Waste/Rock Interactions Technology
Program

Status Report on LWR Spent- Fuel Leach Tests

Y. B. Katayama
D. J. Bradley
C. O. Harvey

November 1980

Prepared for the
Office of Nuclear Waste Isolation
Under its Contract with the
U.S. Department of Energy

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the Department of Energy, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

The views, opinions and conclusions contained in this report are those of the contractor and do not necessarily represent those of the United States Government or the United States Department of Energy.

PACIFIC NORTHWEST LABORATORY
operated by
BATTELLE
for the
UNITED STATES DEPARTMENT OF ENERGY
Under Contract DE-AC06-76RLO 1830

Printed in the United States of America
Available from
National Technical Information Service
United States Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151

Price: Printed Copy \$ _____*; Microfiche \$3.00

*Pages	NTIS Selling Price
001-025	\$4.00
026-050	\$4.50
051-075	\$5.25
076-100	\$6.00
101-125	\$6.50
126-150	\$7.25
151-175	\$8.00
176-200	\$9.00
201-225	\$9.25
226-250	\$9.50
251-275	\$10.75
276-300	\$11.00

3 3679 00055 2705

Waste/Rock Interactions Technology Program

STATUS REPORT ON LWR SPENT-
FUEL LEACH TESTS

Y. B. Katayama
D. J. Bradley
C. O. Harvey

November 1980

Prepared for the
Office of Nuclear Waste Isolation
under its contract with the
U.S. Department of Energy

Pacific Northwest Laboratory
Richland, Washington 99352

ACKNOWLEDGMENTS

We gratefully acknowledge the assistance of H. T. Cartmell, J. E. Gose, R. C. Britton, J. C. Nelson, and C. E. Bigelow in conducting the experiments. We are especially thankful for the analytical work performed on the leach solutions by A. C. Leaf of the Hanford Engineering Development Laboratory.

This research is supported by the Waste/Rock Interactions Technology Program,^(a) which is being conducted by the Pacific Northwest Laboratory. The program is sponsored by the Office of Nuclear Waste Isolation, managed by Battelle Memorial Institute for the U.S. Department of Energy under Contract DE-AC06-76RLO 1830.

(a) On October 1, 1979, the Waste Isolation Safety Assessment Program (WISAP) became the Waste/Rock Interactions Technology (WRIT) Program and the Assessment of the Effectiveness of Geologic Isolation Systems (AEGIS) Program. The tests reported in this document were started under the WISAP program and are continuing under the WRIT program, which issued this report.

SUMMARY

Leach testing of spent fuels is an ongoing work element in the Waste/Rock Interactions Technology Program at the Pacific Northwest Laboratory. The purpose of these leach tests is to gather radionuclide release data to aid in the development of credible models for spent-fuel leaching. These models will be used in radionuclide release consequence analysis.

Spent fuels with burnups of 9,000, 28,000, and 54,500 MWd/MTU have been leach tested at 25°C. Three leach-test procedures, Paige, International Atomic Energy Agency and static, have been used and are being reported here. International Atomic Energy Agency and static tests were conducted in five different solutions: deionized water, sodium bicarbonate, sodium chloride, calcium chloride and Waste Isolation Pilot Plant "B" brine solutions. Elemental leach data are reported based on the release of ^{90}Sr , ^{90}Y , ^{106}Ru , ^{137}Cs , ^{144}Ce , ^{154}Eu , $^{239+240}\text{Pu}$, ^{125}Sb , ^{244}Cm , ^{129}I , ^{99}Tc , and total uranium.

This is the first time Pacific Northwest Laboratory has reported on ^{129}I and ^{99}Tc from spent fuel. These long half-life elements are important for safety assessments because under oxidizing conditions they may migrate with ground-water velocities (i.e., they exhibit very little retardation).

The termination of the Paige test showed that the plateout (radionuclide adsorption) on the test apparatus had negligible effect on the leach rate of cesium and plutonium, but a major (up to a factor of 50 times) effect on the curium leach rate. Three-hundred additional days of leach testing by the IAEA procedure, from 467 to 769 d, showed a continuation of the leaching trends observed during the first 467 d. Results from the first two static leach test series, 2 and 8 d, gave us the ^{129}I and ^{99}Tc release numbers.

CONTENTS

ACKNOWLEDGMENTS	iii
SUMMARY	v
TABLES	viii
INTRODUCTION	1
CONCLUSIONS	2
EXPERIMENTAL METHODS AND MATERIALS	3
PAIGE TESTS	3
MODIFIED IAEA TESTS	3
STATIC LEACH TEST	6
MATERIAL	6
ANALYTICAL PROCEDURES	6
CALCULATIONS	8
RESULTS AND DISCUSSION	9
PAIGE TESTS	9
MODIFIED IAEA TESTS	12
STATIC LEACH TESTS	19
REFERENCES	25
APPENDIX - IAEA LEACH DATA	29

TABLES

1	Hanford Ground-Water Composition in ppm	4
2	Chemical Analysis, Instant Ocean Synthetic Seawater Solution .	4
3	Chemical and Ionic Composition of Salt Brine	5
4	Percent of Cm, Pu and Cs Released During Paige Tests	10
5	Paige Leach Rates (g of fuel/m ² -d) for the Final Increment Leach Period Based on Selected Radionuclides at 25°C	11
6	Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in Deionized Water After 467 Days and 769 Days of Cumulative Leaching	14
7	Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in 0.03M NaCl After 467 Days and 769 Days of Cumulative Leaching	14
8	Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in 0.015M CaCl ₂ After 467 Days and 769 Days of Cumulative Leaching	15
9	Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in WIPP "B" Brine after 467 Days and 769 Days of Cumulative Leaching	16
10	Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in 0.03M NaHCO ₃ After 467 Days and 769 Days of Cumulative Leaching	16
11	Leach Solution Concentration (μg/mL) and Radionuclide Release Rate (μg of radionuclide/m ² -d) During Series 35 (769 days of Cumulative Leaching) of IAEA Leach Tests with 28,000 MWd/MTU Spent Fuel at 25°C	18
12	Leach Solution Concentration (dpm/mL) After Static Leach Testing Spent Fuel with a Burnup of 28,000 MWd/MTU at 25°C	20
13	Static Test Spent Fuel Release, in Normalized g of Fuel/m ² , for 28,000 MWd/MTU Spent Fuel at 25°C Based on the Release of Selected Radionuclides	21
14	Leach Solution Concentration (μg/mL) and Radionuclide Released (μg/m ²) During Static Leach Testing Spent Fuel with a Burnup of 28,000 MWd/MTU at 25°C	22

A.1	IAEA Leach Rate Based on ^{137}Cs , in g of Fuel/ $\text{m}^2\text{-d}$ for 28,000 MWd/MTU Spent Fuel at 25°C	29
A.2	IAEA Leach Rate Based on U in g of Fuel/ $\text{m}^2\text{-d}$ for 28,000 MWd/MTU Spent Fuel at 25°C	30
A.3	IAEA Leach Rate Based on ^{239}Pu in g of Fuel/ $\text{m}^2\text{-d}$ for 28,000 MWd/MTU Spent Fuel at 25°C	31
A.4	IAEA Leach Rate Based on ^{244}Cm in g of Fuel/ $\text{m}^2\text{-d}$ for 28,000 MWd/MTU Spent Fuel at 25°C	32
A.5	IAEA Leach Rate Based on $^{90}\text{Sr}+^{90}\text{Y}$ in g of Fuel/ $\text{m}^2\text{-d}$ for 28,000 MWd/MTU Spent Fuel at 25°C	33
A.6	IAEA Leach Rate Based on ^{125}Sb in g of Fuel/ $\text{m}^2\text{-d}$ for 28,000 MWd/MTU Spent Fuel at 25°C	34
A.7	IAEA Leach Rate Based on ^{106}Ru in g of Fuel/ $\text{m}^2\text{-d}$ for 28,000 MWd/MTU Spent Fuel at 25°C	35
A.8	IAEA Leach Rate Based on ^{244}Ce in g of Fuel/ $\text{m}^2\text{-d}$ for 28,000 MWd/MTU Spent Fuel at 25°C	36
A.9	IAEA Leach Rate Based on ^{254}Eu in g of Fuel/ $\text{m}^2\text{-d}$ for 28,000 MWd/MTU Spent Fuel at 25°C	37

INTRODUCTION

At the Pacific Northwest Laboratory (PNL)^(a) researchers in the Waste/Rock Interactions Technology (WRIT) Program are gathering radionuclide leach-rate data from spent fuel for the Office of Nuclear Waste Isolation. Radionuclide release information from spent fuel, a candidate waste form for geologic disposal, is needed to evaluate the safety of repository storage of spent fuels. These data and solutions are used both in release consequence modeling, a portion of overall safety assessment calculations, and as source material for radionuclide/rock interaction experiments.

