	FPPP
X (A and B)	ILW <sup>a</sup>	Pot Calcination	New Building
XI	ILW + HTGR	Pot Calcination	New Building
XII	ILW + HLW <sup>c</sup>	Pot Calcination	New Building
XIII	HLW <sup>c</sup>	Pot Calcination	FPPP
XIV	HTGR	Pot Calcination	FPPP

<sup>a</sup>ORNL Intermediate-Level Waste (ILW) assumed to include the present ILW, TRU wastes plus the wastes from the proposed Process Waste Treatment Plant.

<sup>b</sup>ORNL Fission Product Pilot Plant.

<sup>c</sup>HLW-High-Level Waste (HTGR, LMFBR and LWBR).

The above cases were set up to determine the incremental costs of disposing of high-level wastes such as HTGR or HLW along with ORNL intermediate-level wastes (ILW) using several types of processes. The processes considered include hydrofracturing and solidification by pot and fluidized bed calcination. Results indicate that the total quantity of HTGR waste could be processed in one year by blending it with the ILW normally generated at ORNL. If hydrofracturing is used, a maximum level of 10 curies per gallon\* would be maintained so the volume of hydrofracture waste would be proportional to the activity level. If solidification processes are used (pot or fluid bed calcination), the high-level waste would be blended with the ILW, but very little increase in final solids volume would result since the solids content of the HTGR waste would be small in comparison with the total solids produced. The advantage of this method lies in the economy of placing more activity in a given pot for shipment to the repository. The same number of pots per year would be required for either ILW or ILW plus HTGR solidification.

If HLW is blended with the ILW, the duration of the operation period would be increased to three years for any of the disposal processes considered. The volume of hydrofracture waste would be increased over that required for ILW or ILW plus HTGR, but the annual number of pots containing solidified waste would be the same as for ILW or ILW plus HTGR.

The determination of incremental costs required that a cost basis be established for each type of mixed waste disposal. For HTGR waste blended with ILW, the capital and operating costs for ILW disposal over one year were used as the basis (Cases I-A, IV-A, VII-A, and X-A in the above table). For HLW blended with ILW, the capital cost plus three years of operating costs for ILW disposal served as the basis (Cases I-B, IV-B, VII-B, and X-B in the above table).

Incremental costs have been estimated for the following cases:

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\*10 Ci/gal was assumed for this analysis, although 20 Ci/gal is our final recommendation.

<u>Case No.</u>	<u>Incremental Cost</u>
II	II over I-A
III	III over I-B
V	V over IV-A
VI	VI over IV-B
VIII	VIII over VII-A
IX	IX over VII-B
XI	XI over X-A
XII	XII over X-B

The resulting total and incremental costs (capital costs plus operating costs over the operating period) are presented in Table 24. The overall results may be summarized as follows:

- (1) Hydrofracturing is lower in cost than any of the solidification systems considered for the disposal of pilot plant quantities of high-level waste blended with ORNL ILW (Sects. 6.5.1.8).
- (2) The capital cost for solidification of ILW will be about \$3.5 million in an existing building and \$6.9 million in a new building (1975 costs). If high-level wastes are blended with the ILW, an additional 0.7 to \$1 million should be added to provide for additional equipment and facilities. Solidification of this blended waste will require about 3 years. The annual operating costs for solidification of ILW are about \$1.7 million for ILW and \$1.8 million for ILW + HLW (1975 dollars). The annual operating cost for solidification of tank sludges plus ILW, assuming a volume reduction of 4.0 for sludges containing 25 wt % solids, would be about the same and would require about 10 years. The solidification of ILW-HLW could be postponed until solidification of the tank sludges was completed.

Table 24. Summary of Waste Disposal Cost Estimates (In Thousands of Dollars)

Case Number	Waste	Disposal Method	Location	Operating Time (years)	Capital Cost	Operating Cost Over Operating Period	Total Cost for Operating Period	Case Used for Incremental Cost Basis	Incremental Cost
I-A	ILW	Hydrofracture	ORNL	1	1197	58	1255	a	a
I-B	ILW	Hydrofracture	ORNL	3	1197	174	1371	a	a
II	ILW + HTGR	Hydrofracture	ORNL	1	1297	309	1606	I-A	351
III	ILW + HLW	Hydrofracture	ORNL	3	1297	930	2227	I-13	856
IV-A	ILW	Pot Calcination	FPPP	1	2442	1181	3623	a	a
IV-B	ILW	Pot Calcination	FPPP	3	2442	3543	5985	a	a
V	ILW + HTGR	Pot Calcination	FPPP	1	3210	1243	4453	IV-A	880
VI	ILW + HLW	Pot Calcination	FPPP	3	3210	3729	6939	IV-B	954
VII-A	ILW	Fluid Bed Calcination	FPPP	1	2476	1124	3600	a	a
VII-B	ILW	Fluid Bed Calcination	FPPP	3	2476	3372	5848	a	a
VIII	ILW + HTGR	Fluid Bed Calcination	FPPP	1	3187	1186	4373	VII-A	776
IX	ILW + HLW	Fluid Bed Calcination	FPPP	3	3187	3558	6745	VII-B	887
X-A	ILW	Pot Calcination	New Building	1	4822	1181	6003	a	a
X-B	ILW	Pot Calcination	New Building	3	4822	3543	8365	a	a
XI	ILW + HTGR	Pot Calcination	New Building	1	5590	1243	6833	X-A	830
XII	ILW + HLW	Pot Calcination	New Building	3	5590	3729	9319	X-B	954
XIII	HLW	Pot Calcination	FPPP	1	2119	798	2917	a	a
XIV	HTGR	Pot Calcination	FPPP	1	1995	642	2637	a	a

°Not applicable.

