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UC-70

**Nuclear Waste Management
and Transportation
Quarterly Progress Report
April Through June 1976**

**Compiled by
A. M. Platt**

October 1976

**Prepared for the Energy Research
and Development Administration
under Contract E(45-1)-1830**

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Waste Management

NUCLEAR WASTE MANAGEMENT AND
TRANSPORTATION QUARTERLY PROGRESS
REPORT APRIL THROUGH JUNE 1976

Compiled by
A. M. Platt
Nuclear Waste Technology Program

October 1976

BATTELLE
Pacific Northwest Laboratories
Richland, Washington 99352

FOREWORD

This document is one of a series of technical progress reports designed to report on commercial radioactive waste management programs at Battelle, Pacific Northwest Laboratories. These programs are funded by the Nuclear Fuel Cycle and Production and the Environmental Control Technology Divisions of ERDA. The Commercial Radioactive Waste Fixation Program is reported elsewhere. These reports are issued on a quarterly basis.

The prior report in this series is Nuclear Waste Management and Transportation Quarterly Progress Report January Through March 1976, BNWL-2029.

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1.0 ALTERNATIVE WASTE MANAGEMENT SYSTEMS

J. W. Bartlett - Project Manager

The objectives of this program are to identify and evaluate alternative systems for treatment, interim storage, and final disposition of commercial radioactive wastes.

SUMMARY

The development of remote sensing techniques for regional reconnaissance continued. These techniques have possible application to the selection of waste repository sites.

A draft of an Energy Knowledge Test and a literature review of past attempts at communication of energy information were completed. A draft was completed of a topical report assessing public values as they pertain to complex technological decisions.

Studies of the reconcentration phenomenon and the fuel cycle wastes in perspective continued. The drafts of two papers describing these studies were completed.

ADVANCED GEOLOGIC MONITORING - REMOTE SENSING - H. P. Foote

The Land Use Program was improved and expanded to two versions. The original version could classify a 1024 x 1024 pixel region while the second version works on a full LANDSAT frame 3120 x 3120 pixels. The program requires about 2 hr to classify a full frame into 15 categories. The classi-

fication portion of the Land Use Program was rewritten in assembly language for greater speed. This improvement reduced the classification time to about 60% of its original value.

Gould electrostatic output was added to the training sample identification portion of the Land Use Program. This option allows printing out a strip 259 pixels wide of arbitrary length with each row and column identified by its pixel coordinates. Thus a training site can be specified exactly in the input phase of the program. Later this strip will be used with the digitizer table for choosing the training samples. An additional advantage is the hard copy record of each training area.

A second phase was added to the main Land Use Program. After the initial classification scan, there are frequently many one- or two-pixel "holes," or unidentified pixels. The new program scans the picture and fills in the holes by assigning to the unclassified pixel the class that predominates in its immediate neighborhood. This procedure makes a

classification map more informative and considerably improves its appearance.

A lineprinter plotting routine was developed for the color enhancement program to produce intensity histograms of the selected picture. The histogram permits selection of a contrast table more easily and accurately than in the original version. Assembly language subroutines and double buffered output were also added to this program to reduce the picture making time. It now takes about 5 min per color for a picture regardless of resolution. This time is fixed by the DICOMED color image recorder itself and not the program.

PUBLIC VALUES AND ATTITUDES REGARDING NUCLEAR WASTE DISPOSAL AND MANAGEMENT - John Hébert (HARC)

During the past quarter, activities continued in the study of communication of energy information to the public and public perceptions of risk associated with nuclear waste management.

In the communication studies, two major tasks were accomplished during the quarter. First a draft of an Energy Knowledge Test was completed. This instrument in its present form consists of over 80 items. These measure the public's knowledge about alternative energy sources, health, safety, and environmental impacts of nuclear energy, the management of nuclear wastes, and safeguards. Limited pretests of the instrument revealed the need for revisions to reduce the level of difficulty of the questions.

Secondly, a review of literature on energy information communication has produced a summary matrix of information sources and communication targets, as a way of integrating past and current attempts at communication with the public.

The major activity for this quarter in the risk perception studies has been the review and integration of research approaches to the study of public perceptions of technological risk. The emphasis of this effort has been to categorize the approaches taken to study the problem of risk perception and to identify those quantitative and qualitative variables influencing risk preference in nuclear waste management. A draft of a topical report, Assessment of Citizen Values for Complex Technological Decisions, was completed and is being circulated for comment prior to submission for publication in the open literature.

THE RECONCENTRATION PHENOMENON OF RADIONUCLIDE CHAIN MIGRATION -

M. O. Cloninger and H. C. Burkholder

The previous quarterly report described a radionuclide migration phenomenon whose occurrence had apparently not been postulated before. This situation, called the reconcentration phenomenon, can occur following a hypothetical leach incident at an underground nuclear waste disposal site for certain radionuclide decay products in particular disposal environments. As a result of the reconcentration effect, the predicted maximum decay product discharge rates

to the biosphere for these particular nuclides can increase above their corresponding values for surface storage. Thus geologic isolation can increase the largest dose that any individual is projected to receive throughout geologic time above doses from surface storage even though it has significantly delayed the discharge of those nuclides to the biosphere. Conversely, doses averaged over all geologic time are decreased from surface storage values by geologic isolation.

Work during the last quarter explored the effects of geosphere path length and leach time on the reconcentration phenomenon for the second and third members of the three number chain, $^{234}\text{U} \rightarrow ^{239}\text{Th} \rightarrow ^{226}\text{Ra}$. Because the sorption equilibrium constants (K values) for the various chain members can vary by orders of magnitude for various geologic media, the effect of varying K_{Ra} was also investigated. The methods and data used in these investigations were described in the previous quarterly.

Work during the present quarter explored the effects of varying other important parameters. Figure 1 shows the effect of varying K_{Th} on the reconcentration phenomenon. The leach time is 10,000 years, and K_{Ra} is 500. As K_{Th} is increased from 50,000 to 500,000, the reconcentration becomes significant at very short path lengths (0.001 to 0.03 mile), but the effect is unchanged at path lengths greater than 0.03 mile. As K_{Th} is decreased from 50,000 to 500, the reconcentration effect disappears. The adsorption equilibrium constants for ^{230}Th

FIGURE 1. The Equilibrium Constant of the First Member of a Two-Member Chain on the Reconcentration Phenomenon

and ^{226}Ra are both equal to 500 for this situation. Thus Figure 1 shows that the effect disappears when both chain members migrate at the same velocity. As K_{Th} is further decreased from 500 to 50, the reconcentration phenomenon does not reappear, showing that the phenomenon occurs only when K_{Th} is greater than K_{Ra} (i.e., when the second chain member migrates faster than its precursor). The inflection in the curve for $K_{\text{Th}} = 50$ occurs as the controlling source of ^{226}Ra switches from the ^{226}Ra to the ^{230}Th initial inventory. For path

lengths up to about 0.3 mile the maximum ^{226}Ra discharge rate is controlled by the ^{226}Ra initial inventory; for path lengths greater than about 3 miles the discharge rate is controlled by the ^{226}Ra which has decayed from the ^{230}Th initial inventory. Both inventories are important contributors to the maximum ^{226}Ra discharge rate between 0.3 and 3 miles.