At PNL, spent fuels have been leach tested by three different procedures: Paige tests, static leach tests, and the modified International Atomic Energy Agency (IAEA) procedure. This status report presents the final results of the Paige tests, interim results of the IAEA tests, and initial results from the static leach tests.

(a) Operated by Battelle Memorial Institute for the Department of Energy.

CONCLUSIONS

Paige leach tests of spent fuel with burnup of 9,000, 28,000 and 54,500 MWd/MTU and leached in deionized water for 822, 822 and 1399 d, respectively, were terminated and radionuclide plateout on the apparatus measured. The increase in the leach-rate value, to include plateout, for cesium and plutonium fall within the range of the experimental uncertainties for our Paige leach rates. Curium leach rates, previously reported, may be affected (up to a factor of 50 times) by addition of the plateout contribution. There is a need to determine the chemical distribution of curium in spent fuel. Based on the release of cesium, fuel burnup has no effect on the leach rate of spent fuel.

The additional 300 d of IAEA leaching (from 467 to 769 d) have shown no major changes. Deionized water continues to give the highest leach rates for all elements except antimony. Antimony had a higher leach rate in 0.03M NaHCO_3 . 0.015M CaCl_2 gave the lowest leach rate for all elements except plutonium. Plutonium had a lower leach rate in 0.03M NaHCO_3 . Cesium had the highest leach rate in three solutions (deionized water, 0.03M NaCl and 0.015M CaCl_2). Antimony had the highest leach rate in 0.03M NaHCO_3 and plutonium the highest in WIPP (Waste Isolation Pilot Plant) "B" brine. Ruthenium had the lowest leach rate in four solutions (deionized water, 0.03M NaCl , 0.03M NaHCO_3 and WIPP "B" brine). Uranium had the lowest leach rate in 0.015M CaCl_2 .

The first two series of static leach tests, 2 and 8 d, shows that 0.03M NaHCO_3 solution gives the highest release of spent fuel. This is consistent with IAEA leach test results for the same time period.

Release data for ^{129}I and ^{99}Tc , two of the long half-life radionuclides in spent fuel, are reported for the first time. Technetium-99 concentration in deionized water, 0.03M NaHCO_3 , WIPP "B" brine, 0.03M NaCl and 0.015M CaCl_2 ranged from 1200 to 2000 dpm/mL. Iodine-129 concentration in 0.03M NaHCO_3 solution after 8 d was 5.2 dpm/mL, and in deionized water after 8 d the concentration was 0.58 dpm/mL. These low retardation radionuclides have significant concentrations in the leach solutions.

In all leach tests (Paige, IAEA and static) spent fuel does not leach congruently.

EXPERIMENTAL METHODS AND MATERIALS

The three experimental methods used for determining the leach rate of spent fuel are the Paige procedure, the modified IAEA procedure and the static leach procedure. Detailed descriptions of these procedures can be found in earlier publications: Katayama (1976; 1979) (Paige tests); Katayama, Bradley and Harvey (1980) (modified IAEA leach tests); and Bradley, McVay and Coles (1980) (static leach tests).

PAIGE TESTS

Leach rates were measured by this method at ambient temperature (approximately 25°C) in a shielded facility with an airlift-pumped recirculating flow-rate of 75 L/d, which corresponds to 150 volume displacements in the leach apparatus/d. The leachant was changed at the time of sampling. Sampling was done on a daily basis for the first week, on a weekly basis for 3 wk, and on a monthly basis thereafter. Plateout samples for the Paige apparatus were not taken at each time of sampling, and radionuclides could accumulate with time on the apparatus. However, analytical samples were acidified to prevent plateout in the sample bottles.

The solutions used for the Paige test were deionized water, Hanford ground water, and synthetic sea-brine solution. The compositions of the Hanford ground water and synthetic sea brine are shown in Tables 1 and 2, respectively. Results of the short-term, approximately 1 yr, test in Hanford ground water was reported in 1977 (Katayama and Mendel 1977). The termination of the synthetic sea-brine tests after 1039 d and deionized water tests after 822 and 1399 d will be discussed in this report.

MODIFIED IAEA TESTS

The WRIT-modified IAEA leach test involves the immersion of a spent-fuel sample in a solution according to a fixed ratio of 1:10 (cm^2/cm^3) of geometric surface area of sample to volume of solution. The solution is then left in contact with the sample, at approximately 25°C ambient temperature, for progressively longer time intervals. The solution and polypropylene container is changed after each time interval. Between each sampling period the system is

TABLE 1. Hanford Ground-Water Composition
in ppm

<u>Element/Compound</u>	<u>ppm</u>
Sulfate	20.0
Nitrate	4.2
Chloride	3.1
Bicarbonate	168.0
Sodium	25.0
Potassium	5.4
Calcium	31.0
Magnesium	7.1
pH	7.9

TABLE 2. Chemical Analysis, Instant Ocean® Synthetic
Seawater Solution

<u>Element/ Compound</u>	<u>Content, ppm</u>	<u>Element/ Compound</u>	<u>Content, ppm</u>
Cl	18,400	Sr	6.0
Na	10,220	SiO ₃	3.0
SO ₄	2,518	Mn	1.3
Mg	1,238	PO ₄	1.2
K	390	F	1.0
Ca	370	MoO ₄	0.6
HCO ₃	142	S ₂ O ₃	0.3
Br	60	Li	0.2
H ₃ BO ₃	25		

®Instant Ocean is a registered name for a synthetic
sea salt manufactured by Aquarium Systems, Inc.

not disturbed by stirring or shaking. The sampling frequency is daily for the first 4 d, weekly for 8 wk, and then monthly until the tests are stopped. The plateout material is dissolved from the leach containers and analyzed.

The solutions used for the IAEA spent-fuel leach tests were:

- salt brine, WIPP "B" salt brine (Dosch and Lynch 1978)
- synthetic high-ionic-strength calcium ground water (0.015M CaCl_2)
- synthetic high-ionic-strength bicarbonate ground water (0.03M NaHCO_3)
- synthetic high-ionic-strength sodium ground water (0.03M NaCl)
- deionized water.

Table 3 shows the composition of WIPP "B" salt brine.

TABLE 3. Chemical and Ionic Composition of Salt Brine(a)

Compound	Concentration, g/L	Ion	Concentration, mole/L
NaCl	287.0	Na^+	5.0
Na_2SO_4	6.2	K^+	0.00038
$\text{Na}_2\text{B}_4\text{O}_7 \cdot 10\text{H}_2\text{O}$	0.0160	Rb^+	0.000012
NaHCO_3	0.0140	Cs^+	0.000008
NaBr	0.5200	Mg^{++}	0.00041
KCl	0.0290	Ca^{++}	0.022
KI	0.0130	Sr^{++}	0.00017
MgCl_2	0.0400	Fe^{+++}	0.000036
$\text{CaCl}_2 \cdot 2\text{H}_2\text{O}$	0.0033	Cl^-	4.94
FeCl_3	0.0060	Br^-	0.0050
$\text{SrCl}_2 \cdot 2\text{H}_2\text{O}$	0.0330	I^-	0.000079
Rb_2SO_4	0.0016	HCO_3^-	0.00016
CsCl	0.0013	SO_4	0.036
Total dissolved solids	287.2	B (BO_3^{---})	0.00017

(a) pH (adjusted) = 6.5

STATIC LEACH TEST

As the name implies, no sample agitation, solution flow, or solution replenishment is involved in the static leach tests. Six replicates per type of leach solution, started at the same time, serve as the complete test series over time. When a sampling time was reached, one replicate was sacrificed for analysis. The sampling times for ambient-temperature (approximately 25°C), spent-fuel static leach tests are 2, 8, 32, 106, 341 and 1095 d. Fused silica containers and the five leach solutions previously described for the IAEA tests are used in the static tests. Static leach tests at 75°C are now being started. These tests, combined with autoclave leach tests at 150°C and 250°C, are designed to determine the temperature effect on spent-fuel leaching.

