**Nuclear Waste Management
and Transportation Quarterly
Progress Report
April Through June 1975**

September 1975

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Pacific Northwest Laboratories

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NUCLEAR WASTE MANAGEMENT AND
TRANSPORTATION QUARTERLY PROGRESS
REPORT APRIL THROUGH JUNE 1975

Compiled by

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Nuclear Waste Technology Program

September 1975

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FOREWORD

This document is one of a series of technical progress reports designed to report on all Division of Waste Management and Transportation (DWMT) funded programs at Battelle, Pacific Northwest Laboratories, with the exception of the Commercial Radioactive Waste Fixation Program. These reports are issued on a quarterly basis.

The prior report in this series is Nuclear Waste Management and Transportation Quarterly Progress Report January through March 1975, BNWL-1913.

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1.0 ADVANCED METHODS FOR MANAGEMENT AND DISPOSAL OF RADIOACTIVE WASTES

J. W. Bartlett - Project Manager

The objectives of this program are to 1) develop comprehensive plans for a program to select and implement methods for disposal of radioactive wastes and 2) extend the base for the plans and program via selected topical studies.

SUMMARY

Revisions to the draft PNL plan for R&D on disposal of high-level and transuranic wastes will be based on the recent decision to accelerate demonstration of disposal (at Carlsbad, New Mexico) and a comprehensive assessment of alternatives for the back-end of the nuclear fuel cycle.

Engineering concepts and an R&D plan for a seabed disposal system were developed at a workshop held in Seattle.

A mathematical model was developed for estimating migration of nuclides from salt formations.

The survey of public attitudes toward alternatives for waste disposal was administered nationwide to more than 400 persons. A preliminary compilation of results is complete.

A report is being prepared on waste management methods for I-129 and Tc-99. Isotopic dilution and geologic storage appear to be acceptable disposal methods.

The possibilities and consequences of loss of waste at sea in conjunction with seabed disposal have been evaluated. The assessment found no factors precluding seabed disposal.

Assessment of the consequences of long-term (up to several decades)

storage of spent fuel elements was started. A list of factors to consider was developed.

Study of disposal by dispersion in geologic formations was started. Such a method could reduce potential hazards resulting from subsequent geologic disturbances. Volumes of geologic media required were estimated and evaluation of concentration limits was begun.

A program plan development was begun to assess alternatives for waste management operations prior to disposal. The program will encompass alternatives for the back-end of the fuel cycle.

TASK 1 - DEVELOPMENT OF ADVANCED WASTE MANAGEMENT PLANS -

J. W. Bartlett

The objective for this task is to prepare plans for R&D programs on disposal of commercial radioactive wastes. The basic procedure for the work is to convene workshops composed of persons who collectively possess the spectrum of expertise needed for the plans. Results from these workshops are then used in plans prepared by PNL.

R&D Plans for Disposal of High-Level and Transuranic Wastes

Revision was started on the draft PNL plan for R&D on disposal of high-level and transuranic wastes. Changes will be based on comments on the draft and on recent events that impact our approach to disposal R&D.

Substantive changes in the thrust and content of the draft plan are not expected. Comments and further study have shown that the basic concept of the plan, i.e., investigation and winnowing of continental and seabed geologic disposal options, is sound. However, it is now possible to place the disposal R&D plan into perspective as part of a master strategy for waste management R&D. The revised plan will reflect this approach.

Strategic factors to be considered in the revision include:

- A decision was made to accelerate demonstration of disposal in the near term at a pilot facility to be located near Carlsbad, New Mexico. This action implements one of the options within the framework of the draft R&D plan.
- In addition to the Carlsbad project, which utilizes known technology, the time frame for generation of commercial wastes permits parallel investigation and evaluation of alternative disposal methods and sites which could be based on advanced technology and operable near year 2000.
- As recognized by Dr. Seaman's statements concerning revision of WASH-1539⁽¹⁾ to include comprehensive evaluation of alternatives for the back-end of the nuclear fuel cycle, pre-disposal operations may

impact disposal itself. A systematic assessment of pre-disposal waste management alternatives and their interaction with disposal operations is necessary.

- The United States has more options for disposal than many nations, but our R&D programs can be viewed as part of an international effort to explore alternatives and exchange information. As far as possible our plans should reflect this viewpoint.

Within this strategic framework, the overall objective of the R&D plans are to 1) provide as soon as possible an acceptable method of disposal for use if necessary and 2) provide, for future decision making, an agenda of alternatives--including assessments thereof--for commercial radioactive waste management practice. We estimate that these decisions will be necessary and possible during the 1990s. This schedule is consistent with that given in the draft R&D plan.

It is seen that in developing and assessing an agenda of alternatives all options are concerned with fission power by-product management. Numerous alternatives, including indefinite storage of spent fuel, use of heat and resource values in high-level waste, and partitioning, have been suggested. Some have been investigated, but there has been no systematic evaluation to assess factors such as a) exclusion of one action by another; b) impacts on industrial economics and incentives; c) impacts on long-range potential hazards; d) impacts on occupational exposure; and e) impacts on ultimate disposal and ultimate waste forms.

The revised R&D plan will describe a program to assess these alternatives. The plans for R&D on disposal will be described in context as part of an overall plan for fission by-product management.

Workshop Activities

1. Evaluation Methods

In a meeting at Sandia Laboratories, Albuquerque, April 9, a workshop was planned to develop methods for evaluation of disposal alternatives. The objectives of the workshop would be to a) review and critique existing evaluation methods and b) define needs for additional methods. Ultimate output from the workshop would be a detailed program plan for R&D to develop evaluation methods.

The projected workshop was structured to involve participants with three types of interests (data inputs, model development, and applications) in two basic categories: geospheric transport and biospheric transport. With this scope, the workshop is expected to involve 25 to 30 participants. This may be too large a number for effective work; the workshop may be split into two parts. The workshop(s) are tentatively planned for the Fall of 1975. Details are yet to be established; the location is expected to be Denver, and the current list of potential participants is tentative.

2. Seabed Disposal

A workshop to develop generic waste management system concepts for seabed disposal was held in Seattle on May 16 and 17. Participants were A. M. Platt and J. W. Bartlett, PNL;

D. R. Anderson, W. P. Bishop, D. Talbert, and Mark Davis of Sandia; Charles Hollister and Allyn Vine of Woods Hole Oceanographic Institute; D. Gibson of Scripps, and Armand Silva of Worcester Polytechnic Institute. Drs. Hollister, Vine, Gibson, and Silva are consultants to Sandia on their seabed disposal program.

The workshop considered a variety of concepts for the seabed disposal waste management system. Major conclusions concerning the system were:

- Penetrometer (i.e., self-burial) emplacement of waste containers in sediment is attractive and feasible.
- The port facility can be quite simple and small. A key feature of operations would be to have the waste-carrying portion of the ship separable from the crew and propulsion sections.
- Having the waste in golf-ball size glass spheres might simplify waste handling. This waste form would also permit a trenching-type emplacement operation.
- All aspects of seabed disposal are feasible with current technology.

However, R&D is needed on:

- seabed characterization
- canister-to-sediment heat transfer
- nuclide migration in sediments
- biological phenomena
- hydrodynamics
- canister durability
- safety
- institutional considerations
- system design

An outline of an R&D program that would permit a go-no go decision on

seabed disposal by 1980 was developed. The critical path in this program is authorization and funding for cruise ships. Details of the plan will be developed by Sandia Laboratories within their seabed disposal program.

TASK 2 - DEVELOPMENT OF EVALUATION METHODOLOGY

RADIONUCLIDE MIGRATION FROM NONSALT FORMATIONS - H. C. Burkholder and M. O. Cloninger

Work on this subtask is reported in Section 2.0, "Separation of Actinide Elements from Radioactive Waste," in this quarterly report.

RADIONUCLIDE MIGRATION FROM SALT FORMATIONS - A. G. Gibbs and H. C. Burkholder

To compare risks of disposal in salt formations with those in non-salt formations, a method is needed to estimate the migration of nuclides from salt formations. A model for this purpose was developed in the previous quarterly report. The solution to this model for the no dispersion case has been obtained by the method of characteristics.

For the situation when saltwater is injected into the soil column before, during, and after waste release, the salt concentration at any position and time is given by

$$C_S(n, \theta) = C_S^0 \left\{ u(\theta - K_S n) - u(\theta - TV/L - K_S n) \right\}$$

where:

C_S^0 = salt concentration at the soil column inlet

K_S = sorption equilibrium constant for salt

L = soil column length

t = time after salt first starts leaching out

T = time when salt stops leaching out

V = water velocity

Z = distance from soil column inlet

$\theta = tV/L$ = dimensionless time

$\eta = Z/L$ = dimensionless distance

$u(\)$ = unit step function

The nuclide discharge rate is described by

$$N_1(\eta, \dots) = \frac{N_1^0}{T_2 - T_1} \exp(-R_1) \left\{ u(f_2(\)) - K_1 \right\} - u(f_1(\)) - K_1 \right\}$$

where:

$$f_1(\theta) = \begin{cases} \theta - T_2 V/L & \text{for } \theta < \theta_a \\ \eta_a + \frac{K_1}{K_0} (\theta - \theta_a) & \text{for } \theta > \theta_a \end{cases}$$

$$f_2(\theta) = \begin{cases} \theta - T_1 V/L & \text{for } \theta < \theta_b \\ \eta_b + \frac{K_1}{K_0} (\theta - \theta_b) & \text{for } \theta > \theta_b \end{cases}$$

$$\eta_a = \theta_a - T_2 V/L$$

$$\eta_b = \theta_b - T_1 V/L$$

$$\theta_a = (TV/L - K_S T_2 V/K_1 L) / (1 - K_S/K_1)$$

$$\theta_b = (TV/L - K_S T_1 V/K_1 L) / (1 - K_S/K_1)$$

K_0 = sorption equilibrium constant for nuclide when no salt is present in leachant

$$K_1 = 1 + \frac{\rho}{\epsilon} \exp(C_1) C_S^{-C_2}$$

= sorption equilibrium constant for nuclide when leachant has the inlet salt concentration

N_1^0 = nuclide inventory at the time the salt first starts leaching out of the formation

$R_1 = \lambda_1 L/V$ = decay number

T_1 = time after salt starts leaching out when nuclide starts leaching out

T_2 = time after salt starts leaching out when nuclide stops leaching out

ρ = bulk density of soil

ϵ = soil void fraction

C_1 and C_2 are empirical constants

These analytical results can now be evaluated in a suitable computer program to estimate the migration of single nuclides from salt formations for the no dispersion case.

PUBLIC ATTITUDES TOWARD NUCLEAR WASTE MANAGEMENT - W. S. Maynard (HARC)

Work completed during the past quarter has included:

- Final revision of waste disposal information film
- Final revision of waste disposal survey questionnaire
- Sponsor review and clearance to proceed with survey
- Administration of survey to respondent groups in five regions of the United States
- Completion of initial data analyses which summarize the results both for each respondent group and for the entire sample of respondents.

SURVEY METHOD

Questionnaire. The primary focus of the survey instrument was to measure the relative importance which respondents attached to four aspects of nuclear waste disposal. The four issues selected for the study were:

- Short-Term Safety
- Long-Term Safety
- Cost
- Accident Detection and Recovery

The selection of these four issues was dictated by the need to incorporate dimensions which would be of value to decision-makers and planners as well as the need to use issues which would be relevant to participants in the study.

The questionnaire contained six parts. Three sections were devoted to measuring the relative importance of the four issues using different measurement procedures. A fourth section focused on respondents' evaluations of various levels of risk and cost. The fifth part of the questionnaire dealt with respondents' attitudes toward nuclear waste disposal and with related topics such as the perceived efficacy of nuclear power and alternative power sources, and the relationship between national power needs and the economic well-being of the United States. The final section comprised an evaluation and critique by respondents of the methodological approach.

Sample. The objective of the sampling was to obtain responses from a broad spectrum that would vary as to age, sex, occupation, specialized technical training, environmental orientation, and geographic region.

Administration. The questionnaire was administered in a group setting. Prior to completing the survey, respondents viewed a 15-min film. The intent of the film was to present a brief introduction to nuclear waste disposal and to highlight the four issues on which the questionnaire was based.

SURVEY RESULTS

Table 1 summarizes characteristics for each group of respondents. Twenty groups were surveyed and the summary results are based on a total sample size of 434 respondents. In addition to demographic and biographical characteristics, the average responses to two questionnaire items were used to help characterize each group: self-rating of knowledge about nuclear waste disposal, and attitude toward nuclear power as an energy source. The data in Table 1 indicate that the total sample: spans a fairly broad age range, is roughly balanced between men and women (male/female ratio = 60/40), and varies considerably as to knowledge of the topic and orientation toward nuclear power. When the high school and university groups are removed, the remaining sample can be characterized as well above the national norm of educational attainment.

In the first section of the questionnaire, respondents were asked to rank order the four waste disposal factors from "Most Important" to "Least Important." The results of this ranking procedure for the total sample are presented in Figure 1. The height of the bars indicates the extent to which each factor was ranked first, second, third, and fourth in importance. The results suggest that Cost was almost universally seen as least important. Long-Term Safety was generally seen as the most important of the four factors. Across the total sample, there was considerable variation as to the relative importance which should be attached to Short-Term Safety and Accident Detection and

Recovery. Significant numbers of respondents saw Long-Term Safety, Short-Term Safety, and Accident Detection and Recovery as requiring highest priority.

Part III of the questionnaire presented several levels of the four waste disposal factors and asked respondents to indicate how satisfactory or unsatisfactory they would find each level. The purpose of this section was to obtain information on how respondents evaluate cost and risk considerations. As an illustration, Figure 2 presents the average responses to the four factors by two groups of respondents: nuclear engineers and members of a West coast environmental organization. Examination of Figure 2 suggests the two groups differed substantially in their evaluation of Short-Term Risks, differed moderately in their evaluation of Long-Term Risks, and evaluated rather similarly the levels of Cost and Accident Detection and Recovery.

Part IV of the questionnaire consisted of 30 attitude statements with which respondents were asked to indicate their agreement or disagreement. Some of these statements dealt with the four waste disposal factors and were written with the expectation that the results would supplement the results from other sections of the questionnaire. Other attitude statements were focused on a constellation of issues related to the topic of nuclear waste disposal. Table 2 shows the responses for the total respondent sample. Beside each statement the number and percentage of respondents who endorsed each response category is

TABLE 1. Characteristics of Respondent Groups

Respondent Group	Number of Completed Questionnaires	Incomplete Questionnaires(2)	Average Age	Age Range	Male/Female Ratio	Average Years of Education	Average Knowledge of Topic(3)	Orientation Toward Nuclear Power(4)
1. Engineering Scientists and Public Officials (NW)(1)	20	0	41.6	27-62	19/1	16.2	6.55	4.25
2. Corporate Nuclear Engineers (NW)	31	1	44.6	29-60	31/0	18.4	5.42	4.45
3. National Environmental Organization (MW)	5	1	27.0	25-30	4/1	18.0	5.00	2.00
4. National Environmental Organization (NE)	12	0	30.6	22-48	9/3	19.3	4.83	2.17
5. Local Environmental Organization (W)	18	1	50.6	22-78	2/16	15.9	4.56	2.50
6. Public Interest Lobbyists (MW)	6	0	23.3	19-30	6/0	16.0	5.33	1.33
7. Womens' University Organization (NW)	20	3	41.2	26-63	0/20	16.7	4.05	2.95
8. Recreationists (MW)	9	3	40.1	23-64	8/1	14.9	4.33	3.78
9. Civic Organization (W)	18	0	45.3	27-65	15/3	16.1	4.0	3.78
10. Civic Organization (MW)	10	2	33.1	22-50	9/1	16.0	2.90	3.10
11. Church Members (NW)	26	4	44.6	29-60	12/14	16.7	3.77	3.38
12. Church Members (NE)	28	9	45.5	23-83	2/20	16.7	3.55	3.13
13. Junior High School Students (NW)	32	3	13.8	12-15	18/14	8.1	3.50	3.22
14. High School Physics Class (MW)	15	1	17.8	17-18	10/5	12.0	4.50	3.23
15. High School Environmental Science Class (S)	35	7	15.9	14-19	23/12	10.5	2.37	3.26
16. Undergraduate Physics Class (W)	19	2	21.5	17-28	17/2	15.1	5.21	3.26
17. Undergraduate Civil Engineering Class (S)	23	1	19.9	18-32	23/0	13.7	3.39	3.04
18. Undergraduate Psychology Class (NE)	20	1	20.9	18-28	6/14	14.8	3.0	2.90
19. Continuing Education College Students (NE)	12	0	31.2	21-50	4/8	16.1	3.5	2.83
20. Undergraduate Civil Engineering Class (NW)	75	41	27.3	18-41	40/35	15.2	3.95	3.19
	434	80			264/170			

1. Region of U.S. is shown in parentheses following group name. NW = Northwest; MW = Midwest; S = South; W = West; NE = Northeast.

2. Number of incomplete questionnaires not included in analysis.

3. Knowledge based on participants' response to question, "Before you participated in this study, how well-informed were you about the subject of nuclear waste disposal?", where 1 = Not at all informed; 7 = Very well informed.

4. Nuclear power orientation is based on extent of agreement with the statement, "Based on past operating experience, nuclear energy has been demonstrated to be a clean and safe source of electrical power." 1 = Strongly disagree; 2 = Disagree; 3 = Neither agree nor disagree; 4 = Agree; 5 = Strongly agree.

FIGURE 1. Results of Questionnaire Ranking Procedure

presented. Detailed analyses of individual items, clusters of items, and comparisons across groups will be included in the final report. Examination of responses to items 11, 19, and 28, however, will illustrate the way that this information may supplement the results from other sections of the questionnaire. Response patterns to items 11 and 28 support the emphasis placed on the factor, Long-Term Safety, but this support is tempered somewhat by responses to item 19, where there is a more even distribution of responses across the five response categories.

The final section of the questionnaire consisted of an evaluation and critique by respondents of the methodological approach used in the study. Specific and general reactions to the survey were solicited, and respondents were invited to add comments of their own. Table 3 presents the responses of the total sample to three of these questions, together with sample verbatim comments. Approximately 75 to 80% of the respondents approved of the research objectives and methods. Most of the questionnaires contain write-in comments; these remarks will be content analyzed and presented in the final report.

FIGURE 2. Example Average Responses from Two Groups

The final report now in preparation will include more extensive analyses of the data, comparisons of results across respondent groups, and verification of results by comparisons made among the different measurement methods.

Other reporting activities in the first two months of FY-1976 will involve preparation of a series of topical reports focusing on specific components of the questionnaire content

and methodology. This series will include reports on: risk perceptions associated with nuclear technology; applicability of survey research methods to assessment of technology; the relationship between stated estimates of priority and implicit applications of policy; and the appropriateness of the information-questionnaire approach to assess attitudes and values toward technological issues.