Using the $^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ chain as an example, we can now elaborate on the physical interpretation described in the previous quarterly. As ^{230}Th moves through the geologic medium, it continually decays to ^{226}Ra , which moves increasingly ahead of its ^{230}Th parent because of its lower K value and hence faster migration. Thus ^{226}Ra created by decay of ^{230}Th released at a given time during the leaching period overtakes ^{230}Th which was leached from the disposal site earlier. Because this ^{230}Th continually creates additional ^{226}Ra from its own decay, the local ^{226}Ra concentration increases above the initial equilibrium value. As the original plus the additional ^{226}Ra moves further ahead of its ^{230}Th parent it encounters more newly created ^{226}Ra , and the local ^{226}Ra concentration becomes even larger. If the path length is long enough for significant reconcentration to occur but short enough that the reconcentrated ^{226}Ra does not decay before discharging to the biosphere, the phenomenon is observed. Since the reconcentration of the ^{226}Ra decay product depends on a diluted ^{230}Th parent spread out over a significant portion of the path length, the

effect disappears as the leach time approaches zero (impulse release).

The reconcentration effect can be further illustrated using Figure 2, which shows the moving band of ^{230}Th in the geologic medium at various times. We assume a coordinate system which moves with the migration velocity of ^{230}Th . At time "t" (Frame 1) some ^{226}Ra is created by decay of the ^{230}Th , as shown by the vertical arrow. At time "t + Δt " (Frame 2) this ^{226}Ra has moved through the migrating ^{230}Th a short distance because of its smaller K value and hence higher migration velocity relative to ^{230}Th . The original ^{226}Ra now merges with additional ^{226}Ra which is created at this location by ^{230}Th decay. From this time on, both amounts of ^{226}Ra migrate together. At time "t + 2 Δt " (Frame 3), the combined ^{226}Ra inventory meets additional newly created

FIGURE 2. Physical Interpretation of the Reconcentration Phenomenon

^{226}Ra and the further concentrated ^{226}Ra moves on through the ^{230}Th band. This process is repeated for each differential period of time until the ^{226}Ra either moves ahead of the ^{230}Th band or discharges to the biosphere.

Understanding the effect of the nuclide half-lives on reconcentration is important in predicting how the phenomenon would occur for other decay chains. Figure 3 shows the effect of varying the half-life of the second member, H_2 . The bold curve for $H_2 = 1,600$ years describes the $^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ situation. As H_2 decreases from 1,600 to 1.6 years, the reconcentra-

tion effect greatly increases, and the path length at which the maximum effect is predicted decreases slightly. When $H_2 = 1.6$ years, the maximum discharge to the biosphere increases a factor of about 100. As H_2 increases from 1,600 to 16,000 years, the reconcentration effect decreases. The effect presumably disappears for large H_2 (slow decaying member). We note that the path length necessary to overcome the effect is almost constant for half-lives in the range of 1.6 to 1,600 years. The nuclide inventories in curies are held constant while H_2 varies. Hence the g-atoms of the second chain member in the initial inventory decreases linearly as the H_2 decreases.

FIGURE 3. The Effect of the Half-Life of the Second Member of a Two-Member Chain on the Reconcentration Phenomenon

Figure 4 shows the effect of varying H_1 on the reconcentration phenomenon. The bold curve for $H_1 = 77,000$ years describes the $^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ situation. As the H_1 decreases from 77,000 to 7,700 years, both the magnitude and range of the reconcentration effect decrease. The effect presumably disappears for small values of H_1 (fast decaying parent). This makes good physical sense because the very fast decay of the parent would not allow time for the reconcentration to occur. As H_2 increases from 77,000 to 770,000 years, the magnitude of the effect remains unchanged but the range increases significantly.

We now consider the three-member chain $^{234}\text{U} \rightarrow ^{230}\text{Th} \rightarrow ^{226}\text{Ra}$. Figure 5 shows the effect of varying the adsorption equilibrium constants of the third member on the reconcentration phenomenon. In Figure 5 the magnitude and range of the effect increase as

FIGURE 4. The Effect of the Half-Life of the First Member of a Two-Member Chain on the Reconcentration Phenomenon

K_{Ra} decreases. However, both characteristics are less strongly affected by the K values for the three-member chain $^{234}\text{U} \rightarrow ^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ than for the two-member chain $^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ shown in Figure 1. Comparison of the $K_{Th} = 50,000$ curves in Figures 1 and 5 shows that the presence of ^{234}U does not change the magnitude of the reconcentration phenomenon but does increase the range by about half an order of magnitude. The latter result occurs because $K_U = 14,000$ and $K_{Th} = 50,000$ so that the ^{234}U migrates faster than its ^{230}Th decay product. Hence the ^{226}Ra which comes from the

FIGURE 5. The Effect of the Equilibrium Constant of the Second Member of a Three-Member Chain on the Reconcentration Phenomenon

^{234}U initial inventory will discharge at longer path lengths without decaying significantly.

Figure 6 compares the contributions from each of the initial inventories in the $^{234}\text{U} \rightarrow ^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ chain to the maximum ^{226}Ra discharge. For the hypothetical situation displayed, each decay product moves faster than its parent ($K_U = 14,000$, $K_{Th} = 500$, $K_{Ra} = 5$). The results show that the ^{234}U initial inventory causes little change in the magnitude of the reconcentration phenomenon even though the situation at first seems ideal for an enhanced effect. The reconcentration

FIGURE 6. Comparison of the Contributions from Each of the Initial Inventories to the Maximum Discharge Rate

effect requires a wide band of radionuclide parent in the migration path. But after reconcentration has occurred between the first two members of a decay chain, a wide distribution of the second member (^{230}Th in this case) does not exist. Thus the potential for further reconcentration between the second and third members is partly eliminated.

A paper entitled "The Reconcentration Phenomenon of Radionuclide Chain Migration" will be presented at the 69th Annual Meeting of the American Institute of Chemical Engineers in Chicago, Illinois, November 28 - December 2, 1976.

A second paper entitled "A Management Perspective on Nuclear Fuel Cycle Wastes" will be presented at the Winter American Nuclear Society meeting, Washington, DC, November 14-19, 1976.

