Nuclear Waste Management and Transportation Quarterly Progress Report January Through March 1976

**Compiled by
A. M. Platt**

June 1976

**Prepared for the Energy Research
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Battelle
Pacific Northwest Laboratories

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NUCLEAR WASTE MANAGEMENT AND
TRANSPORTATION QUARTERLY PROGRESS
REPORT JANUARY THROUGH MARCH 1976

Compiled by

A. M. Platt
Nuclear Waste Technology Program

June 1976

Battelle
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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 October through December 1975, BNWL-1978.

CONTENTS

FOREWORD	iii
1.0 ALTERNATIVE WASTE MANAGEMENT SYSTEMS	1.1
Summary	1.1
Disposal Media Studies	1.1
Advanced Geologic Monitoring - Remote Sensing.	1.2
Public Attitudes Survey	1.6
Perceived Risk	1.6
Communication With the Public.	1.8
The Reconcentration Phenomenon of Radionuclide Chain Migration.	1.8
Management Perspectives for Nuclear Fuel Cycle Wastes.	1.16
References	1.24
2.0 DECONTAMINATION AND DENSIFICATION OF CHOP-LEACH CLADDING RESIDUES.	2.1
Summary	2.1
Technical Progress	2.1
References	2.8
3.0 TRITIUM SEPARATION AND FIXATION	3.1
Summary	3.1
Technical Progress	3.1
References	3.4
4.0 DISPOSITION (D&D) OF RETIRED CONTAMINATED FACILITIES AT HANFORD	4.1
Summary	4.1
Technical Progress	4.1
5.0 TRANSPORTATION SAFETY STUDIES.	5.1
Summary	5.1
Technical Progress	5.1
6.0 TRANSPORT PROBLEMS, 1976-2000.	6.1
Summary	6.1
Technical Progress	6.1
7.0 SAFETY AND ECONOMIC STUDY OF SPECIAL TRAINS	7.1
Summary	7.1
Technical Progress	7.1
8.0 DEVELOPMENT OF HIGH-LEVEL WASTE SHIPPING CASK MODELS	8.1
Summary	8.1
Technical Progress	8.1
Future Plans	8.2

LIST OF FIGURES

1	LANDSAT-1 Satellite Imagery, Southeastern New Mexico, October 18, 1974, Image Width 115 Miles	1.4
2	LANDSAT-1 Satellite Image Subsection Enlargement, Showing Detail, Southeastern New Mexico, October 18, 1974, Image Width ~38 Miles	1.5
3	Preliminary Surface Materials Classification, LANDSAT-1, Image of the Area Covered by Figure 2	1.7
4	Waste Management Control Surface for Incremental Background Dose	1.10
5	The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 5$	1.14
6	The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 50$	1.14
7	The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 500$	1.14
8	The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 5000$	1.14
9	The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 50,000$	1.15
10	Waste Management Control Surface for Background Dose from Spent Fuel Elements	1.17
11	Waste Management Control Surface for Background Dose from TRU Solids	1.17
12	Waste Management Control Surface for Background Dose from TRU Contaminated Cladding Hulls	1.19
13	Waste Management Control Surface for Background Dose from Tritium	1.19
14	Waste Management Control Surface for Background Dose from Carbon	1.20
15	Waste Management Control Surface for Background Dose from Iodine	1.20
16	Waste Management Control Surface for Background Dose from Miscellaneous α , β , γ Wastes	1.22
17	Waste Management Control Surface for Background Dose from Miscellaneous β , γ Wastes	1.22
18	Waste Management Control Surface for Background Dose from Ore Tailings	1.23
19	Waste Management Control Surface for Background Dose from Ore Tailings for Year 2000 Only	1.23
20	Oxidized Zircaloy-4 Melted by Inductoslag-USBM Number 529	2.5
21	Oxidized Zircaloy-Stainless Steel-Inconel Alloys Melted by Inductoslag-USBM Number 534	2.6
22	Representative Zirconium-Hydride Samples for Various Conditions of Preparations	3.2
23	Zirconium-Hydride Sample Showing Intricate Cracking and Unhydrided Zircaloy-2 Sample	3.2
24	Pressure-Composition Isotherms (Composite Data)	3.3
25	Effluent System, D and DR Reactors	4.4

LIST OF TABLES

1	Characteristics of the Geosphere Transport Model	1.9
2	Parameter Values for ^{230}Th and ^{226}Ra	1.11
3	Summary of Nuclide Inventory Estimation Methods	1.18
4	Chemical Analyses of High Oxygen Zircaloy-4 Ingot Number USBM 529, Values Given in ppm	2.4
5	Chemical Analyses of Slag Used for Melting Zircaloy-4 Ingot Number USBM 529	2.4
6	Summary of Melting Rates	2.7
7	Radionuclide Inventories (Estimated) Used as Basis for "Worst Case" Radiological Dose Calculations	4.2
8	107-DR Retention Basin "Worst Case" Doses	4.2
9	Crib 216-S1 & 2 "Worst Case" Doses, Airborne Release	4.3
10	Estimates of Decontamination and Decommissioning Costs, 107-D/DR Retention Basin System, in Thousands of 1976 Dollars	4.5
11	Information on Projected Shipments of Radioactive Fuel Cycle Materials in the U.S.	6.2

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

Documentation of the workshop on actinide-sediment reactions was completed. An annotated bibliography on this topic, to be issued as BNWL-1983, is in press.

Development was started on remote sensing techniques for regional reconnaissance in connection with selecting waste repository sites.

Work on interaction with public interests included extension of analysis of data from the prior study of public attitudes and initiation of studies on perceived risk and public communications.

Studies of nuclide migration in geologic media were extended to investigation of reconcentration phenomena and non-high-level wastes.

DISPOSAL MEDIA STUDIES - R. J. Serne, D. Rai, and L. L. Ames

Documentation of the papers and question and answer sessions for the "Actinide-Sediment Reactions Working Meeting" is in the final stages of editing. It will be published as an ERDA symposium document during the next quarter. The document entitled A Review of Actinide-Sediment Reactions, with an Annotated Bibliography, BNWL-1983,

is in press and will be available around June 1, 1976.

A paper, "Plutonium Activities in Soil Solution and the Stability and Formation of Selected Plutonium Minerals," authored by D. Rai and R. J. Serne was presented at the Environmental Chemistry and Cycling Process Symposia sponsored by ERDA, the University of Georgia, and the Savannah River Ecology Laboratory at Augusta, Georgia, April 28-May 1, 1976. This paper and an accompanying one on americium are being submitted to the Journal of Environmental Quality.

Laboratory work is well under way to investigate the mechanisms which bound the plutonium and americium from the wastes disposed to the Z-9 crib, Z-12 crib, and Z-1a trench on the Hanford Reservation. From selective extraction experiments on the contaminated soils from the Z-9 crib, large portions of these transuranics appear to be extracted by acid ammonium oxalate and DTPA. Acid ammonium oxalate dissolves iron and portions of aluminum hydrous oxides while DTPA acts as a strong complexant and will strip elements which are bound with moderate strength to

organic and other inorganic sediment phases. Thirty-five to forty-five percent of the plutonium and 25 to 30% of the americium was not removed even after 8N HNO_3 acid treatment, possibly signifying the presence of discrete PuO_2 particles. About 7% of the americium was found to be readily mobilized from the soil with a 1N MgCl_2 solution whereas only 0.1 to 0.2% of the plutonium was released. Results of selective extractions on the Z-12 crib soils will be available within 2 weeks. Experimentation will begin on Z-1a trench soils upon their receipt from Atlantic Richfield Hanford Company.

Initial investigations on determining the leach rates of both synthetic and actual high-level waste glass have started. Concentrations of Al, Cs, Eu, Fe, K, Na, Ru, Si, Sr, and Zn in a distilled water leachate after 30 days' contact have been determined for a synthetic glass. Radiochemical analyses of an actual high-level glass leachate were begun the week of May 17. Utilizing gamma, beta, alpha, and nondestructive photon spectroscopy techniques coupled with wet chemical separations, the concentrations of various isotopes of Cs, Sr, Ru, Tc, Pm, Eu, U, Np, Pu, Am, Cm and others hopefully will be determined. Initial work will determine the feasibility, analytical techniques, and costs needed to determine the leach rates of radioactive waste glass under various conditions.

ADVANCED GEOLOGIC MONITORING - REMOTE SENSING - H. P. Foote and D. D. Hostettler

A program for identifying optimum sites for long-term storage of radioactive waste materials requires at the outset the development of techniques for rapid, efficient reconnaissance of large land areas. The recent availability of high-quality satellite imagery permits the development of remote sensing techniques providing this needed capability. A pilot study has been started to develop and demonstrate techniques designed specifically to utilize multispectral satellite imagery in these surveys for ERDA's Waste Storage Program. This preliminary study will focus on geologic and hydrologic factors such as geologic structure, faulting, and surface water distribution. The test area, a portion of the Permian Basin in the south-central United States, includes the bedded salt site now being developed by Sandia Laboratories.

Preliminary computer imagery processing capabilities for analysis of LANDSAT imagery have been developed. These capabilities include:

- Transfer image from tapes to disc - ~20 min
- Correct striping (detector gain inconsistencies) - ~30 min
- Interpolate to square pixels (59.3 m^2)
 - a) Using nearest neighbor technique - ~50 min

- b) Using bilinear technique -
~4 hr
- Save corrected image to tape
(3120 pixels x 3120 pixels) -
~20 min
- Supervise spectral classification
- Enhance image.

These capabilities provide the basic framework for conducting regional studies with the satellite data.

The image enhancements, Figures 1 and 2, were generated with a program that gives the user considerable control over contrast, magnification, and color. The whole LANDSAT image is stored on the disc so that any portion of it, up to the entire image, may be used to make a picture. A histogram of intensity for the selected subpicture can be printed for control of contrast. There are two automatic and one manual option for contrast stretching. Thus, the contrast in each band of a color composite can be selected separately to produce the best overall picture. A full frame (3120 x 3120) color composite takes about 20 to 25 min to expose on the DICOMED, Figure 1. A single band requires about 6 min. The examples of color pictures here were shot on 4 x 5 Kodak Ektacolor® film and prints were made from the negatives. These images are in the test area in southeastern New Mexico.

The striping correction removes the appearance of scan lines caused

by detector gain inconsistencies. Striping is first characterized by calculating the means and standard deviations of pixel levels for each of the four bands and six sensors. Usually several test areas are selected within an image to properly characterize the striping. The main striping problems often are less than one level off. The correction technique involves randomly adding or subtracting a level from the proper band and sensor based on its deviation from the mean for a specific band. Example: If band 4 and sensor 3 had a mean error of +0.82 levels (as compared to the five other sensors), then 82% of these pixels would have one level randomly subtracted from the original pixel level.