TABLE 2. Attitude Response Frequencies for Total Sample

	<u>SA</u>	<u>A</u>	<u>AD</u>	<u>D</u>	<u>SD</u> ⁽¹⁾
1. Overpopulation and economic conditions are more serious threats to our way of life than the problem of nuclear waste disposal.	96 22%	137 32%	90 21%	87 20%	19 4%
2. Nuclear wastes should be disposed of in such a way that no one will ever be able to disturb or recover them.	137 32%	118 28%	64 15%	89 21%	21 5%
3. In dealing with nuclear wastes, cost should be of no concern.	57 13%	90 21%	70 16%	171 40%	41 10%
4. Based on the information I got from this study, the hazards of nuclear wastes are less serious than I thought.	11 3%	68 16%	118 28%	140 33%	92 21%
5. In the next 10 years, solar power could solve the problem of electrical power in the United States.	21 5%	95 22%	116 27%	136 32%	61 14%
6. The loss of even one life from nuclear waste is unacceptable.	34 8%	66 15%	72 17%	168 39%	89 21%
7. The United States can maintain a strong economy without having nuclear power.	33 8%	54 13%	98 23%	149 35%	95 22%
8. The construction of nuclear power plants should be slowed until an absolutely safe waste disposal system has been developed and thoroughly evaluated.	94 22%	140 33%	45 11%	101 24%	49 11%
9. Nuclear waste is one of the most serious threats facing the world.	41 10%	97 23%	86 20%	140 33%	65 15%
10. The radioactive hazards in nuclear power are no greater than the environmental hazards from coal burning power plants.	34 8%	67 16%	81 19%	166 39%	81 19%
11. Future generations must be <u>totally</u> safe from our nuclear waste.	102 24%	155 36%	77 18%	74 17%	21 5%
12. In my opinion, the information contained in this study seriously underestimates the real risks of nuclear wastes.	37 9%	73 17%	144 34%	151 35%	24 6%
13. The benefits of nuclear power more than outweigh the hazards in nuclear waste storage and disposal.	54 13%	132 31%	89 21%	103 24%	51 12%
14. Nuclear power is the most promising source of electrical power for the next 30 years.	71 17%	168 39%	104 24%	50 12%	36 8%
15. Whatever disposal system is devised, there must be a way to detect leaks and recover the waste.	235 55%	148 34%	25 6%	16 4%	5 1%

TABLE 2. (Continued)

	SA	A	AD	D	SD ⁽¹⁾
16. Before I got involved in this study, I knew very little about the problem of nuclear wastes.	51 12%	107 25%	53 12%	152 35%	66 15%
17. The construction of additional electrical power plants will not be necessary for the United States to maintain its present standard of living.	14 3%	41 10%	80 19%	174 41%	120 28%
18. I would not want to have nuclear wastes disposed of in my region of the country.	96 22%	108 25%	79 18%	109 25%	37 9%
19. The generation that uses nuclear power should take all the risks for waste disposal.	51 12%	104 24%	74 17%	144 34%	56 13%
20. Government and industry have done a good job of keeping the public informed about the advantages and disadvantages of nuclear power.	11 3%	41 10%	45 10%	180 42%	152 35%
21. Based on past operating experience, nuclear energy has been demonstrated to be a clean and safe source of electrical power.	41 10%	157 37%	126 29%	78 18%	27 6%
22. Participating in this study makes me realize just how frightening nuclear waste is.	31 7%	111 26%	100 23%	125 29%	62 14%
23. In the next 10 years, geothermal power could solve the problem of electrical power in the United States.	9 2%	53 12%	171 40%	125 29%	71 17%
24. Using more coal is far less harmful to man than using nuclear power.	11 3%	75 17%	138 32%	153 36%	52 12%
25. To satisfy critics, the United States nuclear waste system will probably cost far more than is necessary.	43 10%	162 38%	85 20%	96 22%	43 10%
26. Paying a little more attention to conservation would make more electrical power unnecessary.	44 10%	89 21%	91 21%	160 37%	45 10%
27. We should immediately stop developing nuclear technology as an energy source.	10 2%	21 5%	46 11%	170 40%	182 42%
28. The main concern of nuclear waste planners should be a safe system for the short term.	24 6%	84 20%	64 15%	202 47%	55 13%
29. Compared to other problems that future generations will face, nuclear waste monitoring is insignificant.	22 5%	44 10%	90 21%	178 42%	95 22%
30. If the United States does not increase the production of electricity, our economy will decline.	59 14%	182 42%	112 26%	60 14%	16 4%

1. SA = Strongly Agree; A = Agree; AD = Neither Agree nor Disagree; D = Disagree; SD = Strongly Disagree

TABLE 3. Respondents' Evaluation of Survey

Total responses to the question, "Do you think it is wise to consult the general public in the United States about decision-making on nuclear waste disposal?" Yes - 351 (81%); No - 80 (19%).

Representative comments:

Yes, it will concern us and our children. If we plan on using the atom we all must be aware of the consequences. The government makes too many decisions without public involvement.

Yes, people should have a hand in molding the future--not just have it handed to them.

No, I think the general public is far more concerned about its own personal comfort and saving money than about hazards and consequences their actions may present to future generations.

Yes, it is likely that the general public will not choose to evaluate the problem as the long-term problem it is but will instead be overly concerned with current cost and short-term safety.

No, nothing could be gained by consulting an uninformed public. Their decisions and answers would be more emotional than factual. Yes, an informed public should be consulted.

Total responses to the question, "Is the approach used in this study (movie and questionnaire) an effective way to gather public input about nuclear waste disposal?" Yes - 318 (74%); No - 113 (26%).

Representative comments:

Yes, working on a more personal basis like this asks for a responsible opinion, and we are given a chance to hear some information on the subject before answering this.

No, the motion picture had very low information content, and the questionnaire was redundant.

Yes, it's basically a reasonable approach but a fifteen-minute film can only begin to scratch the surface of factors to be considered.

Yes, the movie "educates" and focuses attention on the issue. The questionnaire allows these individuals to be concrete in his/her opinions at his/her own pace, in a non-threatening manner.

Total responses to the question, "Were the four factors emphasized in this study a satisfactory way to think about nuclear waste disposal?" Yes - 360 (84%); No - 70 (16%).

Representative comments:

Excellent, especially the schematic, or mathematically logical, nature of many of the questions for comparing relative importance of factors.

Yes, the film. No, the questionnaire. Long-term, short-term, accident detection, cost--I don't know that much about any of them. Except cost, I've never been involved.

No. Factors such as pertinent health hazards should have been emphasized as well as actual incidents of hazards that have already occurred. The risk figures, long-term and short-term, are inadequate to reflect potential hazards.

No, I suppose so but I feel unsatisfied; it is too simplified. I keep feeling I need a lot more information and a lot more reassurance that the scientists involved know what they are talking about.

No, the subjects all were brushed over too lightly. More emphasis on each point was needed.

TASK 3 - SPECIAL EVALUATIONS -

J. W. Bartlett

These studies support and result from plans development work (Task 1) and other waste management programs. Some studies are intended to provide definitive responses to specific problems or questions. Others are to initiate investigation of topics of concern but not yet under study in other programs.

WASTE MANAGEMENT FOR ^{129}I AND ^{99}Tc

- H. G. Rieck

A draft final report of this study is being prepared.

Waste management techniques being considered as acceptable for ^{129}I are isotopic dilution, salt mine storage, extraterrestrial injection and transmutation processes. Based upon the chemical and biological activity of iodine, geologic and seabed injection involve greater risk in contaminating the pathway to human exposure.

Evaluation of waste management methods for ^{99}Tc is difficult because there is minimal quantitative information on the physical properties of technetium compounds. For example no solubility coefficient could be found for Tc_2S_7 , TcS_2 , or TcO_2 except that they are insoluble. Evaluation of methods that involve aqueous solutions is difficult if not impossible. The geologic disposal of technetium would be feasible if the insolubility and stability of the sulfides and/or oxides could be verified. Another candidate technetium compound for geologic disposal may be as a boride if

production is not too costly or difficult. Seabed injection would be an alternative method if future research verifies chemical stability of these technetium compounds.

Transmutation processes and terrestrial injection are possible waste management techniques for ^{99}Tc as well as ^{129}I . Since three times as much technetium is formed, terrestrial disposal would be economically desirable unless production costs of the insolubles would be an offsetting factor.

LOSS OF HIGH-LEVEL WASTE AT SEA -

W. B. Silker

The assessment of possibilities and consequences of loss of high-level waste at sea in seabed disposal was completed during the quarter. The study considered modes of loss, feasibility of recovery, and the fate of waste material if losses were irretrievable.

Conditions of loss that were considered included ship collision, sinking due to high seas or ship-board failures, and accidents during waste handling operations (transfers and disposal emplacement). All such events can occur, but consequences that would result in irretrievable loss of one or more canisters can be minimized by engineering design and procedures. Details will appear in a topical report to be issued.

Capability to locate and retrieve lost waste has been demonstrated implicitly by location and salvage operations such as those involving the submarines Thresher and Scorpion, the lost hydrogen bomb, and the

Glomar Explorer's (partial) recovery of the Russian submarine. In brief, current location and retrieval technology is capable of recovering single or a few waste canisters. Recovery of an entire ship and its waste cargo is improbable. However, recovery of the waste cargo can be enhanced by design of the waste containers, e.g., making them flotation devices.

Preliminary evaluation was made of the consequences of irretrievable loss (detailed studies within Sandia's seabed disposal program are expected). These studies, which considered factors such as dilution, dispersion, and uptake in the food web, projected insignificant radiological consequences. More detailed study and data are needed, however,

In summary, this overview assessment found no factors associated with loss of waste that would preclude seabed disposal. A topical report is being prepared.

IMPACTS OF DEFERRED REPROCESSING - R. L. Murray, North Carolina State University

This study was started this quarter. The project consists essentially of a preliminary technological assessment of the storage of spent nuclear fuel elements for different periods of time prior to reprocessing. The time intervals include five years, twenty years, forty years, and indefinitely as a special limiting case. The first three time periods correspond respectively to current practice, to a date when a successful breeder would demand a great deal of fuel,

and to a time when the status of fusion is clarified. In the fourth case, fusion and/or other energy sources might be ample, and uranium and plutonium not needed.

The work is being carried out by the principal investigator with advice by three consultants: Jerome Kohl, Edward W. Erickson, and Lloyd R. Zumwalt, and with assistance by a graduate student in Nuclear Engineering, K. K. Low.

The first two and a half months of the project were devoted to the identification of the key factors involved in the long-term storage concept.

Factors in the Fuel Storage Concept

Table 4 briefly outlines the factors in the fuel storage concept. It is noted that several of the factors will involve data that can be utilized to estimate dollar costs of different strategies, but that others are much more qualitative in nature, and involve non-economic aspects. Thus the best method of evaluating all factors in the most equitable and understandable manner will have to be determined.

Plans for Future Work

A decision must be made relatively soon as to the exact scope of the study. Among the obvious choices are a) a "once-over-lightly" treatment of all of the factors in Table 2, or b) quantitative investigations of a few factors on the basis of their importance and our local capabilities and interests, or c) identification of what

TABLE 4. Factors in Fuel Storage

1. Composition of fuel elements versus time
 - a. Short-lived fission products
 - b. Intermediate-lived fission products
 - c. Long-lived fission products
 - d. Actinides
2. Accessibility of radioactive materials
 - a. Radiation exposure to workers
 - b. Comparison of storage techniques
 - c. Retention of plutonium
 - d. Transportation hazards
 - e. Safeguards of special nuclear material
 - f. Loss of isotopes for applications
3. Uranium isotope separation capacity
 - a. Effect of storage on demand for new facilities
4. Fuel reprocessing facilities
 - a. De facto storage policy
 - b. Potential for new facilities, costs
5. Spent fuel storage facilities
 - a. Requirements in space and protection against access, criticality, etc.
6. Uranium resources for the future
 - a. Relation of uranium to fossil fuels
 - b. International competition for energy resources
 - c. Interaction of storage with exploration, mining, processing, stockpiling
 - d. Effect of new ore discoveries
7. Alternate approaches in fuel cycle operation
 - a. Present LWR approach
 - b. Plutonium recycle
 - c. Plutonium burnup
 - d. Changes in emphasis
8. Effects of and on new nuclear developments
 - a. Breeder operation - fuel supply problem
 - b. New isotope separation methods (centrifuge, laser, etc.)
 - c. Fusion (magnetic or inertial confinement)
 - d. Waste processing by fusion neutrons
9. Reactor design and refueling cycle changes to accommodate to a storage policy
 - a. Minimize Pu production
 - b. Maximize Pu consumption
 - c. Fuller use of uranium (pseudo breeding)
 - d. Other criteria for change, e.g., ^{241}Pu
10. Potential new developments in the energy field
 - a. Effect of conservation
 - b. Effect of new energy systems - solar, geothermal, etc.
 - c. Electric automobile
 - d. Net energy analysis
11. Public acceptance of nuclear power
 - a. Available public opinion polls
 - b. Status of nuclear opposition and pending political action
 - c. Sensitivity to incidents
 - d. Urgency of need for nuclear energy
 - e. Improvement of hazard indices

detailed investigations should be made in a follow-on study of greater depth. The decision can be delayed during the period when general information is being collected and methodology is being reviewed, but must be made firm shortly thereafter. An iterative process proceeding roughly in order a), b), and c) is expected.

Some of the specific tasks to be performed during the next quarter are as follows:

- a. Collection of references--books, reports, articles--and extraction of useful information.
- b. Acquisition and test of an appropriate set of computer programs on isotope separation, fuel isotopic composition, fuel cycle physics and costs, etc.
- c. Determination of economic model and method of treatment of non-economic aspects, involving various measures such as dollars, energy, and social impact indices.

DISPOSAL AND DISPERSION IN GEOLOGIC FORMATIONS - B. W. Mercer

During the report period, studies were begun to evaluate the disposal and dispersion of radioactive wastes in suitable geologic formations so as to minimize potential hazards related to subsequent geologic activity. The objective of this disposal concept is to dilute the toxic radionuclides with very long half-lives (e.g., ^{237}Np , ^{239}Pu) to concentration levels which would not be hazardous if the geological formation were exposed

after several centuries. This approach would involve considerably less risk in projecting geologic activity in a disposal zone since the time span is several orders of magnitude less than the $\sim 10^6$ years required for decay of the very long-lived radioactivity in concentrated forms of the waste.

The long-lived transuranics require a large, but not prohibitive, volume of soil for their disposal and dispersal to a 10 p Ci/g concentration. Several possible inorganic complexants exist to facilitate the transuranic radioisotope dispersions, including SO_4^{-2} and SCN^- . The use of these complexants would increase mobility of the transuranic radioisotopes to allow adequate dispersion. However, these radioisotopes would be sorbed by the soil eventually due to complexant dilution by groundwater to assure retention in the desired zone. It is estimated that a 100-ft-thick soil zone over a 100 mile²-area could be used to store the transuranic waste from a maximum of 1500 tons of LWR fuel reprocessing waste (composition as given in BNWL-1900 and assuming a 10 p Ci/g limit in the soil) per year for 22 years.

Dispersion of the medium half-life fission products, such as ^{90}Sr , to nonhazardous concentrations would require prohibitive volumes of soil. However, due to the much shorter decay time needed for these radionuclides, dispersion to innocuous concentrations is not nearly as critical as in the case of the long-lived

transuranics. Projecting the integrity of a suitable soil disposal zone over a few centuries can be done with considerably greater confidence than for periods of hundreds of millenia.

Current efforts in this program are focusing on potential soil-related chemical reactions with long-lived transuranics and the significant fission products. To begin this work, an existing sediment column ion-exchange model was used to determine column loadings of strontium and cesium in relation to the distribution of transuranics. The model requires use of distribution coefficients for each radioisotope examined. Reasonable estimates of strontium and cesium distribution coefficients between sediment and solution were made based on past ion-exchange experience. The combined equilibrium distribution between sediment and solution was assumed to equal one. The effects of utilizing a salt or brackish water and river water for waste solution dilution on strontium and cesium loading also were obtained. These data can be used to determine whether or not a heating problem may occur due to concentration and radioactive decay of strontium and cesium radioisotopes.

Maximum sediment loading limits also were examined. We had arbitrarily chosen an upper limit of 10 p Ci/g of sediment for loading of long-lived transuranics, but little justification can be given for use of this concentration value for an upper limit. Radiological Safety personnel refuse to commit themselves to 10 p Ci/g of sediment as a maximum loading limit.

It was decided to examine and compare with this load limit some on-going activities such as uranium milling. The idea behind such a comparison is that while mill tailings are under surveillance in many cases⁽¹⁻⁴⁾, disposal of uranium milling process residues as tailing piles or ponds is a generally accepted practice. Though the total mill throughputs to date were not located, a daily ore processing tonnage was given, along with the daily amount of ^{226}Ra disposed to the tailings pond. We may conservatively estimate the daily tonnage of waste slimes and sands generated as equal to the daily ore processing tonnage because the U_3O_8 content of the ore was usually 0.2 wt%, or less than the probable throughput weighing error. The picocuries per gram of tailings are given in Table 5 for the ^{226}Ra analytical results only.

PROGRAM PLAN FOR R&D ON FISSION BY-PRODUCT MANAGEMENT - A. S. Kubo

As noted in Task 1, recent events have encouraged a systematic, comprehensive assessment of alternatives for the back-end of the LWR fuel cycle. This study to develop an R&D plan for such assessments was started during the quarter.

The study is expected to produce an R&D program plan that will:

- identify alternatives for fission by-product management
- provide a work plan for evaluation of the alternatives
- assess and apply prior work.

Major factors to be considered include:

TABLE 5. Loading of Tailing Sediments with ^{226}Ra

<u>Mill Location</u>	<u>Ore Processed/Day, Tons</u>	<u>Measured Daily Output ^{226}Ra, mg</u>	<u>^{226}Ra, p Ci/g Tailings</u>
Edgemont, SD	500	300	661
Gunnison, CO	330	198	647
Grand Junction, CO	540	176	353
Grants, NM	750	396	570
Grants, NM	1640	750	493

Note that the tailing sediments of Table 5 are generally 35 to 66 times as high in ^{226}Ra ($\text{MPC}_w = 3 \times 10^{-8} \text{ Ci/m}^3$) as the 10 p Ci/g sediment limit used in our dispersion work. We are not implying that the 10 p Ci/g limit is wrong, but only that it is ultraconservative in terms of present uranium ore processing waste disposal practices. For the Edgemont mill tailings to decay to 10 p Ci $^{226}\text{Ra/g}$, for example, would require approximately 10,000 years (^{226}Ra half-life - 1622 years).