2.0 DECONTAMINATION AND DENSIFICATION OF CHOP-LEACH CLADDING RESIDUES

R. L. Dillon and R. S. Kemper

Fuel element cladding residues from the chop-leach head end process constitute a low-density, high-surface area metallic waste with a substantial actinide element contamination. The objective of this study is to determine the feasibility of and to develop methods for removal of long half-life transuranics from the cladding residues and for consolidation of the zirconium and stainless steel residues. The study involves:

1. *Selection and laboratory testing of methods for transuranic removal by*
 - a) *aqueous reagents*
 - b) *molten salt reagents*
 - c) *fluxing during melting.*
2. *Consolidation of the fuel element hulls by melting and casting Zircaloy, stainless steel, Inconel or alloys of the three.*
3. *Testing and evaluating the consolidated fuel hulls for long-term waste management.*
4. *Absorption and fixation of tritium into the waste ingots.*
5. *Consolidation of transuranics removed.*

SUMMARY

The mini-scale chemical decontamination process is testing unirradiated cladding hulls to optimize some of the treatment parameters and equipment design. The flow rates and gas composition necessary to achieve desired oxide removal have been determined.

Experiments aimed at improving the melting rate of the Zircaloy-15 wt% stainless steel alloys were performed. Increasing the crucible diameter from 4 to 5 in. and increasing the CaF_2 to MgF_2 slag ratio showed no significant melting rate improvement.

Graphite crucible melting of the Zircaloy-10 wt% 304 stainless steel-5 wt% Inconel 718 alloy using oxidized melt stock showed that melting required temperatures above the melting points of the stainless steel and Inconel.

DECONTAMINATION PROCESS

The parameters for surface decontamination of oxide-covered Zircaloy hulls, employing gaseous HF and aqueous stripping, are being systematically investigated. Treatment with undiluted HF or a 25% Ar - 75% HF

mixture at 600°C for 60 min followed by treatment with an aqueous decontaminant has proven effective during nonradioactive evaluation. Table 1 shows the effect of some parameters in oxide removal.

After a temperature excursion of about 200°C in 2 min was noted during the gas-phase treatment, the role of hydrogen as produced by the reaction of HF with zirconium is being evaluated. The excursion was easily controlled by flooding the reaction vessel with argon. Metallographic evaluation of the samples indicated massive zirconium hydride formation due to a hydrogen buildup in the reaction vessel. The usefulness of hydride formation in the removal of oxide from Zircaloy cladding was sub-

sequently tested. These tests showed the excess hydrogen to be somewhat detrimental, apparently reducing the protective nature of the film formed on the Zircaloy. A change in reaction vessel design so that HF flows in the bottom and out the top of the vessel has been shown to be effective in displacing the hydrogen.

The removal of about 2.6 g/dm² from the oxidized sample after aqueous treatment appears to be the minimum requirement for complete descaling of the Zircaloy. Achievement of near 100% oxide removal in the reaction vessel currently in use requires an excess HF flow of about 27 cc/min/dm² Zircaloy if the HF flow is from the bottom of the reaction vessel to the top.

TABLE 1. Effect of Gas Composition on Oxide Removal

Run No.	Inlet Gas, cc/min/dm ²	Effluent Gas, cc/min/dm ²	Exposure Time, min	Weight Loss in Oxidized Samples After Aqueous Strip, g/dm ²	Estimated Percent Oxide Removal	Weight Loss in Unoxidized Samples After Aqueous Strip, g/dm ²
8(a)	HF 80	HF 12	60	2.56	99+	2.86
		H ₂ ---				
15	HF 79	HF 27	60	2.62	98+	2.98
	Ar 25	Ar 25				
		H ₂ 27				
16	HF 81	HF 27	60	2.54	90	3.11
	H ₂ 81	H ₂ 69				
17	HF 41	HF 11	60	1.93	50	3.16
	Ar 26	Ar 26				
		H ₂ 10				

a. In contrast to the other runs, gas flow was from top to bottom on this run.

DENSIFICATION PROCESS

A series of melts was made in a 5-in.-diam Inductoslag crucible to determine the effect of crucible diameter on melting rate for the Zircaloy-15 wt% 304 stainless steel alloy. Details of slag composition and observed melting rates for these 5-in. runs are shown in Table 2. The following comments summarize the observations and conclusions for these heats.

1. The larger ingot size at near constant power did not yield a higher melting rate. In fact, melting rates were lower than for the 4-in. ingots.
2. Variations in slag composition yielded no significant changes in melting rates. In no cases was the ingot completely coated with slag; contact between the crucible wall and the ingot was evident.

This metal-to-metal contact causes high rates of heat loss from the metal pool and may have contributed substantially to the low melting rates. It may be that adding MgF_2 (to the CaF_2) causes changes in physical properties of the slag (such as viscosity) and that these changes are more important than changes in the melting point of the slag.

3. Bridging of the molten pool with unmelted Zircaloy was observed to occur when unmelted pieces of metal already in the pool floated to the top and solidified. This probably happens in the 4-in. runs also but was more obvious in the 5-in. crucible.
4. With the 4-in. crucible, changes in pool temperature with changes in composition of the molten pool were noted. With the 5-in. crucible, pool temperatures remained

TABLE 2. Zircaloy-Stainless Steel Alloy Melt Rate Experiments by Inductoslag Process

Melt No.	Ingot Diam	Slag Composition, wt%	Slag Melting Point, °C	Metal Composition, wt%	Weight Metal Melted, lb	Power Input, kW	Time, min(a)	Melt Rate, lb/hr	Comments
538-1	4	67 CaF_2 /33 MgF_2	1100	Zr-15Fe + Cr + Ni(b)	11	85	21	31	Fuel hulls + 304 SS to make 15% Fe alloy
538-2	4	73 CaF_2 /27 MgF_2	1150	Zr-15Fe + Cr + Ni(b)	11	85	39	17	Fuel hulls + 304 SS to make 15% Fe alloy
538-3	4	62 CaF_2 /33 MgF_2	1100	Zr-15 SS	11	80	60	11	Fuel hulls + 15% 304 SS
538-4	4	67 CaF_2 /33 MgF_2	1100	Zr-15 SS	--	85	--	--	Run terminated, pool dumped into bottom of furnace
548-1	5	46 CaF_2 /48 MgF_2	950	Zr-15 SS	--	95	--	--	Bridged
548-2	5	52 CaF_2 /48 MgF_2	950	Zr-15 SS	11.5	90	71	9.7	Chopped Zr sheet + 15% 304 SS
548-3	5	67 CaF_2 /33 MgF_2	1100	Zr-15 SS	12.7	95	100	7.6	Chopped Zr sheet + 15% 304 SS

a. Time measured from start of feeding scrap to an established molten pool until last scrap added was completely melted.

b. Sufficient 304 SS added to make a 15% Fe alloy.

uniformly low and melting rates were uniformly slow.

In the preparation of a series of 4-in.-diam ingots of the Zircaloy-15% stainless steel, variations in slag composition (Run 538) did not improve melting rates. Part 1 of this run melted at a rate of 31 lb/hr, but this high rate is believed to result from the high zirconium content of the pool. This was the result of the influence of a zirconium starting stub. Other parts of this run made with the same slag composition melted at a much lower rate.