For interpolation to square pixels, the nearest neighbor technique seems adequate. Bilinear interpolation can be used though it is over four times longer.

The supervised classification program operates on a 7 x 1024 x 1024 array stored on the disc. Four of the 1024 x 1024 arrays contain the original LANDSAT bands, 4-7. The other arrays are used for various types of output or for additional input arrays to aid classification. These might be band ratios generated from the original four bands or from bands selected at other times. The training samples are selected with a digitizer and cathode ray tube (CRT). The mean vector and covariance matrix are calculated for each training sample. The eigen values and eigen

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FIGURE 1. LANDSAT-1 Satellite Imagery, Southeastern New Mexico, October 18, 1974, Image Width 115 Miles

FIGURE 2. LANDSAT-1 Satellite Image Subsection Enlarge-
ment, Showing Detail, Southeastern New Mexico, October 18,
1974, Image Width ~38 Miles

vectors of the covariance matrix are also calculated and displayed along with the correlation matrix. Two-dimensional scatter plots and histograms of the samples may also be displayed on the CRT or plotted if the user desires.

Currently the program uses only the mean and standard deviation of the training sample for classification. Soon it will be able to use a reference table approach based on the complete covariance matrix when required. The present version of the program can classify a 1024 x 1024 pixel region into 15 classes in about 10 min using four bands. The classification results can be checked on a memory scope or photographed in color on the DICOMED recorder. Figure 3 is an example of output from the classification program for the area covered by Figure 2. Usually several iterations through this procedure are required to produce the final classification map. A color coded classification key with areas of each class is also produced. Figure 3, for example, shows vegetation as green and bodies of water as deep blue. Major factors determining the colors are the presence of water, density and variations of vegetation types, soil types, and rock types.

PUBLIC ATTITUDES SURVEY - John Hébert (HARC)

Review of the report on the public attitudes survey was completed. The report, BNWL-1997, Public Attitudes Associated with Nuclear Waste Disposal, is now in press.

The major activity for this quarter has been further review of decision and policy analysis models for the comparative assessment of nuclear waste management strategies. An integrated review of these models to be completed during the coming quarter will address such issues as the strengths and weaknesses of decision models aids for generating, screening, and evaluating design alternatives. Other topics will include the relative comprehensiveness of the different approaches, their ability to assist in identifying the sources of technical uncertainty, and the degree to which they lend themselves to resolution of conflict over technical and social values.

Other activities included two papers delivered at the Western Psychological Association meeting which presented the data from the public attitudes report. In addition, a condensed, relatively non-technical version of this report was prepared for distribution to participating organizations and consultants as well as for broader popular circulation.

PERCEIVED RISK - John Hébert (HARC)

Work has just begun on outlining a program of research on the relationship between individuals' perceptions of risk and formal risk assessment techniques. Research instruments to probe the individual's risk assessment process are being developed in this phase of the work. It is hoped that some understanding of informational needs will be identified by

FIGURE 3. Preliminary Surface Materials Classification,
LANDSAT-1, Image of the Area Covered by Figure 2

comparison of the formal and the individual risk assessment methods.

COMMUNICATION WITH THE PUBLIC -
John Hébert (HARC)

Work has recently begun on a study of communication with the public on nuclear waste issues. Present efforts involve compiling materials and analyzing previous attempts by public and private agencies to inform the public about nuclear waste issues. Also under way is the construction of a test of knowledge about nuclear waste issues and technology. Secondary content areas include nuclear plant safety, nuclear materials safeguards, and knowledge of alternative power sources. This work is expected to yield a background on which to build better means of providing the public balanced information about nuclear waste issues.

THE RECONCENTRATION PHENOMENON OF
RADIONUCLIDE CHAIN MIGRATION -
H. C. Burkholder

Summary

The potential reconcentration of radionuclide decay products during their transport by flowing groundwater from underground geologic nuclear waste disposal sites to the biosphere is analyzed. The calculations show that the predicted maximum (but not the average) nuclide discharge rates to the biosphere for certain nuclides in particular disposal environments can increase above their corresponding values for surface storage. Thus

as a result of geologic isolation the largest dose that any individual receives throughout geologic time from these particular nuclides can be increased above doses from surface storage even though the discharge of the nuclides to the biosphere has been significantly delayed. Conversely, the doses averaged over all geologic time are decreased from surface storage doses as a result of the geologic isolation. Both the magnitude and range of occurrence of the phenomenon are functions of the adsorption properties of the geologic medium, the migration path length to the biosphere, and the leach rate of the waste form. The results of this investigation emphasize the need to perform a thorough and comprehensive safety analysis of nuclear waste management systems and to establish the acceptability criteria for geologic isolation systems with great care.

Background

Three mathematical models to describe the migration of radionuclides from underground geologic isolation under various conditions have been completely or partly developed under this task.⁽¹⁻⁴⁾ A computing scheme (named GETOUT) was developed to evaluate numerically the analytical equations of the model described in Reference 1, and this computing scheme was applied (together with a biosphere transport model) to examine the incentives for partitioning high-level U.S. nuclear power economy waste.⁽⁵⁾ The results of these combined geosphere-biosphere transport

calculations using the geosphere model described in Table 1 are summarized in Figure 4.

The axes on this three-dimensional plot are measurement scales for three barriers which man can select or control when designing a management system for nuclear waste. The time of initial release after the Year 2000 is a measure of canister integrity and site stability from water penetration; the leach time is a measure of the resistance of the waste form to leaching; and the path length is a measure of geosphere isolation, i.e., migration distance to the biosphere. An increase in any of these parameters results in greater isolation of the

waste from man's environment. A surface can be located on this three-dimensional plot which corresponds to any assumed incremental radiation dose level (i.e., background in this case). Combinations of path length, leach time, and time of initial release which define points located inside or under the surface (i.e., between the surface and the origin) have incremental doses above background; combinations located outside or above have doses below background.

An Example from Previous Work

The so-called ^{226}Ra ridge shown in Figure 4 provides an example of what is herein called the reconcentration

TABLE 1. Characteristics of the Geosphere Transport Model

1. The radionuclide source is contained in a nonsalt repository surrounded by a western U.S. desert geologic medium.
 2. The waste is contacted by groundwater from a typical U.S. desert starting at varying times between the Year 2000 and the Year 10,002,000.
 3. The waste is leached by that groundwater at varying rates between 0.00003 and 100%/yr.
 4. The groundwater moves from the repository through a one-dimensional column of the medium and discharges into a surface water body.
 5. The sorption equilibrium constants are based on measurements and estimates for U.S. desert subsoils and are assumed valid for the geologic medium.
 6. The decay of the nuclides (including chain decay of the actinides) is modeled both in the repository and in the geologic medium.
 7. The groundwater velocity is 1 ft/day.
 8. The path length from the repository to the surface water body varies from 0 to 100 miles.
 9. The axial dispersion coefficient is $0.008 \text{ cm}^2/\text{min}$.
-

FIGURE 4. Waste Management Control Surface for Incremental Background Dose

phenomenon. Using conditions on the ridge, for a path length of 0 miles and a time of initial release of 100,000 years after the Year 2000, a leach time of about 150,000 years is required to reduce the maximum ²²⁶Ra dose to background levels (6000 mrem for a 50-year accumulated dose). However, at a path length of 0.5 miles and a release which begins in the Year 2000, a leach time of about 3,000,000 years is required to produce a ²²⁶Ra dose equal to background. The arrows in Figure 4 denote the location of these scenarios. For both scenarios, the "maximum" individual receiving those doses lives 100,000

years after the Year 2000. Thus, as indicated by the ratio of the required leach times, the received dose to the individual living in the Year 102,000 is about 20 times greater (3,000,000/150,000) if the nuclides migrated through a geologic medium instead of being released directly to the biosphere. Hence the presence of the geologic medium between the underground repository and the biosphere made the projected ²²⁶Ra dose greater than the dose with no geologic medium at all.

The reason for these results cannot be clearly discerned from the migration calculations reported in Reference 5 because the ²²⁶Ra and its

precursors are not in decay equilibrium with one another at the Year 2000 (the radionuclide source was high-level waste). The ^{226}Ra inventory increases with time by decay from its precursors so that waste leached from the repository at later times has a higher ^{226}Ra concentration than that released at earlier times. This ^{226}Ra ingrowth possibly distorts the reconcentration phenomenon.

Description of the Present Study

To avoid the potentially complicating effect of ^{226}Ra ingrowth and thus better understand the reconcentration phenomenon, we have investigated a different radionuclide source which contains the same nuclides that cause the ^{226}Ra ridge in Figure 4 but which is in decay equilibrium. We considered uranium ore necessary to support the projected U.S. nuclear power economy in the Year 2000 and its contained ^{230}Th and ^{226}Ra inventories. These are also the same ^{230}Th and ^{226}Ra inventories which are ultimately contained in the tailings from the ore milling operation.

Thus, as long as we consider only the $^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ chain, we can refer to the ore and the ore tailings interchangeably. As shown in Table 2, these two nuclides are in decay equilibrium with one another and, thus, have the same curie (but not mass) inventories. We assume that all of the uranium ore is located underground in a nonsalt geologic formation whose nuclide sorption characteristics for ^{230}Th and ^{226}Ra are the same as those estimated for western U.S. desert subsoils.⁽⁶⁾ We neglect axial dispersion (which simplifies the model described in Table 1, but does not substantially affect the calculated discharges for the situation of this study) and assume that the release begins in the Year 2100. Using the GETOUT computing scheme, we calculate the discharge of ^{226}Ra to the biosphere for various path lengths and leach times. The path length is the distance that the nuclides traverse in migrating from the underground disposal site to the biosphere. The leach time is the time necessary for complete dissolution of the waste form.

TABLE 2. Parameter Values for ^{230}Th and ^{226}Ra			
Nuclide	Inventory, Ci*	Half-Life, Yr	Sorption Equilibrium Constant**
^{230}Th	2.81×10^4	7.7×10^4	50,000
^{226}Ra	2.81×10^4	1.6×10^3	500

* Uranium ore necessary to support the projected U.S. nuclear power economy in the Year 2000.