1. Alternatives for spent fuel management such as immediate reprocessing, indefinite storage, and use for heat generation.
2. Alternatives for fuel reprocessing that consider:
 - a. Should plutonium be recycled and if so who should do the processing?
 - b. Should multivalue extractions be made that add to the operational risk of fuel reprocessing?
 - c. If multivalued extractions occur, could the currently unprofitable and sensitive extractions occur at a Federal By-Product Repository?
 - d. How will large regional nuclear-energy centers or Integrated (Dispersed) Fuel Cycle Facilities impact the By-Product Management Program?
3. Alternatives for storage disposal of cladding hulls involving advanced technologies to use the material as waste containers or potential resource materials.
4. Alternatives for waste disposal to insure compatibility with the overall By-Product Management Program.
5. Assessment of the future environment to discern impacts on industry, anticipate new technologies, and forecast fuel cycle throughputs.
6. Regulatory and operational control of materials.
7. Impacts on radiation exposure (operating personnel and public).
8. Characterization of a fission by-product management facility that can
 - reclaim selected valuable by-products

- store, for future use, potentially valuable by-products
- prepare by-products of no discernable value for disposal
- dispose of by-products as necessary.

This approach views direct-to-storage/disposal management of high-level wastes and other fission power by-products as one of several alternatives. There are potential advantages and disadvantages for the alternatives that involve processing, transport and other logistics, and use prior to disposal. Generically, these include:

Potential Advantages

1. Economic and material value of the fission by-products as fissile, mineral, heat and irradiation sources.
2. Reduction of long-term potential waste hazards.
3. Future value of extractants and program flexibility.

Potential Disadvantages

1. Operational hazards owing to increased processing, system complexity, and dispersed inventories.
2. Accessibility of retained by-product materials, to errors and the irrational acts of man and nature.

3. Commitment of resources to support and monitor the expanded use of fission by-products.
4. Lack of sufficient economic incentives or demands for recovered by-product values.

This study is expected to continue through the first quarter of FY-1976. Results will be incorporated into R&D plans for management of commercial radioactive wastes (output of Task 1).

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2.0 SEPARATION OF ACTINIDE ELEMENTS FROM RADIOACTIVE WASTE

E. J. Wheelwright - Project Manager

The primary FY-75 objective is to determine the effect of actinide partitioning upon advanced waste management concepts and thereby demonstrate the incentives or lack thereof for partitioning. A secondary objective is to develop the necessary technology for performing the partitioning processes and thus determine the technical feasibility.

SUMMARY

A sample of irradiated oxide fuel from the "Yankee" reactor has been dissolved in nitric acid, and the residual solids removed by filtering the dissolver solution through a 5- μ pore size filter. The solids were weighed, then solubilized by leaching with HNO_3 -HF followed by a NaHSO_4 fusion, and the solutions analyzed for actinide content. In this particular experiment, 0.32 wt% of the fuel remained undissolved in the HNO_3 dissolution step, and the solids contained 0.71 wt% of the plutonium in the irradiated oxide.

The dose calculations have been completed for each radionuclide on 20 parametric variations of the base-case conditions for release of high-activity waste through the soil from a geologic disposal site. The parameter variation ranges are:

- Time of initial release after the year 2000 --- 0 to 1,000,000 years
- Path length --- 0 to 10 miles
- Leach rate --- 0.3 to 0.00003 wt%/year

Experimental details, conclusions, and recommendations from the six project tasks will be found in the annual

report, Partitioning of Long-Lived Nuclides from Radioactive Waste - FY 1975 Annual Report, BNWL-1926, to be issued about the same time as this quarterly report.

TECHNICAL PROGRESS

TASK 1 - CHARACTERIZATION AND TREATMENT OF SOLIDS IN HIGH-LEVEL WASTES - F. P. Roberts

Finely divided solids are known to be present in dissolver solution and in other high-activity process streams of fuel reprocessing and waste treatment facilities. The solids originate from undissolved fuel particles, cladding fines, and precipitates formed during reprocessing and waste treatment.

The objective of this task was to evaluate the effects of the solids on partitioning by determining the amount of plutonium, americium, and curium associated with the solids at various points in the reprocessing of reactor fuels and the treatment of the wastes, and by measuring the volume, particle size, and other characteristics of the solids needed to develop processes for their treatment and handling. Because work on this task was terminated in February and funding withdrawn, progress toward

the objective was restricted to a single experiment in which a small quantity of highly irradiated light-water reactor fuel was dissolved and the solid residue analyzed to determine the plutonium, americium, and curium content.

The procedure used to dissolve 120 g of "Yankee" reactor fuel which had been irradiated to a nominal 39,000 MWd/T, remove the solids, and then solubilize the solids in preparation for analysis was described in the previous quarterly report of this series. The total weight of solids obtained from the nitric acid dissolution of the fuel was 290 mg and represented 0.32 wt% of the uranium metal present. This is in reasonable agreement with results obtained at HNL with a stainless steel clad mixed-oxide fuel. Under similar dissolution conditions the HNL workers observed 0.4 to 0.5 wt% solids for fuel with exposures of 25,000 to 50,000 MWd/T.⁽¹⁾

Table 6 summarizes the analyses of the clarified dissolver solution and the solution obtained by dissolving the solids. The principal alpha activity found in the solids is plutonium, but the minor amounts of americium and curium indicate some tendency of these elements to adhere to the solids.

As a part of the high-level radioactive waste fixation program, 2810 g of irradiated oxide (54,000 MWd/T) from the Zorita reactor have been dissolved in HNO_3 , and 29.5 g of undissolved solids removed by filtration through 5, 0.65, and then 0.025- μ millipore filters. The value of

1.04 wt% undissolved solids is a factor of about 3 higher than the solids recovered from the "Yankee" oxide and may indicate that the "Yankee" results are low because of poor filtration. These results do show that solids treatment is required if high plutonium decontamination factors are needed in partitioning.

TASK 2 - REMOVAL OF LONG-LIVED NUCLIDES FROM SOLIDIFIED WASTE -

J. A. Partridge

The objective of this task was to develop a partitioning flowsheet based on starting with solidified high-level waste, e.g., a calcine. Laboratory and hot-cell work on this task was terminated in February and the results described in the January-March quarterly progress report. Task 2 input for the annual report was completed during the current quarter.

TASK 3 - REMOVAL OF LONG-LIVED NUCLIDES FROM NON-HIGH-LEVEL WASTES -

J. H. Westsik

The objective of Task 3 was to evaluate the applicability of the partitioning concept to non-high-level nuclear wastes and to develop the technology necessary to apply partitioning to these wastes, where desirable, to integrate them into the overall advanced waste management program.

During Fiscal Year 1975, Task 3 was limited to surveying the general area of non-high-level waste to determine the scope of the problem, and the need, if any, for partitioning. A literature search had been started but was canceled when directions were

TABLE 6. Analyses of Clear Dissolver Solution and Solubilized Solids

	<u>Contents of Dissolver Solution</u>		<u>Contents of Solubilized Solids</u>	
	<u>Amount</u>	<u>% of Total in Fuel</u>	<u>Amount</u>	<u>% of Total in Fuel</u>
Pu	1.00 g	99.3	0.0072 g	0.71
^{241}Am	4.06×10^{11} d/m	99.9	3.5×10^8 d/m	0.086
^{244}Cm	3.28×10^{11} d/m	99.99	4.19×10^7 d/m	0.013

received in February to terminate work on this task. There is nothing to report in this quarter.

TASK 4 - DEVELOPMENT OF LOW-ADDITIVE FLOWSHEETS FOR REMOVING LONG-LIVED NUCLIDES FROM THE FISSION PRODUCTS -

J. L. Swanson

The objective of this task was to develop and test methods for purifying the high-level fission product waste from potentially hazardous, long-lived nuclides (primarily actinides) by processes which do not add excessive amounts of inert additives to the waste. This task was also canceled in February and the accomplishments summarized in the January-March quarterly report. Task 4 input for the annual report was completed during the current report period.

TASK 5 - EVALUATION OF INCENTIVES FOR PARTITIONING - H. C. Burkholder, D. A. Baker, and M. O. Cloninger

The purpose of this task is to determine the conditions under which incentives exist for partitioning the high-activity wastes before ultimate disposal/storage. The scope is limited to evaluation of the effects of partitioning on the safety of various

storage forms and storage modes after the interim storage phase and after all handling operations for ultimate disposal are completed. The task deals with the incentives for partitioning by evaluating the health risk over an indefinite future period.

During the current quarter, the leach rate and path length have been varied to determine the effect of these parameters on the 50-year accumulated dose to a "maximum" individual. Table 7 summarizes the parameter values used for this study. As shown, the parametric variations that have been used for calculations with the integrated geosphere-biosphere transport model include the time of initial release after the year 2000, the leach rate, and the soil path length. These three parameters are indices of canister integrity-site stability, waste form stability, and geospheric isolation, respectively. All are variables whose values can, within limits, be selected by man in designing a system for managing radioactive waste, and it is appropriate to consider the incentives for partitioning in that context. Tables 8 through 23 show summaries of the 50-year accumulated dose to a

TABLE 7. Parameter Values for Geosphere Transport Calculations

Time of Initial Release After Yr-2000, yr	Leach Rate, %/yr	Path Length, Mi
100	0.3	10
100	0.3	3
100	0.3	1
100	0.3	0.3
100	0.3	0.1
100	0.3	0
100	0.03	10
100	0.003	10
100	0.0003	10
100	0.03	0.1
100	0.003	0.1
100	0.0003	0.1
100	0.0003	1
100	0.00003	0.3
100	0.00003	1
100	0.00003	3
0	0.3	10
1,000	0.3	10
10,000	0.3	10
100,000	0.3	10
1,000,000	0.3	10

"maximum" individual at various times after burial of the waste for the parametric combinations shown in Table 7.

The results from Tables 8, 10, 11, and 12 for bone dose (which is usually the controlling dose) are shown graphically in Figure 3. Thus Figure 3 shows the effect of path length on the bone dose versus time curve. As the path length decreases, the bone dose versus time curve moves to lower time values. Hence the dose increases because nuclides are not as completely separated by differences in migration rates and because decay has less time to occur. The great value of increasing the geospheric isolation to the safety of ultimate disposal of waste is clearly shown.

The results from Tables 8, 13, 15, and 16 for bone dose are shown graphically in Figure 4 as the effect of leach rate on the bone dose versus time curve. As the leach rate decreases, the dose generally decreases. But at the lowest leach rates (0.003 and 0.0003 wt%/yr) the dose sometimes increases because significant quantities of ^{14}C and ^{237}Np then exit the soil column at times which had low doses from other nuclides when the leach rates were greater. Decreasing the leach rate has only a small effect on the uranium peak at 2×10^6 years (at least for the range shown) because the major dose contributor at that time, ^{226}Ra , was created by decay of precursors after the precursors left the repository. Hence

TABLE 8. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.3%/Yr
Path Length: 10 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem						Remarks		
		Skin	Body	GI-LLI	Thyroid	Bone	Liver		Lung	Kidney
2.50E2	³ H, ⁹⁹ Tc, ¹²⁹ I	1.3E1	2.3E2	2.6E4	1.4E4	5.4E2	8.0E2	7.5E0	1.7E1	GI-LLI: 100% ⁹⁹ Tc
1.55E3	¹⁴ C	8.0E-3	6.3E4	3.8E4	6.2E4	3.1E5	6.2E4	6.2E4	6.2E4	Bone: 100% ¹⁴ C
3.73E3	⁹³ Mo	4.4E-7	3.5E-1	2.1E0	2.5E-7	2.5E-7	1.3E1	2.5E-7	2.5E-7	Liver: 100% ⁹³ Mo
1.46E4	⁴¹ Ca, ⁷⁹ Se, ²³⁷ Np, ⁹⁰ Sr	3.1E2	1.3E3	2.0E3	2.7E2	1.2E4	3.5E3	2.7E2	2.7E2	Bone: 96% ²³⁷ Np+D
4.85E4	¹⁰ Be, ⁶⁰ Co, ⁵⁹ Ni	1.0E-2	4.5E1	2.0E1	8.6E-3	2.7E2	9.5E1	8.6E-3	2.2E-2	Bone: 100% ⁵⁹ Ni
7.26E4	⁸⁷ Rb, ²²⁶ Ra	6.2E-3	9.1E-2	7.1E-2	5.1E-3	5.7E-1	1.2E-1	5.1E-3	1.7E-2	Bone: 58% ²²⁹ Th, 31% ²²⁵ Ra+D
1.45E5	¹³⁵ Cs	6.7E-3	3.5E2	2.0E1	3.8E-3	9.4E2	8.4E2	6.5E2	1.2E-2	Bone, Liver, Lung: 100% ¹³⁵ Cs
1.61E5	¹⁰⁷ Pd, ¹²⁶ Sn	7.5E2	8.2E2	2.4E3	4.2E2	4.6E3	8.6E2	7.0E2	7.0E2	Bone: 100% ¹²⁶ Sn+D
3.62E5	¹⁶⁶ Ho	1.8E-3	2.1E-2	2.7E-2	1.5E-3	2.3E-1	5.5E-3	1.5E-3	4.8E-3	Bone: 61% ²²⁹ Th, 33% ²²⁵ Ra+D
4.82E5	²⁴⁷ Cm	1.3E-3	1.3E-2	1.7E-2	1.1E-3	1.4E-1	7.5E-3	1.1E-3	3.1E-3	Bone: 56% ²²⁹ Th, 31% ²²⁵ Ra+D
1.45E6	⁹³ Zr, ^{113M} Cd, ²⁴³ Am, ²⁴⁴ Pu	3.5E-3	7.5E-1	6.6E2	4.4E-4	7.5E0	1.8E0	4.4E-4	2.8E-1	GI-LLI: 100% ^{93M} Nb: Bone: 56% ^{93M} Nb, 34% ²⁴² Pu
2.07E6	²³⁸ U	8.9E-1	1.1E3	2.9E1	7.5E-1	2.4E3	2.9E2	7.5E-1	1.2E1	Bone: 86% ²²⁶ Ra+D, 7% ²¹⁰ Pb+D
2.41E6	²³¹ Pa	2.4E-2	3.4E1	7.3E-1	2.1E-2	6.8E1	8.7E0	2.1E-2	2.1E-2	Bone: 88% ²²⁶ Ra+D, 7% ²¹⁰ Pb+D
7.23E6	²³² Th	2.5E-4	6.3E-3	3.7E-2	2.3E-4	9.6E-2	3.1E-3	2.3E-4	2.3E-4	Bone: 44% ²³² Th+D, 35% ²²⁴ Ra+D 21% ²³⁸ Th+D

*LST008 Dose Input file

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 9. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.3%/Yr
Path Length: 3 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem								Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	Kidney	
1.4E2	^3H , ^{99}Tc , ^{129}I	1.3E1	2.3E2	2.6E4	1.4E4	5.4E2	8.0E2	9.6E0	1.9E1	GI-LLI: 100% ^{99}Tc
5.4E2	^{14}C	9.0E-3	7.1E4	4.3E4	7.0E4	3.5E5	7.0E4	7.0E4	7.0E4	Bone: 100% ^{14}C
1.2E3	^{93}Mo	8.0E-7	6.4E-1	3.8E0	4.5E-7	4.5E-7	2.4E1	4.5E-7	4.5E-7	Liver: 100% ^{99}Mo
4.5E3	^{41}Ca , ^{79}Se , ^{90}Sr , ^{237}Np	3.1E2	1.4E3	2.1E3	2.7E2	1.2E4	3.8E3	2.7E2	2.7E2	Bone: 96% $^{237}\text{Np}+\text{D}$
1.5E4	^{10}Be , ^{60}Co , ^{159}N	3.0E-2	6.1E1	2.7E1	2.6E-2	3.6E2	1.3E2	2.6E-2	4.0E-2	Bone: ~100% ^{59}Ni
2.2E4	^{87}Rb , ^{226}Ra	1.9E-2	1.7E-1	1.5E-1	1.6E-2	1.2E0	1.8E-1	1.6E-2	3.0E-2	Bone: 38% $^{237}\text{Np}+\text{D}$, 32% ^{229}Th , 17% $^{225}\text{Ra}+\text{D}$
4.4E4	^{135}Cs	1.9E-2	4.0E2	2.3E1	1.4E-2	1.1E3	9.7E2	7.3E2	2.4E-2	Bone: 99% ^{135}Cs
4.8E4	^{107}Pd , ^{126}Sn	2.0E3	2.0E3	5.3E3	1.8E3	1.2E4	2.0E3	1.8E3	1.8E3	Bone: ~100% $^{126}\text{Sn}+\text{D}$
1.1E5	^{166}Ho	4.5E-1	5.6E1	9.0E-1	3.9E-2	1.1E2	8.8E0	3.9E-2	4.5E-2	Bone: 92% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
1.4E5	^{247}Cm	7.9E-1	1.0E2	1.6E0	6.8E-2	2.1E2	1.6E1	6.8E-2	1.2E-1	Bone: 92% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
4.3E5	^{93}Zr , ^{113}M , ^{243}Am , ^{244}Pu	3.3E0	4.6E3	2.5E3	2.8E0	9.2E3	7.3E2	2.8E0	5.9E0	Bone: 92% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
6.2E5	^{238}U	3.9E1	4.9E4	7.9E2	3.2E1	9.9E4	7.8E3	3.2E1	2.4E2	Bone: 92% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
7.2E5	^{231}Pa	2.3E1	3.3E4	4.7E2	2.0E1	6.5E4	5.0E3	2.0E1	2.0E1	Bone: 92% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
2.2E6	^{232}Th	4.6E-4	1.1E-2	6.6E-2	4.1E-4	1.8E-1	5.6E-3	4.1E-4	4.1E-4	Bone: 46% $^{232}\text{Th}+\text{D}$, 34% $^{224}\text{Ra}+\text{D}$, 20% $^{228}\text{Th}+\text{D}$