Two 5-kg heats of 85 wt% Zircaloy-4 10 wt% 304 stainless steel and 5 wt% Inconel 718 were made using the graphite crucible process. Oxidized sheet material was used as melt stock to determine the effect of the oxidized surfaces on the temperature required for melting. Temperatures in the range of 1550 to 1600°C were required to melt the material. This is above the melting point of the stainless steel and Inconel. Previous work^(1,2) has shown that clean melt stock of this composition will melt in the 1300 to 1320°C range.

The first ingot was allowed to cool in the crucible, resulting in extensive crucible reaction. The second ingot (Figure 7) was poured into a graphite mold. While crucible reac-

FIGURE 7. Graphite Crucible Melted Zircaloy-10 wt% 304 Stainless Steel 5 wt% Inconel 718 Using Oxidized Melt Stock: 8.9 cm diam

tion is apparent, both the crucible and mold were reusable. The ingot cracked. Ingots of this composition melted with clean material and under similar vacuum conditions did not crack.

Fabrication of the PNL Inductoslag crucible has been completed and is shown in Figure 8.

FIGURE 8. Inductoslag Melting Crucible

REFERENCES

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3.0 DISPOSITION (D&D) OF RETIRED CONTAMINATED FACILITIES AT HANFORD

K. M. Harmon - Project Manager

*The program has the following objectives for FY-1975-1976:
to establish methods, costs and priorities for the disposi-
tion of retired (excessed) Hanford facilities; and to identify
and perform required facility disposition R&D.*

SUMMARY

Analyses of surface rust and deposits in the reactor retention basin pipelines indicate alpha contamination ranges of 20 to 400 pCi/g. The lines will have to be cleaned before they can be released.

Decontamination and Decommissioning (D&D) scenarios for Hanford have been redefined, and certain decommissioning modes which fit these scenarios are suggested. Preliminary cost estimates and relative D&D priorities for selected facilities are presented.

Radiological criteria governing the release and use of 107-DR Retention Basin and 216-S-1, 2 crib sites are given.

A prototypic water cannon, for use in spalling contaminated concrete surfaces, has been assembled by Exotech Incorporated, tested, and shipped to PNL for further evaluation.

TASK 1 - ESTABLISH METHODS, COSTS AND PRIORITIES FOR D&D AT HANFORD -

K. M. Harmon

Site Characterization - A. P. Larrick
and W. K. Kratzer (UNI)

United Nuclear Industries has begun a short-term study of methods to remove radioactive sludge and corrosion products from inside reactor retention basin system pipelines.

The first step has been to sample and analyze the deposits in the effluent lines at the B, C, DR, F, H and KE reactors. Analyses of the surface rusts and deposits show that alpha contamination ranges from 20 to 400 pCi/g, indicating that all lines will require cleaning before release. The deposits in all effluent lines, except for the KE Reactor effluent line deposits, were very loose. This suggests that a high-pressure water jet may be the most desirable means of

cleaning. Soak tests of the deposits are in progress to determine the leaching qualities and to ascertain whether or not special handling of the water will be required.

The KE Reactor effluent line deposits are hard and tenacious, perhaps requiring mechanical or chemical methods of decontamination.

The gamma-emitting radionuclides found in the deposits are predominantly ^{152}Eu , ^{154}Eu , ^{155}Eu and ^{60}Co . Cesium-137 was found occasionally, and shorter half-lived ^{65}Zn and ^{125}Sb were found in the KE Reactor effluent line deposits, the most recent reactor to be shut down. Gamma radionuclide concentrations ranged from $5 \times 10^{-4} \mu\text{Ci/g}$ to $2 \mu\text{Ci/g}$.

D&D Scenarios - K. M. Harmon

The alternative scenarios for decommissioning retired ERDA facilities at Hanford, as identified in the quarterly report for July-Sept 1975 (BNWL-1952), have been redefined. The following four cases are under current evaluation.

Case I. Prepare all retired sites so that unrestricted use can be permitted within 100 years:

- Decontaminate and/or dismantle contaminated structures.
- Remove contaminated soil and buried materials which will still exceed the release limits for unrestricted use at the end of the 100-year period.
- Correct any potential safety hazards; wherever final D&D of a facility will be delayed long enough to increase the potential

for release of contamination significantly, prepare the site or facility for safe storage during the interim period.

Case II. Prepare all retired sites so that unrestricted use can be permitted within 500 years. Complete all D&D project activities within 100 years:

- Decontaminate and/or dismantle all structures.
- Transfer all TRU-contaminated soil and buried materials to authorized storage or disposal sites.
- Prepare contaminated, TRU-free ground sites for safe storage (protective storage or entombment modes) until they can be released.
- Correct any potential safety hazards; wherever final D&D of a facility will be delayed long enough to increase the potential for release of contamination significantly, prepare the site for safe storage during the interim period.

Case III. Prepare selected sites for long-term safe storage of their radionuclide inventory; prepare all other sites so that unrestricted use can be permitted within 500 years. Complete all D&D project activities within 100 years:

- Entomb selected facilities and sites (e.g., reactor blocks, fuels reprocessing plant process cells).
- Decontaminate and/or dismantle other structures.
- Transfer all TRU-contaminated soil and buried materials from the 100 Areas and other "shoreline" sites to authorized storage or disposal sites.

- Prepare other contaminated ground sites for safe long-term storage.
- Correct any potential safety hazards; wherever final D&D of a facility will be delayed long enough to increase the potential for release of contamination significantly, prepare the site for safe storage during the interim period.

Case IV. Provide indefinite-term maintenance, surveillance, and access control:

- Upgrade, seal and isolate structures as necessary.
- Control human, animal and plant access to contaminated soils and burial grounds.
- Clean up or cover surface contamination.
- Provide surveillance.

Cases I to III involve the following assumptions:

- Layaway action will have to be taken with many sites, to provide safe storage until their turn comes for longer-term action.
- D&D action will be taken in the sequence established in the master plan for D&D at Hanford.
- Unrestricted use of a site requires that residual contamination not exceed the unrestricted release criteria for that site and that potential safety hazards be corrected.
- Contaminated D&D wastes will be transferred to burial or storage sites in the 200 Areas.

Prioritization - J. W. Litchfield, J. C. King, and C. L. McDonald

A preliminary priority listing, which indicates the relative urgency

for undertaking some type of D&D action with each facility, has been prepared for 366 inactive or standby contaminated ERDA facilities at Hanford. A facility's position on the list depends on a partially subjective evaluation of the ranking of that facility, in its present condition, by four independent factors:

- potential for causing an offsite radiological hazard
- potential for becoming a hazard to persons onsite
- cost of maintenance and surveillance in its present condition
- compatibility of the facility, in its present status, with projected use of the environs.

The relative ranking of a number of facilities, as determined by their priority indices, is shown in Table 3.

Radiological Disposition Criteria - D. A. Waite

Radiological criteria governing the release and use of decommissioned nuclear facilities were updated for the 107-DR Reactor Retention Basin System in accord with recent changes in release fraction methodology, and criteria were calculated for the 216 S-1, 2 crib. These criteria are reported in Tables 4 and 5.