** Reference 6, pp. 3-54.

Results and Discussion

Figure 7 presents the maximum ^{226}Ra discharge rate to the biosphere as functions of path length and leach time for the situation described by Table 2. The reconcentration of ^{226}Ra by the geologic medium is evident in the curves for the 1,000, 10,000 and 1,000,000 year leach times. Using the curve for a 100,000 year leach time as an example, the maximum discharge rate of ^{226}Ra to the biosphere is 14 times greater for a 0.1 mile path length than for a 0 mile path length. Thus, designing a waste management system for ore tailings to have 0.1 miles of western U.S. desert subsoil or equivalent as the geologic barrier causes the maximum ^{226}Ra discharge to the biosphere to be 14 times greater than it would have been if the ore tailings had remained on the surface and the leach time is 100,000 years. Therefore, depending on the leach rate of ore tailings, near surface disposal may have larger maximum discharges to the biosphere from a leach incident than surface disposal when only the groundwater transport pathway is concerned. The burial of ore tailings may be greatly beneficial in decreasing nuclide transport by the air and surface water transport pathways. Of course, increasing the leach time (decreasing the leach rate) is clearly beneficial in reducing radionuclide discharges to the biosphere, as Figure 7 shows. As path length increases beyond 0.1 mile, the reconcentration effect decreases, and for path lengths greater than 1 mile (for

the 100,000 year leach time curve) the normal dilution behavior is observed. As shown in Figure 7, the magnitude of the reconcentration effect decreases as leach time decreases and essentially disappears for a leach time of 100 years. Furthermore, as leach time increases, the magnitude of the reconcentration effect approaches a maximum and constant value of about 14. Thus, the reconcentration phenomenon is affected by both geosphere path length and leach time. It is also affected by the sorption equilibrium constants as discussed below.

Because the values of the sorption equilibrium constants can vary by orders of magnitude for various geologic media, it is important to understand the effect of their value on the reconcentration phenomenon. Figures 5, 6, 8 and 9 show curves similar to those in Figure 7, but for different values of the ^{226}Ra sorption equilibrium constant ($K_{\text{Ra-226}}$). Figures 5 through 9 are arranged in order of increasing $K_{\text{Ra-226}}$. We note that as $K_{\text{Ra-226}}$ increases, the migration velocity of ^{226}Ra decreases. Comparison of Figures 5 and 6 with Figure 7 shows that as $K_{\text{Ra-226}}$ decreases, the maximum value of the reconcentration effect increases slightly from 14 to 17.5, but more importantly, the path length necessary to counteract the reconcentration effect increases by two orders of magnitude. As shown in Figure 5, a path length of 100 miles is required to counteract the reconcentration effect of the leach time is 10,000 or 1,000,000 years and $K_{\text{Ra-226}}$ is 5. Comparison of

Figures 8 and 9 with Figure 7 shows that as $K_{\text{Ra-226}}$ increases, the maximum value of the reconcentration effect decreases and finally disappears completely when $K_{\text{Ra-226}} = K_{\text{Th-230}}$. Thus the sorption properties of the geologic medium can have an effect on both the magnitude and range of occurrence of the reconcentration phenomenon.

Physical Interpretation

We can give the reconcentration phenomenon a physical interpretation using the $^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ chain as an example. 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 leaching overtakes ^{230}Th which was leached from the 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. If this combined ^{226}Ra inventory exits from the geologic medium and enters the biosphere before it decays, the phenomenon is observed.

Conclusions and Implications to Waste Management

From the results presented here several conclusions about the reconcentration phenomenon can be made.

1. The phenomenon can occur only when the second member of a decay chain migrates faster than its precursor. (K for the second member is less than that of its precursor.)

2. Increasing the geosphere path length (from zero) first increases and then decreases the magnitude of the effect.
3. Increasing the leach time (i.e., decreasing leach rate) increases the magnitude of the effect but at the same time reduces the discharge rates to the biosphere.
4. Decreasing the sorption equilibrium constant of the second chain member (i.e., faster migration) increases both the magnitude and the range of occurrence of the phenomenon.

The results presented here for the $^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ chain show that a small degree of geologic isolation can under certain circumstances make the maximum ^{226}Ra discharge rates to the biosphere (and hence the maximum dose to man) larger than no isolation at all. (However, doses averaged over geologic time will be decreased by geologic isolation.) Further, the results show that isolation in geologic media where ^{226}Ra moves very fast relative to ^{230}Th makes the range of occurrence of the reconcentration phenomenon so large (100 miles) that its effect probably cannot be overcome in a practical sense by greater degrees of isolation in that media. Looking at the results in another way, the sorption properties of the geologic medium can partially negate (under certain circumstances) the benefits of a low leach rate waste form for the $^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ chain if water-induced leaching and migration occur.

The importance of the reconcentration phenomenon to nuclear waste management is also dependent on the

FIGURE 5. The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 5$

FIGURE 6. The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 50$

FIGURE 7. The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 500$

FIGURE 8. The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 5000$

FIGURE 9. The Effects of the Geosphere Path Length and the Leach Time on the Reconcentration Phenomenon for $K_{Ra} = 50,000$

transport of nuclides in the biosphere where accumulation (concentration) and dispersion (dilution) phenomena are simultaneously active. If the biosphere dispersion phenomena are dominant, nuclides entering the local biosphere from a breached nuclear waste repository will be biologically active for only a short period (we presently assume a 50-year accumulation followed by a 50-year exposure to that accumulation) before being dispersed at low levels into the regional, national, and international biosphere. For this situation the reconcentration phenomenon could control potential local consequences from the $^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ chain. However, if biosphere accumulation phenomena are dominant, nuclides discharging to the biosphere will be

accumulated locally over long time periods. For this situation the biosphere accumulation will overwhelm any geosphere reconcentration effects, and the geosphere reconcentration will be insignificant by comparison.

Research Needs and Plans for Future Work

The reconcentration phenomenon emphasizes the need to:

1. Measure in the laboratory and in situ the sorption equilibrium constants for nuclides in various geologic media, and develop a better fundamental understanding of actinide migration phenomena.
2. Improve the biosphere transport model to include the long-term (>100 years) accumulation and dispersion of nuclides in the biosphere, and develop a better fundamental understanding of nuclide biosphere concentration phenomena.
3. Measure in the laboratory and in situ the leaching behavior of the waste forms to be used in nuclear waste management systems.

Many questions on the reconcentration phenomenon remain to be investigated, among which are:

1. What combinations of half-lives, sorption equilibrium constants, leach time, and path length maximize the magnitude of the effects?
2. How much would the effect be increased for the $^{234}\text{U} \rightarrow ^{230}\text{Th} \rightarrow ^{226}\text{Ra}$ chain, geologically isolated under conditions where $K_{\text{U-234}} > K_{\text{Th-230}} > K_{\text{Ra-226}}$?

3. What other decay chains of importance to nuclear waste management exhibit the phenomenon?

These questions will be addressed in future reporting periods.

MANAGEMENT PERSPECTIVES FOR NUCLEAR FUEL CYCLE WASTES - H. C. Burkholder

As described in the preceding section, a method has been developed to evaluate the effectiveness of geologic repository systems in keeping nuclear wastes isolated from man. This method was used to generate Figure 4 (see preceding section), which presents the waste management control surface for the accumulated high-level waste of the U.S. nuclear power economy through the Year 2000 (including all tritium, carbon, and iodine from spent fuel and all activation products from the cladding). To develop a management perspective on potential wastes from the nuclear fuel cycle, control surfaces for other wastes have been drawn on about the same basis as Figure 4.

The inventories of the projected Year 2000 nuclear fuel cycle wastes used in this study were derived using References 5 and 7 and two ORIGEN calculations for the decay of natural uranium and the decay of the descendants of natural uranium. Table 3 summarizes the methods for estimating the inventories.

Figure 10 shows the waste management control surface for spent fuel (incremental dose level is "background"). Except for the small portion of the surface near a 100-yr initial release time after the Year

2000 on the left wall of the plot, the entire surface is controlled by the ^{226}Ra dose. If we accept the somewhat conservative (i.e., dose maximizing) assumptions made in calculating the doses as correct (see Reference 5), 1 ft-diam glass monoliths with a leach rate of about 10^{-6} g/cm²/day and a path length of 100 miles are necessary to reduce the dose to "background" when the release begins in the Year 2000. If the release does not begin for 10,000,000 years, a glass with a leach rate of about 10^{-6} g/cm/day and no geologic isolation are needed. The hidden depression (bounded by the dotted line) in Figure 10 is a result of the reconcentration phenomenon discussed in the previous section. Thus, if our method of calculating the dose is correct (i.e., 50-year accumulation followed by 50-year exposure to that accumulation, and followed finally by complete dilution and dispersal with no subsequent dose), a third management method would store the spent fuel on the surface (zero path length) in the form of a 10^{-7} g/cm²/day glass for 1000 years and then purposely begin leaching it into a surface water body.

Figure 11 shows the control surface for the TRU solids. The location of the entire surface is controlled by ^{226}Ra . Figure 11 is identical to Figure 10 except near a 100-yr initial release time. The figures would also differ if the control surfaces had been drawn at larger values of the leach time. The hidden depression is

FIGURE 10. Waste Management Control Surface for Back-ground Dose from Spent Fuel Elements

FIGURE 11. Waste Management Control Surface for Back-ground Dose from TRU Solids

TABLE 3. Summary of Nuclide Inventory
Estimation Methods

Potential Waste	Procedure
Spent Elements	High-level inventory from Reference 5 (which included tritium, carbon, iodine, and activation products) + $\frac{0.995}{0.005} \times (\text{U and Pu inventories from high-level})$
TRU Solids	$0.994 \times (\text{U and Pu inventories from spent fuel})$
TRU-Contaminated Cladding Hulls	Activation products + $0.001 \times (\text{U, Pu, other transuranics, and fission product inventories from spent fuel})$
Tritium Solids	All tritium from spent fuel
Carbon Solids	All carbon from spent fuel
Iodine Solids	All iodine from spent fuel
Misc. α , β , γ Solids	High-level inventory (except iodine) $\times \left[\frac{20 \times 10^6 \text{ (total } \alpha \text{ activity from misc. } \alpha, \beta, \gamma \text{ solids from Reference 7)}}{1.61 \times 10^9 \text{ (total } \alpha \text{ activity for high-level from Reference 5)}} \right]$
Misc. β , γ Solids	High-level fission product inventory (except iodine) $\times \left[\frac{8 \times 10^5 \text{ (total activity from misc. } \beta, \gamma \text{ solids from Reference 7)}}{1.38 \times 10^{11} \text{ (total fission product activity for high-level from Reference 5)}} \right]$
Ore Tailings	5% of the uranium and 100% of the uranium decay products from the ore adjusted to have the same total activity as the ore tailings in Reference 7

a result of the reconcentration phenomenon. The management requirements for the TRU solids are essentially the same as for the spent fuel.

Figure 12 shows the control surface for TRU-contaminated cladding hulls. As shown, a path length of 5 miles is needed to reduce the dose to "background" if the release begins at the Year 2000 and the hulls are isolated in their natural form.