MATERIAL

All the leach tests to date have been with unclad fuel in the form of fuel fragments of the kind shown in Figure 1. The fuels used in the leach tests were from three different reactors and had different average burnups. The Zorita reactor fuel had a burnup of 54,500 MWd/MTU, the HB Robinson-2 reactor fuel had a burnup of 28,000 MWd/MTU, and the Quad Cities-1 reactor fuel had a burnup of 9,000 MWd/MTU. Samples from all three of these fuels were leach tested by the Paige procedure. The IAEA and static leach tests were with the HB Robinson-2 reactor fuels.

ANALYTICAL PROCEDURES

The measurements of radionuclides were all made using standard radiochemical analysis methods (Katayama, Bradley and Harvey 1980). For the static leach tests the following radionuclides are being analyzed: ^{137}Cs , U (total), $^{239+240}\text{Pu}$, ^{244}Cm , $^{90}\text{Sr}+^{90}\text{Y}$, ^{125}Sb , ^{106}Ru , ^{144}Ce , ^{154}Eu , ^{99}Tc and ^{129}I . These analyses are made for both the leachate solution and the acid-dissolved plate-out material, except for ^{99}Tc and ^{129}I which are only analyzed in the leach solution. Plateout analysis for ^{99}Tc and ^{129}I will be included in future, long-period, static leach tests.

FIGURE 1. Spent LWR Fuel Fragments Photographed Through Hot-Cell Periscope
(each division on scale equals 1/16 in.)

In the IAEA leach tests, all radionuclides except ^{99}Tc and ^{129}I are analyzed in the leachate solutions and plateout solutions. In the Paige tests ^{137}Cs , $^{239+240}\text{Pu}$, ^{244}Cm and U (total) were analyzed.

CALCULATIONS

All the leach rates in this report are incremental leach rates and are average leach rates for the sampling interval. The equation used to calculate the incremental leach rate is as follows:

$$R_i = \frac{a_i}{A_0 S t} \quad (1)$$

where

R_i = incremental leach rate, g of fuel $\text{m}^{-2} \text{d}^{-1}$

a_i = activity of isotope in leachate + plateout, counts s^{-1}

A_0 = specific activity of isotope in sample before leaching, counts $\text{s}^{-1} \text{g}^{-1}$

S = geometric surface area of sample, m^2

t = incremental leaching period, d.

Leach rates are reported for the Paige and IAEA leach tests. For the static leach tests the release of spent fuel based on the measured radionuclides are reported as g of fuel/ m^2 of fuel geometric surface area, or

$$\text{radionuclide release} = \frac{a_i}{A_0 S} = \text{g fuel/m}^2 \quad (2)$$

RESULTS AND DISCUSSION

PAIGE TESTS

Paige leach tests with spent fuels were all terminated October 10, 1979, and the glass apparatus flushed with 5M HNO_3 + 0.5M HF solution to remove any radionuclides that had adhered (plateout) to the leach test apparatus walls. Plateouts calculated from the solution analysis are shown in Table 4 for ^{244}Cm , ^{137}Cs , and $^{239+240}\text{Pu}$. In all cases, ^{244}Cm has the greatest percentage (plateout versus total radionuclide released) of plateout. The plateout for the three radionuclides ranged from 0 to 43% of the cumulative fractional release of the radionuclide. Quad Cities-1 (9,000 MWd/MTU) fuel replicate sample number 2 had the highest plateout and corresponded to the highest fractional release to the leach solution (deionized water). This sample may have contained fuel fragments highly enriched with respect to curium. The chemical distribution of curium in the fuel fragment could not be determined by x-ray microprobe analysis due to the interference from the many x-ray lines of uranium.

The Paige test leach rates based on cesium, plutonium and curium were reported to be lower, by about one order of magnitude than the IAEA leach rate values (Katayama, Bradley and Harvey 1980). This difference was thought to be due to the absence of plateout measurements for the Paige tests. Cumulative plateout results should allow the calculation of corrected incremental leach rates (including plateout) if the fractional amount of this plateout for each sampling period could be calculated. The plateout during each incremental sampling period is expected to be dependent on the concentration of the radionuclide in solution and the length of the incremental period. This dependency is not known. However, by using a simple approach of plateout depending linearly with time, we can calculate a conservative incremental leach rate for the last sampling period, when the solution concentrations were the lowest and the leaching period equal to the longest (1 mo). Table 5 lists the last incremental Paige leach rates based on solution analysis and a conservatively estimated plateout. The addition of this conservatively calculated plateout does not account for the one order of magnitude difference between the Paige test and IAEA test leach rates based on cesium and plutonium. The corrected Paige

TABLE 4. Percent of Cm, Pu and Cs Released During Paige Tests

Reactor Sample Number & Fuel Burnup MWd/MTU(a)	Weight, g	Geometric Surface Area, cm ²	Solution Volume, mL	Volume/ Surface Area	Days Leached	Cumulative Totals					
						% Released to Solution			% "Plateout" on Container Walls		
						Cm	Pu	Cs	Cm	Pu	Cs
Zorita-6 54,500	5.516	12.6	500	40	1399	0.29	0.38	2.00	0.02	0.02	0.01
Zorita-7 54,500	4.431	10.0	500	50	1399	0.33	0.67	2.17	0.02	0.02	0.02
H.B. Robinson-1 28,000	4.944	4.87	500	103	822	0.06	0.17	0.40	0.01	0.00	0.01
H.B. Robinson-2 28,000	5.109	9.44	500	53	822	0.05	0.13	0.34	0.03	0.01	0.02
Quad Cities-1 9,000	4.853	5.62	500	89	822	0.31	0.17	0.43	0.02	0.00	0.00
Quad Cities-2 9,000	4.846	9.48	500	53	822	2.64 ^(d)	0.11	0.46	1.98 ^(d)	0.02	0.08
Zorita-8 ^(b) 54,500	2.722	37.60	500	13	1039	(c)	(c)	2.71	(c)	0.01	0.07

(a) Deionized water leach solution

(b) Synthetic sea-brine leach solution

(c) Not detectable

(d) May reflect uncertainties in chemical distribution of Cm in fuel

**TABLE 5. Paige Leach Rates (g of fuel/m²-d) for the Final Incremental Leach Period
Based on Selected Radionuclides at 25°C**

Reactor Sample Number & Fuel Burnup Mwd/MTU ^(a)	Days Leached	¹³⁷ Cs Based Leach Rate		²³⁹⁺²⁴⁰ Pu Based Leach Rate		²⁴⁴ Cm Based Leach Rate	
		Solution	Solution + Plateout ^(d)	Solution	Solution + Plateout ^(d)	Solution	Solution + Plateout ^(d)
Zorita-6 54,500	1399	2.3 E-2	2.4 E-2	3.4 E-3	4.2 E-3	4.5 E-4	1.2 E-3
Zorita-7 54,500	1399	4.9 E-2	5.0 E-2	4.7 E-3	5.2 E-3	5.4 E-4	1.2 E-3
H.B. Robinson-1 28,000	822	1.3 E-2	1.5 E-2	8.2 E-4	1.3 E-3	3.9 E-5	7.7 E-4
H.B. Robinson-2 28,000	822	1.3 E-2	1.4 E-2	5.1 E-2	5.6 E-2	3.1 E-4	2.6 E-3
Quad Cities-1 9,000	822	8.1 E-3	9.0 E-3	3.6 E-4	6.7 E-4	5.9 E-4	3.0 E-3
Quad Cities-2 9,000	822	9.9 E-3	1.5 E-2	6.3 E-4	1.6 E-3	2.4 E-3	1.3 E-1 ^(e)
Zorita-8 ^(b) 54,500	1039	4.0 E-3	4.3 E-3	(c)	(c)	(c)	(c)

(a) Deionized water leach solution

(b) Synthetic sea-brine leach solution

(c) Not detectable

(d) Assumed that plateout accumulation was linear with time

(e) May reflect uncertainties in chemical distribution of Cm in fuel

curium-based leach rates have increased by a factor of 2 to 54, indicating uncertainties in the chemical distribution of curium in the fuel samples.

In summary, the Paige leach tests results show:

- Cesium has the highest leach rate of the radionuclides analyzed.
- Based on the release of cesium, fuel burnup has no effect on the leach rate of spent fuel.
- Conservatively calculated plateout additions to past calculated leach rates of plutonium and cesium increase these leach rates less than the range of estimated experimental uncertainties.
- The curium-based leach rate results indicate the need to study the chemical distribution of curium in spent fuel.