TASK 3 - EXPLORATORY R&D

"Water Cannon" Evaluation -
Saul Mandel and Robert G. Sullivan

Exotech Incorporated has fabricated and shipped a prototype water cannon for Hanford tests of its use in spalling contaminated concrete surfaces.

TABLE 3. Preliminary Relative Priorities for Decommissioning Selected Inactive or Standby Contaminated ERDA Facilities at Hanford

Facility No.	Facility Name	Priority Index
105-B, C, D, DR, F, H	Production Reactors	760-860
318-11	300 Wye Burial Ground	610
107-C, KE, KW	Reactor Retention Basins (steel tanks)	510-590
218-W-4A	Burial Ground (200 W)	520
118-K	100 Area Burial Ground	490
216-B-7A & 7B	Class II Crib (TRU & FP)	470
216-S-17	Redox Pond	450
107-B, D, DR, F, H	Reactor Retention Basin (concrete)	350-450
116-K-2	Class V Crib (100 Area)	440
216-Z-9	Class IA Crib (TRU)	400
116-H	Exhaust Stack	360
118-H-1	100 Area Burial Ground	350
115-D	Gas Recirculation Bldg.	340
117-C, F, H	Filter Buildings	300-310
116-B, C, D, K	Exhaust Stacks	270
107-D, DR	Effluent Pipelines	170
207-U	U Plant Retention Basin	150
116-DR-5	Outfall Structure	80
119-H	Air Sample Building	10

Factory tests prior to shipment gave the following observations:

- The "falling block action" loading mechanism worked well and apparently will lend itself to modification for automatic operation.
- In one test, the glycerine jet produced a spalled area of 5 in. x

4 in. x 0.9 in. deep at the apex. In a two-shot test, the spalled area measured about 7 in. diam x 1-1/2 in. deep.

- The shield affixed to the end of the muzzle collected and contained nearly all the spalled concrete, broken into pieces less than

TABLE 4. Disposition Criteria, 107 DR^(a)

		Surface Contamination	Activity/Mass Concentration
Release Mode	Unrestricted	$4\alpha^{(b)} + \beta\gamma^{(c)} = 10^3 \text{ dis/min/100 cm}^2 \text{ fixed}$ $4\alpha + \beta\gamma = 10^3 \text{ dis/min/m}^2 \text{ smearable}$	$4\alpha + \beta\gamma = 45 \text{ pCi/g}$
	Conditional Nonsurveillance	$4\alpha + \beta\gamma = 10^4 \text{ dis/min/100 cm}^2 \text{ fixed}$ $4\alpha + \beta\gamma = 10^3 \text{ dis/min/m}^2 \text{ smearable}$	$4\alpha + \beta\gamma = 450 \text{ pCi/g}$
	Conditional Surveillance	$4\alpha + \beta\gamma = 10^5 \text{ dis/min/100 cm}^2 \text{ fixed}$ $4\alpha + \beta\gamma = 10^3 \text{ dis/min/m}^2 \text{ smearable}$	$4\alpha + \beta\gamma = 4.5 \text{ nCi/g}$
	Restricted	$4\alpha + \beta\gamma > 10^5 \text{ dis/min/100 cm}^2 \text{ fixed}$ $4\alpha + \beta\gamma > 10 \text{ dis/min/m}^2 \text{ smearable}$	$4\alpha + \beta\gamma > 4.5 \text{ nCi/g}$

a. Radionuclide content of 3-in. sludge layer in 107 DR (as of 12/31/73)

2200 pCi/g ^{60}Co
 210 pCi/g ^{90}Sr
 1600 pCi/g ^{137}Cs
 12,000 pCi/g ^{152}Eu
 3200 pCi/g ^{154}Eu
 130 pCi/g ^{155}Eu
 1.3 pCi/g ^{238}Pu
 48 pCi/g $^{239}\text{Pu} + ^{240}\text{Pu}$

b. Alpha dis/min.

c. Beta-gamma dis/min.

3/4 in. in size. The shield with the vacuum mechanism attached did not contain the smoke from the firing.

- Some of the spalled concrete pieces stuck to the inside surfaces of the shield, probably because of the glycerine projectiles.

4.0 TRANSPORTATION SAFETY STUDIES

R. J. Hall - Project Manager

The objective of the Transportation Safety Studies Program is to develop and use a model for calculating the risks associated with the shipping of radioactive materials.

SUMMARY

The draft report of the plutonium shipment by rail risk assessment was issued for comments.

Work continued on the study of the loss of spent fuel casks and plutonium packages at sea. A paper has been prepared and accepted for presentation at the IAEA Conference in Vienna this summer.

The major portion of the accident sequence model for shipment of gasoline by truck has been completed. The first of the risk assessment report is being assembled and will soon be issued for comments.

Survey questionnaires, covering the condition of UF_6 packages during transport, have all been returned. Work has begun on tabulating the results.

The risk involved in shipment of plutonium oxide by air has been assessed. Sensitivity studies showed that the impact portion of the accident environment played the major role in the release of plutonium oxide. The first draft is being prepared for release.

An assessment of the risk in shipment of chlorine by rail has been

started. A fault tree has been developed and certain failure thresholds calculated.

RISK ASSESSMENT OF PLUTONIUM SHIPMENT BY RAIL

The draft report of the risk assessment has been issued for comment. Plans are being made to obtain a review similar to the one conducted for BNWL-1846. The draft report is entitled An Assessment of the Risk of Transporting Plutonium Oxide and Liquid Plutonium Nitrate by Train, BNWL-1996 (Draft, June 1976).

STUDY OF LOSS OF SPENT FUEL CASK AND PLUTONIUM PACKAGES AT SEA

Long-term (hundreds of years) release rates from spent fuel in seawater were estimated, and additional calculations to assess the impact from release of contaminated cooling water from the spent fuel cask were performed. A paper, scheduled to be presented at the IAEA Convention in Vienna, has been completed and accepted. Work on the risk assessment report was continued.

RISK ASSESSMENT OF GASOLINE SHIPMENT BY TRUCK

The major portion of the consequence model was completed. Models were developed for gasoline pool fires, gasoline vapor dispersion, gasoline explosions, urban population areas and the spread of fires in urban areas. The fault tree has undergone revision following internal review, and work was completed on the calculation of the tank failure thresholds. Detailed descriptions of the tank truck relief valves, manway covers and outlet valve assembly were obtained from a tank truck manufacturer and used in the calculation of the threshold values. The first draft report is presently being assembled.

RISK ASSESSMENT OF UF₆ SHIPMENT BY TRUCK AND RAIL

All participants in the survey of UF₆ packaging conditions completed and returned their questionnaire forms. A list of the facilities contacted is included with this report. Tabulation of the information began, and a copy of the results of the survey is also attached.