Figures 13, 14, and 15 present the control surfaces for tritium, carbon, and iodine. As shown in Figure 13, the tritium dose can be reduced to "background" by a 5 mile path length even if an impulse release occurs at the Year 2000. As shown in Figure 14, the carbon dose can be reduced to "background" by 1 ft-diam glass monoliths with a leach rate of about 10^{-5} g/cm²/day regardless of the time of initial release or the path length.

FIGURE 12. Waste Management Control Surface for Background Dose from TRU Contaminated Cladding Hulls

FIGURE 13. Waste Management Control Surface for Background Dose from Tritium

FIGURE 14. Waste Management Control Surface for Back-ground Dose from Carbon

FIGURE 15. Waste Management Control Surface for Back-ground Dose from Iodine

The iodine dose shown in Figure 15 can be reduced to "background" levels by a glass with a leach rate of about 10^{-4} g/cm²/day.

Figures 16 and 17 present the control surfaces for miscellaneous α , β , γ and miscellaneous β , γ solid wastes. If the miscellaneous α , β , γ solid wastes have a relatively high leach rate, they will require a path length of 5 miles for releases beginning in the Year 2000 but only a very small path length for releases beginning in the Year 12,000. The miscellaneous β , γ wastes require a path length 0.3 miles or less (depending on the time of initial release) even if an impulse release occurs.

Figure 18 shows the waste management control surface for ore tailings. The reconcentration phenomenon discussed in the preceding section is strikingly evident by the greater leach time required to reduce the dose to "background" at a path length of 0.1 mile than at a path length of 0 mile. As shown, a path length of 1 mile would be adequate to reduce the doses to "background" if the ore tailings were contained in 1-ft-diam x 10-ft long borosilicate glass monoliths with a 10^{-7} g/cm² day leach rate. Considering the tremendous volume of ore tailings, such a management system may not be practical. Of course, the accumulated ore tailings from the Year 2000 U.S. nuclear power economy are not located in one place

but are spread over many sites in the western U.S. A large tailings pile by present standards would contain only about 1% of the accumulated total nuclide inventory used in drawing Figure 18.

Figure 19 shows the control surface for the ore tailings from the projected U.S. nuclear power economy for the Year 2000 only. The assumed nuclide inventory used to draw Figure 19 is 15 times smaller than that used to draw Figure 18. Comparing the two figures, the strongly nonlinear relationship between the total activity level at a given ore tailings disposal site and the required leach rate to reduce the maximum dose to background is shown. Considering a path length of 3 miles as an example, Figure 18 shows that a leach time of almost 1,000,000 years is required to reduce the maximum incremental dose to "background," while Figure 19 shows that only a 0 year leach time (impulse release) is needed to produce the same result. These results using the ore tailings suggest that multisite disposal of nuclear wastes may offer substantial advantages over single site disposal because of the greatly reduced waste form leach resistance required to meet yet to be determined site acceptability criteria. This suggestion needs to be considered as the U.S. Geologic Isolation Program is planned.

FIGURE 16. Waste Management Control Surface for Background Dose from Miscellaneous α , β , γ Wastes

FIGURE 17. Waste Management Control Surface for Background Dose from Miscellaneous β , γ Wastes

FIGURE 18. Waste Management Control Surface for Back-ground Dose from Ore Tailings

FIGURE 19. Waste Management Control Surface for Back-ground Dose from Ore Tailings for Year 2000 Only

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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 HF-argon treatment for zirconium oxide removal from zirconium metal is being optimized before the mini-scale decontamination process is moved into a cell for treatment of irradiated Zircaloy fuel cladding. The equipment to determine the extent of removal of TRU elements from molten zirconium alloys has been set up in a hood and is ready to begin operation. Plutonium distribution between molten zirconium alloys and calcium fluoride flux will be measured.

The melting of Zircaloy with surface oxide of varying thickness has been demonstrated using the Inducto-slag process. Melts of material with

up to 10,000 ppm oxygen proceeded smoothly and produced satisfactory ingots. Attempts to melt material in the 20,000- to 40,000-ppm oxygen range resulted in excessive splattering and were terminated. Melting of Zircaloy-Inconel-stainless steel alloys with oxide surface layers was tested and found to be possible with the Inductoslag process.

TECHNICAL PROGRESS

Decontamination Processes - B. Griggs

The mini-sized process to treat fuel cladding hulls in 600°C HF-argon mixture has been subjected to nonradioactive evaluation and optimization. In general, the reaction of oxidized

Zircaloy with HF in this facility has been the same as the experiments on single pieces. Stirring of the reaction mixture is difficult but is being improved to avoid unreacted spots of zirconium oxide at the contact points. Unirradiated samples of oxidized Zircaloy-4 fuel tubing have been treated with various mixtures of HF-argon for 40 to 60 min at 600°C. Generally, oxide removal during the subsequent treatment in 0.4M ammonium oxalate-0.16M ammonium citrate-0.1M ammonium fluoride-0.3M hydrogen peroxide at 95°C for 1 hr depended on the gas phase pretreatment. It was increased by higher HF concentrations, longer times, or lighter oxide films during the gas phase treatment.

Even though all equipment problems have not been completely resolved, the problems with nonuniform heating and with burned out heaters have been eliminated, and the process is being prepared for 1706-KEL cave installation. In-cell testing will determine the decontamination on actual fuel hulls and produce realistic side-streams for recycling and treatment. Experience with the mini-process is being utilized to help design the pilot process to be installed in D-cell of 324 Building.

Procurement of irradiated Zircaloy cladding from a BWR is still in process. Typical BWR fuel cladding is needed in order to test the process and to determine the TRU element distribution in the cladding as had previously been done for PWR cladding.⁽¹⁾ A sufficient stock of PWR clad hulls is on hand for the mini-process testing.

Transuranic elements left on the hulls from the surface decontamination together with the transuranic elements produced in the metal by irradiation of uranium and thorium impurities must be removed to produce metal with less than 10 nCi/g of TRU elements. The only opportunity to remove the low residual transuranics from hulls is during melting. The extraction of the high free energy fluorides of many TRU elements into a calcium fluoride flux is feasible. But the exact distribution coefficients for the TRU elements present (largely ²⁴¹Pu) between the flux and melt at low concentrations must be determined. To accomplish this a small vacuum melting apparatus, previously described,⁽²⁾ was constructed in a hood and is being optimized. Problems with thermal insulation and insufficient power input for melting Zircaloy have been solved by using a double wrap of a graphite felt and thinning all heat conductors from the heated regions. These modifications make it easy to reach 2000°C with the 15 kW-10 kHz induction generator. Experimental distribution determinations will begin shortly.

Densification Process - R. G. Nelson

Heat Number USBM 529 melted last quarter⁽²⁾ has been sectioned and analyzed. The purpose of this melt was to demonstrate the Inductoslag process' ability to melt highly oxidized Zircaloy representing uncleaned chop-leach cladding residues. Four batches of Zircaloy-4 clad tubing were autoclaved to total oxygen levels of 4100, 4800, 5000, and 9000 ppm. Each batch was melted separately, one atop

TABLE 4. Chemical Analyses of High Oxygen Zircaloy-4
Ingot Number USBM 529, Values Given in ppm

	Melt Stock ^(a) Batch Number				Ingot 529- Location			
	1	2	3	4	28	20	12	4
C	148	127	142	178	108	106	92	143
H	258	221	286	265	134	147	319	149
O	4,787 ^(b)	4,998 ^(b)	8,947 ^(b)	4,131 ^(b)	4,220	5,040	9,800	4,630
N	39	40	32	45	48	45	43	55
Fe	1,300	1,300	1,300	1,300	1,200	1,200	1,300	1,300
Cr	1,100	1,100	1,100	1,100	1,000	1,000	1,100	1,100
Sn	14,300	14,300	14,300	14,300	13,500	13,500	14,000	14,000
Hf								
Al	<20	<20	<20	25	<20	<20	<20	40
B	4	4	4	4	4	4	10	15
Cd	<0.5	<0.5	<0.5	<0.5	<0.5	<0.5	<0.5	<0.5
Co	<5	<5	<5	<5	<5	<5	<5	<5
Cu	<10	<10	10	<10	10	15	200	20
Mg	<5	80	<5	<5	5	<5	<5	30
Mn	40	60	30	60	30	35	20	60
Ni	15	15	15	10	60	40	40	40
Pb	15	10	15	<10	15	15	30	30
Si	150	200	200	200	300	300	250	300
Ti	<20	<20	<20	<20	<20	<20	<20	<20
V	<20	<20	<20	<20	<20	<20	<20	<20
Mo	100	80	80	60	80	100	80	100
W	<40	<40	<40	<40	<40	<40	<40	<40
Ca	<10	<10	<10	<10	<10	<10	<10	<10
F	RESULTS NOT AVAILABLE				40	20	20	16
Zn	50	<50	<50	<50	<50	<50	<50	<50
Brinell Hardness Number ^(c)					265	283	375	271

- a. Melt stock 1 corresponds to ingot location 28; 2 to 20; 3 to 12; and 4 to 4.
b. Calculated from autoclave weight gain data and the original ingot analysis of 1267 ppm O₂.
c. Original ingot hardness 175.

TABLE 5. Chemical Analyses of Slag Used for Melting
Zircaloy-4 Ingot Number USBM 529

	Starting Slag	Ingot Locations			
		28	20	12	4
Ca	50.4%	50.6%	50.6%	50.4%	50.6%
F	45.0%	47.6%	46.1%	46.1%	46.5%
C	10	75	38	22	12
N	5	5	5	5	5
H	60	60	60	60	60
O	1405	927	685	920	522

Values in ppm except where noted.

the other, to produce a single 10-cm-diam ingot about 80 cm long.

The high (9000 ppm)-oxygen batch showed more splattering and smoke than the others; however, melting was not inhibited. Melting at 85 kW resulted in a melt rate of about 0.34 kg/min. The overall appearance of the ingot is nearly identical to one melted with clean hulls, with the surface finish of the high-oxygen material being less shiny and more matte-like. This experiment's results indicated that the Inductoslag process could melt without difficulty highly oxidized Zircaloy cladding.

Table 4 provides chemical analyses for each of the four autoclave batches of Zircaloy-4 along with the respective ingot analyses. Table 5 shows chemical analyses of the CaF_2 slag before and after melting. Comparison of these tables shows that carbon is slightly reduced in the ingot while the carbon content of the slag increases. Oxygen increases slightly in the ingot with a decrease of oxygen in the slag. Silicon increases slightly in the ingot; however, its presence in the slag has not been determined. The reason for the high boron content in both melt stock and ingot is being investigated.