MODIFIED IAEA TESTS

IAEA leach test data up to 467 d was reported earlier this year (Katayama, Bradley and Harvey 1980). Additional analytical data, up to 769 d of cumulative leaching, is now being reported. Figure 2 shows the leach rate for 28,000 MWd/MTU spent fuel in deionized water at 25°C based on the nine elements analyzed. On a logarithmic time scale the addition of five data points appears like a small contribution to the leach curves. Since there are no major changes in the trend of the leach rates, the curves for the other four solutions are not shown. Tables in the Appendix contain all available IAEA leach rate data.

Table 6 compares the IAEA leach rate data at 467 and 769 d of cumulative leaching in deionized water at 25°C. Elements are listed in order (highest to lowest) of their observed release at 467 d. The leach rate trends are all decreasing, with ^{137}Cs having the highest leach rate after 769 d. Curium-244, ^{125}Sb , and ^{154}Eu had higher (compared to ^{137}Cs) leach rates at 467 d, but had greater decreases. The ^{106}Ru leach rate remained the lowest among the nine elements measured.

Table 7 compares the IAEA leach rates at 467 and 769 d of cumulative leaching in 0.03M NaCl solution at 25°C. As in deionized water, all the leach

FIGURE 2. IAEA Leach Rate of 28,000 MWd/MTU Spent Fuel in Deionized Water at 25°C Based on Selected Elements

TABLE 6. Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in Deionized Water After 467 Days and 769 Days of Cumulative Leaching

Elements	Incremental Leach Rate, g of fuel/m ² /d		Trend with Time
	467 d	769 d	
¹⁵⁴ Eu	1.5 x 10 ⁻¹	4.6 x 10 ⁻²	Decrease
²⁴⁴ Cm	1.5 x 10 ⁻¹	3.9 x 10 ⁻²	Decrease
¹²⁵ Sb	1.3 x 10 ⁻¹	5.1 x 10 ⁻²	Decrease
¹⁴⁴ Ce	1.2 x 10 ⁻¹	3.1 x 10 ⁻²	Decrease
¹³⁷ Cs	1.2 x 10 ⁻¹	5.2 x 10 ⁻²	Decrease
²³⁹⁺²⁴⁰ Pu	1.1 x 10 ⁻¹	3.9 x 10 ⁻²	Decrease
⁹⁰ Sr+ ⁹⁰ Y	9.6 x 10 ⁻²	4.2 x 10 ⁻²	Decrease
U	5 x 10 ⁻²	3.0 x 10 ⁻²	Decrease
¹⁰⁶ Ru	1.8 x 10 ⁻²	7.2 x 10 ⁻³	Decrease

TABLE 7. Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in 0.03M NaCl After 467 Days and 769 Days of Cumulative Leaching

Elements	Incremental Leach Rate, g of fuel/m ² /d		Trend with Time
	467 d	769 d	
¹⁵⁴ Eu	6.9 x 10 ⁻²	2.4 x 10 ⁻³	Decrease
¹³⁷ Cs	6.1 x 10 ⁻²	2.0 x 10 ⁻²	Decrease
²⁴⁴ Cm	4.0 x 10 ⁻²	3.3 x 10 ⁻³	Decrease
¹⁴⁴ Ce	3.5 x 10 ⁻²	1.6 x 10 ⁻³	Decrease
¹²⁵ Sb	3.5 x 10 ⁻²	1.6 x 10 ⁻³	Decrease
²³⁹⁺²⁴⁰ Pu	3.0 x 10 ⁻²	2.8 x 10 ⁻³	Decrease
⁹⁰ Sr+ ⁹⁰ Y	3.3 x 10 ⁻²	2.1 x 10 ⁻³	Decrease
U	2 x 10 ⁻²	6 x 10 ⁻⁴	Decrease
¹⁰⁶ Ru	5.1 x 10 ⁻³	1.5 x 10 ⁻⁴	Decrease

rate trends are decreasing. The ^{137}Cs leach rate moved from second highest to highest; ^{106}Ru continues to have the lowest leach rate.

Table 8 compares the IAEA leach rates at 467 and 769 d of cumulative leaching in 0.015M CaCl_2 solution at 25°C. The ^{125}Sb and ^{154}Eu analytical data were not available to be included in this report. For all the other elements, the trend is decreasing as in deionized water and 0.03M NaCl solution. The ^{137}Cs leach rate continues to be the highest, and the uranium leach rate continues to be the lowest.

Table 9 compares the IAEA leach rate data at 467 and 769 d of cumulative leaching in WIPP "B" brine solution at 25°C. The leach rate trends are all decreasing except for uranium which is increasing. A cyclic behavior of the uranium leach rates was reported previously (Katayama, Bradley and Harvey 1980). For WIPP "B" brine solution, the uranium leach rates appear to be on the increasing portion of such a cycle.

Table 10 compares the IAEA leach rate data at 467 and 769 d of cumulative leaching in 0.03 M NaHCO_3 solution at 25°C. Five of the elemental leach rates

TABLE 8. Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in 0.015M CaCl_2 After 467 Days and 769 Days of Cumulative Leaching

Elements	Incremental Leach Rate, g of fuel/m ² /d		Trend with Time
	467 d	769 d	
^{137}Cs	8.9×10^{-3}	4.7×10^{-3}	Decrease
^{106}Ru	3.7×10^{-3}	1.9×10^{-4}	Decrease
$^{239+240}\text{Pu}$	1.6×10^{-3}	8.5×10^{-4}	Decrease
$^{90}\text{Sr}+^{90}\text{Y}$	1.4×10^{-3}	8.3×10^{-4}	Decrease
^{154}Eu	1.2×10^{-3}	NA	---
^{144}Ce	1.1×10^{-3}	2.1×10^{-4}	Decrease
^{244}Cm	9.1×10^{-4}	2.9×10^{-4}	Decrease
^{125}Sb	7.6×10^{-4}	NA	---
U	3×10^{-4}	8×10^{-5}	Decrease

NA = analytical results not available at time of publication.

TABLE 9. Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in WIPP "B" Brine After 467 Days and 769 Days of Cumulative Leaching

Elements	Incremental Leach Rate, g of fuel/m ² /d		Trend with Time
	467 d	769 d	
¹³⁷ Cs	3.8 x 10 ⁻²	2.8 x 10 ⁻²	Decrease
¹²⁵ Sb	3.8 x 10 ⁻²	2.7 x 10 ⁻²	Decrease
²³⁹⁺²⁴⁰ Pu	3.7 x 10 ⁻²	3.2 x 10 ⁻²	Decrease
⁹⁰ Sr+ ⁹⁰ Y	3.1 x 10 ⁻²	2.1 x 10 ⁻²	Decrease
²⁴⁴ Cm	3.1 x 10 ⁻²	9.7 x 10 ⁻³	Decrease
¹⁵⁴ Eu	2.2 x 10 ⁻²	1.2 x 10 ⁻²	Decrease
¹⁴⁴ Ce	9.4 x 10 ⁻³	4.4 x 10 ⁻³	Decrease
U	7 x 10 ⁻³	2 x 10 ⁻²	Increase
¹⁰⁶ Ru	3.8 x 10 ⁻³	2.1 x 10 ⁻³	Decrease

TABLE 10. Incremental Leach Rate (IAEA at 25°C) of Spent Fuel (28,000 MWd/MTU) in 0.03M NaHCO₃ After 467 Days and 769 Days of Cumulative Leaching

Elements	Incremental Leach Rate, g of fuel/m ² /d		Trend with Time
	467 d	769 d	
¹²⁵ Sb	7.0 x 10 ⁻²	7.8 x 10 ⁻²	Increase
¹³⁷ Cs	3.3 x 10 ⁻²	3.5 x 10 ⁻²	Increase
U	1 x 10 ⁻²	2 x 10 ⁻²	Increase
⁹⁰ Sr+ ⁹⁰ Y	8.2 x 10 ⁻³	1.0 x 10 ⁻²	Increase
²³⁹⁺²⁴⁰ Pu	2.1 x 10 ⁻³	7.4 x 10 ⁻⁴	Decrease
²⁴⁴ Cm	1.5 x 10 ⁻³	7.2 x 10 ⁻⁴	Decrease
¹⁰⁶ Ru	1.3 x 10 ⁻³	6.9 x 10 ⁻⁴	Decrease
¹⁵⁴ Eu	9.9 x 10 ⁻⁴	1.2 x 10 ⁻³	Increase
¹⁴⁴ Ce	1.0 x 10 ⁻³	NA	---

NA = analytical results not available at time of publication.

show an increasing trend (Cs, U, Sr+Y, Sb and Eu), and three of the elemental leach rates show a decreasing trend (Pu, Cm and Ru). These three decreasing leach rate elements now have the lowest leach rates. Antimony still has the highest leach rate.