RISK ASSESSMENT OF PLUTONIUM OXIDE SHIPMENT BY AIR

Work has been completed on the first draft report of the risk assessment of plutonium shipment by air. In the analysis, accident sequence cut sets were generated using the computer code FAULT.⁽¹⁾ Release frac-

tions were assigned to the basic events, and the computer code RAFT⁽¹⁾ was used to calculate release fractions. Inspection of the release sequence risks showed that the puncture environment was contributing more to the overall risk than expected. This resulted from a poor model of the puncture environment in which some estimates proved to be too conservative. A more adequate puncture environment was obtained from Sandia, and when rerun, the results appeared more reasonable. The Transportation Risk Evaluation Code (TREC - formerly HEADA) was used to translate the release sequence probabilities and release fractions into risk spectra. Sensitivity analyses and a multiple container analysis were completed and included in the report, which is now being edited and typed.

RISK ASSESSMENT OF CHLORINE RAIL SHIPMENT

The basic chlorine release fault tree was developed and has been periodically modified in response to suggestions made during the internal review. Probabilities have been assigned to basic events, and release sequence evaluation has started. A copy of a detailed mechanical analysis of chlorine tank cars, done by Battelle-Columbus for the Association of American Railroads, was obtained to assist in refining tank car failure threshold calculations. A preliminary consequences model has been developed; however, this model may be modified later.

REFERENCE

1. T. H. Smith et al., A Risk-Based Fault Tree Analysis Method for Identification, Preliminary Evaluations, and Screening of Po-

tential Accidental Release Sequences in Nuclear Fuel Cycle Operations, BNWL-1959, Battelle, Pacific Northwest Laboratories, Richland, Washington, January 1976.

PARTICIPANTS IN THE SURVEY OF UF₆ PACKAGING CONDITIONS

Fuel Fabrication Plants:

Westinghouse Electric Corp.
Columbia, SC

Exxon Nuclear Co.
Richland, WA

Kerr-McGee Nuclear Corp.
Cimarron, OK

Nuclear Fuel Services
Erwin, TN

Combustion Engineering, Inc.
Hematite, MO

General Electric Co.
Wilmington, NC

Babcock and Wilcox
Apollo, PA

Enrichment Plants:

Union Carbide Corp.
Paducah, KY

Union Carbide Corp.
Oak Ridge, TN

Goodyear Atomic Corp.
Portsmouth, OH

SURVEY OF UF₆ PACKAGING CONDITIONS DURING TRANSPORT⁽¹⁾

	2 1/2-TON MODEL		10-TON MODEL	14-TON MODEL
	30A	30B	48X	48Y
A. PACKAGES RECEIVED				
1975	<u>476</u>	<u>526</u>	<u>1104</u>	<u>1967</u>
1974	<u>584</u>	<u>691</u>	<u>825</u>	<u>1447</u>
1973	<u>461</u>	<u>688</u>	<u>1032</u>	<u>1029</u>
1972	<u>324</u>	<u>750</u>	<u>1383</u>	<u>1205</u>
1971	<u>695</u>	<u>517</u>	<u>2644</u>	<u>663</u>
Total	2540	3172	6988	6311
B. GENERAL CONDITION OF SHIPMENT (1971-1975)				
1. DO YOU FORMALLY EXAMINE AND RECORD THE CONDITION OF RECEIVED CYLINDERS?	YES =8	1=No Ans.	NO =1	(Inspect but
2. IS THE H/U RATIO VERIFIED UPON RECEIPT?	YES =2	2=No Ans.	NO =6	do not record)
3. NUMBER OF TIMES H/U RATIO EXCEEDED SPECIFIED LIMITS.	<u>0</u>	<u>0</u>	<u>0</u>	<u>0</u>
4. NUMBER OF TIMES ENRICHMENT EXCEEDED SPECIFIED LIMITS.	<u>0</u>	<u>1</u>	<u>0</u>	<u>0</u>
5. NUMBER OF CYLINDERS RECEIVED OVERFILLED.	<u>0</u>	<u>0</u>	<u>0</u>	<u>0</u>
6. NUMBER OF CYLINDERS RECEIVED OVERPRESSURIZED.	<u>1</u>	<u>0</u>	<u>20</u>	<u>56</u>
7. NUMBER RECEIVED WITH CONTAMINATION (UF ₆ , UO ₂ F ₂ , ETC.) OUTSIDE CYLINDER.	<u>0</u>	<u>0</u>	<u>0</u>	<u>0</u>
C. CYLINDER CONDITION (1971-1975)				
1. NUMBER OF CYLINDERS REMOVED FROM SERVICE DUE TO DETERIORATION.	<u>1</u>	<u>2</u>	<u>0</u>	<u>0</u>
2. NUMBER OF CYLINDERS WITH BADLY CORRODED SHELLS.	<u>0</u>	<u>0</u>	<u>0</u>	<u>0</u>
3. NUMBER OF CYLINDERS RECEIVED WITHOUT REQUIRED PROTECTIVE VALVE COVER.	<u>0</u>	<u>0</u>	<u>0</u>	<u>0</u>
4. NUMBER OF CYLINDERS RECEIVED WITH WELDS DAMAGED (EXAMPLE, VESSEL FRESHLY DENTED IN AREA OF WELD RESULTING IN POSSIBLE PARTIAL WELD FAILURE, ETC.)	<u>7</u>	<u>0</u>	<u>1</u>	<u>0</u>
D. VALVE CONDITION (1971-1975)				
1. NUMBER OF CYLINDERS REQUIRING VALVE REPLACEMENT.	<u>13</u>	<u>28</u>	<u>60</u>	<u>86</u>
2. NUMBER OF CYLINDERS WITH VALVE AND/OR VESSEL THREADS PARTIALLY STRIPPED.	<u>0</u>	<u>5</u>	<u>0</u>	<u>0</u>
3. NUMBER OF CYLINDERS WITH VALVE CROSSTHREADED.	<u>0</u>	<u>7</u>	<u>0</u>	<u>0</u>
4. NUMBER OF CYLINDERS WITH VALVE NOT COMPLETELY CLOSED.	<u>0</u>	<u>0</u>	<u>0</u>	<u>0</u>
5. NUMBER OF CYLINDERS WITH VALVE LOOSE OR INSUFFICIENT THREADS ENGAGED.	<u>0</u>	<u>0</u>	<u>7</u>	<u>19</u>
6. NUMBER OF CYLINDERS RECEIVED WITH VALVE DAMAGED (EXAMPLE, VALVE BENT AND FRESH METAL EXPOSED AT COUPLING, ETC.)	<u>0</u>	<u>14</u>	<u>1</u>	<u>5</u>

⁽¹⁾ IF ACCURATE NUMBERS ARE NOT AVAILABLE, APPROXIMATE VALUES OR ESTIMATES BASED ON BEST RECOLLECTIONS CAN BE USED AND ARE REQUESTED.