The Brinell hardness readings of the high-oxygen material are typical of oxygen contaminated zirconium. Figure 20 portrays typical etched sections of the ingot, and typical as-cast microstructures. The grain size is smaller than that of the low-oxygen ingot (USBM 431).

While the melting of the 9000-ppm O_2 material described above was suc-

cessful, attempts to melt material with oxygen in the 20,000-to 40,000-ppm range failed. Inductoslag melting of this material resulted in excessive spatter as the cold charge entered the molten pool. Melting was stopped when it was evident that the amount of spattering was beyond any reasonable melting practice. Spattering is probably due to the hydrogen content of the highly oxidized material. It is not expected that irradiated Zircaloy will have oxygen levels this high; however, premelt heat treatment of these hulls would reduce the hydrogen level to a range compatible with melting.

In this period Inductoslag melting experiments have been performed on combinations of oxidized zirconium with oxidized 304 stainless steel and oxidized Inconel. Heat Number USBM 534 consisted of three separate compositions:

- Zircaloy-4 with 3000 to 8000 ppm O_2 with 4 wt% oxidized Inconel 718 and Al_2O_3 pellets
- Zircaloy-4 with 3000 to 8000 ppm O_2 with 10 wt% oxidized Inconel 718 and Al_2O_3 pellets
- Zircaloy-4 with 3000 to 8000 ppm O_2 with 10 wt% oxidized 304 stainless steel and 5 wt% oxidized Inconel 718 and Al_2O_3 pellets.

These compositions were chosen to represent: 1) waste streams where the stainless end fittings are removed, leaving only the Zircaloy and Inconel for melting, and 2) the ternary mix previously melted with clean surfaces. The Zircaloy was oxidized by autoclaving and the stainless steel and Inconel by hot rolling. The Al_2O_3 additions represented insulator material in the waste stream.

FIGURE 20. Oxidized Zircaloy-4 Melted by Inductoslag
USBM Number 529

The melting of Zircaloy-4 with 10 and 4 wt% Inconel and Zircaloy-4-10 wt% stainless steel-5 wt% Inconel 718 using oxidized materials showed no difference in melting characteristics over those of clean materials. Preliminary ingot sectioning reveals that these alloys contain consider-

able amounts of unmelted material including unreacted Al_2O_3 pellets "potted" in a matrix of low-melting eutectic. The ingot with these three compositions is shown in Figure 21. Its density and integrity make it suitable for direct storage.

FIGURE 21. Oxidized Zircaloy-Stainless Steel-Inconel Alloys Melted by Inductoslag USBM Number 534

The problem with these alloys, clean or oxidized, is the relatively slow melting rate. No attempt was made to achieve complete melting for these alloys. Feeding was as rapid as possible without causing bridging. Observations during melting revealed that the low-melting eutectic increased the time required for dissolution and that excessive semimolten slag on top of the pool restricted contact of the melt to the unmelted material. Examination of the ingot after melting showed no visible slag on the ingot side wall. However, no electrical contact between the copper crucible and the melt was observed. Nonhomogeneous feeding of the Zircaloy and Inconel to the relatively small molten pool results in considerable melting point fluctuation of the pool.

Table 6 summarizes the melting rates of the alloys melted for this program. Additional development is needed to increase the melting rate of these Zircaloy-Inconel and Zircaloy-stainless steel-Inconel alloys. It is believed that this can be accomplished by one or more of the following:

- adjustment of slag composition
- independent slag feeder
- reduction of slag addition
- independent stainless steel-Inconel feeder
- bridge removing device
- increase of power input.

Preliminary results using a (CaF_2 , MgF_2) slag composition richer in CaF_2 have indicated that improved melting rates for these alloys may be possible.

TABLE 6. Summary of Melting Rates

Composition wt%	USBM Heat Number	Approximate Melting Rate	
		kg/min	lb/hr
Low O ₂ pure Zircaloy-4	431	0.3	38
High O ₂ pure Zircaloy-4	529	0.34	45
85 wt% Zircaloy-4 10 wt% SS; 5 wt% Inconel 718	443	0.044	4.7
304 stainless	460	0.46	60.7
67 wt% 304 SS - 33% Inconel 718	460	0.25	33
Inconel 718	460	0.26	34.3
High O ₂ Zircaloy-4 4 wt% oxidized Inconel 718	534	0.078	10.3
High O ₂ Zircaloy-4 10 wt% oxidized Inconel 718	534	0.079	10.4
High O ₂ Zircaloy-4 10 wt% oxidized 304 SS 5 wt% oxidized Inconel 718	534	0.128	16.9

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3.0 TRITIUM SEPARATION AND FIXATION

L. L. Burger - Project Manager

The goals of the program are to investigate isotopic separation as a means of concentrating tritiated aqueous wastes, to consider alternative procedures such as increased recycle of water, and to examine methods of fixing tritiated water for permanent storage.

SUMMARY

Zirconium hydride samples having an atom ratio, H/Zr, of 1.5 to 1.9 and showing good physical integrity have been obtained with reaction times of about 30 min. Zircaloy tubing (cladding scraps) cracked less than 1/4-in. rod.

Laboratory work has terminated, and short summary reports are being written on fixation by metal hydrides and on organic polymer fixation.

TECHNICAL PROGRESS

Metal Hydride Fixation - R. D. Scheele

A study of the factors affecting the hydriding rate was continued. Most of these were identified in the previous quarterly report, BNWL-1978.⁽¹⁾ Included among these are temperature, H₂ pressure, and degassing prior to hydridization. For the present studies, the purity of the hydrogen used was emphasized.

Metal preparation was also studied. Samples cleaned in reagent grade acetone and abraded with silicon carbide paper showed no improvement over untreated metal samples. The effects of degassing the metal prior to hydriding were found to be negligible

as were effects from annealing after a preliminary hydriding treatment. The form of the Zircaloy proved to have a significant effect on the rate of reaction and the degree of sample cracking. The 0.50-in. tubing was found to react more rapidly and to stress-relieve itself better than 0.25-in. rod.

The effects of pressure and temperature were studied by hydride preparations isothermally at 400, 600, and 900°C at pressures of 200 or 760 torr H₂ pressure. Hydrides were also prepared by hydrogen addition at 760 torr as the temperature was increased from room temperature to 600°C, also as temperature was increased from 600°C. Zirconium hydrides with high H:Zr ratios (1.5 to 1.9) were obtained for isothermal preparations at 600 and 900°C at 760 torr H₂. To obtain the high ratio for the latter temperature a cooling cycle was included. Hydrogen addition as the temperature was increased from 600°C at 760 torr H₂ also produced comparable hydride composition. L. D. LaGrange et al.⁽²⁾ found the reaction to proceed very slowly below 550°C. Workers at Brookhaven National

Laboratory⁽³⁾ observed slow reaction rates at 100 to 200 torr H_2 pressures. We found similar behavior at these conditions.

Figures 22 and 23 are pictures of zirconium hydride samples prepared at various conditions. Each sample, ex-

cept for No. 36, which was 99.99% pure Zr, was prepared from Zircaloy rod or tubing. Sample No. 20 was prepared at 600°C and 200 torr H_2 ; 22, 31, 32, and 36 were prepared at 600°C and 760 torr H_2 . No. 23 was

FIGURE 22. Representative Zirconium-Hydride Samples for Various Conditions of Preparations

FIGURE 23. Zirconium-Hydride Sample Showing Intricate Cracking and Unhydrided Zircaloy-2 Sample

prepared at 400°C and 760 torr H₂, No. 26 at 900°C and 760 torr H₂, and No. 34 at 760 torr H₂ under conditions of increasing temperature starting at 600°C. Figure 23 provides a closer view of No. 31 and of an unhydrided piece of Zircaloy tubing.

The degree of hydride cracking is related to the temperature of preparation. The higher the temperature of isothermal preparation, the less cracking. An improvement in sample integrity is found for the hydride preparation as the temperature is increased from 600°C at 760 torr. For these conditions no cracking could be detected under a magnification of 215 X.

The time needed to prepare a hydride with high H:Zr ratio is equal for the isothermal preparation at 600°C and for the procedure employing an increasing temperature from 600°C, both at 760 torr H₂ pressure. The rate of reaction is more rapid at higher temperatures, but the maximum obtainable H:Zr ratio is lower (see Figure 24)⁽⁴⁾ due to increased hydrogen vapor pressure. A typical hydride preparation takes about 30 min.

A short topical report is being written summarizing the hydriding studies.

Organic Polymers - J. A. Franz

Supplement I to the report Polymeric Media for Tritium Fixation has been prepared.

Two procedures are described for the fixation of tritium as TH or THO. Fixation of THO in a polyureylene-polyurethane copolymer is achieved by stirring an acetone solution contain-

FIGURE 24. Pressure-Composition Isotherms (Composite Data)⁽⁴⁾

ing 5 equivalents tolylene-2,4-diisocyanate, 4 equivalents of water, and 1 equivalent of ethylene glycol for a few minutes, evaporating the solvent, and air-drying the polymer overnight. The resulting powder is cast into solid cylindrical forms by heating at 220°C under 22,000 psi for 20 min. The resulting material exhibits negligible loss of tritium in leach tests, despite rapid NH exchange occurring with unmolded urea polymers. Fixation of TH is achieved by catalytic tritiation under 1 atmosphere pressure, 20°C, of polystyrene in ethanol/ethylacetate over a rhodium-on-alumina catalyst. The resulting polystyrene-polyvinyl-cyclohexane exhibits no loss of tritium on leaching with water. Both procedures are simple, quantitative, and rapid and are carried out under mild conditions.

Material properties of the polymers (thermal stability, resistance to tritium leaching) are excellent. But costs are high (up to \$5/lb water fixed), thus limiting the procedures

to highly concentrated waste. More work in the area of organic media is needed, specifically for efficient use of acetylene, the least expensive organic monomer.

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4.0 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 disposition of retired (excessed) Hanford facilities; to formulate plans and specifications for a project to accomplish permanent disposition of a specific Hanford facility; and to identify required facility disposition R&D.

SUMMARY

"Worst case" calculations have been made of the potential radiological dose to individuals and to the regional populace in event of the release of radioactivity from the 216-S1 & 2 crib and from the 107-DR retention basin.

Preliminary estimates indicate that decommissioning the 100-D/DR reactor effluent retention basin system could cost in the range of \$2 to 5 million if the system is placed in protective storage, \$8 to 10 million if it is dismantled and all radioactivity transferred to burial or storage in the 200 Areas.

Tests have been made of a modified rock splitter for removal of contaminated surface layers of concrete.