The uranium leach rate currently shows a cyclic behavior as in WIPP "B" brine and appears to be near the top part of a cycle. The antimony leach rate tends to follow the general trend of the uranium leach rate although it is about 1/2 to 1 order of magnitude higher.

The spread in the IAEA leach rates at 769 d for all the solutions is about 1-1/2 to 2 orders of magnitude. In the IAEA tests, as in the Paige test, spent fuel does not leach congruently.

In Table 11 the IAEA leach test results for 769 d of cumulative leaching are tabulated in units for safety assessment calculations. The radionuclide release rate is shown as μg of radionuclide/ $\text{m}^2\text{-d}$ (includes leach solution plus plateout) and the radionuclide concentration in the leach solution is shown as $\mu\text{g/mL}$.

In summary, the IAEA test leach rate results show:

- Spent fuel does not leach congruently.
- Cesium has the highest leach rate in three solutions (deionized water, 0.03M NaCl and 0.015M CaCl_2). Antimony has the highest leach rate in 0.03M NaHCO_3 and plutonium the highest in WIPP "B" brine solution.
- Ruthenium has the lowest leach rate in four solutions (deionized water, 0.03M NaCl, 0.03M NaHCO_3 and WIPP "B" brine). Uranium has the lowest leach rate in 0.015M CaCl_2 solution.
- Deionized water tends to give the highest leach rates for all elements.
- 0.015M CaCl_2 solution tends to give the lowest leach rate for all the elements. Plutonium has a lower leach rate in 0.03M NaHCO_3 .

TABLE 11. Leach Solution Concentration ($\mu\text{g/mL}$) and Radionuclide Release Rate (μg of radionuclide/ $\text{m}^2\text{-d}$) During Series 35 (769 Days of Cumulative Leaching) of IAEA Tests with 28,000 MWd/MTU Spent Fuel at 25°C

Element		Deionized Water	0.03M NaHCO_3	WIPP "B" Brine	0.03M NaCl	0.015M CaCl_2
^{137}Cs	(a)	1.1E-2	7.8E-3	6.2E-3	4.3E-3	1.4E-3
	(b)	3.8E+1	2.6E+1	2.1E+1	1.4E+1	4.8E+0
Uranium	(a)	7.6E+0	4.3E+0	4.7E+0	1.1E-1	1.0E-2
	(b)	2.5E+4	1.4E+4	1.6E+4	5.3E+2	6.7E+1
$^{239+240}\text{Pu}$	(a)	7.1E-2	1.2E-3	5.0E-2	5.0E-3	1.5E-3
	(b)	2.4E+2	4.5E+0	2.0E+2	1.7E+1	5.2E+0
^{244}Cm	(a)	1.7E-4	2.1E-6	4.0E-5	1.3E-5	1.0E-6
	(b)	5.6E-1	1.0E-2	1.4E-2	4.6E-2	4.3E-3
$^{90}\text{Sr}+^{90}\text{Y}$	(a)	2.7E-3	9.7E-4	2.1E-3	2.1E-4	7.6E-5
	(b)	1.4E+1	3.2E+0	6.9E+0	6.9E-1	2.7E-1
^{125}Sb	(a)	1.2E-5	1.8E-5	6.4E-6	3.8E-7	(c)
	(b)	4.0E-2	6.1E-2	2.1E-2	1.3E-3	
^{106}Ru	(a)	2.4E-6	2.0E-7	6.6E-7	2.3E-7	6.7E-8
	(b)	8.5E-3	8.0E-4	2.4E-3	7.6E-4	2.2E-4
^{144}Ce	(a)	9.0E-6	(c)	1.2E-6	4.7E-7	6.2E-8
	(b)	3.0E-2		4.3E-3	1.6E-3	2.1E-4
^{154}Eu	(a)	1.6E-4	4.3E-6	4.3E-5	8.7E-6	(c)
	(b)	5.6E-1	1.4E-2	1.4E-1	2.9E-2	

(a) Radionuclide concentration in solution in $\mu\text{g/mL}$.

(b) Radionuclide released (solution + plateout) in μg of radionuclides/ $\text{m}^2\text{-d}$.

(c) Analytical data not available at time of publication.

STATIC LEACH TESTS

Table 12 lists available static leach solution analysis data from the first two series of tests, 2 and 8 d. Solution concentrations are shown for ^{137}Cs , total U, $^{239+240}\text{Pu}$, ^{244}Cm , $^{90}\text{Sr}+^{90}\text{Y}$, ^{125}Sb , ^{106}Ru , ^{144}Ce , ^{154}Eu , ^{99}Tc and ^{129}I . This is the first reporting by PNL on the release of ^{99}Tc and ^{129}I in spent fuel leach tests. The ^{99}Tc concentration in the solutions are greater than 1000 dpm/mL and are easily counted. The ^{129}I concentrations were measured as 7.0×10^{15} atoms/L in deionized water and as 6.3×10^{16} atoms/L in 0.03M NaHCO_3 solution. These results were converted to the dpm/mL units shown in Table 12. Solution concentration numbers are dependent on the surface area of the sample exposed to the leach solution. Therefore, in comparing or using the listed concentrations, surface areas must be considered. Sample surface areas are listed in Table 13 along with the spent fuel release, in g of fuel/m² of sample surface area, based on the release of the 11 radionuclides. Spent fuel release numbers have been adjusted to include the plateout measurements and normalized to fuel composition (Equation 2). The differences in the spent fuel release values based on each element indicate that the spent fuel is not leaching congruently.

In static leach tests, 0.03M NaHCO_3 solution gives the highest spent fuel release. Iodine-129 and ^{99}Tc , two of the long half-life radionuclides, also give high spent-fuel release in bicarbonate solution. The high-spent fuel release in bicarbonate solution is consistent with the IAEA test in which all elements tested had their highest release rate for the first 10 d of cumulative leaching in bicarbonate solution (Katayama, Bradley and Harvey 1980).

The static leach test allows study of the approach to solubility limits for various radionuclides and matrix elements, which is important in determining thermodynamic solubility constraints. In Table 14 the static leach test results are shown in units used in safety assessment calculations. The release of the radionuclide from 28,000 MWD/MTU spent fuel at 25°C is in μg of radionuclide/m² (leach solution plus plateout), and the radionuclide concentration in the leach solution is in $\mu\text{g/mL}$.

TABLE 12. Leach Solution Concentration (dpm/mL) After Static Leach Testing Spent-Fuel with a Burnup of 28,000 MWd/MTU at 25°C

Element	Deionized Water		0.03M NaHCO ₃		WIPP "B" Brine		0.03M NaCl		0.015M CaCl ₂	
	2 d	8 d	2 d	8 d	2 d	8 d	2 d	8 d	2 d	8 d
¹³⁷ Cs	9.7E+6	1.7E+7	2.4E+7	2.5E+7	7.7E+6	1.6E+7	1.2E+7	1.9E+7	1.5E+7	1.1E+7
Uranium ^(a)	4.2	2.7	62	57	4.7	10	9.8	8.4	4.8	7.6
²³⁹⁺²⁴⁰ Pu	1.0E+3	4.0E+3	1.8E+4	1.4E+4	8.8E+2	2.9E+3	1.7E+3	3.2E+3	1.2E+3	4.2E+3
²⁴⁴ Cm	7.0E+4	1.3E+3	1.4E+5	8.9E+4	8.9E+4	2.4E+5	1.5E+5	1.6E+5	1.0E+5	9.9E+4
⁹⁰ Sr+ ⁹⁰ Y	7.7E+6	(b)	1.3E+7	(b)	6.5E+6	(b)	1.0E+7	(b)	1.1E+7	(b)
¹²⁵ Sb	9.6E+3	(b)	3.4E+5	(b)	1.1E+4	(b)	1.2E+4	(b)	2.3E+4	(b)
¹⁰⁶ Ru	7.6E+4	(b)	4.2E+5	(b)	8.7E+4	(b)	1.7E+5	(b)	6.6E+5	(b)
¹⁴⁴ Ce	1.4E+5	(b)	1.9E+5	(b)	2.6E+5	(b)	2.9E+5	(b)	2.9E+5	(b)
¹⁵⁴ Eu	1.8E+5	(b)	2.7E+5	(b)	2.2E+5	(b)	3.2E+5	(b)	3.5E+5	(b)
⁹⁹ Tc	(c)	2.0E+3	(c)	2.0E+3	(c)	1.4E+3	(c)	1.2E+3	(c)	1.3E+3
¹²⁹ I	(c)	0.58	(c)	5.2E+0	(c)	(c)	(c)	(c)	(c)	(c)
		5.8E-1								

(a) µg/mL.