*LST020 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 10. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.3%/Yr
Path Length: 1 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem							Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	
1.2E2	^3H , ^{99}Tc , ^{129}I	1.3E1	2.4E2	2.6E4	1.4E4	5.4E2	8.1E2	1.9E1	GI-LLI: 100% ^{99}Tc ; Thyroid: 100% ^{129}I
2.5E2	^{14}C	1.3E1	7.4E4	7.0E4	8.7E4	3.7E5	7.4E4	7.4E4	Bone: 100% ^{14}C
4.7E2	^{93}Mo	9.5E-3	7.1E4	4.3E4	7.1E4	3.6E5	7.1E4	7.1E4	Bone: 100% ^{14}C
1.6E3	^{41}Ca , ^{79}Se , ^{90}Sr , ^{237}Np	2.9E2	1.3E3	2.0E3	2.6E2	1.1E4	3.8E3	2.6E2	Bone: 96% $^{237}\text{Np}+\text{D}$
5.0E3	^{10}Be , ^{59}Ni , ^{60}Co	2.0E-1	6.6E1	3.0E1	1.7E-1	4.0E2	1.4E2	1.7E-1	Bone: 98% ^{59}Ni
7.4E3	^{226}Ra	6.1E-2	9.1E0	4.5E-1	5.3E-2	2.1E1	1.7E0	5.3E-2	Bone: 82% $^{226}\text{Ra}+\text{D}$, 9% $^{237}\text{Np}+\text{D}$
7.5E3	^{87}Rb	7.1E-2	3.6E1	7.9E-1	6.2E-2	2.3E1	5.8E0	6.2E-2	Bone: 90% $^{226}\text{Ra}+\text{D}$, 5% $^{210}\text{Pb}+\text{D}$
1.5E4	^{135}Cs	2.1E0	3.5E3	6.7E1	2.0E0	7.3E3	1.5E3	7.3E2	Bone: 79% $^{226}\text{Ra}+\text{D}$, 15% ^{135}Cs
1.6E4	^{107}Pd , ^{126}Sn	2.5E3	6.4E3	6.6E3	2.3E3	2.2E4	3.1E3	2.2E3	Bone: 65% $^{126}\text{Sn}+\text{D}$, 32% $^{226}\text{Ra}+\text{D}$
3.6E4	^{166}Ho	8.3E0	1.2E4	1.7E2	7.5E0	2.4E4	1.9E3	7.5E0	Bone: 92% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
4.8E4	^{247}Cm	2.0E1	1.8E4	3.7E2	1.6E1	3.6E4	3.1E3	1.6E1	Bone: 90% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
1.4E5	^{93}Zr , ^{113}M , ^{243}Am , ^{244}Pu	5.6E1	7.7E4	6.2E3	4.8E1	1.6E5	1.2E4	4.8E1	Bone: 92% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
2.1E5	^{238}U	1.3E2	1.6E5	2.5E3	1.1E2	3.3E5	2.6E4	1.1E2	Bone: 92% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
2.4E5	^{231}Pa	8.2E1	1.1E5	1.6E3	7.1E1	2.2E5	1.8E4	7.1E1	Bone: 92% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$
7.2E5	^{232}Th	2.7E-3	2.6E0	1.5E-1	2.4E-3	5.5E0	4.1E-1	2.4E-3	Bone: 88% $^{226}\text{Ra}+\text{D}$, 4% $^{210}\text{Pb}+\text{D}$

*LST012 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 11. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.3%/Yr
Path Length: 0.3 Mile

Years Since Burial	Peak Nuclides	Skin	Body	GI-LLI	Organ Dose, mrem				Remarks
				Thyroid	Bone	Liver	Lung	Kidney	
1.0E2	³ H, ⁹⁹ Tc, ¹²⁹ I	1.3E1	2.5E2	2.6E4	5.4E2	8.2E2	2.7E1	3.7E1	GI-LLI: 100% ⁹⁹ Tc
1.4E2	¹⁴ C	1.3E1	7.4E4	7.0E4	3.7E5	7.5E4	7.5E4	7.5E4	Bone: 100% ¹⁴ C
2.1E2	⁹³ Mo	1.3E1	7.4E4	7.0E4	3.7E5	7.4E4	7.4E4	7.4E4	Bone: 100% ¹⁴ C
5.4E2	⁹⁰ Sr, ⁴¹ Ca, ⁷⁹ Se, ²³⁷ Np	3.1E2	2.5E4	4.5E3	2.7E2	3.9E3	2.7E2	2.7E2	Bone: 89% ⁹⁰ Sr, 11% ²³⁷ Np+D
1.6E3	¹⁰ Be, ⁵⁹ Ni	6.5E0	8.5E1	6.5E1	6.0E1	1.7E2	6.0E0	6.0E0	Bone: 61% ⁵⁹ Ni, 38% ²³⁷ Np+D
1.7E3	⁶⁰ Co	4.6E0	7.9E1	5.4E1	4.0E0	1.6E2	4.0E0	4.0E0	Bone: 70% ⁵⁹ Ni, 30% ²³⁷ Np+D
2.3E3	²²⁶ Ra	2.2E0	4.2E1	1.2E1	1.9E0	1.5E1	1.9E0	1.9E0	Bone: 64% ²³⁷ Np+D, 31% ²²⁶ Ra+D
2.4E3	⁸⁷ Rb	1.9E0	2.9E2	1.3E1	1.7E0	5.4E1	1.7E0	1.7E0	Bone: 84% ²²⁶ Ra+D, 9% ²³⁷ Np+D
4.5E3	¹³⁵ Cs	3.4E0	5.0E3	9.0E1	2.9E0	1.7E3	7.3E2	2.3E0	Bone: 83% ²²⁶ Ra+D, 11% ¹³⁵ Cs
5.0E3	¹⁰⁷ Pd, ¹²⁶ Sn	2.7E3	8.3E3	7.1E3	2.4E3	3.6E3	2.4E3	2.4E3	Bone: 58% ¹²⁶ Sn+D, 38% ²²⁶ Ra+D
1.1E4	¹⁶⁶ Mn, ¹⁶⁶ Ho	1.3E1	1.8E4	2.5E2	1.1E1	3.5E4	1.1E1	1.2E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
1.5E4	²⁴⁷ Cm	1.5E2	2.6E4	2.6E3	1.2E2	1.1E4	1.2E2	3.3E3	Bone: 71% ²²⁶ Ra+D, 12% ²⁴¹ Am, 9% ²⁴⁵ Cm+D
4.4E4	⁹³ Zr, ¹¹³ M, ²⁴³ Am, ²⁴⁴ Pu	2.1E2	2.1E4	1.5E4	1.6E2	1.0E4	1.6E2	2.1E3	Bone: 56% ²²⁶ Ra+D, 26% ²³⁹ Pu, 12% ²⁴³ Am+D
6.2E4	²³⁸ U	3.9E2	3.3E5	1.0E4	3.3E2	7.1E5	3.1E2	1.9E3	Bone: 96% ²²⁶ Ra+D, 5% ²²⁹ Th
7.3E4	²³¹ Pa	4.2E2	5.6E5	7.7E3	3.5E2	1.1E6	3.5E2	3.5E2	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
2.2E5	²³² Th	8.7E-2	8.7E1	1.4E0	7.5E-2	1.7E2	7.5E-2	7.5E-2	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D

*LST018 Dose Input File
Background: 6000 mrem
ICRP Radiation Limit: 75,000 mrem

TABLE 12. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.3%/Yr
Path Length: 0.1 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem							Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	
1.0E2	³ H, ⁹⁹ Tc, ¹²⁹ I	1.3E1	2.5E2	2.6E4	1.4E4	5.4E2	8.2E2	2.7E1	GI-LLI: 100% ⁹⁹ Tc
1.2E2	¹⁴ C	1.3E1	7.4E4	7.0E4	8.8E4	3.7E5	7.5E4	1.8E1	Bone: 100% ¹⁴ C
1.4E2	⁹³ Mo	1.3E1	7.4E4	7.0E4	8.8E4	3.7E5	7.5E4	1.0E1	Bone: 100% ¹⁴ C
2.5E2	⁴¹ Ca, ⁷⁹ Se, ⁹⁰ Sr, ²³⁷ Np	3.2E2	2.6E7	2.6E6	8.8E4	1.0E8	7.8E4	2.7E2	Bone: 100% ⁹⁰ Sr
5.8E2	⁶⁰ Co	1.8E1	1.5E2	1.4E2	1.5E1	2.6E3	3.0E2	1.5E1	Bone: 64% ⁶³ Ni, 25% ²³⁷ Np+D
6.0E2	¹⁰⁸ Be, ⁵⁹ Ni	1.7E1	1.8E2	1.5E2	1.5E1	3.1E3	3.6E2	1.5E1	Bone: 66% ⁶³ Ni, 21% ²³⁷ Np+D, 13% ⁵⁹ Ni
8.4E2	²²⁶ Ra	1.2E1	1.9E2	9.8E1	1.0E1	1.3E3	2.3E2	1.0E1	Bone: 33% ²³⁷ Np+D, 31% ⁵⁹ Ni, 29% ⁶³ Ni
9.9E2	⁸⁷ Rb	9.8E0	6.1E2	5.6E1	8.3E0	1.6E3	1.3E2	8.4E0	Bone: 68% ²²⁶ Ra+D, 22% ²³⁷ Np+D
1.6E3	¹³⁵ Cs	6.9E0	5.2E3	1.2E2	6.0E0	1.1E4	1.7E3	7.3E2	Bone: 83% ²²⁶ Ra+D, 10% ¹³⁵ Cs
1.8E3	¹²⁶ Sn, ¹⁰⁷ Pd	2.7E3	9.1E3	7.3E3	2.5E3	2.9E4	4.7E3	3.1E3	Bone: 56% ¹²⁶ Sn+D, 37% ²²⁶ Ra+D
3.8E3	¹⁶⁶ Mn, ¹⁶⁶ Ho	1.5E1	2.0E4	2.9E2	1.3E1	4.1E4	3.2E3	1.3E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
5.1E3	²⁴⁷ Cm	3.0E2	3.1E4	5.5E3	2.4E2	9.4E4	2.0E4	2.4E2	Bone: 59% ²²⁶ Ra+D, 19% ²⁴⁵ Cm+D, 15% ²⁴¹ Am
1.5E4	⁹³ Zr, ²⁴³ Am, ²⁴⁴ Pu, ^{113M} Cd	3.8E3	1.2E5	6.9E4	3.2E3	4.3E5	1.1E5	3.2E3	Bone: 41% ²⁴³ Am+D, 43% ²²⁶ Ra+D, 9% ²³⁹ Pu
2.1E4	²³⁸ U	1.4E2	8.2E4	1.8E3	1.2E2	1.7E5	1.3E4	1.2E2	Bone: 87% ²²⁶ Ra+D, 5% ²³⁴ U
2.4E4	²³¹ Pa	1.1E2	7.5E4	2.5E3	9.1E1	1.5E5	1.2E4	9.1E1	Bone: 88% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
7.3E4	²³² Th	8.5E-1	7.6E2	1.1E1	7.4E-1	1.5E3	1.2E2	7.4E-1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D

*LST013 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 13. Summary of 50-Year Accumulated Dose to Maximum Individual*
 Time of Initial Release After Yr-2000: 100 Yr
 Leach Rate: 0.3%/Yr
 Path Length: 0 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem								Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	Kidney	
1.0E2	All	1.2E7	9.7E8	1.3E8	9.7E6	3.6E9	1.6E8	9.6E6	6.7E7	Bone: 96% ⁹⁰ Sr

*LST017 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 14. Summary of 50-Year Accumulated Dose to Maximum Individual*
 Time of Initial Release After Yr-2000: 100 Yr
 Leach Rate: 0.03%/Yr
 Path Length: 10 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem								Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	Kidney	
2.5E2	³ H, ⁹⁹ Tc, ¹²⁹ I	1.3E0	2.3E1	2.6E3	1.4E3	5.4E1	8.0E0	7.5E-1	1.7E0	GI-LLI: 100% ⁹⁹ Tc
1.6E3	¹⁴ C	8.0E-4	6.2E3	3.8E3	6.2E3	3.1E4	6.2E3	---	---	Bone: 100% ¹⁴ C
3.7E3	⁹³ Mo	6.5E-4	4.8E3	2.9E3	4.8E3	2.4E4	4.8E3	2.5E-8	2.5E-8	Bone: 100% ¹⁴ C
1.5E4	⁴¹ Ca, ⁷⁹ Se, ²³⁷ Np, ⁹⁰ Sr	3.0E1	1.3E2	2.0E2	2.6E1	1.1E3	3.4E2	2.6E1	2.6E1	Bone: 96% ²³⁷ Np+D
4.9E4	¹⁰ Be, ⁵⁹ Ni, ⁶⁰ Co	8.6E-3	4.6E0	2.0E0	7.2E-3	2.7E1	9.4E0	7.2E-3	1.9E-2	Bone: 98% ⁵⁹ Ni
7.3E4	²²⁶ Ra, ⁸⁷ Rb	5.3E-3	4.2E-2	4.6E-2	4.4E-3	4.0E-1	2.3E-2	4.4E-3	1.5E-2	Bone: 55% ²²⁹ Th, 30% ²²⁵ Ra+D
1.4E5	¹³⁵ Cs	4.2E-3	3.8E1	2.3E0	3.3E-3	1.0E2	9.7E1	7.2E1	1.1E-2	Bone, Liver, Lung: 100% ¹³⁵ Cs
1.6E5	¹⁰⁷ Pd, ¹²⁶ Sn	9.0E1	9.3E1	2.8E2	8.3E1	5.3E2	9.8E1	8.0E1	8.0E1	Bone: 100% ¹²⁶ Sn+D
3.6E5	¹⁶⁶ Ho	1.5E-3	1.6E-2	2.0E-2	1.3E-3	1.0E-1	4.4E-3	1.3E-3	4.2E-3	Bone: 60% ²²⁹ Th, 33% ²²⁵ Ra+D
4.8E5	²⁴⁷ Cm	9.9E-4	9.8E-3	1.2E-2	7.9E-4	1.1E-1	3.7E-3	7.9E-4	2.5E-3	Bone: 59% ²²⁹ Th, 32% ²²⁵ Ra+D
1.4E6	⁹³ Zr, ¹¹³ M, ²⁴³ Am, ²⁴⁴ Pu	2.5E-3	5.5E-1	4.3E2	3.4E-4	5.1E0	1.2E0	3.4E-4	1.9E-1	GI-LLI: 100% ⁹³ M, ⁹³ Nb, 36% ²⁴² Pu Bone 55% ²²⁹ Th, 32% ²²⁵ Ra+D
2.1E6	²³⁸ U	9.0E-1	1.2E3	2.4E1	7.3E-1	2.4E3	2.2E2	7.3E-1	7.9E0	Bone: 90% ²²⁶ Ra+D, 5% ²¹⁰ Pb+D
2.4E6	²³¹ Pa	2.4E-2	3.3E1	5.7E-1	2.1E-2	6.5E1	6.2E0	2.1E-2	2.4E-2	Bone: 91% ²²⁶ Ra+D, 5% ²¹⁰ Pb+D
7.2E6	²³² Th	2.5E-4	6.2E-3	3.7E-2	2.2E-4	9.5E-2	3.1E-3	2.2E-4	2.2E-4	Bone: 44% ²³² Th+D, 35% ²²⁴ Ra+D, 21% ²²⁸ Th+D

*LST009 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 15. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.003%/Yr
Path Length: 10 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem					Remarks		
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	Kidney
2.5E2	³ H, ⁹⁹ Tc, ¹²⁹ I	1.3E-1	2.3E0	2.6E2	1.4E2	5.4E0	8.0E0	7.5E-2	1.7E-1
1.7E3	¹⁴ C	8.0E-5	6.1E2	3.7E2	6.1E2	3.0E3	6.1E2	---	---
3.9E3	⁹³ Nb	6.0E-5	4.7E2	2.9E2	4.7E2	2.4E3	4.7E2	2.4E-9	2.4E-9
1.5E4	⁴¹ Ca, ⁷⁹ Se, ²³⁷ Np	3.3E0	1.4E2	9.7E1	1.3E2	7.5E2	1.6E2	2.9E0	2.9E0
3.1E4	⁹⁰ Sr	2.9E0	2.9E1	2.9E1	2.0E1	2.2E2	4.8E1	2.5E0	2.5E0
4.8E4	¹⁰ Be, ⁵⁹ Ni	7.7E-3	5.0E-1	2.6E-1	6.4E-3	3.2E0	9.6E-1	6.4E-3	1.8E-2
6.5E4	⁶⁰ Co	6.2E-3	4.4E-1	2.3E-1	5.1E-3	2.8E0	8.3E-1	5.1E-3	1.6E-2
7.3E4	²²⁶ Ra	5.7E-3	4.1E-1	2.2E-1	4.7E-3	2.7E0	7.8E-1	4.7E-3	1.6E-2
8.9E4	⁸⁷ Rb	5.1E-3	4.3E-2	5.2E-2	4.1E-3	4.5E-1	1.3E-2	4.1E-3	1.4E-2
1.4E5	¹³⁵ Cs	4.0E-3	3.9E0	2.7E-1	3.3E-3	1.1E1	9.7E0	7.2E0	1.1E-2
1.6E5	¹⁰⁷ Pd, ¹²⁶ Sn	9.0E0	1.3E1	2.8E1	8.3E0	6.4E1	1.9E1	1.5E1	8.0E0
3.8E5	¹⁶⁶ Mn	1.4E-3	1.5E-2	1.8E-2	1.2E-3	1.6E-1	3.9E-3	1.2E-3	4.0E-3
4.8E5	²⁴⁷ Cm	9.1E-4	9.3E-3	1.1E-2	7.8E-4	9.9E-2	2.6E-3	7.8E-4	2.6E-3
1.5E6	⁹³ Zr, ¹¹³ M, ²⁴³ Am, ²⁴⁴ Pu	4.3E-4	2.8E-1	4.9E1	1.8E-4	1.0E0	1.8E-1	1.8E-4	2.1E-2
2.1E6	²³⁸ U	7.1E-1	1.0E3	1.7E1	6.3E-1	2.0E3	1.9E2	6.3E-1	1.6E0
2.4E6	²³¹ Pa	2.2E-2	3.2E1	5.3E-1	1.9E-2	6.3E1	5.8E0	1.9E-2	2.2E-2
7.2E6	²³² Th	9.3E-5	2.3E-3	1.4E-2	8.2E-5	3.6E-2	1.2E-3	8.2E-5	8.2E-5