TECHNICAL PROGRESS

TASK 1. Establish Methods, Costs and Priorities for D&D Hanford -

K. M. Harmon

Radiological Studies - D. A. Waite

"Worst case" radiological dose calculations have been made for a 200 West Area crib (216-S1 & 2) and a 100 Area retention basin (107-DR) in their present condition. These cal-

culations are part of the effort to develop an "as is" risk index system for use in prioritizing the D&D of retired Hanford facilities. Using available radionuclide inventory data (Table 7), potential doses were calculated, assuming different release or exposure mechanisms, for:

- an individual placed at the nearest maximum exposure location ("maximum individual")
- the residents within a 50-mile radius of the site
- an occupational or 8-hr residence dose.

A 1-year dose and a 50-year dose commitment were calculated for both the maximum individual and the regional population.

The release mechanisms studied included:

- Airborne release from both sites over periods of 1 day and 1 year. Depending on whether the facility is subsurface or above ground, it is possible in short intervals to release different percents of the total inventory. Percents determined and used in the dose calculations were 60% for surface contamination such as in 107-DR; 0.6% for

TABLE 7. Radionuclide Inventories (Estimated) Used as Basis for "Worst Case" Radiological Dose Calculations

Estimated Radio- nuclide Inventory, Ci	107-DR (Retention Basin - 100 D Area)	216-S1 & 2 (Crib - 200 W Area)
⁶⁰ Co	11.4	276
⁹⁰ Sr	0.747	1932
⁹⁰ Y	--	1932
¹²⁵ Sb	--	46
¹³⁷ Cs	7.51	1288
¹⁵² Eu	38.8	--
¹⁵⁴ Eu	10.5	--
¹⁵⁵ Eu	0.532	--
²³⁸ Pu	0.00489	--
²³⁹ Pu	} 0.127	70
²⁴⁰ Pu		14
²⁴¹ Pu	--	250
²⁴¹ Am	--	0.24

subsurface such as in an underground crib; and 0.1 to 1% for an aboveground structure. The same total release was assumed for the two cases, release for 1 day and release for 1 year. Exposure from inhalation of dust and from submer-
sion in the cloud of contamination were both assumed.

- Aquatic release from the 107-DR basin, via the Columbia River, over a 1-year period and during a 24-hr catastrophic flood. The predominant sources of exposure are eating contaminated fish and drinking contaminated water.

The results of the calculations are summarized in Tables 8 and 9.

TABLE 8. 107-DR Retention Basin "Worst Case" Doses

Liquid Release			
	1-Year Dose	50-Year Dose Commitment	Critical Organ
<u>24-Hr Release</u>			
Maximum Individual	4×10^{-1} mrem	8×10^{-1} mrem	Bone
<u>1-Year Release</u>			
Maximum Individual	6×10^{-2} mrem	10^{-1} mrem	Bone
Population	4×10^{-1} person-rem	2 person-rem	Bone
Airborne Release			
<u>1-Year Release</u>			
Maximum Individual	34 mrem	380 mrem	Lung, Bone
Population	0.7 person-rem		Bone
<u>8-Hr Exposure</u>	160 mrem		

TABLE 9. Crib 216-S1 & 2 "Worst Case" Doses, Airborne Release

	<u>1-Year Dose</u>	<u>50-Year Dose Commitment</u>	<u>Critical Organ</u>
<u>24-Hr Release</u>			
Maximum Individual	1.7×10^2 mrem	3.9×10^3 mrem	Bone
	6.6×10^1 mrem	8.4×10^2 mrem	Total Body
	1.0×10^2 mrem		GI-LLI
		5.2×10^1	Lung
Population	0.01 person-rem	0.51 person-rem	Total Body
<u>1-Year Release</u>			
Maximum Individual	1.6×10^1 mrem	1.0×10^3 mrem	Bone
	5.3 mrem	9.6×10^1 mrem	Total Body
	8.4 mrem		GI-LLI
		9.6×10^1 mrem	Liver
	2.5×10^1 mrem		Lung
		7.2×10^1 mrem	Kidney
<u>8-Hr Exposure</u>	160 mrem		

Cost Estimating - J. C. King

Cost estimates were made by Vitro Engineering for two different alternative D&D modes for the 100 D/DR Area reactor effluent system: Protective Storage and Dismantling. The system, shown in Figure 25, includes the following major components:

- 107-D Retention Basin: Concrete lined, 230 x 467 x 20 ft deep
- 107-DR Retention Basin: Concrete lined, 230 x 600 x 20 ft deep
- 116-D-5 Outfall Structure
- 116-DR-5 Outfall Structure

Carbon Steel Pipe

- 15 in. diam 140 linear ft
- 21 in. diam 70 linear ft
- 24 in. diam 1400 linear ft
- 42 in. diam 3720 linear ft
- 60 in. diam 9900 linear ft

Concrete Pipe

- 30 in. diam 300 linear ft
- 42 in. diam 400 linear ft
- 60 in. diam 2340 linear ft

Both basins contain contaminated soil, and the dirt surrounding the basins is contaminated.

The cost estimates (in 1976 dollars) are summarized in Table 10.

FIGURE 25. Effluent System, D and DR Reactors

TABLE 10. Estimates of Decontamination and Decommissioning Costs, 107-D/DR Retention Basin System, in Thousands of 1976 Dollars

<u>Facility</u>	<u>Protective Storage(a)</u>	<u>Dismantling(b)</u>
107-D Retention Basin	557	1,380
107-DR Retention Basin	614	1,900
116-D-5 Outfall Structure and Pipelines into River	15	596
116-DR-5 Outfall Structure and Pipelines into River	15	582
Common Use Temporary Facilities at 100-D	—	663
Common Use Temporary Facilities at 200-W	—	370
107-D/DR Effluent Piping	<u>490</u>	<u>2,620</u>
Total	1,691	8,111

- a. Protective Storage: Knock down top 6 ft of walls in retention basins, install a soft "biobarrier"--3 ft layer of cobbles, 1 ft layer of sand, 2 ft layer of sandy loam, and plant to cheatgrass; remove sludge and scale from pipelines (remove to burial in 200 Areas), install concrete slabs over expansion, gate and junction boxes, seal openings in pipelines with concrete; demolish spillway; fill outfall structure with soil and cap it with concrete.
- b. Dismantling: Dismantle basins, pipelines, outfall structures and spillways. Transport all contaminated materials and soil to 200 Area for burial.

TASK 3. Exploratory R&D -
K. M. Harmon

Concrete Decontamination - Use of Modified Rock Splitter - C. H. Allen

Investigation of the Darda Splitter® revealed that it is intended for splitting thick rocks, concrete walls, etc., but is not usable for the removal of surface layers of concrete. A new experimental tool, based on the rock-splitter principle and termed a "concrete scarfer," was designed and tested.

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NJ 07407.

In this method, a pattern of 1-in.-diam holes is drilled into the concrete and the scarfing tool is inserted into each hole in turn. The tool is a split expandable collet. The collet is expanded in the drilled hole by a tapered mandrel, which is drawn through a hole in the center of the collet. The mandrel forces a sharp edge out against the concrete, causing the concrete to fracture from the surface in a roughly circular pattern around the drilled hole.

Limited tests to prove the feasibility of the concept were run with an experimental tool operated by tightening a nut on a screw thread. Using

4.6

holes drilled in approximately 32-year-old concrete, three holes were scarfed out with the tool.

In these tests, the sharp edge of the tool was inserted in the concrete at three different depths, 0.60, 0.69 and 0.94 in. The area of concrete removed was about 6.2, 13.6 and 14.7 in.² respectively. The concrete shears away from the surface at about 18 to 20°. At this angle the amount of concrete removed in the second test (13.6 in.² surface area) was about 6.1 in.³. With this amount approximately 60 in.³/ft² of surface area would be removed.

One problem noted was that in only one of the three tests did the concrete break away concentrically with the drilled hole. In the other two tests the drilled hole was near the

edge of the scarfed concrete. In regular use, if this occurred, additional holes would have to be drilled and scarfed to allow for the eccentric pattern.

In follow-up tests, the scarfing tool was converted to use air pressure for operation. A 2.5-in. bore cylinder was estimated to be adequate at 100 psia air pressure but when tested did not provide enough force. To correct this problem, a 4-in. bore cylinder has been ordered, the lightest-weight cylinder that could be found. Unfortunately the tool will be considerably heavier with this large cylinder and may be too heavy for manual operation. Further testing after the tool is again modified will help determine if the additional weight is a significant problem.

5.0 TRANSPORTATION SAFETY STUDIES

L. D. Williams - 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

Comments from the internal review of the plutonium shipment by rail risk assessment were incorporated in the draft, which is now in final editing and typing.

A study was started of leakage from radioactive material shipping packages at the bottom of the sea. The loss of typical shipping packages for LWR spent fuel and plutonium dioxide is being investigated, focusing on leakage from an undamaged container and from a fire-damaged container.

An assessment of the risk in shipment of gasoline by truck has been initiated. A fault tree has been drawn of the ways that gasoline can escape during transport, and other system description information has been gathered.

Work on the assessment of the risk in transporting UF_6 by truck and rail continued. The system description has been updated, and the accident environment is being determined. A survey questionnaire has been circulated to UF_6 receivers.

The assessment of the risk involved in the truck portion of the shipment of plutonium by air has been analyzed. The fault tree has been extensively modified to more fully use available accident environment

information. A draft of the preliminary report is being prepared.

TECHNICAL PROGRESS

Risk Assessment of Plutonium Transport by Rail

An internal review, made on a preliminary draft of the risk assessment report, generated several suggestions for improving the analysis. The suggestions have been incorporated, and the report is currently in final editing and typing.

Study of Leakage from Shipping Packages at the Bottom of the Sea

In ocean shipment of radioactive materials it is conceivable that accidents could occur which would result in loss of the materials into the sea. The effects of two postulated loss situations for typical shipping packages for LWR spent fuel or plutonium dioxide are being analyzed. These are loss to the sea of an undamaged package and that of a fire-damaged package. The shipping package for LWR spent fuel is assumed to be a large 100-ton mechanically cooled cask. The plutonium dioxide package is a 15-gal specification 6M container with a standard 2R inner containment vessel. In each loss situation the

effects are dependent on the ocean depth where the loss occurs. Two ocean locations are considered, the continental shelf (200-m depth) and the deep ocean (5000-m depth).

The accident, containment failure mechanisms, release rates, and the effects of releases are being examined. The deep ocean failure mechanisms for the spent fuel cask and plutonium package have been identified, and work is progressing on shallow water failure mechanisms and preloss fire damage effects. Volatile radioactive isotopes in the spent fuel have been quantified, and long-term leach rates are being evaluated. A model for determining population doses has been identified.