IF YOU HAVE ANY QUESTIONS ON COMPLETING THIS FORM, PLEASE CONTACT

JIM JOHNSON, BATTELLE NORTHWEST, RICHLAND, WASH 99352 (509)946 2073 FTS 444-7411
(EXT 946 2073)

	2 1/2-TON MODEL	10-TON MODEL	14-TON MODEL	
	30A	30B	48X	48Y
E. PLUG CONDITION (1971-1975)				
1. NUMBER OF CYLINDERS REQUIRING PLUG REPLACEMENT.	0	0	0	0
2. NUMBER OF CYLINDERS WITH PLUG LOOSE OR WITH INSUFFICIENT THREADS ENGAGED	0	0	0	0
3. NUMBER OF VESSELS WITH PLUG AND/OR VESSEL THREADS PARTIALLY STRIPPED.	0	0	0	0
4. NUMBER OF VESSELS WITH PLUG CROSSTHREADED	0	0	0	0
5. NUMBER OF CYLINDERS RECEIVED WITH PLUG DAMAGED FROM ROUGH TREATMENT DURING HANDLING OR SHIPPING (EXAMPLE, PLUG BENT AND FRESH METAL EXPOSED AT COUPLING, ETC)	0	0	0	0
F. OVERPACK CONDITIONS (1971-1975)				
1. NUMBER OF CYLINDERS RECEIVED IN OVERPACK.	2480	3164	878	0
2. NUMBER OF CYLINDERS RECEIVED WITHOUT OVERPACK WHEN REQUIRED.	0	0	0	0
3. NUMBER OF CYLINDERS RECEIVED WITH GROSS DAMAGE TO OVERPACK.	0	0	0	0
4. NUMBER OF CYLINDERS RECEIVED WITHOUT SEALING GASKET IN OVERPACK.	0	0	0	0
5. NUMBER RECEIVED WITH DETERIORATED GASKET.	3	1	0	0
6. NUMBER RECEIVED WITH CLOSURE BOLTS LOOSE.	~40	~40	0	0
7. NUMBER WITH MISSING CLOSURE BOLTS:				
a. ALL CLOSURE BOLTS MISSING	0	0	0	0
b. SOME CLOSURE BOLTS MISSING	~20	~16	0	0
8. NUMBER RECEIVED WITH DAMAGED OVERPACK MATING SURFACE	7	5	0	0
G. TRANSPORTATION MODE				
APPROXIMATE PERCENTAGE OF CYLINDERS RECEIVED BY:				
1. RAIL	0	3	12	14
2. TRUCK	88	97	88	62
3. PIGGY-BACK (TRUCK ON RAIL CAR)	12	0	0	24
4. OTHER (PLEASE SPECIFY)	0	0	0	0
TOTAL	100%	100%	100%	100%
H. DO FUTURE PLANS (NEXT 10 YEARS) INCLUDE FACILITIES TO HANDLE 10- AND/OR 14-TON UF₆ CYLINDERS IF SUCH FACILITIES DO NOT CURRENTLY EXIST?				
YES <u>3</u> NO <u>7</u>				

COMMENTS _____

5.0 TRANSPORT PROBLEMS, 1976-2000

R. J. Hall - Project Manager

The objective of this program is to examine future (1976-2000) transportation systems, trends and problems associated with transport of radioactive and fuel cycle materials to assure a more orderly problem solving approach.

SUMMARY

Working papers on the workshop titled "Nuclear Materials Transportation, 1976-2000" have been prepared by Pacific Northwest Laboratory and Battelle-Columbus Laboratory. All workshop participants were mailed summaries of the workshop results. The papers presented at the Atomic Industrial Forum Conference on "Transportation of Radioactive Materials" are being analyzed and integrated into the study. An overview of the logistics of the nuclear fuel cycle is being prepared. It includes the cost of each link of transportation in the fuel cycle. Plans for completion of this phase of the study are being prepared.

TECHNICAL PROGRESS

The individual sessions at the "Nuclear Materials Transportation, 1976-2000" workshop have been summarized and analyzed. These summaries include the conclusions reached at the sessions as well as general comments made during the sessions. BCL has prepared a commentary on the work-

shop and suggestions on the output of the workshop. Simultaneously PNL has prepared written discussions of the workshop sessions. Both of these will be integrated into a proceedings that could be used as the workshop's final output. The workshop stressed trends and events external to the nuclear industry that could affect transport of radioactive material.

The major links in transportation of nuclear fuel cycle materials have been examined in terms of cost and transportation system logistics. While there are numerous links in the transportation cycle, only a few involve material that requires unusual handling and is sufficiently critical that it might lead to problems in operation of the nuclear fuel cycle. The more costly links are found to be transport of spent fuel from reactors to reprocessing sites, of cladding waste from reprocessing sites, and of high-level waste. These account for the major share of transportation costs and are most subject to scrutiny because of their radioactivity.

In addition to cost, the availability and requirements for storage and containers have been examined.

5.2

Again, as with cost, the spent fuel, high-level waste, and cladding waste are seen as the more critical links in the overall transportation logistics system.

To gain a better understanding of trends, events, and actions within the nuclear power industry, papers presented at the AIF conference titled "Transportation of Radioactive

Materials" are being examined and analyzed. Information from the conference may be combined with results of the workshop, which stressed events, trends, and actions outside the nuclear industry. The combined information may provide a balanced viewpoint of potential problems in the transport of radioactive materials.

6.0 SAFETY AND ECONOMIC STUDY OF SPECIAL TRAINS

R. J. Hall - Project Manager

The objective of this program is to compare the safety and economics of special trains with those of regular trains for shipment of nuclear fuel cycle material.

SUMMARY

Analyses of train accident data have been expanded to include 1972, 1973 and 1974. Correlations have been obtained for accident frequency versus speed; accident frequency versus train length; dollar damage to rail property versus train speed at time of accident; involvement with hazardous material; accident cause; and other pertinent data. Those data are being analyzed to determine effects of speed, passing, and freight mixing restrictions on safety of spent fuel transport.

Route profiles of special trains and conventional freight trains are being established to determine differences between the two types of rail transport.

The spent fuel railroad network for the model is being used to determine minimum shipping cost schedule for the spent fuel cycle. Once the schedule is determined, a series of cases is planned to evaluate effects of variations in parameters such as train speed, turnaround time, size of shipment, and cask utilization.

SAFETY ANALYSIS

Freight train accident data for Class I line-haul railroads for the years 1972-1974 have been studied to determine number of accidents as a function of train speed and dollar damage to railroad property. For freight train accidents in a given speed range, the fraction of accidents resulting in high dollar damage increases as the speed increases. For example, the 1972-1974 data indicate that only 0.7% of accidents occurring at speeds between 20 and 29 mph resulted in greater than \$250,000 damage to railroad property, whereas in the speed range 60 to 69 mph the percentage was 9.2%. For a given dollar damage threshold, the fraction of accidents occurring at high speeds increases as the damage threshold is increased. For example, of accidents with greater than \$75,000 damage to railroad property, 18.5% occurred at speeds in the 20-29 mph range and 5.1% occurred at speeds in the 60-69 mph range. For accidents with greater than \$250,000 damage to railroad property, 8.7% occurred at speeds in the

60-69 mph range. The increase in dollar damage to railroad property associated with high-speed freight train accidents is believed due both to an increase in the average accident severity (in terms of forces involved) and to the greater number of cars involved in high-speed accidents.