*LST010 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 16. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.0003%/Yr
Path Length: 10 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem								Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	Kidney	
2.5E2	^3H , ^{99}Tc , ^{129}I	1.3E-2	2.3E-1	2.6E-1	1.4E-1	5.4E-1	8.0E-1	7.5E-3	1.7E-2	GI-LLI: 100% ^{99}Tc
2.0E3	^{14}C	1.3E-2	5.9E1	6.1E1	7.3E1	3.0E2	6.0E1	7.5E-3	1.7E-2	Bone: 100% ^{14}C
4.0E3	^{93}Mo	1.3E-2	4.5E1	5.3E1	5.9E1	2.2E2	4.6E1	7.5E-3	1.7E-2	Bone: 100% ^{14}C
1.6E4	^{41}Ca , ^{79}Se , ^{237}Np	3.0E-1	1.2E1	3.3E1	2.5E+1	6.4E1	1.5E1	2.6E-1	2.7E-1	Bone: 82% ^{14}C , 17% $^{237}\text{Np}+\text{D}$
5.0E4	^{10}Be , ^{59}Ni	3.0E-1	1.5E0	2.5E1	1.5E+1	1.3E1	3.7E0	2.6E-1	2.7E-1	Bone: 84% $^{237}\text{Np}+\text{D}$
7.4E4	^{87}Rb	2.9E-1	1.3E0	2.4E1	1.4E1	1.2E1	3.1E0	2.5E-1	2.6E-1	Bone: 90% $^{237}\text{Np}+\text{D}$
1.5E5	^{135}Cs	2.9E-1	1.4E0	2.0E1	1.4E1	1.2E1	3.2E0	9.6E-1	2.6E-1	Bone: 84% $^{237}\text{Np}+\text{D}$, 8% ^{135}Cs
1.6E5	^{107}Pd , ^{126}Sn	1.2E0	2.3E0	2.2E1	1.5E0	1.7E1	4.1E0	1.8E0	1.1E0	Bone: 59% $^{237}\text{Np}+\text{D}$, 30% $^{126}\text{Sn}+\text{D}$
1.8E5	^{90}Sr	1.0E0	2.2E0	2.0E1	1.5E1	1.7E1	3.9E0	1.7E0	9.6E-1	Bone: 62% $^{237}\text{Np}+\text{D}$, 27% $^{126}\text{Sn}+\text{D}$
2.2E5	^{60}Co	8.8E-1	1.9E0	1.8E1	1.5E1	1.5E1	3.5E0	1.5E0	8.1E-1	Bone: 65% $^{237}\text{Np}+\text{D}$, 24% $^{126}\text{Sn}+\text{D}$
2.4E5	^{226}Ra	7.7E-1	1.7E0	3.3E0	7.1E-1	1.5E1	2.8E0	1.4E0	7.1E-1	Bone: 69% $^{237}\text{Np}+\text{D}$, 21% $^{126}\text{Sn}+\text{D}$
4.8E5	^{247}Cm	9.7E-2	1.2E-1	7.9E-1	8.9E-2	7.6E-1	1.8E-1	8.7E-2	9.0E-2	Bone: 75% $^{126}\text{Sn}+\text{D}$, 8% $^{225}\text{Ra}+\text{D}$
5.3E5	^{166}Ho	1.5E-3	1.5E-2	1.8E-2	1.2E-3	1.6E-1	4.0E-3	1.2E-3	4.1E-3	Bone: 59% ^{229}Th , 33% $^{225}\text{Ra}+\text{D}$
1.4E6	^{93}Zr , ^{244}Pu	9.8E-5	7.2E-2	4.9E0	6.4E-5	2.0E-1	2.6E-2	6.4E-5	2.2E-3	GI-LLI: 100% $^{93}\text{M}_{\text{Nb}}$ Bone: 66% $^{226}\text{Ra}+\text{D}$, 16% $^{93}\text{M}_{\text{Nb}}$
1.6E6	^{113}Cd , ^{243}Am	4.2E-4	5.4E-1	4.4E0	3.5E-4	1.1E0	1.1E-1	3.5E-4	1.9E-3	GI-LLI: 100% $^{93}\text{M}_{\text{Nb}}$ Bone: 88% $^{226}\text{Ra}+\text{D}$
2.1E6	^{238}U	1.8E-1	2.5E2	4.3E0	1.5E-1	5.0E2	4.7E1	1.5E-1	2.5E-1	Bone: 91% $^{226}\text{Ra}+\text{D}$
2.4E6	^{231}Pa	1.2E-1	1.8E2	2.9E0	1.1E-1	3.5E2	3.2E1	1.1E-1	1.1E-1	Bone: 91% ^{226}Ra
7.3E6	^{232}Th	9.3E-6	2.4E-4	1.4E-3	8.2E-6	3.6E-3	1.2E-4	8.2E-6	8.2E-6	Bone: 45% $^{232}\text{Th}+\text{D}$, 35% $^{224}\text{Ra}+\text{D}$, 21% $^{228}\text{Th}+\text{D}$

*LST011 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 17. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.03%/Yr
Path Length: 0.1 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem							Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	
1.0E2	³ H, ⁹⁹ Tc, ¹²⁹ I	1.3E0	7.4E3	7.0E3	8.8E3	3.7E4	7.5E3	7.5E3	Bone: 100% ¹⁴ C
1.4E2	¹⁴ C, ⁹³ Mo	1.3E0	7.4E3	7.0E3	8.8E3	3.7E4	7.5E3	7.5E3	Bone: 100% ¹⁴ C
2.6E2	⁴¹ Ca, ⁷⁹ Se, ⁹⁰ Sr, ²³⁷ Np	3.1E1	2.0E6	2.0E5	8.8E3	7.8E6	7.8E3	7.8E3	Bone: 99% ⁹⁰ Sr
6.2E2	¹⁰ Be, ⁵⁹ Ni	3.0E1	7.6E3	7.0E3	8.4E3	3.8E4	7.5E3	7.5E3	Bone: 92% ¹⁴ C
8.6E2	²²⁶ Ra	3.0E1	7.0E3	6.9E3	8.2E3	3.5E4	7.3E3	7.3E3	Bone: 96% ¹⁴ C
1.6E3	¹³⁵ Cs	3.1E1	6.9E3	6.5E3	7.6E3	3.3E4	6.9E3	6.9E3	Bone: 93% ¹⁴ C
1.8E3	¹²⁶ Sn, ¹⁰⁷ Pd	3.0E2	7.2E3	7.2E3	7.8E3	3.5E4	7.1E3	7.1E3	Bone: 88% ¹⁴ C
2.2E3	⁶⁰ Co	3.0E2	7.8E3	7.0E3	7.4E3	3.7E4	7.0E3	7.0E3	Bone: 78% ¹⁴ C, 13% ²²⁶ Ra+D
2.5E3	⁸⁷ Rb	3.0E2	8.1E3	6.9E3	7.3E3	3.7E4	6.9E3	6.9E3	Bone: 76% ¹⁴ C, 16% ²²⁶ Ra+D
3.8E3	^{166m} Ho	2.7E2	8.5E3	8.3E2	2.5E2	1.8E4	1.7E3	2.4E2	Bone: 84% ²²⁶ Ra+D, 9% ¹²⁶ Sn+D
5.1E3	²⁴⁷ Cm	3.3E1	6.8E3	6.0E2	2.5E1	1.7E4	2.6E3	2.5E1	Bone: 73% ²²⁶ Ra+D, 11% ²²⁵ Cm+D, 8% ²⁴¹ Am
1.5E4	²⁴³ Am, ⁹³ Zr, ²⁴⁴ Pu, ^{113m} Cd	4.3E2	4.7E4	8.0E3	3.7E2	1.1E5	1.7E4	3.7E2	Bone: 72% ²²⁶ Ra+D, 18% ²⁴³ Am+D
2.2E4	²³⁸ U	6.9E1	8.1E4	1.3E3	5.9E1	1.6E5	1.2E4	5.9E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
2.5E4	²³¹ Pa	6.0E1	6.9E4	1.3E3	5.1E1	1.4E5	1.1E4	5.1E1	Bone: 91% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
7.4E4	²³² Th	7.1E-1	9.4E2	1.4E1	6.2E-1	1.9E3	1.5E2	6.2E-1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D

*LST014 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 18. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.003%/Yr
Path Length: 0.1 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem							Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	
1.0E2	³ H, ⁹⁹ Tc, ¹²⁹ I	3.0E0	7.3E2	7.1E2	8.6E2	3.7E3	7.6E2	7.6E2	Bone: 96% ¹⁴ C
2.5E2	⁹⁰ Sr	3.2E0	2.6E5	2.6E4	8.4E2	1.0E6	7.5E2	7.5E2	Bone: 99% ⁹⁰ Sr
5.0E2	¹⁴ C, ⁹³ Mo	3.0E0	1.4E3	7.7E2	8.5E2	6.5E3	7.6E2	7.6E2	Bone: 55% ¹⁴ C, 43% ⁹⁰ Sr
6.0E2	⁶⁰ Co, ⁷⁹ Se, ⁴¹ Ca, ²³⁷ Np, ¹⁰ Be, ⁵⁹ Ni	3.0E0	7.7E2	7.1E2	8.4E2	3.8E3	7.5E2	7.5E2	Bone: 91% ¹⁴ C, 6% ⁵⁰ Sr
1.2E3	²²⁶ Ra, ⁸⁷ Rb	3.1E0	6.9E2	6.7E2	7.9E2	3.4E3	7.0E2	7.0E2	Bone: 95% ¹⁴ C
1.7E3	¹³⁵ Cs	3.0E1	7.3E2	7.2E2	7.8E2	3.5E3	7.2E2	7.2E2	Bone: 87% ¹⁴ C, 5% ¹²⁶ Sn+D
1.5E3	¹⁰⁷ Pd, ¹²⁶ Sn	3.0E1	7.4E2	7.1E2	7.6E2	3.6E3	7.0E2	7.0E2	Bone: 84% ¹⁴ C, 8% ²²⁶ Ra+D 5% ¹²⁶ Sn+D
3.8E3	¹⁶⁶ Mn	3.0E1	1.5E3	6.4E2	6.4E2	4.6E3	7.1E2	7.1E2	Bone: 51% ¹⁴ C, 39% ²²⁶ Ra+D
5.3E3	²⁴⁷ Cm	3.4E1	2.7E3	6.7E2	6.6E2	7.0E3	9.8E2	3.7E1	Bone: 58% ²²⁶ Ra+D, 29% ¹⁴ C
1.5E4	⁹³ Zr, ²⁴³ Am, ²⁴⁴ Pu	8.5E1	2.3E4	1.4E3	3.3E2	4.7E4	4.7E3	2.8E1	Bone: 86% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D, 4% ²⁴³ Am+D
2.2E4	²³⁸ U	8.1E1	4.4E4	1.5E3	2.6E2	9.2E4	7.7E3	6.9E1	Bone: 90% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
2.5E4	²³¹ Pa	8.5E1	5.6E4	1.5E3	2.4E2	1.1E5	9.3E3	8.1E1	Bone: 91% ²²⁶ Ra+D, 5% ²¹⁰ Pb+D
3.1E4	¹¹³ Cd	9.6E1	8.2E4	1.8E3	2.2E2	1.7E5	1.3E4	9.0E1	Bone: 92% ²²⁶ Ra+D, 5% ²¹⁰ Pb+D
7.6E4	²³² Th	1.2E1	1.6E4	2.2E2	1.0E1	3.2E4	2.5E3	1.0E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D

*LST015 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 19. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.0003%/Yr
Path Length: 0.1 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem								Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	Kidney	
1.0E2	³ H, ⁹⁹ Tc, ¹²⁹ I	3.0E-1	4.8E1	5.5E1	6.0E1	2.4E2	5.0E1	5.0E1	5.0E1	Bone: 95% ¹⁴ C
2.5E2		⁹⁰ Sr	3.2E-1	2.5E4	2.5E3	6.0E1	1.0E5	5.0E1	5.0E1	5.0E1
5.9E2	⁶⁰ Co	3.0E-1	5.6E1	5.6E1	6.0E1	2.8E2	5.1E1	5.1E1	5.1E1	Bone: 83% ¹⁴ C, 12% ⁹⁰ Sr
6.4E2	⁵⁹ Ni	3.0E-1	6.0E1	5.6E1	6.0E1	2.5E2	5.1E1	5.1E1	5.1E1	Bone: 91% ¹⁴ C, 4% ²³⁷ Np+D, 4% ⁹⁰ Sr
8.2E2	⁸⁷ Rb	3.0E-1	5.0E1	5.6E1	6.0E1	2.5E2	5.1E1	5.1E1	5.1E1	Bone: 91% ¹⁴ C, 4% ²³⁷ Np+D, 4% ⁹⁰ Sr
1.6E3	¹³⁵ Cs	3.0E-1	4.8E1	5.5E1	6.0E1	2.4E2	5.2E1	5.2E1	5.2E1	Bone: 94% ¹⁴ C, 4% ²³⁷ Np+D
1.8E3	¹⁰⁷ Pd, ¹²⁶ Sn	3.0E0	5.1E1	6.3E1	6.3E1	2.6E2	5.4E1	5.4E1	5.4E1	Bone: 89% ¹⁴ C, 6% ¹²⁶ Sn+D
4.0E3	¹⁶⁶ Mn, ¹⁴ C, ⁹⁹ Mo	3.0E0	5.1E1	6.2E1	6.3E1	2.6E2	5.4E1	5.4E1	5.4E1	Bone: 89% ¹⁴ C, 6% ¹²⁶ Sn+D
5.1E3	⁷⁹ Se, ¹⁰ Be, ²²⁶ Ra, ⁴¹ Ca, ²³⁷ Np, ²⁴⁷ Cm	3.0E0	4.6E1	5.9E1	5.8E1	2.3E2	5.0E1	5.0E1	5.0E1	Bone: 88% ¹⁴ C, 7% ¹²⁶ Sn+D
1.6E4	²⁴³ Am, ²⁴⁴ Pu, ⁹³ Zr, ¹¹³ Mn, ¹¹³ Cd	7.7E0	2.4E3	1.4E2	3.2E1	5.1E3	4.8E2	7.5E0	1.3E1	Bone: 87% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
2.3E4	²³⁸ U	6.1E0	5.3E3	1.5E2	2.4E1	1.1E4	8.8E2	5.2E0	1.2E1	Bone: 91% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
2.6E4	²³¹ Pa	6.5E0	6.4E3	1.5E2	2.3E1	1.3E4	1.0E3	5.7E0	1.3E1	Bone: 91% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
7.9E4	²³² Th	2.1E1	2.8E4	4.1E2	3.2E1	5.5E4	4.3E3	1.8E1	2.1E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D

*LST016 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 20. Summary of 50-Year Accumulated Dose to Maximum Individual*
Time of Initial Release After Yr-2000: 100 Yr
Leach Rate: 0.0003%/Yr
Path Length: 1 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem							Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	
1.2E2	³ H, ⁹⁹ Tc, ¹²⁹ I	3.0E-1	4.8E1	5.5E1	6.0E1	2.4E2	5.1E1	5.1E1	Bone: 95% ¹⁴ C
4.1E3	¹⁴ C, ⁹³ Mo	3.0E-1	4.8E1	5.5E1	6.0E1	2.4E2	5.1E1	5.1E1	Bone: 95% ¹⁴ C
5.1E3	⁴¹ Ca, ⁷⁹ Se, ²³⁷ Np	3.0E-1	4.3E1	5.2E1	5.6E1	2.2E2	4.6E1	4.6E1	Bone: 94% ¹⁴ C
5.2E3	¹⁰ Be, ⁵⁹ Ni	3.0E-1	4.3E1	5.2E1	5.6E1	2.2E2	4.6E1	4.6E1	Bone: 94% ¹⁴ C
1.0E4	²²⁶ Ra	3.0E-1	2.4E1	4.0E1	3.7E1	1.2E2	2.7E1	2.7E1	Bone: 90% ¹⁴ C, 9% ²³⁷ Np+D
1.1E4	⁸⁷ Rb	3.1E-1	3.6E1	3.9E1	3.4E1	1.4E2	2.6E1	2.6E1	Bone: 71% ¹⁴ C, 19% ²²⁶ Ra+D
1.8E4	¹³⁵ Cs, ¹⁰⁷ Pd, ¹²⁶ Sn	2.7E0	8.9E1	3.9E1	2.5E1	2.3E2	2.8E1	3.2E0	Bone: 63% ²²⁶ Ra+D, 19% ¹⁴ C, 6% ¹²⁶ Sn+D
5.0E4	²⁴⁷ Cm	2.7E0	1.1E3	4.3E1	1.7E1	2.2E3	1.8E2	3.1E0	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
1.5E5	⁹³ Zr, ²⁴³ Am, ²⁴⁴ Pu	1.0E1	1.2E4	2.0E2	2.0E1	2.5E4	1.9E3	9.4E0	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
2.0E5	¹⁶⁶ Mn	2.1E1	3.0E4	4.1E2	3.2E1	5.7E4	4.5E3	1.9E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
2.1E5	²³⁸ U	2.4E1	3.1E4	4.5E2	3.4E1	6.3E4	4.9E3	2.1E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
2.4E5	²³¹ Pa	2.9E1	4.1E4	5.7E2	4.0E1	8.2E4	6.5E3	2.7E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
3.1E5	^{113m} Cd	3.4E1	4.9E4	6.7E2	4.5E1	9.7E4	7.7E3	3.1E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D
7.3E5	²³² Th	1.4E1	2.0E4	2.7E2	1.2E1	3.9E4	3.1E3	1.2E1	Bone: 92% ²²⁶ Ra+D, 4% ²¹⁰ Pb+D

*LST019 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 21. Summary of 50-Year Accumulated Dose to Maximum Individual*
 Time of Initial Release After Yr-2000: 10,000
 Leach Rate: 0.3%/Yr
 Path Length: 10 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem								Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	Kidney	
1.0E4	^3H , ^{99}Tc , ^{129}I	1.3E1	2.2E2	2.5E4	1.4E4	5.2E2	7.7E2	7.5E0	1.7E1	GI-LLI: 100; ^{99}Tc
1.1E4	^{14}C	2.5E-3	1.9E4	1.1E4	1.9E4	9.4E4	1.9E4	1.9E4	1.9E4	Bone: 100; ^{14}C
1.4E4	^{93}Mo	2.1E-7	1.7E-1	1.0E0	1.2E-7	1.2E-7	6.1E0	1.2E-7	1.2E-7	Liver: 100; ^{93}Mo
2.5E4	^{90}Sr , ^{41}Ca , ^{79}Se , ^{237}Np	3.5E2	1.4E3	2.2E3	3.0E2	1.3E4	3.4E3	3.0E2	3.0E2	Bone: 97; $^{237}\text{Np}+\text{D}$
5.8E4	^{59}Ni , ^{10}Be , ^{60}Co	8.9E-3	4.2E1	1.8E1	7.1E-3	2.5E2	8.7E1	7.1E-3	2.1E-2	Bone: 100; ^{59}Ni
8.2E4	^{87}Rb , ^{226}Ra	5.8E-3	7.2E-2	4.8E-2	4.8E-3	3.6E-1	1.2E-1	4.8E-3	1.7E-2	Bone: 53; ^{229}Th , 29; $^{225}\text{Ra}+\text{D}$, 15; ^{233}U
1.6E5	^{135}Cs	6.5E-3	3.3E2	1.9E1	3.8E-3	8.7E2	8.4E2	6.3E2	1.3E2	Bone: 100; ^{135}Cs
1.7E5	^{126}Sn , ^{107}Pd	7.0E2	7.6E2	2.3E3	6.7E2	4.3E3	8.0E2	6.5E2	6.5E2	Bone: 100; $^{126}\text{Sn}+\text{D}$
4.9E5	^{247}Cm	1.3E-3	1.1E-2	1.4E-2	1.1E-3	1.2E-1	7.0E-3	1.1E-3	3.1E-3	Bone: 55; ^{223}Th , 30; $^{225}\text{Ra}+\text{D}$
1.5E6	^{244}Pu , ^{93}Zr , $^{113\text{M}}\text{Cd}$, ^{243}Am	3.6E-3	8.6E-1	6.6E2	5.2E-4	7.6E0	1.8E0	5.2E-4	2.9E-1	Bone: 55; $^{93\text{M}}\text{Nb}$, 35; ^{242}Pu
2.1E6	^{238}U	1.2E0	1.7E3	2.9E1	1.0E0	3.4E3	3.0E2	1.0E0	9.7E0	Bone: 91; $^{226}\text{Ra}+\text{D}$, 5; $^{210}\text{Pb}+\text{D}$
7.2E6	^{232}Th	2.5E-4	6.2E-3	3.7E-2	2.2E-4	9.5E-2	3.1E-3	2.2E-4	2.2E-4	Bone: 44; $^{232}\text{Th}+\text{D}$, 35; $^{224}\text{Ra}+\text{D}$, 21; $^{228}\text{Th}+\text{D}$