(b) Analytical data not available at time of publication.

(c) Analysis not requested.

TABLE 13. Static Test Spent Fuel Release, in Normalized g of Fuel/m², for
28,000 MWd/MTU Spent Fuel at 25°C Based on the Release of
Selected Radionuclides

Leachant time, d	Deionized Water		0.03M NaHCO ₃		WIPP "B" Brine		0.03M NaCl		0.015M CaCl ₂	
	2 d	8 d	2 d	8 d	2 d	8 d	2 d	8 d	2 d	8 d
Surface area, cm ²	2.129	3.568	3.626	2.742	3.045	3.013	3.439	3.407	3.297	2.955
Sample wt, g	2.594	2.603	2.551	2.492	2.561	2.483	2.583	2.447	2.435	2.629
Solution volume, mL	80.0	80.0	80.0	80.0	80.0	80.0	80.0	80.0	80.0	80.0
Solution pH	5.6	4.4	8.8	8.7	6.3	6.5	5.9	5.6	5.8	5.4
¹³⁷ Cs	26.0	31.0	38.0	54.0	28.0	32.0	20.0	31.0	19.0	21.0
Uranium	2.5	11.0	17.0	20.0	1.6	3.5	3.1	3.1	1.5	3.0
²³⁹⁺²⁴⁰ Pu	0.75	5.7	3.4	3.7	1.6	0.44	0.57	0.62	0.35	1.1
²⁴⁴ Cm	11.0	14.0	12.0	12.0	17.0	27.0	14.0	15.0	10.0	10.0
⁹⁰ Sr+ ⁹⁰ Y	15.0	(a)	14.0	(a)	11.0	(a)	12.0	(a)	(a)	(a)
¹²⁵ Sb	2.5	(a)	41.0	(a)	2.2	(a)	1.8	(a)	3.1	(a)
¹⁰⁶ Ru	3.7	(a)	11.0	(a)	3.4	(a)	4.9	(a)	19.0	(a)
¹⁴⁴ Ce	8.4	(a)	6.9	(a)	11.0	(a)	10.0	(a)	11.0	(a)
¹⁵⁴ Eu	18.0	(a)	16.0	(a)	17.0	(a)	19.0	(a)	23.0	(a)
⁹⁹ Tc	(b)	16.0	(b)	22.0	(b)	13.0	(b)	10.0	(b)	13.0
¹²⁹ I	(b)	1.6	(b)	18.6	(b)	(b)	(b)	(b)	(b)	(b)

(a) Analysis not available at time of publication.

(b) Analysis not requested.

TABLE 14. Leach Solution Concentration ($\mu\text{g/mL}$) and Radionuclide Released ($\mu\text{g/m}^2$) During Static Leach Testing Spent Fuel with a Burnup of 28,000 MWd/MTU at 25°C

Element		Deionized Water		0.03M NaHCO_3		WIPP "B" Brine		0.03M NaCl		0.015M CaCl_2	
		2 d	8 d	2 d	8 d	2 d	8 d	2 d	8 d	2 d	8 d
^{137}Cs	(a)	5.0E-2	8.8E-2	1.2E-1	1.3E-1	3.4E-2	7.1E-2	6.2E-2	9.8E-2	7.8E-2	5.7E-2
	(b)	1.9E+4	2.3E+4	2.7E+4	3.9E+4	2.0E+4	2.3E+4	1.4E+4	2.3E+4	1.4E+4	1.5E+4
Uranium	(a)	4.2E+0	2.7E+0	6.2E+1	5.7E+1	4.7E+0	1.0E+1	9.8E+0	8.4E+0	4.8E+0	7.6E+0
	(b)	2.1E+6	9.4E+6	1.4E+7	1.7E+7	1.3E+6	2.9E+6	2.6E+6	2.6E+6	1.2E+6	2.6E+6
$^{239+240}\text{Pu}$	(a)	4.2E-3	1.7E-2	7.5E-2	5.9E-2	3.2E-3	1.0E-2	7.1E-3	1.3E-2	5.0E-3	1.8E-2
	(b)	4.4E+3	3.5E+4	2.0E+4	2.2E+4	9.5E+3	4.6E+3	3.4E+3	3.7E+3	2.1E+3	6.9E+3
^{244}Cm	(a)	3.9E-4	7.2E-6	7.8E-4	4.9E-4	4.2E-4	1.1E-3	8.3E-4	8.9E-4	5.6E-4	5.5E-4
	(b)	1.5E+2	2.0E+2	1.8E+2	1.6E+2	2.3E+2	3.8E+2	1.9E+2	2.1E+2	1.5E+2	1.5E+2
$^{90}\text{Sr}^{+90}\text{Y}$	(a)	1.2E-2	(c)	2.1E-2	(c)	8.6E-3	(c)	1.6E-2	(c)	1.8E-2	(c)
	(b)	4.6E+3	---	4.6E+3	---	3.4E+3	---	3.7E+3		4.2E+3	
^{125}Sb	(a)	4.1E-6	(c)	1.4E-4	(c)	4.0E-6	(c)	5.1E-6	(c)	9.8E-6	(c)
	(b)	1.9E+0		3.2E+1		1.6E+0		1.4E+0		2.5E+0	
^{106}Ru	(a)	1.0E-5	(c)	5.6E-5	(c)	1.0E-5	(c)	2.3E-5	(c)	8.8E-5	(c)
	(b)	4.3E+0		1.3E+1		3.8E+0		5.6E+0		2.2E+1	
^{144}Ce	(a)	2.1E-5	(c)	2.8E-5	(c)	3.3E-5	(c)	4.3E-5	(c)	4.3E-5	(c)
	(b)	8.3E+0		6.8E+0		1.0E+1		1.0E+1		1.1E+1	
^{154}Eu	(a)	5.6E-4	(c)	8.4E-4	(c)	5.9E-4	(c)	9.9E-4	(c)	1.1E-3	(c)
	(b)	2.2E+2		1.9E+2		2.1E+2		2.3E+2		2.8E+2	
^{99}Tc	(a)	(d)	5.3E-2	(d)	5.3E-2	(d)	3.2E-2	(d)	3.2E-2	(d)	3.4E-2
	(b)		1.2E+4		1.5E+4		9.2E+3		7.5E+3		9.3E+3
^{129}I	(a)	(d)	1.6E-3	(d)	1.4E-2	(d)	(d)	(d)		(d)	(d)
	(b)		3.6E+2		4.2E+3						

(a) Radionuclide concentration in solution in $\mu\text{g/mL}$.
 (b) Radionuclide released (solution + plateout) in $\mu\text{g/m}^2$.
 (c) Analytical data not available at time of publication.
 (d) Analysis not requested.

Tests are in progress to determine the temperature dependency of spent fuel leach and spent-fuel solubilities. Static leach tests under reducing conditions are being planned.

In summary, the static leach test results show:

- Spent fuel does not leach congruently.
- The ^{99}Tc concentration in the leach solutions falls between that of ^{137}Cs and $^{239+240}\text{Pu}$ except for 0.03M NaHCO_3 solution where it is slightly lower than the plutonium concentration.
- Iodine-129 concentration in 0.03M NaHCO_3 leach solution is nine times greater than in deionized water.

REFERENCES

- Bradley, D. J., J. L. McVay and D. G. Coles. 1980. Leach Test Methodology for the Waste/Rock Interactions Technology Program. PNL-3326, Pacific Northwest Laboratory, Richland, Washington.
- Dosch, R. G., and A. W. Lynch. 1978. Interaction of Radionuclides with Geomedia Associated with the Waste Isolation Pilot Plant (WIPP) Site in New Mexico. SAND 78-0297, Sandia Laboratory, Albuquerque, New Mexico.
- Katayama, Y. B. 1976. Leaching of Irradiated LWR Fuel Pellets in Deionized and Typical Groundwater. BNWL-2057, Pacific Northwest Laboratory, Richland, Washington.
- Katayama, Y. B. 1979. Spent LWR Fuel Leach Tests. PNL-2982, Pacific Northwest Laboratory, Richland, Washington.
- Katayama, Y. B., D. J. Bradley and C. O. Harvey. 1980. Status Report on LWR Spent Fuel IAEA Leach Tests. PNL-3173, Pacific Northwest Laboratory, Richland, Washington.
- Katayama, Y. B., and J. E. Mendel. 1977. Leaching of Irradiated Fuel Pellets in Deionized Water, Sea Brine, and Typical Ground Water. PNL-SA-6416, Pacific Northwest Laboratory, Richland, Washington.