The failure mode for the deep ocean is hydrostatic crushing. In the spent fuel shipping package, hydrostatic pressure first fails the cask seals. At the depths of the deep ocean floor the pressure is expected to be great enough to also collapse the fuel rod cladding. The outer drum of the 6M package is assumed to leak. This allows hydrostatic pressure to be applied directly to the inner vessel. In shallower water hydrostatic pressure plays a lesser role, and corrosion predominates. For those events with preloss fire damage, the effect of this damage on the failure mode is considered.

Releases after containment failure take two forms. Volatile radioactive material is immediately available for release. Leaching by sea water accounts for long-term releases. Both of these mechanisms are being considered. Using the postulated release

rates for each loss event, the total population dose to man via the marine food chain will be calculated.

Risk Assessment for Low-Level Beta-Gamma Waste Shipment by Truck

Work on this task was suspended this quarter to concentrate efforts on higher priority tasks.

Risk Assessment of Gasoline Shipment by Truck

A preliminary gasoline release fault tree has been developed. The tree is being used to identify those events which could contribute to accidental release of gasoline from a tank truck. Data on the likelihood of occurrence for the identified events (e.g., likelihood of an accident involving impact) are being developed. At this stage of the analysis three possible effects of gasoline release are being considered: fire, explosion, creation of toxic atmosphere. Consequence models are being developed.

Risk Assessment of UF₆ Shipment by Rail and Truck

Efforts are continuing in evaluation of truck and rail accident environments involving large UF₆ cylinders. Sandia's accident analysis is being used as a starting point, with special emphasis placed on the impact and fire portions of the accident environment.

The UF₆ system description has been updated by including more complete information on the amounts of UF₆ exported to and imported from foreign countries. However, only

5.3

transportation of this material within the United States will be considered in the analysis.

The survey questionnaire on UF_6 shipping conditions was circulated for internal comment prior to mailing to the UF_6 receivers. Several suggestions directed at improving the clarity of the survey questions were received, and the questionnaire was revised accordingly. The revised questionnaire was distributed, and analysis of the release sequence awaits the return of the completed survey. A copy of the finalized questionnaire, as sent to the UF_6 receivers, is included.

Risk Assessment of Plutonium Oxide Shipment by Air

Shipment of plutonium by air will involve multimode transport. It is assumed that the plutonium will be shipped by truck to and from airports having air freight service. The risk in this truck transport is a part of the overall risk in shipment by air. Using the risk analysis code HEAD, an initial calculation has been made of

the risk in trucking PuO_2 from the reprocessing plant to the originating airport and from the destination airport to the fuel fabrication plant.

Review of the initial fault tree constructed for air shipment led to the conclusion that more detail would be desirable. Therefore, a new tree has been described in Sandia's aircraft impact accident environment analysis. These cover mid-air breakup accidents, container free fall accidents, accidents in which airplanes impact into the ground or mountainside, and accidents occurring during takeoff and landing. This revised tree uses the accident environment data more completely, resulting in greater detail than in the previous tree.

The occurrence frequencies for the basic events in the fault tree were reevaluated and adjusted to reflect recently obtained information. The results of recent packaging tests at Sandia are being used to normalize calculated failure thresholds for the 6M container used in PuO_2 shipment. A preliminary draft report is being prepared.

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	_____	_____	_____	_____
1974	_____	_____	_____	_____
1973	_____	_____	_____	_____
1972	_____	_____	_____	_____
1971	_____	_____	_____	_____
B. GENERAL CONDITION OF SHIPMENT (1971-1975)				
1. DO YOU FORMALLY EXAMINE AND RECORD THE CON- DITION OF RECEIVED CYLINDERS?	YES		NO	
2. IS THE H/U RATIO VERIFIED UPON RECEIPT?	YES		NO	
3. NUMBER OF TIMES H/U RATIO EXCEEDED SPECIFIED LIMITS.	_____	_____	_____	_____
4. NUMBER OF TIMES ENRICHMENT EXCEEDED SPECIFIED LIMITS.	_____	_____	_____	_____
5. NUMBER OF CYLINDERS RECEIVED OVERFILLED.	_____	_____	_____	_____
6. NUMBER OF CYLINDERS RECEIVED OVERPRESSURIZED.	_____	_____	_____	_____
7. NUMBER RECEIVED WITH CONTAMINATION (UF ₆ , UO ₂ F ₂ , ETC.) OUTSIDE CYLINDER.	_____	_____	_____	_____
C. CYLINDER CONDITION (1971-1975)				
1. NUMBER OF CYLINDERS REMOVED FROM SERVICE DUE TO DETERIORATION.	_____	_____	_____	_____
2. NUMBER OF CYLINDERS WITH BADLY CORRODED SHELLS.	_____	_____	_____	_____
3. NUMBER OF CYLINDERS RECEIVED WITHOUT REQUIRED PROTECTIVE VALVE COVER.	_____	_____	_____	_____
4. NUMBER OF CYLINDERS RECEIVED WITH WELDS DAMAGED (EXAMPLE, VESSEL FRESHLY DENTED IN AREA OF WELD RESULTING IN POSSIBLE PARTIAL WELD FAILURE, ETC.)	_____	_____	_____	_____
D. VALVE CONDITION (1971-1975)				
1. NUMBER OF CYLINDERS REQUIRING VALVE REPLACEMENT.	_____	_____	_____	_____
2. NUMBER OF CYLINDERS WITH VALVE AND/OR VESSEL THREADS PARTIALLY STRIPPED.	_____	_____	_____	_____
3. NUMBER OF CYLINDERS WITH VALVE CROSSTHREADED.	_____	_____	_____	_____
4. NUMBER OF CYLINDERS WITH VALVE NOT COMPLETELY CLOSED.	_____	_____	_____	_____
5. NUMBER OF CYLINDERS WITH VALVE LOOSE OR INSUFFICIENT THREADS ENGAGED.	_____	_____	_____	_____
6. NUMBER OF CYLINDERS RECEIVED WITH VALVE DAMAGED (EXAMPLE, VALVE BENT AND FRESH METAL EXPOSED AT COUPLING, ETC.)	_____	_____	_____	_____

⁽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, BATTLE 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	_____	_____	_____	_____
2. NUMBER OF CYLINDERS WITH PLUG LOOSE OR WITH INSUFFICIENT THREADS ENGAGED	_____	_____	_____	_____
3. NUMBER OF VESSELS WITH PLUG AND OR VESSEL THREADS PARTIALLY STRIPPED	_____	_____	_____	_____
4. NUMBER OF VESSELS WITH PLUG CROSSTHREADED	_____	_____	_____	_____
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.)	_____	_____	_____	_____
F. OVERPACK CONDITIONS (1971-1975)				
1. NUMBER OF CYLINDERS RECEIVED IN OVERPACK	_____	_____	_____	_____
2. NUMBER OF CYLINDERS RECEIVED WITHOUT OVERPACK WHEN REQUIRED	_____	_____	_____	_____
3. NUMBER OF CYLINDERS RECEIVED WITH GROSS DAMAGE TO OVERPACK	_____	_____	_____	_____
4. NUMBER OF CYLINDERS RECEIVED WITHOUT SEALING GASKET IN OVERPACK	_____	_____	_____	_____
5. NUMBER RECEIVED WITH DETERIORATED GASKET	_____	_____	_____	_____
6. NUMBER RECEIVED WITH CLOSURE BOLTS LOOSE	_____	_____	_____	_____
7. NUMBER WITH MISSING CLOSURE BOLTS:				
a. ALL CLOSURE BOLTS MISSING	_____	_____	_____	_____
b. SOME CLOSURE BOLTS MISSING	_____	_____	_____	_____
8. NUMBER RECEIVED WITH DAMAGED OVERPACK MATING SURFACE	_____	_____	_____	_____
G. TRANSPORTATION MODE				
APPROXIMATE PERCENTAGE OF CYLINDERS RECEIVED BY:				
1. RAIL	_____	_____	_____	_____
2. TRUCK	_____	_____	_____	_____
3. PIGGY-BACK (TRUCK ON RAIL CAR)	_____	_____	_____	_____
4. OTHER (PLEASE SPECIFY)	_____	_____	_____	_____
	=====	=====	=====	=====
	TOTAL	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 _____ NO _____				

COMMENTS _____

6.0 TRANSPORT PROBLEMS, 1976-2000

L. D. Williams - 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

Characteristics of current nuclear fuel cycle materials transport system were identified. Shipment needs through the Year 2000 were projected for a likely nuclear industry growth scenario.

A workshop was held to identify possible trends, events and changes outside the nuclear industry in the rest of the century which could affect nuclear fuel cycle materials transportation.

TECHNICAL PROGRESS

The initial effort in this project was establishment of the characteristics of current nuclear fuel cycle shipments and projection of shipment needs through the Year 2000. The following information was collected for each step of the fuel cycle, from mining through waste disposal: fuel cycle sites, material form and general characteristics, transportation regulations, annual shipping requirements, transport modes, special equipment and personnel requirements, quantity per shipment, and shipping package characteristics. Shipment needs through the Year 2000 using a likely nuclear industry growth scenario were projected. An internal report on this

portion of the study was prepared. Table 11 summarizes some of the more pertinent shipment system characteristics.

Identification of potential problems in five broad, interrelated areas is being sought in this project. These areas are: safety, cost, logistics, security, and public acceptance. For this study, we have categorized potential problems that could develop in this century in transport of fuel cycle materials as resulting from: 1) characteristics, trends, and changes of the nuclear industry and 2) trends, events, and changes outside the nuclear industry.