*LST004 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 22. Summary of 50-Year Accumulated Dose to Maximum Individual*
 Time of Initial Release After Yr-2000: 100,000
 Leach Rate: 0.3%/Yr
 Path Length: 10 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem								Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	Kidney	
1.0E5	^3H , ^{99}Tc , ^{129}I	1.3E1	1.7E2	1.8E4	1.4E4	3.9E2	5.8E2	7.5E0	1.7E1	GI-LLI: 100; ^{99}Tc
1.01E5	^{14}C	4.6E-8	3.5E-1	2.1E-1	3.5E-1	1.8E0	3.5E-1	3.5E-1	3.5E-1	Bone: 100; ^{14}C
1.2E5	^{90}Sr , ^{41}Ca , ^{79}Se , ^{237}Np	3.3E2	1.0E3	1.9E3	2.9E2	1.3E4	2.2E3	2.9E2	2.9E2	Bone: 98; $^{237}\text{Np}+\text{D}$
1.5E5	^{59}Ni , ^{60}Co , ^{10}Be	5.9E-3	1.9E1	8.5E0	5.0E-3	1.1E2	4.0E1	5.0E-3	1.8E-2	Bone: 100; ^{59}Ni
2.4E5	^{135}Cs	6.4E-3	3.3E2	1.9E1	3.6E-3	8.7E2	8.4E2	6.1E2	1.2E-2	Bone, Liver: 100; ^{135}Cs
2.6E5	^{126}Sn , ^{107}Pd	3.9E2	4.1E2	1.4E3	3.6E2	2.3E3	4.7E2	3.5E2	3.5E2	Bone: 100; $^{126}\text{Sn}+\text{D}$
2.17E6	^{238}U	1.1E0	1.4E3	2.6E1	9.5E-1	2.9E3	2.5E2	9.5E-1	9.9E0	Bone: 90; $^{226}\text{Ra}+\text{D}$, 5; $^{210}\text{Pb}+\text{D}$

*LST005 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

TABLE 23. Summary of 50-Year Accumulated Dose to Maximum Individual*
 Time of Initial Release After Yr-2000: 1,000,000
 Leach Rate: 0.3%/Yr
 Path Length: 10 Mile

Years Since Burial	Peak Nuclides	Organ Dose, mrem								Remarks
		Skin	Body	GI-LLI	Thyroid	Bone	Liver	Lung	Kidney	
1.0E6	^3H , ^{99}Tc , ^{129}I	1.3E1	2.7E1	9.8E2	1.4E4	3.2E1	4.1E1	7.5E0	1.6E1	Thyroid: 100 ^{129}I
1.01E6	^{41}Ca , ^{79}Se , ^{90}Sr , ^{237}Np	2.5E2	5.9E2	1.3E3	2.2E2	9.3E3	1.0E3	2.2E2	2.2E2	Bone: 100 $^{237}\text{Np}+\text{D}$
1.05E6	^{10}Be , ^{60}Co , ^{59}Ni	4.2E-3	2.6E-2	2.5E-2	3.4E-3	2.2E-1	2.2E-2	3.4E-3	1.2E-2	Bone: 38 ^{229}Th , 21 ^{59}Ni , 21 $^{225}\text{Ra}+\text{D}$, 20 ^{233}U
1.14E6	^{135}Cs	4.9E-3	2.7E2	1.6E1	2.7E-3	7.2E2	6.5E2	5.0E2	9.2E-3	Bone: 100 ^{135}Cs
1.16E6	^{107}Pd , ^{126}Sn	7.5E-1	5.3E0	4.3E2	7.3E-1	4.8E0	7.0E1	7.0E-1	7.1E-1	GI-LLI: 100 ^{107}Pd
3.07E6	^{238}U	5.6E-1	5.7E2	1.3E1	4.6E-1	9.3E2	1.1E2	4.6E-1	8.8E0	Bone: 85 $^{226}\text{Ra}+\text{D}$, 7 $^{210}\text{Pb}+\text{D}$

*LST006 Dose Input File

Background: 6000 mrem

ICRP Radiation Limit: 75,000 mrem

FIGURE 3. The Effect of Path Length on the Bone Dose Versus Time Curve

FIGURE 4. The Effect of Leach Rate on the Bone Dose Versus Time Curve

the ^{226}Ra dose is so spread out in time from the differences in migration rates between it and its precursors that decreases in leach rate have no effect for the situation shown in Figure 4. Thus, decreasing the leach rate of the waste has great value for reducing the dose from single nuclides but has somewhat less value for reducing the dose from nuclides created predominantly by decay during nuclide transport through the soil.

The results from Tables 8, 21, 22, and 23, showing the effect of the

time of initial release from the disposal site on the doses at various times after release, were given in the previous quarterly report.⁽²⁾

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1. Oak Ridge National Laboratory, Aqueous Fuel Reprocessing Quarterly Report for Period Ending December 31, 1973, USAEC Report ORNL-TM-4488, 1964.
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3.0 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 distribution of transuranic elements in a piece of high-exposure PWR fuel cladding is being determined. Initial results are as expected, with significant fuel incorporation in the internal corrosion product layer and essentially no fuel diffusion into the metal. A substantial amount of fission product cesium was associated with the corrosion product layer.

A small piece of the high-exposure PWR cladding was decontaminated by a 45-min exposure in 600°C Ar-HF followed by an aqueous fluoride solution treatment. Decontamination factors in excess of 10^4 were achieved with removal of about a 55- μ m surface layer.

Corrosion by the molten (Li, Na, Zr) F salt system on bare and oxidized Zircaloy surfaces has been examined by Battelle-Columbus Laboratory. Briefly, they found the presence of hafnium in the zirconium salt about doubled the corrosion rate of the Zircaloy. The sparging of the melt with HF increased corrosion rate of Zircaloy about sixfold. Nearly parabolic corrosion rates were observed on Zircaloy with unsparged molten salt while nearly linear rates were ob-

An alloy of Zircaloy-4, 10 wt% 304 stainless steel - 5 wt% Inconel[®] 718 stainless steel - 5 wt% Inconel 718 has been melted by the U.S. Bureau of Mines Inductoslag process into an ingot ~ 10 cm diam x ~ 14 kg (Heat

Number USBM 443). Inductoslag melting of these eutectic alloys requires use of the 48 wt% CaF_2 - 52 wt% MgF_2 slag. Melting rates are slow, and some unmelted material remains in the ingot. Surface finish and ingot soundness are adequate for direct storage.

Heat Number USBM 460 was melted by Inductoslag process. This heat demonstrates the capability of the process to remelt a) alloy CF3 (304 SS casting grade), b) a 2 to 1 (by weight) mix of CF3 and Inconel 718, and c) pure Inconel 718. An attempt to melt using the pure CaF_2 slag failed; successful melting was achieved using the 48 wt% CaF_2 - 52 wt% MgF_2 eutectic mixture. The melting rate was about 1 lb/min with a power input of 75 to 80 kW.

A corrosion test on candidate materials for pilot plant construction was conducted with the proposed aqueous solution (ammonium oxalate, ammonium citrate, ammonium fluoride and hydrogen peroxide). After 400 hr the 304L, 316 stainless steels and Inconel 600 appear acceptable. On the basis of acceptable corrosion behavior and excellent mechanical properties, Inconel 600 was selected as the material for construction. Plate, sheet, and bar stock has been purchased and delivered. Inconel 600 tubing is scheduled for delivery in August 1975.

A small scaleup of the film stripping/decontamination process system has started. This scaled up system is to be installed in an existing shielded facility and will be capable of handling several fuel hulls per test.

A second prototype demonstration is to be built and proof-tested using a series of nonradioactive test runs. This system will be full size but not at present involve remote operation.

TECHNICAL PROGRESS

TRANSURANIC ELEMENT DISTRIBUTION IN CHOP-LEACH CLADDING RESIDUES -

B. Griggs

Several small pieces of highly irradiated fuel exposed in a typical PWR were leached in a fuel dissolving operation (20 hr in nitric acid initial concentration 8M) followed by 4 hr in 8M HNO_3 at 99°C and by a water rinse. The leached cladding was further leached in a 0.4M ammonium oxalate, 0.16M ammonium citrate, 0.1M ammonium fluoride, 0.3M hydrogen peroxide at 95°C for a total of 12 hr in five fresh 30 ml aliquots of solution. The extent of total alpha activity removal by the special leaching procedure is shown in Figure 5. This alpha activity is presumably fuel that has been incorporated with the zirconium dioxide corrosion product layer such that it is insoluble in nitric acid. The presence of fluoride in the leaching liquor causes it to dissolve a small quantity of ZrO_2 and promote the dissolution of trapped actinides.

The leached cladding was mechanically stripped of surface oxide at one end and a 0.67 cm piece of metal cut off. This piece was etched to remove surface contamination and then dissolved for analysis. A similar 0.96-cm piece from the other end was dissolved without treatment. A comparison of analyses normalized to internal surface area gives information

FIGURE 5. Rate of Decontamination by Leaching* of Irradiated Fuel Cladding

on surface activity, activity in ZrO_2 layer and finally in base metal. These data are presented in Table 24 and show that 56% of total alpha activity was removed by leaching, 44% is in surface oxide, and less than 0.02% is in base metal.

Preliminary Decontamination and Densification Process

The candidate processes for decontamination and densification of the chop-leach cladding have been discussed in a previous publication.⁽¹⁾ The processes selected for investigation and development consist of: 1) a gas phase pretreatment of oxidized Zircaloy in HF at about 600°C followed by aqueous fluoride etchants. A molten salt decontamination process for zirconium oxide removal is a backup process. 2) Melting of the cleaned and decontaminated Zircaloy with the stainless steel and Inconel fuel element hardware to obtain a dense, corrosion resistant alloy. A small additional decontamination during melting

by flux treatment, absorption or volatilization may be possible. As an alternative to 2) the Zircaloy, stainless steel, and Inconel may be melted separately with a possible reuse of the metal after short half-life isotopes have decayed. The process steps requiring the most immediate attention are the oxide removal from zirconium alloys, and melting techniques which produce additional decontamination of the alloys from transuranic contamination.

TABLE 24. Gross Alpha Count of Fuel Clad Samples

Summation of 5 leaches	$1.89 \cdot 10^7$ dis/min/cm ^{2(a)}
Residual dissolution	$1.47 \cdot 10^7$ dis/min/cm ²
Base metal after surface cleaning	$4.2 \cdot 10^3$ dis/min/cm ²

- a. It is assumed that most alpha activity comes from the internal surface of the fuel clad, and thus the values are stated per cm² of the fuel clad inner surface.

DECONTAMINATION PROCESS - B. Griggs

A 2.5-cm piece of high-exposure PWR Zircaloy fuel cladding was stripped chemically to simulate the results of one potential decontaminating process. The piece of cladding had been leached in a nitric acid fuel dissolving operation and rinsed with dilute nitric acid. The piece was weighed, then treated in a slow flow of argon and HF (about 50% HF) at 595 to 625°C for 45 min. The treated piece was immersed in a 0.4M ammonium oxalate - 0.16M ammonium

citrate - 0.1M ammonium fluoride - 0.3M hydrogen peroxide solution for 45 min at 90 to 95°C. This process removed all visual oxide from the surface except for two (< 1 mm diam) spots. An additional 45-min treatment did not improve its visual appearance. Weight loss from the combined treatments corresponded to 3.2 g/dm² or 0.05 mm. The first aqueous stripping solution was decanted in order to apply a more concentrated HF solution to completely dissolve the fine black solid for analyses. These solutions and a solution of dissolved metal were analyzed for total alpha count to determine the degree of decontamination achieved. The decontamination factor was greater than 10⁴ with a substantial portion of the residual alpha activity in the base metal originating from irradiation of impurity uranium which is not available to surface decontamination. Analyses of isotopes in the base metal are not yet complete. The residual alpha count in the base metal after surface decontamination was 21 nCi/g.

Decontamination Process - Abstracted from a Battelle-Columbus Report⁽²⁾

The corrosion of plain and oxidized Zircaloy-2 was tested in a molten salt bath of 37 mole % NaF, 37 mole % LiF, 26 mole % ZrF₄ at 704°C. The effects of hafnium contamination (~3%) of the zirconium fluoride, time, hydrogen fluoride sparge, and surface oxide on the Zircaloy were examined. It was found that the corrosion rate was nearly parabolic without the HF sparge and

was relatively unaffected by an oxidized surface layer (~83 mg/dm²/wt gain from autoclaving). Sparging with HF increases the corrosion rate substantially and produces nearly linear dissolution kinetics. The presence of hafnium impurity in ZrF₄ caused a two to threefold increase in corrosion rate. The salt adhering to the removed specimen was nearly proportional to the Zr corrosion.

Analyses of the molten salt used for obtaining the data reported in the previous quarterly report⁽³⁾ have shown substantial hafnium contamination of the molten salt bath. This would partly explain the higher corrosion rates obtained in that work compared to previously published corrosion rates.⁽⁴⁾ Because of differences in temperature or equipment it is difficult to compare the rates quantitatively. However, Reference 4 showed rates of 25 to 50 μm/hr on Zr-2 at 650°C with HF sparge. Reference 3 had rates of about 30 μm/min on Zr-2 at about 700°C with a HF sparge while Reference 2 gave rates of 7 to 12 μm/min on Zr-2 at 704°C.

SHIELDED FILM STRIPPING/DECONTAMINATION FACILITY - R. G. Moles

Preliminary work has begun on construction of an inline film stripping/decontamination facility to be installed in an existing lead-shielded cave. The lead-shielded cave has 8-in. thick shielding, windows, and two overhead manipulators.

The film stripping/decontamination system when installed will be capable of cleaning 0.2 liter of fuel hulls per test and will permit optimization

of side waste stream cleanup. This system will be operated in an inline operational mode and is to be a proof-in-principle test prior to fabricating a full-scale decontamination facility.

Proof Testing Facility

Plans are proceeding on designing, fabricating, and installing a full-scale film stripping/decontamination facility. This facility is to be installed in a hot cell mock-up area but will not at present include remote handling equipment or remote process operation.

DENSIFICATION PROCESS - R. G. Nelson

A 14-kg heat of Zircaloy-4 10 wt% 304 stainless steel - 5 wt% Inconel 718 was made by the "Inductoslag" process (Figure 6). Charge material consisted of chopped 1-in. long Zircaloy fuel cladding; 1-in. long x 0.25-in.

diam, 304L stainless steel rod; and bent 1-in. squares of 0.020-in. Inconel 718 sheet. Initial melting was tried using the CaF_2 slag. This attempt failed because the low melting ($\sim 1100^\circ\text{C}$) eutectic alloy ran out the bottom of the crucible before the CaF_2 slag melted (1360°C). Using a eutectic mixture of 48 wt% CaF_2 - 52 wt% MgF_2 (MP $\sim 1000^\circ\text{C}$) resulted in successful melting. About 7 hr were required to melt the 14-kg ingot. This was due to the long time required for the solid charge material to dissolve in the low-temperature liquid. Increased power to the pool resulted in a larger pool volume with no apparent increase in pool temperature. The resulting crack-free ingot consisted of 5 to 10 wt% of unmelted Zircaloy, stainless steel, and Inconel "potted" in a matrix of solidified eutectic mixtures (Figure 7). Ingot hardness varied from 255 to

Neg. No. 75 2898-1

FIGURE 6. As-Cast Inductoslag Melted Ingot of Zircaloy 4 - 10 wt% 304 SS - 5 wt% Inconel 718 Heat Number USBM 443

Neg. No. 75 3379-1

Neg. No. 75 3379-2

FIGURE 7. Sections of 10-cm-Diam Zircaloy-4 - 10 wt% 304L SS - 5 wt% Inconel 718 Inductoslag Melted Ingot Heat Number USBM 443

364 BHN. High-speed steel hacksaw blades were adequate for ingot sectioning. Surface finish and ingot soundness are adequate for direct storage. Melting times could be shortened by an additional heat source concentrated at the top of the crucible. An induction concentrator coil, plasma torch, arc, or laser beam could provide this.

A 34-kg heat consisting of three sections--a) about 7 kg of alloy CF3 (304 stainless steel casting grade), b) 14 kg of a 2 to 1 by weight mixture of CF3 and Inconel 718, and c) 14 kg of pure Inconel 718--was melted by the U.S. Bureau of Mines Inductoslag process. Charge material consisted of cut pieces of reject tie plates (alloy grade CF3) and chopped 1-in. squares of 3/16-in. thick Inconel 718.

Fourteen kg of the CF3 alloy were charged into the feed hopper with 3 wt% CaF_2 . Initial melting resulted in a run out of molten metal because of insufficient fluidity of the CaF_2 slag. The remaining 7 kg of CF3 alloy was melted using the 48 wt% CaF_2 - 52 wt% MgF_2 eutectic mixture. The ingot was allowed to cool and the hopper recharged with 14 kg of the 2 to 1 mix of CF3 and Inconel 718 and 3 wt% CaF_2 - MgF_2 slag. Melting was then started on the top of the previously melted CF3 ingot. This

procedure was repeated with the Inconel 718 material resulting in a three-component ingot of about 34 kg (Figure 8). The melting rate of all three components was about 1/2 kg/min using a power setting of 75 to 80 kW. The Inconel-CF3 alloy and the pure Inconel ingot material have gold-colored surfaces. This is apparently associated with a film that appears to float on the molten pool during melting. The surface of the pure CF3 ingot is bright, and the pool was free of any film during melting.

The final draft of a report sponsored by this program by M. P. Schlienger and E. V. Tiesenhansen has been received. This report, Analysis for Selection of a Melting Furnace for Consolidation of Nuclear Fuel Hulls into an Ingot for Remanufacture, compares ten different cold-mold melting systems and their adaptability to this program. The high rating this report gives to the Inductoslag process plus the excellent results achieved by the U.S. Bureau of Mines in fabricating and melting ingots for this program have culminated in a decision to use the Inductoslag process for in-cell work. Complete texts of the Schlienger report and the engineering criteria from which it was written will be published in a separate BNWL document.