The literature contains several estimates of the average number of freight cars involved in derailment accidents as a function of train speed. These include an empirical study of accidents in one railroad made by Sandia Corporation; a study involving a computer model of freight train accidents made by the Pullman Company for the Association of American Railroads; and a study of tank car derailment accidents performed jointly by the Railroad Progress Institute and the Association of American Railroads. In general, there is a significant increase with speed in the number of freight cars involved in freight train derailment accidents. The three studies cited above indicate that the average number of freight cars derailed in accidents occurring at 60 mph is about three times the average derailed at 20 mph.

To gain more information on the relationship between speed and accident frequency, the number of accidents occurring at a given speed range was estimated as a function of total freight train miles traveled at that speed. A weighted estimate was made, for a representative route, of the length of track on which a freight train would travel at a speed not greater than 35 mph as compared to the total length of track. With this

approach it was estimated that about two-thirds of all freight train miles are traveled at speeds less than 35 mph. This estimate was confirmed by examining the speed distribution of derailment accidents caused by factors that would be expected to be only weakly coupled to train speed.

COST ANALYSIS

Three reference cases from the 63 routes in the shipment model have been analyzed. These represent a short distance category (0-600 miles), an intermediate distance category (600-1400 miles), and long distance category (more than 1400 miles). The general approach was to establish a set of reference parameters for economic comparison of regular and special train shipments and then to perform sensitivity studies to determine the effects of various cost factors. Major results indicate that for special train shipments costs for single-cask shipments are minimum for short distances, costs for two-cask shipments are minimum for intermediate distances, and costs for three-cask shipments are minimum for long distances. In all cases analyzed so far, special train requirements result in increases in shipping costs from 35 to 100%.

Analysis is near completion on the interactions of variables impacting economics of special trains for the entire spent fuel rail transportation system. A program has been constructed and is now being updated and improved. The program calculates the lowest cost shipping schedule for

6.3

each reactor. The algorithm currently assumes no time schedule for these shipments. The optimized cost-per-reactor algorithm used may not necessarily give the lowest system cost; the latter will vary, depending on scheduling and slack cask-time considerations. No algorithm has yet been developed to deal with this extremely complex situation; therefore, the optimized cost-per-reactor algo-

rithm is used. This type of optimization is also likely to be used by the reprocessor unless a complex scheduling model can be built.

The program is currently being modified to calculate the minimum number of casks required by the system and the associated costs so that cost advantages of previous decision logic can be estimated.

7.0 DEVELOPMENT OF HIGH-LEVEL WASTE SHIPPING CASK MODELS

R. J. Hall - Project Manager

The objective of this program is to conceptually design and fabricate models of a cask for transporting high-level waste by railroad.

SUMMARY

Two types of models of a high-level waste shipping cask are to be fabricated in this project: 1) a model of the cask mounted on an H0-gauge rail car (1:87 scale) and 2) a table model of the cask alone for use in portable displays.

The design of a conceptual solidified high-level waste shipping cask system was completed. The 1/10th scale table display model of the cask is currently being fabricated.

TECHNICAL PROGRESS

The system design was completed, and a 1/10th scale table display model of a conceptual solidified high-level

waste shipping cask is being fabricated. The cask for the table model will be about 1 ft diam by 2 ft long and will include a cutaway section showing the high-level waste canisters inside. The table display model is expected to be ready early in July. A display poster to accompany the model has been prepared to further illustrate the conceptual design and to characterize the solidified high-level waste transported. Fabrication of the H0-gauge cask car system model has been initiated.

The concept drawings and sketches used for model shop fabrication are being refined. These drawings will be sent to various ERDA laboratories for review and comment.

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A. A. Churm
ERDA Chicago Patent Group
9800 South Cass Avenue
Argonne, IL 60439

S. H. Smiley
Deputy Director for
Fuels and Materials
NRC Directorate of Licensing
for Fuels and Materials
4915 St. Elmo Ave.
Bethesda, MD 20014

W. P. Bishop
Chief, Waste Management Branch
NRC Division of Materials and
Fuel Cycle Facility Licensing
Washington, DC 20555

W. G. Belter
ERDA Division of Biomedical and
Environmental Research
Earth Sciences Branch
Washington, DC 20545

W. A. Brobst
ERDA Division of Environmental
Control Technology
Washington, DC 20545

2 W. E. Mott
ERDA Division of Environmental
Control Technology
Washington, DC 20545

3 F. P. Baranowski
ERDA Division of Nuclear Fuel
Cycle and Production
Washington, DC 20545

C. R. Cooley
ERDA Division of Nuclear Fuel
Cycle and Production
Washington, DC 20545

R. W. Ramsey
ERDA Division of Nuclear Fuel
Cycle and Production
Washington, DC 20545

R. A. Wolfe
ERDA Division of Nuclear Fuel
Cycle and Production
Washington, DC 20545

G. Wehmann
ERDA Idaho Operations Office
P. O. Box 2108
Idaho Falls, ID 83401

ERDA Oak Ridge Operations Office
P. O. Box X
Oak Ridge, TN 37830

246 ERDA Technical Information Center

2 Allied Chemical Corporation
550 - 2nd Street
Idaho Falls, ID 83401

J. A. Buckham
Allied-General Nuclear Services
P. O. Box 847
Barnwell, SC 29812

A. Williams
Allied-General Nuclear Services
P. O. Box 847
Barnwell, SC 29812

M. J. Steindler/
L. E. Trevorow
Argonne National Laboratory
9700 South Cass Ave.
Argonne, IL 60439

Battelle Memorial Institute
505 King Ave.
Columbus, OH 43201

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Combustion Engineering, Inc.
Windsor, CT 06095

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Rockwell International
Rocky Flats Division
Golden, CO 80401

M. D. Boersma
duPont Company, Aiken (ERDA)
E. I. duPont DeNemours and Co.
Savannah River Laboratory
Aiken, SC 29801

C. H. Ice
duPont Company, Aiken (ERDA)
E. I. duPont DeNemours and Co.
Savannah River Laboratory
Aiken, SC 29801

A. S. Jennings
duPont Company, Aiken (ERDA)
E. I. duPont DeNemours and Co.
Savannah River Laboratory
Aiken, SC 29801

No. of
Copies

P. H. Permar
duPont Company, Aiken (ERDA)
E. I. duPont DeNemours and Co.
Savannah River Laboratory
Aiken, SC 29801

L. Henning
Electric Power Research Institute
3412 Hillview Ave.
P. O. Box 10412
Palo Alto, CA 94304

Environmental Protection Agency
Technology Assessment Div. (AW-559)
Office of Radiation Programs
U. S. Environmental Protection
Agency
Washington, DC 20460

R. G. Barnes
General Electric Company
175 Curtner Ave.
(M/C 160)
San Jose, CA 95125

L. H. Brooks
Gulf Energy and Environmental
Systems
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