APPENDIX

IAEA LEACH DATA

TABLE A.1. IAEA Leach Rate Based on ^{137}Cs , in g of Fuel/m²-d for
28,000 MWd/MTU Spent Fuel at 25°C

Series	Cumulative Days	WIPP Brine	0.015M CaCl ₂ Solution	0.03M NaCl Solution	0.03M NaHCO ₃ Solution	Deionized Water
1	1	$1.2 \times 10^{+1}$	$1.2 \times 10^{+1}$	$1.1 \times 10^{+1}$	$1.4 \times 10^{+1}$	$3.8 \times 10^{+2}$
2	2	5.9×10^{-1}	8.1×10^{-1}	6.8×10^{-1}	1.2×10^0	1.3×10^0
3	3	5.1×10^{-1}	3.8×10^{-1}	4.8×10^{-1}	8.9×10^{-1}	7.5×10^{-1}
4	4	2.0×10^{-1}	1.6×10^{-1}	2.4×10^{-1}	6.4×10^{-1}	4.8×10^{-1}
5	11	1.2×10^{-1}	3.2×10^{-1}	2.5×10^{-1}	4.2×10^{-1}	6.6×10^{-1}
7	25	8.9×10^{-2}	2.1×10^{-1}	2.4×10^{-1}	1.9×10^{-1}	2.8×10^{-1}
12	60	6.4×10^{-2}	6.7×10^{-2}	1.3×10^{-1}	1.1×10^{-1}	1.6×10^{-1}
13	91	5.6×10^{-2}	6.1×10^{-2}	1.3×10^{-1}	9.9×10^{-2}	2.2×10^{-1}
15	154	3.8×10^{-2}	7.5×10^{-2}	8.8×10^{-2}	3.7×10^{-2}	2.1×10^{-1}
19	277	7.0×10^{-2}	8.8×10^{-2}	4.8×10^{-2}	2.3×10^{-2}	1.6×10^{-1}
23	404	7.1×10^{-2}	2.0×10^{-2}	3.1×10^{-2}	3.8×10^{-2}	6.8×10^{-2}
25	467	3.8×10^{-2}	8.9×10^{-3}	6.1×10^{-2}	3.3×10^{-2}	1.2×10^{-1}
27	529	4.0×10^{-2}	1.2×10^{-2}	3.4×10^{-2}	2.4×10^{-2}	6.8×10^{-2}
31	648	(a)	2.7×10^{-2}	4.1×10^{-2}	5.0×10^{-2}	7.9×10^{-2}
33	710	1.9×10^{-2}	(a)	(a)	(a)	(a)
35	769	2.8×10^{-2}	4.7×10^{-3}	2.0×10^{-2}	3.5×10^{-2}	5.2×10^{-2}

(a) Not available at time of publication.

TABLE A.2. IAEA Leach Rate Based on U in g of Fuel/m²-d for 28,000 MWd/MTU Spent Fuel at 25°C

Series	Cumulative Days	WIPP Brine	0.015M CaCl ₂ Solution	0.03M NaCl Solution	0.03M NaHCO ₃ Solution	Deionized Water
1	1	5×10^{-1}	7×10^{-1}	8×10^{-1}	2×10^0	9×10^{-1}
2	2	9×10^{-2}	1×10^{-1}	9×10^{-2}	3×10^{-1}	2×10^{-1}
3	3	4×10^{-1}	2×10^{-1}	1×10^{-1}	4×10^{-1}	1×10^{-1}
4	4	2×10^{-1}	2×10^{-1}	3×10^{-2}	1×10^{-1}	7×10^{-2}
5	11	4×10^{-2}	3×10^{-2}	4×10^{-2}	6×10^{-2}	6×10^{-2}
7	25	2×10^{-2}	2×10^{-2}	2×10^{-2}	5×10^{-3}	6×10^{-2}
12	60	2×10^{-2}	1×10^{-2}	2×10^{-2}	3×10^{-3}	5×10^{-2}
13	91	6×10^{-3}	2×10^{-2}	3×10^{-2}	4×10^{-3}	6×10^{-2}
15	154	8×10^{-3}	6×10^{-2}	6×10^{-2}	5×10^{-3}	1×10^{-1}
19	277	6×10^{-2}	6×10^{-2}	1×10^{-2}	7×10^{-3}	8×10^{-2}
23	404	6×10^{-2}	3×10^{-2}	5×10^{-3}	2×10^{-2}	5×10^{-2}
25	467	7×10^{-3}	3×10^{-4}	2×10^{-2}	1×10^{-2}	5×10^{-2}
27	529	2×10^{-2}	7×10^{-4}	5×10^{-3}	1×10^{-2}	3×10^{-2}
31	648	(a)	7×10^{-3}	1×10^{-2}	3×10^{-2}	5×10^{-2}
33	710	8×10^{-3}	(a)	(a)	(a)	(a)
35	769	2×10^{-2}	8×10^{-5}	6×10^{-4}	2×10^{-2}	3×10^{-2}

(a) Not available at time of publication.

TABLE A.3. IAEA Leach Rate Based on ^{239}Pu in g of Fuel/m²-d for
28,000 MWd/MTU Spent Fuel at 25°C

Series	Cumulative Days	WIPP Brine	0.015M CaCl ₂ Solution	0.03M NaCl Solution	0.03M NaHCO ₃ Solution	Deionized Water
1	1	2.4×10^{-1}	3.0×10^{-1}	2.9×10^{-1}	7.1×10^{-1}	5.9×10^{-1}
2	2	9.2×10^{-2}	6.5×10^{-2}	1.4×10^{-1}	2.4×10^{-1}	2.1×10^{-1}
3	3	9.7×10^{-2}	1.1×10^{-1}	1.6×10^{-1}	2.4×10^{-1}	2.4×10^{-1}
4	4	5.9×10^{-2}	7.1×10^{-2}	6.1×10^{-2}	9.7×10^{-2}	1.2×10^{-1}
5	11	2.3×10^{-2}	9.7×10^{-2}	9.2×10^{-2}	2.1×10^{-2}	2.6×10^{-1}
7	25	9.2×10^{-3}	9.1×10^{-2}	9.5×10^{-2}	1.4×10^{-2}	2.3×10^{-1}
12	60	1.4×10^{-2}	4.7×10^{-2}	7.4×10^{-2}	9.5×10^{-3}	1.5×10^{-1}
13	91	1.1×10^{-2}	3.5×10^{-2}	6.5×10^{-2}	2.0×10^{-3}	1.6×10^{-1}
15	154	2.4×10^{-2}	3.1×10^{-2}	2.7×10^{-2}	1.2×10^{-3}	1.7×10^{-1}
19	277	5.8×10^{-2}	6.7×10^{-2}	1.1×10^{-2}	4.8×10^{-3}	1.3×10^{-1}
23	404	5.3×10^{-2}	2.1×10^{-2}	7.5×10^{-3}	6.0×10^{-3}	8.1×10^{-2}
25	467	3.7×10^{-2}	1.6×10^{-3}	3.0×10^{-2}	2.1×10^{-3}	1.1×10^{-1}
27	529	9.3×10^{-3}	3.3×10^{-3}	7.6×10^{-3}	6.3×10^{-3}	6.7×10^{-2}
31	648	(a)	9.3×10^{-3}	2.2×10^{-2}	2.7×10^{-3}	8.2×10^{-2}
33	710	1.1×10^{-2}	(a)	(a)	(a)	(a)
35	769	3.2×10^{-2}	8.5×10^{-4}	2.8×10^{-3}	7.4×10^{-4}	3.9×10^{-2}

(a) Not available at time of publication.