Neg. No. 75 4352-1

FIGURE 8. Three Component Ingot of Inductoslag Melted Stainless Steel and Inconel Heat Number USBM 460

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4.0 PYROLYSIS-INCINERATION OF COMBUSTIBLE ALPHA WASTES

P. C. Walkup and L. K. Mudge - Project Managers

The principal program objective is to develop a pyrolysis-incineration process for safe and economical reduction of the volume of alpha wastes from 10- to 50-fold as a precursor to long-term storage and/or disposal. The technical and economic merits of this process will be defined to allow its comparison to other processes being considered for volume reduction of alpha wastes. The program objective will be pursued via 1) a phased program of bench-scale research leading to supporting pilot plant development activities and 2) systematic and continuing analytical studies to assess the technical and economic merits of the process. Bench-scale and pilot plant activities will be conducted first with noncontaminated simulated wastes and later with actual wastes from Hanford operations.

SUMMARY

Installation of the process demonstration unit was completed during the reporting period. Four separate runs using simulated alpha wastes were completed, the longest of which was 14 hr. Operation without significant channeling was achieved. Condensates produced during the processing were quantified and characterized.

Insufficient ash was generated during the operations to allow further ash fixation studies.

TECHNICAL PROGRESS

PILOT PLANT STUDIES - L. K. Mudge, G. L. Roberts, G. F. Schiefelbein, R. G. Sullivan, and R. A. Walter

Installation of the process demonstration unit (PDU) was completed during May. A photograph of the unit is shown in Figure 9. A schematic diagram of the unit has been presented in previous quarterly reports.

Four runs were made with the process demonstration unit during the last month of the quarter. In each

run the procedure was as follows:

Ash was added to the reactor to cover the grate and insulate it from the hot char combustion zone. Ignited charcoal was then added to the reactor to start combustion, and carbonaceous material in the form of charcoal or coke was added to establish the char zone and preheat the reactor. Air flow to the reactor at the desired rate was then started to ignite the charcoal and coke. After about 2 hr of air flow for combustion of the coke and charcoal, feeding of simulated alpha waste to the reactor was begun. Feed addition was continued to maintain the level of the bed in the reactor within 2 ft of the top of the refractory lining in the reactor. Table 25 shows the initial quantities of materials and air flow rates used for the four runs.

Simulated alpha waste of the composition shown in Table 26 was used in the four runs. Waste components were shredded with a cutter type shredder at the Saturn Manufacturing

FIGURE 9. Process Demonstration Unit

TABLE 25. Initial Run Conditions

Run	Weight of Ash, lb	Weight of Ignited Charcoal, lb	Weight of Unignited Charcoal, lb	Weight of Coke, lb	Air Flow Rate, scfm
1	25	20	--	--	5
2	25	10	--	35	10-14
3	37.5	12	13	23	10-14
4	23	11	--	47	10-14

TABLE 26. Simulated Waste Composition

Component	Wt %
Paper	15
Rags	15
Wood	10
Latex Rubber	5
Hypalon Rubber	10
Neoprene Rubber	10
PVC	20
Polyethylene	15

Facility in Salem, Oregon. Shredded feed components are shown in Figure 10. The components are weighed and hand-mixed in 25-lb batches.

Waste is added to the reactor through the feed lock hopper with the distributor at the top of its traverse. After about 10 to 15 lb of waste are screw-fed into the top of the reactor, the waste is packed and spread with the distributor. When the level of the refuse in the reactor has receded to about 2 to 3 ft from the top of the refractory lining, the distributor is withdrawn and another 10 to 15 lb of feed are screw-fed to the reactor. This procedure is repeated throughout the run.

In the first run with the PDU, the distributor was operated with a force of about 2600 lb/ft^2 exerted on the refuse bed. This force packed the bed so tightly that no air flow was obtained through it. Of the 125 lb of feed added to the reactor, 100 lb remained packed and unprocessed inside the reactor after 8 hr of operation at 5 scfm air flow. In the subsequent runs, the distributor pressure was adjusted to exert a force of about 130 lb/ft^2 . Occasionally a greater force was used to get denser packing of the refuse bed. The extent of the packing was observed by noting changes in the inlet air pressure and inlet air flow. As the bed is packed, the air flow decreases and the inlet pressure increases. During normal operation, the refuse bed was packed and spread so as to give a 1/2 to 1 psi pressure drop across the refuse bed at an air flow rate of about 10 scfm. Air flow varied from

10 to 14 scfm in runs 2, 3 and 4, depending on the pressure drop across the refuse bed.

To avoid channeling of gases around the refuse bed, it is necessary to spread the bed by rotation of the distributor as well as pack the refuse bed. In run 2, the bed was packed and spread intermittently between feed additions. In runs 3 and 4, the bed was spread and packed continuously between feed additions. Improved processing rates resulted from the continuous spreading, indicating a reduction in the amount of channeling. An average rate of 17 lb/hr was attained in run 2, while average rates were 27 and 23 lb/hr in runs 3 and 4, respectively. An average feed rate of 27 lb/hr was achieved over the last 11 hr of run 4, but rates were slow during the first 3 hr, reducing the average rate to 23 lb/hr. The reactor was emptied at the end of each run to determine overall processing rates. Instantaneous rates were higher than the average rates, but the instrumentation was too unreliable to determine these rates accurately. At times during runs 3 and 4, feed addition rates to maintain the desired refuse level in the reactor were 40 lb/hr.

Compositions and heating values of gases generated in runs 2, 3 and 4 are shown in Table 27. The low heating value of the gases generated in run 2 indicates that considerable channeling occurred, while the higher heating values in runs 3 and 4 indicate that channeling can be reduced by continuous distribution of the refuse bed with the mechanical agitator. It

FIGURE 10. Shredded Simulated Waste Components

TABLE 27. Gas Composition in Runs 2 and 3

Run	Time	% by Volume									Heating Value of Gas, 3 Btu/ft.
		H ₂	CO ₂	C ₂ H ₄	C ₂ H ₆	O ₂ -Ar	N ₂	CH ₄	CO	C ₃ H ₈	
2	1200	3.4	20.3	0.1	0.03	0.8	72.3	--	3	--	24
	1300	4.7	18.0	0.07	--	0.8	67.3	--	8.2	--	45
	1400	4.2	15.5	0.07	0.02	0.8	67.2	--	12.2	--	57
3	1320	5.5	8.0	0.1	0.1	0.8	60.1	0.1	26.1	--	112
	1400	3.6	9.1	0.2	0.1	0.8	61.8	0.1	24.3	--	101
	1430	7.1	10.8	1.3	0.2	0.8	60.8	1.0	18.8	--	124
	1615	11.1	14.0	1.8	0.3	0.6	56.6	1.4	12.5	0.4	141
4	1050	3.7	11.2	--	--	0.7	60.3	--	24.1	--	79
	1225	4.0	10.5	0.04	--	0.7	63.0	--	21.7	--	88
	1425	5.4	11.6	0.3	0.1	0.7	61.1	0.3	20.5	--	98
	1700	7.0	13.1	0.2	0.2	1.5	61.5	0.3	16.1	--	89
	2005	8.9	12.6	0.5	0.2	0.9	60.5	0.3	16.1	--	100
	2220	8.6	11.2	0.7	0.4	0.7	61.3	1.3	15.9	--	116

should be noted that the gas compositions vary considerably over short time periods and that the above compositions are representative of the true composition over only a very short period. No attempt was made to identify changes in composition at different times after the charging of a new batch of feed. Concentrations of halogens in the gases are not yet available.

The gases produced were burned at 1200°F in a secondary burner without any visible smoke. The exception to this occurred at the end of run 3 when the refuse bed had completely burned through, and the filtering action of the fresh feed was lost. A white smoke was noted emitting from the stack at this time.

Figure 11 shows temperature profiles in the reactor at different times during run 4. The numbers on the abscissa in Figure 11 refer to thermocouples equally spaced vertically along the wall of the reactor; point 7 is 1 1/2 ft from the bottom flange and point 1 is 1 1/2 ft from the top flange. A considerable change in temperature profile with time is apparent. Longer runs will be needed to determine the course of the temperature changes. Only local temperatures along one side of the reactor are recorded. It is highly probable that the profile varies across the reactor cross-section as well as vertically.

Inputs and outputs of liquid and solid streams for run 4 are shown in Table 28. The carbonaceous residue represents incompletely burned waste that remained in the reactor at the time of shutdown. Inspection of the

residue revealed that pyrolysis of the feed components was complete so that it can be assumed no more tars would result from further processing of the residue. For this particular run, the condensate (or tar) yield was 18% of the weight of the waste fed to the reactor.

The production of tars (condensate) was thought to be the cause of line plugging which led to an early shutdown in run 4. However, inspection of the reactor at the end of run 4 showed that the off-gas line in the reactor was plugged with shredded rags carried to the line by the distributor during waste feed addition and swept to the 2-in. outlet by the high-velocity gases in this region. The off-gas lines themselves were open. Relocation of the off-gas line in the reactor should prevent similar plugging during future operation.

The condensate produced in run 4 had a calorific value of 9380 cal/g and an ash content of 0.3%. Viscosities of the condensate determined at different temperatures using a Brookfield Viscometer are shown in Table 29. Some non-Newtonian behavior is evident from the changes in viscosity with shear rate (spindle rpm). The viscosity data show that the tars can be handled at temperatures above about 50°C. The tars may thus be burned with the off-gases or separated for separate handling.

Condensate from runs 1 through 3 showed a higher viscosity (300 to 400 cp at 75°C) and a lower calorific value (8650 cal/g) than the condensate from run 4. Difficulty in separation of scrub solution from the tars obtained in runs 1 through 3 is

FIGURE 11. Reactor Temperature Profile for Run 4

TABLE 28. Run 4 - Material Inputs and Outputs

	Ash	Coke	Waste	Condensate	Carbonaceous Residue
Input, lb	25	47	355	0	0
Output, lb	28.5	0	0	62.5	60.5

TABLE 29. Viscosities of Condensate (Tars)

Temperature, °C	Spindle Speed, rpm	Viscosity, cp
57	50	87
	20	115
	10	750
38	10	8,400
	5	13,600
	2.5	21,600

believed to be the cause of these differences. The ash content of this tar was 2.6%, probably because of burn-through of the refuse bed in run 3 which resulted in the loss of filtering action by the bed.

ASH FIXATION STUDIES - J. E. Mendel, H. Hollis, and L. K. Mudge

Ash generated in the PDU studies, see Table 28, was not sufficient to yield an identifiable residue for further studies. Results of earlier work were given in the previous quarterly report.

5.0 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

More detailed studies of the preparation of acrylonitrile, phenol-formaldehyde type, and polystyrene have emphasized some of the problems and limitations for a process-scale operation with high levels of tritium. With acrylonitrile, process complexity may rule it out. In principle polystyrene is the most straightforward, but requires a low-pressure catalyst.

Calculations indicate that radiation damage is probably not critical, but further studies must be made if application to greater than 0.1 g of tritium per liter is anticipated.

The laser laboratory has now been air-conditioned, and temperature stability will permit the remaining excitation experiments to be performed.

Topical reports are being issued on Polymer Fixation and on Laser Excitation.

TECHNICAL PROGRESS

TRITIUM FIXATION - J. A. Franz and L. L. Burger

The fixation of tritiated water into organic polymers of the phenolic-formaldehyde and acrylonitrile types and into polystyrene has been described in the previous quarterly report.⁽¹⁾ The latter process requires

prior conversion to the elemental form. Preliminary leaching experiments using tracer-level tritium were also described and indicated negligible tritium loss after the initial water contact.

More detailed studies of the processes have provided some limitations. Although the acrylonitrile product appears adequate, the process complexity, primarily because of purification steps, probably makes the method infeasible. The problems associated with manipulating streams containing high concentrations of tritium dictate that process simplicity is very important. Likewise, the use of tritiated polystyrene--involving perhaps the simplest process--would not be workable under high-pressure operation because of tritium diffusion through materials of construction and thus requires a low-pressure catalyst.

Samples of polystyrene were sent to New England Nuclear Tritium Labeling Laboratory for tritiation and have now been received. The tritium content is about 220 mCi/g and will provide a sufficient concentration of tritium for further feasibility tests. As to the problem of hydrogenation, materials have been obtained for synthesis of a homogeneous hydrogenation catalyst which will be examined.

The polymers under consideration have good chemical and physical stability. However, radiation damage from the tritium beta must be evaluated. Sufficient radiation effects data on polymers are available in the literature to assess the magnitude of the problem. Physically cross-linking and degradation are the results of high-energy radiation. Generally they occur simultaneously and produce opposite effects on hardness, strength, solubility, etc. Significant changes in mechanical properties generally occur in the range of 10^8 to 10^{10} r.^(2,3) The more stable plastics are those containing phenyl groups, nitrile groups, and inert fillers.

The principal concern would be hydrogen release. G (the number of molecules formed for each 100 ev of energy absorbed) values for hydrogen and other gases are listed in Table 30. Polyethylene is included for comparison.

Reprocessing plant aqueous wastes may range in concentration from roughly 0.01 to 0.2 Ci/l depending on whether the first cycle Purex type wastes can be segregated. Assuming that they are and they can be recycled, the level might be brought up to perhaps 5 Ci/l. At this level an order of magnitude estimate for the volume from a 10T/day plant would be 2000 l. To obtain a waste of the order of 100 Ci/l (~ 10 mg T/l) or greater would require isotopic concentration.

A second source might be tritium wastes from nuclear facilities that handle highly concentrated or pure

TABLE 30. Gas Formation from Irradiated Polymers

Type	G Total Gas	G _{H2}
Polyethylene	3.1	2.2 ⁽²⁾
Polystyrene	0.03 ⁽²⁾	0.02 ⁽²⁾
Phenolic (phenol formaldehyde)	0.08 ⁽³⁾	
Graphite-filled phenolic	<0.001 ⁽³⁾	
Urea formaldehyde	0.8 ⁽³⁾	
Aniline formaldehyde	0.02 ⁽³⁾	
Polyacrylonitrile	0.39 ⁽²⁾	0.1 ⁽²⁾ 0.08 ⁽⁴⁾

tritium compounds. These wastes may be higher by an order of magnitude or greater.

A G value for hydrogen formation of 0.1 is a reasonable upper limit for the polymers in question. Reduction by a factor of 10 may be possible through choice of polymer or with the addition of fillers and inhibitors. Storage of tritiated polymers at a few curies per liter thus appears to be no problem. At much higher concentrations further consideration is required. Water at 0.1 g of tritium per liter ($\sim 10^3$ Ci/l) would be incorporated in a plastic at about 300 Ci/l producing 3×10^{20} ev/kg-hr. For a G_{H2} value of 0.1, 5×10^{-7} moles H₂/hr-kg are formed. In a phenolic polymer, 6.4% H, 7.6×10^5 hr or about 9 years, assuming no decay, would be required to liberate 1% of the hydrogen.

There are probably maximum values, and mechanisms exist which may lower the hydrogen or tritium release. However, the quantity is significant and

suggests that experimental measurements should be made if storage at the 0.1 to 1.0 g/l level is contemplated.

Both radiation and thermal stability will be examined in more detail, particularly with respect to gas evolution.

A report summarizing the polymer studies to date and recommendations for further studies is being prepared.

LASER EXCITATION - T. J. Whitaker

After spectra of HDO vapor at 370°K had been obtained which matched known spectra,⁽⁵⁾ attempts were made to hold the laser systems frequency at a particular value. A frequency corresponding to a 000+110 vibrational transition in HDO was chosen. Because the frequency drift due to room temperature variations was unacceptable, further experiments were delayed to get air conditioning in the room. The air conditioner has now been installed. Experiments at 295°K are now in progress.

Further work has been done on a computer program to describe the kinetics of HDO dissociation products. Figures 12 and 13 show part of the results. These plots show the formation of H_2 , D_2 , and HD with respect to time after 10^{14} molecules of HDO have been dissociated. Figure 12 assumes 80% of the dissociated molecules result in deuterium atom formation. Figure 13 assumes 50% deuterium atom formation. As expected, the higher D_2/H_2 ratio is found in Figure 12. A report describing this kinetics program and the experiments performed to date is near completion.

FIGURE 12. HDO Dissociation Products, 80% Deuterium Atoms Formed and 20% Hydrogen Atoms Formed

FIGURE 13. HDO Dissociation Products, Equal Formation of Hydrogen and Deuterium Atoms

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6.0 CHARACTERIZATION OF 300 AREA AND WYE BURIAL GROUNDS

J. R. Eliason - Project Manager

This study is designed to evaluate and define the present status of waste materials disposed to the 300 Area and Wye Burial Grounds and to evaluate the alternatives of designating the sites for permanent storage and/or removal of all or part of the materials.

SUMMARY

Development and evaluation of techniques for characterizing burial grounds during this study period has led to compilation of various task reports. These individual reports will be combined into a topical report covering work done in FY-1975.

Development of data computer analysis techniques for geophysical studies has proceeded which will allow dimensional location of burial waste and sediment boundary structure within the study area. Development of data collection system hardware has also continued.

Log analyses of wells at the 300 North Burial Ground have delineated zones which may potentially impede the flux of radionuclides through sediments.

Neutron probe and thermocouple psychrometer-temperature transducer arrays have been monitored during this quarter. Analysis of data collected during this period has shown nonequilibrium conditions with depth. However, seasonal fluid flux and temperature gradients have been identified near the ground surface.

Studies of burrowing mammals have continued within the 300 Wye Burial Grounds. These studies, in part,

evaluate potential contamination of plant and animal species by buried waste materials.

TECHNICAL PROGRESS

TASK 1 - REVIEW OF HISTORICAL

RECORDS - A. E. Reisenauer

No progress to report.

TASK 2 - WASTE DETECTION AND MAPPING - G. A. Sandness

During the past quarter, analytical techniques for introducing ground penetrating radar, acoustics, and magnetic data have been developed. Various computer codes have been developed to three-dimensionally locate buried waste materials and burial trench boundaries. Work has also progressed in data reduction for a final topical report.

TASK 3 - DRILL SAMPLING WELLS AND ANALYZE SAMPLES - V. L. McGhan

During the past three months, work under this task has aimed at interpreting drillers' logs. These logs have delineated lithologic boundaries which may impede flow of radionuclides potentially transported through sediments.