To obtain information in the latter category, a workshop on "Nuclear Materials Transportation, 1976-2000" was held March 30 and 31. The 14 workshop participants included: representatives of the trucking and railroad industries, staff from the Senate Commerce Committee, a security consultant, a transportation planning consultant, staff from Battelle-Columbus, and PNL staff members assigned to this project. Prior to the workshop, participants were given a copy of the internal report on nuclear fuel cycle transportation system requirements to aid in their preparation. The goal of the

Fuel Cycle Step	Material Transported	Container	Quantity/Container	Transport Mode	Quantity/Shipment	Shipments/Year ^(a)			Destination	Estimated Ave. Shipping Distance (mi)	Total Shipping Miles/Yr ^(b)			Special Equipment Personnel Requirements
						1976	1990	2000			1976	1990	2000	
Milling	U ₃ O ₈	55 Gallon Drum	600 lbs	Truck Rail	55 Drums 110 Drums	550 275	4455 2228	7686 3843	U ₃ O ₈ Conversion	1500	0.42 0.82	4.5 8.8	7.7-15.4	None
UF ₆ Conversion	Natural UF ₆	10 Ton Cylinder	20,000 lbs	Truck	2 Cylinders	834	6791	11750	Enrichment Plant	500	0.6-2.4	4.8-19.4	8.3-33.6	Cylinders, Cradles and Tie-Downs
		14 Ton Cylinder	28,000 lbs	Rail	4 Cylinders	417	3396	5875						
				Truck Rail	1 Cylinder 4 Cylinders	1192 297	9702 2425	16790 4197						
Enrichment	Enriched UF ₆	2-1/2 Ton Cylinder with Overpack	5,000 lbs	Truck	5 Cylinders	174	1480	2507	Fuel Fabricator	750	0.08-0.33	0.7-2.8	1.2-4.7	Cylinders, Overpack, Cradles and Tie-Downs
		10 Ton Cylinder with Overpack	20,000 lbs	Rail	16 Cylinders	55	463	784						
				Truck	1 Cylinder	217	1850	3134						
				Rail	4 Cylinders	55	463	784						
	Depleted UF ₆ ^(c)	10 Ton Cylinder	20,000 lbs	Truck	2 Cylinders	0	1	76	Fuel Fabricator	750	0	0.001	0.04-0.16	Cylinders and Tie-Downs
		14 Ton Cylinder	28,000 lbs	Rail	4 Cylinders	0	0	28						
Fuel Fabrication	Fresh Fuel	Protective Overpack	1 PWR or 2 BWR Fuel Assemblies	Truck	6 Containers	500	2519	5310	Reactor	1000	1.0	5.0	10.6	Overpack, Tie-Downs
	Waste	Mostly Drums and Plywood Boxes. Some in protective overpacks	7.35 ft ³	Truck Rail	64 Drums 140 Drums	30 14	46 22	83 38	Federal Repository	1500	0.01-0.03	0.03-0.07	0.06-0.13	Overpack, Tie-Downs
Power Production	Spent Fuel	Shielded, Cooled Cask	3 PWR Assemblies or 7 BWR Assemblies or 7 BWR Assemblies or 18 BWR Assemblies	Truck Rail	1 Cask 1 Cask	20 11	1460 2100	2397 5157	Reprocessing Plant	1000	0.04 0.02	2.9 4.2	4.8 10.3	Cask, Dedicated Trailer or Flatcar
	Waste	Cylindrical Drums in Shielded Casks	150 ft ³	Truck	1 Cask	2221	7000	14490	Burial Ground	500	2.2	7.0	14.5	Casks and Tie-Downs
Reprocessing	Plutonium	Shielded, Cooled Container	32 Kg	Safeguard-Equipped Truck	7 Containers	0	319	978	Fuel Fabricator	750	0	0.48	1.47	Armored Tractor, Armed Escort, Communic. Equip.
	Solidified High Level Waste	Shielded, Cooled Cask	75 ft ³	Rail	1 Cask	0	58	215	Federal Repository	2000	0	0.2	0.86	Cask, Dedicated Railcar
	Cladding Waste	Shielded, Cooled Cask	95.2 ft ³	Rail	1 Cask	0	168	379	Treatment/Storage	2000	0	.67	2.3	Cask, Dedicated Railcar
	Noble Gases	50 x Cylinder Pressurized to 2200 psi	1 Cylinder in Shielded Cask	Rail	6 Casks	0	73	172	Federal Repository	2000	0	0.3	0.68	Cask, Dedicated Railcar
	Iodine	Mostly Drums in Shielded Overpack	7.35 ft ³	Truck	448 Kg	0	6	15	Burial Ground	500	0	0.003	0.008	Overpack and Tie-Downs
	Tritium	0.16 ft ³ Container in Steel Drum Overpack	0.16 ft ³	Truck	64 Containers	0	293	690	Burial Ground	500	0	0.29	0.69	Overpack and Tie-Downs
	Other TRU Waste	Drums and Boxes. Some in Overpack	7.35 ft ³	Rail Truck	140 Drums 64 Drums	0 0	170 372	364 795	Federal Repository	2000	0	0.34-0.74	0.72-1.59	Overpack and Tie-Downs
	Other Non-TRU Waste	Drums and Boxes. Some in Overpack.	7.35 ft ³	Rail Truck	140 Drums 64 Drums	0 0	75 162	159 347	Burial Ground	500	0	0.04-0.08	0.08-0.17	Overpack and Tie-Downs
	Slightly Enriched UF ₆	10 Ton Cylinder	20,000 lbs	Truck	2 Cylinders	0	611	1286	Enrichment	500	0	0.2-0.88	0.46-1.9	Cylinders and Tie-Downs
		14 Ton Cylinder	28,000 lbs	Rail	4 Cylinders	0	306	643						
				Truck Rail	1 Cylinder 4 Cylinders	0 0	885 221	1862 446						
Miscellaneous	Waste ^(d)	Mostly Drums and Boxes	7.35 ft ³	Truck Rail	64 Drums 140 Drums	600 275	3000 1360	6000 2720	Burial Ground	500	.13-.3	.68-1.5	1.4-3.0	None

(a) Domestic shipments only. Import and export quantities not included. Except for entry on spent fuel, where more than one shipping method is possible, the entry in this column assumes the entire annual quantity is shipped by that method.

(b) Quantities in millions. Includes return shipments for casks, overpacks and other non-throwaway containers. Maximum and minimum distances are given where two or more shipping methods can be used and the mixture of modes has not been determined.

(c) This represents the amount estimated to be required for LMFBR fuel. Depleted uranium rather than natural U₃O₈ or slightly enriched uranium is assumed to be used in LMFBR fuel for the purposes of this table.

(d) Estimated waste from remainder of fuel cycle steps. Most of this waste is expected to be relatively low level.

TABLE 11. Information on Projected Shipments of Radioactive Fuel Cycle Materials in the U.S.

6.3

workshop was to identify possible trends, events, and changes in the following six areas that could affect nuclear fuel cycle transportation:

- institutional, environmental and societal
- economic
- technological
- management and operations
- legal and regulatory
- other.

Items in each area were identified and preliminary rankings of significance and likelihood made.

These identified trends, events, and changes are currently undergoing further study to evaluate whether they could result in problems in nuclear fuel cycle materials transportation. These results will be combined with information on trends and changes in the nuclear industry to obtain ranked lists of potential problems for several growth scenarios that could occur in the remainder of the century.

7.0 SAFETY AND ECONOMIC STUDY OF SPECIAL TRAINS

L. D. Williams - 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

Train accident data for the years 1972 and 1973 have been analyzed for information on frequency, severity, cause, conditions and other relevant variables. A total of 9890 freight train accidents have been included in the analysis. Environmental characteristics (e.g., cliffs) that may affect train safety have also been surveyed and categorized. This portion of the study emphasizes those safety parameters that would differ for special and regular trains.

Route and trip characteristics for 62 previously defined origin-destination paths have been obtained from railroad companies. These include route-specific rail costs and transit times.

In the economics phase of the program, computer codes have been revised to handle detailed cost data on shipments from reactor sites to reprocessing plants, and data has been developed on anticipated cask turnaround times.

TECHNICAL PROGRESS

Federal Railroad Administration (FRA) data on train accidents have been reduced and analyzed for information on frequency, severity, cause,

conditions, and other relevant variables. The main objective was to determine the effects on train safety of the major differences between regular and special trains--speed limit of 35 mph, exclusive cargo, and passing restrictions.

Data for 1972 and 1973 show that 85% of all train accidents occur below the speed of 35 mph. For the initial analysis, dollar damage to railroad equipment was used as a measure of accident severity. The data indicated that of the accidents resulting in greater than \$75,000 damage (~6% of all accidents) about half occurred above 35 mph and half below.

To evaluate the effects of the passing restriction on special trains (one train must be stationary), all collisions in 1972-1973 and the top 6% high-damage collisions for the period were examined. Data show that 95% of all collisions occur at speeds less than 35 mph and nearly 65% of high-damage collisions occur at speeds less than 35 mph. In more than 50% of the high-damage accidents one train was stationary.

Due to physical features of the route environment, the possibility of impact at greater velocities than 35 mph is not eliminated by a 35 mph

7.2

speed limit. A line profile along a selected origin-destination link has been examined for steep grades, bridges, cliffs, and other features that could lead to impact velocities greater than 35 mph. A number have been found to provide the potential for a high-speed impact.

Collection of information on route and trip characteristics for 62 origin-destination links that would most likely carry spent fuel has been completed. Information obtained from railroad companies include mileages, estimates of travel time, regular and

special train tariffs, train densities, distances between passing rails, and other pertinent data.

With completion of route characterization and determination of freight tariffs, the economic phase of this study will begin full operation. In preparation for the data, computer codes to analyze costs of shipments from reactors to reprocessing sites have been revised or have been developed. Several scenarios have been generated for turnaround requirements for multiple cask shipments.

8.0 DEVELOPMENT OF HIGH-LEVEL WASTE SHIPPING CASK MODELS

L. D. Williams - Project Manager

The objective of this program is to conceptually design and to 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-gage rail car (1:87 scale) and 2) a table model of the cask alone for use in portable displays. The cask for the table model will be about 1 ft in diameter by 2 ft long and include a cutaway section showing the high-level waste canisters inside. Graphics will also be provided to further illustrate the conceptual design and to relate the characteristics of the solidified high-level waste transported.

Final system design, based on the design guidelines discussed in the previous report, is in progress. Preliminary concept drawings have been prepared and are currently being reviewed. The effort is divided into two major areas: 1) design of the cask assembly and 2) design of the tie-down and handling system.

TECHNICAL PROGRESS

Contacts were made with personnel at fuel reprocessing facilities to determine potential handling constraints on casks and high-level waste canisters that should be considered in

the design. Present uncertainties about the nature of the canisters and methods of storage and handling at both ends of the transportation step indicate need for a flexible system. The cask is therefore being designed for complete removal from the railcar for loading and unloading.

To ensure the credibility of the cask system design, a minimum amount of structural and thermal analysis is being performed on the design. A preliminary design of the cask was completed and sent to Los Alamos for analysis under impact conditions using the LASL IMPAC1 computer program. Results are expected back in late April. The analysis is to verify the choice of shielding material made in the design. The lead shielding initially chosen may have to be replaced by depleted uranium if the impact analysis does not prove lead to be structurally satisfactory.

Individuals working on the Waste Solidification Program at PNL indicate that the initial assumption that the high-level waste will be shipped in stainless steel cans 1 ft in diameter should be modified to include an over-pack of about 2-in.-thick carbon steel. This modification will change both heat transfer considerations and

8.2

the geometric configuration of the canisters in the center cavity. The system design is being modified to accommodate this change.

FUTURE PLANS

Construction of the models will begin following review of the con-

ceptual design by Larry Shappert of Oak Ridge National Laboratory. Plans are to have the models and graphics complete and available for display at the Denver Waste Management meeting on July 11-16, 1976.

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