Additional details on the assumptions used for this study are given in ref. 9. Note that the cost of disposal of the large amounts of nitric acid and process waste that are generated during solidification are not included in this cost estimate (Sect. 6.5.2.2).

6.5.2.2 Disposal of Nitrates Produced by Calcination of Wastes. — The calcination of ILW and HLW will produce large amounts of nitric acid, waste as a by-product, i.e., ~ 45 tons of nitrate per year (Sect. 6.4.7 and Table 17). This amount of nitrate would raise the concentration in White Oak Creek by about 4.5 ppm. The discharge of large amounts of nitrate to the Creek will probably not be acceptable to the EPA or the State of Tennessee (Sect. 10), although the Public Health Service specification for drinking water is 44 ppm nitrate. Consequently, we recommend that hydrofracturing be retained for the disposal of nitrate and other chemical wastes, regardless of the choice of method for disposal of ORNL contaminated liquid wastes. This cost of disposal of the nitrate waste was not included in the cost estimate for calcination (Sect. 6.5.2.1).

6.5.2.3 Drying and Hydrofracture. — This procedure consists of drying the sludge and hydrofracturing the supernate. All high-level wastes, after appropriate cooling in the stainless steel tanks, would be mixed with ILW and neutralized and the solids allowed to settle in one of the existing tanks, probably tank W-5 initially. On a scheduled basis, the combined new and old sludge would be removed from the tank by hydraulic mining and the solids separated and washed in a continuous centrifuge. The solids would be dried, packaged, and shipped to the salt mine. Packaging could consist of dispersion of the solids in cement, if this proved desirable. The supernate would be hydrofractured as ILW. Advantages of this compromise system are:

- (1) The  $^{239}\text{Pu}$ ,  $^{90}\text{Sr}$ , and  $^{244}\text{Cm}$  and other transplutonium isotopes and most of the fission products would remain in the sludge and be shipped off-site. This observation is confirmed by the analyses of tanks W-5, W-6, and W-8 (Tables 4, 5, and 6) where the supernates contain less than 1% of the  $^{244}\text{Cm}$ ,  $^{90}\text{Sr}$ , gross  $\beta$  and  $^{239}\text{Pu}$ .

Plutonium is not detectable by pulse height analysis. (Some of these materials could be suspended solids rather than soluble compounds.) Samples of the sludge were separated by centrifugation and washed with about three equal volumes of water to simulate the proposed method. Less than 0.01% of the  $^{239}\text{Pu}$ ,  $^{244}\text{Cm}$ , and  $^{90}\text{Sr}$  were lost to the washes. Thus, we conclude that greater than 99% of the alpha radionuclides and  $^{90}\text{Sr}$  and a major fraction of the materials represented by the gross beta analysis can be isolated in the sludge.

- (2) The degree of separation of the sludge from the supernate is not critical within reasonable limits.
- (3) The supernate and washes would contain cesium and very small amounts of  $^{90}\text{Sr}$  and transplutonium isotopes and would be suitable for hydrofracturing. If necessary, the  $^{137}\text{Cs}$  could be removed by sorption on zeolite, as demonstrated at Hanford, and the zeolite shipped to the repository. The supernate would also contain all of the caustic, sodium nitrate, and other inactive soluble salt. The solid precipitate would contain only hydrous oxides and carbonates. Thus, the amount of solids to be shipped is greatly reduced.
- (4) The dried solids would be thermally stable and are expected to be acceptable for storage in the salt mine. The oxides would be radiolytically stable but the carbonates may release some CO or  $\text{CO}_2$ . Further experiments are required to determine the acceptability of these materials for storage in the salt mine.
- (5) The cost of a drying and hydrofracture facility should be much lower than a calcination facility. However, a conceptual cost estimate has not been made.

Factors influencing the cost are:

- (a) The amount of solids to be processed and shipped is much less since the inert salts are not present (they are hydrofractured). Thus, the size of the operating system and the number of shipping containers are greatly reduced.
- (b) The solids need only to be dried and not calcined. A low temperature operation should have lower costs.
- (c) The nitrates will be decomposed during calcining and an appropriate recovery system will be required so that the nitrates can be hydrofractured or discharged to the river (the latter may not be acceptable) (Sect. 6.5.2.2).
- (d) The off-gas system will be much simpler for the drying system since the nitrates are not present.

This system looks promising and we recommend further development and cost estimates, if hydrofracturing is not selected as the sole disposal method for ORNL.

#### 6.6 References

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9. J. M. Holmes, Preliminary Cost Estimate of High-Level Waste Disposal from Proposed Pilot Plants, ORNL CF 71-4-41 (April 28, 1971).