TASK 4 - INSTRUMENTED WELLS AND SOIL CHARACTERIZATION - S. J. Phillips

Monitoring of meteoric fluid transport through the unsaturated zone of the 300 North Burial Ground continues. Soil moisture flux determinations by neutron probe analysis have been made in two vertical arrays adjacent to the burial ground to depths of about 10 m. The moisture content throughout the sediment profile has decreased at all depths, indicating desiccation with time (Figures 14 and 15). This suggests that equilibrium conditions have not been established after backfilling of neutron access arrays. However, seasonal fluid content gradients with depth have been determined. An influx of recent meteoric fluids has also been determined to a depth of about 1.5 m. Two sediment profile samples have also been physically characterized.

These characterization studies will in part permit evaluation of the burial ground site.

Sediment fluid potential and temperature with depth evaluations have been conducted at the study site. The operating system used for these determinations has been tested, and the thermocouple psychrometers and temperature transducer array have been monitored. Soil fluid potential monitoring studies have delineated nonequilibrium conditions of sediments adjacent to installed *in situ* transducers (Figures 16 and 17). The fluid potential, however, is trending toward a stable configuration. Sediment temperature was also found to have a nonequilibrium trend. Yet the seasonal temperature profile to a depth of approximately 2.0 m appears stable (Figure 18).

FIGURE 14. Mean June 1974 Fluid Content - Bulk Density Versus Depth of Sediment Profile at the 300 North Burial Ground, NW Corner Array

FIGURE 15. Mean June 1974 Fluid Content - Bulk Density Versus Depth of Sediment Profile at the 300 North Burial Ground, SE Corner Array

FIGURE 16. Mean June 1974 Potential Versus Depth of Sediment Profile at the 300 North Burial Ground

FIGURE 17. June 1974 Matrix Potential Versus Depth of Sediment Profile at the 300 North Burial Ground, SE Corner Array

FIGURE 18. Mean June 1974 Temperature Versus Depth of Sediment Profile at the 300 North Burial Ground, SE Corner Psychrometer and Thermocouple Transducer Array

TASK 5 - EVALUATION OF MOISTURE MOVEMENT MECHANISMS AND MODELING -

H. P. Foote

Preliminary evaluations of parameters controlling moisture transport in the unsaturated zone and numerical techniques for modeling these were developed. As data become available, from cooperating task investigators, modeling and analysis techniques will be developed.

TASK 6 - EVALUATION OF BIOLOGICAL TRANSPORT MECHANISMS - W. H. Rickard

Studies of small mammal population in burial grounds have been characterized for the parameters of winter and spring emergence and population dynamics. This study will be used to evaluate the potential of burrowing mammals for contamination.

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

SUMMARY

Assembly of information for establishing methods, costs and priorities of D&D at Hanford continued. An annotated bibliography citing over 400 D&D references was completed and cleared for publication.

A preplan for the 233-S Building D&D demonstration was prepared, and drafts of two Activity Specifications (out of six required) were distributed for review.

Exploratory tests of the use of pulsed, small-volume, high-velocity liquid jets for removing contaminated concrete surfaces gave promising results. Tests were started of in situ neutron measurement techniques to determine plutonium inventories in contaminated liquid disposal sites.

exhaust air systems; reactor fuel storage basins; and contaminated solids storage and burial sites. Additional information was collected on individual cribs and trenches: water table and site elevations, and chemical composition of wastes discharged to the ground at these sites.

Additional information on retention basin systems in the 100 Areas was collected for determining: 1) the feasibility of burying the effluent pipe in the basins or 2) the volume of concrete and pipe needing transport and burial at another repository (e.g., 200 Area location) if excavation and removal of these systems are required at a future date. Data pertinent to these disposition alternates are presented in Table . The data show that there is more than adequate space to bury the effluent pipes in the retention basins.

TECHNICAL PROGRESS

TASK 1 - DEVELOP METHODS, COSTS AND PRIORITIES OF D&D AT HANFORD -

K. M. Harmon

SUBTASK 2 - SITE CHARACTERIZATION -

B. W. Mercer

Descriptive information and available radionuclide inventory data were assembled for the following types of contaminated, retired Hanford facilities: reactor recirculating gas and

SUBTASK 3 - DISPOSITION -

M. R. Kreiter

DEVELOPMENT OF METHODOLOGY FOR SETTING D&D PRIORITIES -

J. W. Litchfield and J. C. King

Work continued on the adapting of an existing decision methodology to the problem of setting priorities for the D&D of retired facilities at

TABLE 31. Retention Basin Systems: Estimated Dimensions and Waste Volumes

Retention Basins						Effluent Pipe Lines								
Area	Retention Basins	Dimensions	Total Vol. cu yd	Approx. Back-fill, ft	Concrete Vol. cu yd	Length, Ft.								Total Vol. Displaced
						Steel Pipe Diam, In.					Concrete Pipe Diam, In.			
						12-16	18-24	36-42	60-72	84	30-36	42-48	60-72	
B	107 B	230 ft x 467 ft x 20 ft Deep	80,000	4	4200	180	1445	750	14710	-	2085	3240	50	20,000
	107 C	2 tanks, 330 ft Diam. 16 ft Deep	101,000	4	(Steel)									
D	107 D	230 ft x 467 ft x 20 ft Deep	80,000	2	4200	140	1470	3720	9900	-	300	400	2340	14,000
	107 DR	273 ft x 600 ft x 20 ft Deep	120,000	4	7000									
F	107 F	230 ft x 467 ft x 20 ft Deep	80,000	5	4200	-	-	2605	-	-	470	2300	350	2,900
H	107 H	273 ft x 600 ft x 20 ft Deep	120,000	4	7000	350	1090	-	4400	-	-	-	-	4,400
K	107 KE	3 tanks, 250 ft Diam x 29 ft Deep	158,000	4	(Steel)	6010	410	6725	5380	2600	-	-	835	16,400
	107 KW	3 tanks, 250 ft Diam x 29 ft Deep	158,000	4	(Steel)									

*Includes 30. for voids between pipes and miscellaneous.

Hanford. Development of a comprehensive set of independent criteria (Figure 19) progressed to the point of final review. A representative facility was picked for each facility class included in the study, and plans were made to test the criteria by applying them to the representative facilities. Preparations were made to apply a computerized decision model to the prioritization effort.

SUBTASK 5 - PROCEDURES -

H. T. Fullam

Preliminary assembly of generalized disposition procedures and related waste disposal information was completed for the major classes of facilities.

The index for the reference center was again updated and a new computer printout of catalogued references prepared. Approximately 600 references have been collected and

591 indexed. The index will be updated periodically as additional references are obtained.

FIGURE 19. Prioritization Criteria

The annotated bibliography has been cleared for publication and is soon to be printed as a formal report, BNWL-1917 (UC-70)⁽¹⁾. The bibliography cites 429 unclassified references relating to the decontamination and decommissioning of nuclear facilities. The references are arranged in chronological order and cover the period from 1944 through 1974. An author index and limited subject index are provided.

TASK 2 - D&D DEMONSTRATION -

J. R. Carrell

Planning and preparation of program documentation is in progress for the 233-S (Redox Plutonium Concentration Building) D&D demonstration. The prime objective is to demonstrate advanced technology and develop cost data for disposal of alpha-contaminated facilities. The program will decontaminate and ultimately demolish the facility. However, the option is being held open for terminating the program should the success with the decontamination of the building provide a useful facility for other activities.

The D&D demonstration will be divided into six major activities:

- Activity I. Detailed Characterization of Facility
- Activity II. Preparatory Effort
- Activity III. Process Equipment Dismantling and Removal
- Activity IV. In-Cell Procedure Test Program
- Activity V. Final Decontamination
- Activity VI. Final Demolition

A preplan issued in June provides the basic plan for the demonstration. Included in the preplan is a definition of the interface between ARHCO and Battelle. In brief, Battelle will manage the overall program, prepare specifications for the six major D&D activities, provide detailed procedures for many of the R&D activities to be carried on in 233-S, and prepare an environmental assessment. ARHCO is to prepare detailed operating procedures, provide design and procurement of specialized equipment for the activities, and conduct actual operations within the guidelines of the Activity Specifications. Battelle will review and approve ARHCO's procedures; ARHCO will review and concur in the Activity Specifications and other Battelle documents. ARHCO will continue as landlord of the facility throughout the demonstration.

ACTIVITY SPECIFICATIONS - C. H. Allen and P. L. Koehmstedt

Activity Specifications I and II have been drafted, and it is evident that the characterization of the facility and the preparatory steps for actual removal of equipment from the process hood will require astute planning and execution. Determination of the current status of utilities such as electrical services, breathing mask air system, and floor drainage in a highly contaminated building are examples of the problems.

TASK 3 - RESEARCH AND DEVELOPMENT -

K. M. Harmon

SUBTASK 1 - DECONTAMINATION OF CON- CRETE WITH PULSED, HIGH-VELOCITY LI- QUID JETS - C. H. Allen, S. Mandel, and M. R. Petersen

Exploratory tests have been conducted of the use of pulsed, high-velocity liquid jets to remove contaminated concrete surfaces. Vendor tests have shown that 1) small jets of glycerin activated by 45-caliber blank shells fired in a "water cannon" produce shallow crater-type holes in the surface of the concrete and 2) surface contamination does not appear to be carried to the bottom of the crater.

The contamination-spread tests were carried out by the vendor with cured concrete blocks coated on one surface with compounds which fluoresce under ultraviolet light: zinc sulfide on one set of blocks and sodium fluorescein (uranine) on the other.

The "water cannon" fired pulsed liquid jets of glycerin at the target blocks. One shot produced a 3.5-in. x 2-in. x 0.835-in.-deep crater in the block treated with sodium fluorescein. After the test firings the blocks were coated with clean plastic spray, rewrapped and shipped back to PNL, and visually inspected under ultraviolet light.

A closeup of one crater showed some green fluorescence on rock aggregate just below the surface. This was caused by penetration of sodium

fluorescein solution into the block and around rocks just under the surface. A slight overall penetration of about 1/8 in. could be seen all around the crater. No green fluorescence was visible within the crater.

Zinc fluoride contamination was observed in the bottom of bubble holes formed during casting of the blocks, but there were definitely no ZnS particles in the craters. Evidence from both the sodium fluorescein and zinc sulfide-coated block shows that the concrete is fractured and ejected from the block so that no surface contamination is driven into the block.

SUBTASK 3 - CHARACTERIZATION OF CRIBS - B. W. Mercer and

R. L. Brodzinski

Site characterizations studies were initiated to evaluate neutron monitoring techniques in radioactive liquid waste disposal cribs. Both a directional neutron counter and the metal disc activation method are being evaluated by ARHCO and BNW personnel, respectively, in the 216-S-20 crib. These techniques will hopefully reduce the cost of determining the plutonium inventory in cribs suspected of having significant plutonium inventories.

REFERENCE

1. W. E. Sande, M. S. Hanson, R. L. McKeever and H. D. Freeman, Decontamination and Decommissioning of Nuclear Facilities - A Literature Search, BNWL-1917, Battelle Pacific Northwest Laboratories, Richland, WA, 1975.

8.0 DEVELOPMENT OF PACKAGING TECHNOLOGY FOR $\text{Pu}(\text{NO}_3)_4$ SHIPMENT

H. H. Van Tuy1 - Project Manager

The USAEC has announced that the shipment of $\text{Pu}(\text{NO}_3)_4$ solution will be prohibited as of June 17, 1978. After that date, the only solid Pu form that can be shipped to fuel fabricators as a plutonium fuel material is PuO_2 .

The objectives of this program are:

- 1. To determine if a solid form of plutonium nitrate can be found that is reasonably stable and directly soluble in nitric acid;*
- 2. To define new packaging technology for the shipment of the solid plutonium form; and*
- 3. Based on the success of 1 and 2, to define a commercially feasible process to make the solid plutonium form.*

If plutonium nitrate in powder form can be stored and shipped safely, made economically, and is readily soluble in nitric acid, then plutonium nitrate powder is likely to be superior to PuO_2 for shipment to plutonium fuel fabricators.

The compounds investigated are $\text{Pu}(\text{NO}_3)_4 \cdot x\text{H}_2\text{O}$, $\text{K}_2\text{Pu}(\text{NO}_3)_6$, and $(\text{NH}_4)_2\text{Pu}(\text{NO}_3)_6$.

SUMMARY

The first kilogram-scale production run of $\text{Pu}(\text{NO}_3)_4 \cdot x\text{H}_2\text{O}$ has been accomplished producing a large mass of solid plutonium nitrate. Some equipment modifications were made prior to the second run to improve heating and removal from the vessel. Analytical results indicate the compound $\text{Pu}(\text{NO}_3)_4 \cdot 6\text{H}_2\text{O}$ was produced during the first production run.

TECHNICAL PROGRESS

A briefing for the nuclear industry on the progress to date of solid plutonium nitrate shipping forms was held in Richland, Washington, on May 21, 1975. Industry representatives were from Allied General Nuclear Services, Babcock and Wilcox

(NUMEC), Exxon Nuclear, General Electric, and Kerr-McGee. A representative of Dow at Rocky Flats also attended. Local attendees were from ERDA-RL, ERDA-FFTF Project Office, ARHCO, and Battelle. Westinghouse (HEDL) was invited but did not attend.

The first large-scale production of solid plutonium nitrate has been accomplished with a total of 920 g of plutonium being converted to $\text{Pu}(\text{NO}_3)_4 \cdot x\text{H}_2\text{O}$. The resultant solid was a large, dark green mass that was taken as a single piece from the evaporation vessel. However, the vessel had to be broken for removal of the compound due to its adherence to the glass surface. The solid was noted to be hygroscopic when left for extended periods exposed to the glove box atmosphere.

The original solution was 5M in nitric acid, with a total volume of 4 liters. The isotopic composition and impurities are shown in Tables 32 and 33. Evaporation to one liter was accomplished at 180 torr. The final evaporation to dryness with use of a vacuum pump varied from 9 torr to 2 torr. Several days were required to complete the evaporation as the process was not continuous, and a few modifications to the equipment were required. The solution was never allowed to exceed 80°C during the final stages of the evaporation to insure the $\text{Pu}(\text{NO}_3)_4 \cdot \text{XH}_2\text{O}$ was not converted to an insoluble form. Some foaming was noted during the evaporation as the solution became more viscous.

Analytical results of the first run plutonium nitrate solid indicated an excessive amount of water, probably due to insufficient evaporation time. A portion of this material was then dried for 48 hr at 1 torr and ambient temperature. The material was then found to contain 41% plutonium by weight as determined by chemical analysis. Calcination to an oxide determined its molecular weight to be about 577 g/mole, indicating 6 waters of hydration. The solid was found to be completely soluble in 4M HNO_3 . It was also found to be soluble in water; however, it rapidly converted to plutonium polymer.

A second kilogram-scale evaporation is in progress and will be taken to dryness at 1 torr and 80°C over a longer period than with the first to determine if more water can be removed from the solid. A metal evaporation vessel is also being used to facilitate removal of the solid.

Thermal conductivity measurements of the solid plutonium nitrate are in progress.

TABLE 32. Isotopic Composition of Plutonium Nitrate Feed

Isotope	% Abundance
^{238}Pu	0.062
^{239}Pu	89.667
^{240}Pu	8.907
^{241}Pu	1.159
^{242}Pu	0.205

TABLE 33. Plutonium Nitrate Feed Impurities

Element	PPM	Element	PPM
Ag	<8	Mg	100
Al	250	Mn	25
As	<100	Mo	<10
B	<2	Na	400
Ba	<100	Nb	<8
Be	<2	Ni	<50
Bi	<5	P	250
Ca	140	Pb	<10
Cd	<2	Si	<40
Co	1	Sn	<15
Cr	125	Ta	<40
Cu	<20	Ti	400
Fe	270	V	<50
Ga	<4	W	<10
K	2000	Zn	<20
Li	<10	Zr	200

9.0 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

Final revision and typing of the report BNWL-1846, The Risk of Transporting Plutonium Oxide and Liquid Plutonium Nitrate by Truck are in progress. The report will be issued at the end of July.

Work is progressing on an assessment of the risk in the shipment by rail of plutonium oxide and liquid plutonium nitrate. The assessment is based on the same plutonium shipping packages as used in the analysis for truck shipment. Preliminary results indicate that the risk of shipment of plutonium nitrate in L-10 containers is greater in the rail mode than in the truck mode while the reverse is true for plutonium oxide shipment in 6M containers. The major factor in the plutonium nitrate shipment risk increase is the increased number of packages that may be carried in a railcar. A reduced puncture environment contributes to the reduced risk in plutonium oxide shipment by rail.

The computer code HEAD, developed to perform the risk calculations, has been modified to improve the calculation of the spatial distribution of the air concentration of material released in an accident. The more rigorous treatment increases the calculated risk less than 10% over that obtained with the original approximation.

TECHNICAL PROGRESS

Risk Assessment of Plutonium Transport by Truck

(BNWL-1846, The Risk of Transporting Plutonium Oxide and Liquid Plutonium Nitrate by Truck)

The report on the first risk assessment completed in this program is currently undergoing final revision and typing. The report incorporates many fine suggestions made by reviewers of a draft distributed in December 1974. It is felt that the revisions have improved the clarity of the presentations. Plans are to issue the report at the end of July.

Risk Assessment of Plutonium Transport by Rail

Work continued on an assessment of the risk in the shipment by rail of plutonium oxide and liquid plutonium nitrate. Much of the information developed in the truck shipment risk assessment (e.g., container failure characteristics, consequences of a unit release) applies to the rail shipment analysis. Other needed information is being obtained or generated. Newly developed information on the rail accident environment has been obtained from Sandia Laboratories.

An analysis of the factors that could limit the amount of plutonium that could be transported in a rail-

car was completed. It indicated that the DOT limits on the permitted radiation levels external to the railcar will limit the number of containers per shipment more than volume, gross weight or criticality considerations.

It was determined that 90 6M containers, arranged in two groups of 45 could be transported in a railcar. The group of 45 would be arranged in 15 rows of 3 containers across, located in opposite ends of the railcar. There would be about a 10-ft separation between groups. It was also found that 100 L-10 containers, arranged in 25 rows of 4 containers across could be transported in a railcar under the radiation level limits.

Preliminary calculations of the risk in rail shipment under these loading conditions indicate that the risk in plutonium nitrate shipment is greater in the rail mode than in the truck mode while the reverse is true for plutonium oxide shipment. The

major factor in the plutonium nitrate risk increase is the greater number of packages that can be carried in a railcar. This results in a greater inertial crush environment. A reduced puncture environment in rail transport contributes to the reduced plutonium oxide shipment risk.

Risk Analysis Code Development

The risk analysis computer code, HEAD, was modified to improve the calculation of isopleth areas to represent more accurately the dispersion of material released to the atmosphere in a transport accident. The new method of calculation is not only more rigorous but also permits extending the analysis three orders of magnitude lower in air concentration of the released material. A test case run with the code demonstrated that the modification increased the calculated risk less than 10%.

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