TABLE A.4. IAEA Leach Rate Based on ^{244}Cm in g of Fuel/m²-d for 28,000 MWd/MTU Spent Fuel at 25°C

Series	Cumulative Days	WIPP Brine	0.015M CaCl_2 Solution	0.03M NaCl Solution	0.03M NaHCO_3 Solution	Deionized Water
1	1	1.2×10^0	3.0×10^0	3.0×10^0	1.8×10^0	4.1×10^0
2	2	1.2×10^{-1}	1.9×10^{-1}	1.8×10^{-1}	4.7×10^{-1}	6.7×10^{-1}
3	3	2.6×10^{-1}	1.3×10^{-1}	2.5×10^{-1}	4.0×10^{-1}	3.1×10^{-1}
4	4	1.1×10^{-1}	4.3×10^{-2}	1.1×10^{-1}	2.0×10^{-1}	1.4×10^{-1}
5	11	7.1×10^{-2}	3.5×10^{-1}	2.6×10^{-1}	2.1×10^{-2}	7.2×10^{-1}
7	25	2.5×10^{-2}	2.1×10^{-1}	1.7×10^{-1}	1.1×10^{-2}	2.8×10^{-1}
12	60	2.3×10^{-2}	7.6×10^{-2}	1.8×10^{-1}	1.4×10^{-2}	1.7×10^{-1}
13	91	1.1×10^{-2}	5.4×10^{-2}	1.3×10^{-1}	2.6×10^{-3}	2.6×10^{-1}
15	154	2.6×10^{-2}	4.5×10^{-2}	4.9×10^{-2}	1.6×10^{-3}	2.4×10^{-1}
19	277	2.8×10^{-2}	6.2×10^{-2}	2.3×10^{-2}	3.5×10^{-3}	1.9×10^{-1}
23	404	1.1×10^{-2}	6.7×10^{-4}	6.8×10^{-3}	6.8×10^{-3}	1.0×10^{-1}
25	467	3.1×10^{-2}	9.1×10^{-4}	4.0×10^{-2}	1.5×10^{-3}	1.5×10^{-1}
27	529	4.7×10^{-3}	2.6×10^{-3}	1.1×10^{-2}	8.3×10^{-3}	6.7×10^{-2}
31	648	(a)	1.2×10^{-2}	2.4×10^{-2}	2.9×10^{-3}	9.0×10^{-2}
33	710	1.1×10^{-2}	(a)	(a)	(a)	(a)
35	769	9.7×10^{-3}	2.9×10^{-4}	3.3×10^{-3}	7.2×10^{-4}	3.9×10^{-2}

(a) Not available at time of publication.

TABLE A.5. IAEA Leach Rate Based on $^{90}\text{Sr}+^{90}\text{Y}$ in g of Fuel/m²-d for 28,000 MWd/MTU Spent Fuel at 25°C

Series	Cumulative Days	WIPP Brine	0.015M CaCl_2 Solution	0.03M NaCl Solution	0.03M NaHCO_3 Solution	Deionized Water
1	1	1.5×10^0	1.5×10^0	1.4×10^0	1.7×10^0	2.3×10^0
2	2	2.1×10^{-1}	2.0×10^{-1}	2.2×10^{-1}	3.9×10^{-1}	5.4×10^{-1}
3	3	2.5×10^{-1}	1.9×10^{-1}	2.7×10^{-1}	3.4×10^{-1}	3.1×10^{-1}
4	4	1.2×10^{-2}	1.4×10^{-1}	1.5×10^{-1}	3.4×10^{-1}	1.4×10^{-1}
5	11	6.8×10^{-2}	2.3×10^{-1}	1.7×10^{-1}	2.4×10^{-2}	3.1×10^{-2}
7	25	6.3×10^{-2}	1.4×10^{-1}	1.9×10^{-1}	1.1×10^{-2}	8.1×10^{-2}
12	60	3.5×10^{-2}	2.4×10^{-2}	6.8×10^{-2}	1.4×10^{-2}	8.1×10^{-2}
13	91	3.9×10^{-2}	2.9×10^{-2}	5.5×10^{-2}	5.7×10^{-3}	1.2×10^{-1}
15	154	5.2×10^{-2}	3.6×10^{-2}	3.9×10^{-2}	5.7×10^{-3}	1.4×10^{-1}
19	277	2.8×10^{-2}	3.4×10^{-2}	7.6×10^{-3}	3.5×10^{-3}	5.8×10^{-2}
23	404	2.8×10^{-2}	7.2×10^{-3}	3.6×10^{-3}	7.1×10^{-3}	3.9×10^{-2}
25	467	3.1×10^{-2}	1.4×10^{-3}	3.3×10^{-2}	8.2×10^{-3}	9.6×10^{-2}
27	529	1.4×10^{-2}	1.7×10^{-3}	4.5×10^{-3}	5.3×10^{-3}	2.4×10^{-2}
31	648	(a)	(a)	(a)	(a)	(a)
33	710	(a)	(a)	(a)	(a)	(a)
35	769	2.1×10^{-2}	8.3×10^{-4}	2.1×10^{-3}	1.0×10^{-2}	4.2×10^{-2}

(a) Not available at time of publication.

TABLE A.6. IAEA Leach Rate Based on ^{125}Sb in g of Fuel/m²-d for 28,000 MWd/MTU Spent Fuel at 25°C

Series	Cumulative Days	WIPP Brine	0.015M CaCl_2 Solution	0.03M NaCl Solution	0.03M NaHCO_3 Solution	Deionized Water
1	1	1.1×10^0	1.4×10^0	1.4×10^0	4.6×10^0	2.0×10^0
2	2	4.6×10^{-1}	2.4×10^{-1}	5.1×10^{-1}	6.9×10^{-1}	7.9×10^{-1}
3	3	2.9×10^{-1}	2.3×10^{-1}	3.0×10^{-1}	4.5×10^{-1}	3.6×10^{-1}
4	4	1.6×10^{-1}	9.8×10^{-2}	1.3×10^{-1}	2.3×10^{-1}	2.3×10^{-1}
5	11	6.2×10^{-2}	1.7×10^{-1}	1.4×10^{-1}	1.1×10^{-1}	3.4×10^{-1}
7	25	5.0×10^{-2}	1.3×10^{-1}	1.5×10^{-1}	2.9×10^{-2}	1.4×10^{-1}
12	60	3.0×10^{-2}	2.1×10^{-2}	5.4×10^{-2}	5.6×10^{-2}	1.1×10^{-1}
13	91	3.0×10^{-2}	1.5×10^{-2}	3.1×10^{-2}	2.8×10^{-2}	1.3×10^{-1}
15	154	2.9×10^{-2}	1.6×10^{-2}	1.9×10^{-2}	2.7×10^{-2}	1.8×10^{-1}
19	277	8.1×10^{-2}	1.0×10^{-1}	1.8×10^{-2}	3.7×10^{-2}	1.7×10^{-1}
23	404	8.2×10^{-2}	1.8×10^{-2}	6.7×10^{-3}	7.7×10^{-2}	1.0×10^{-1}
25	467	3.8×10^{-2}	7.6×10^{-4}	3.5×10^{-2}	7.0×10^{-2}	1.3×10^{-1}
27	529	3.5×10^{-2}	3.4×10^{-3}	1.1×10^{-2}	5.6×10^{-2}	6.3×10^{-2}
31	648	(a)	(a)	(a)	(a)	(a)
33	710	(a)	(a)	(a)	(a)	(a)
35	769	2.7×10^{-2}	(a)	1.6×10^{-3}	7.8×10^{-2}	5.1×10^{-2}

(a) Not available at time of publication.

TABLE A.7. IAEA Leach Rate Based on ^{106}Ru in g of Fuel/m²-d for 28,000 MWd/MTU Spent Fuel at 25°C

Series	Cumulative Days	WIPP Brine	0.015M CaCl_2 Solution	0.03M NaCl Solution	0.03M NaHCO_3 Solution	Deionized Water
1	1	3.1×10^{-1}	2.8×10^{-1}	3.5×10^{-1}	8.7×10^{-1}	3.7×10^{-1}
2	2	5.1×10^{-2}	1.2×10^{-1}	4.8×10^{-2}	1.4×10^{-1}	1.5×10^{-1}
3	3	2.0×10^{-1}	4.2×10^{-2}	6.0×10^{-2}	1.2×10^{-1}	4.8×10^{-2}
4	4	8.9×10^{-2}	2.3×10^{-2}	4.3×10^{-2}	7.6×10^{-2}	3.0×10^{-2}
5	11	7.3×10^{-2}	4.4×10^{-2}	3.8×10^{-2}	1.8×10^{-2}	2.1×10^{-2}
7	25	5.0×10^{-2}	1.9×10^{-2}	2.1×10^{-2}	1.1×10^{-2}	2.5×10^{-2}
12	60	5.0×10^{-2}	4.8×10^{-3}	8.9×10^{-3}	5.5×10^{-3}	1.5×10^{-2}
13	91	1.9×10^{-2}	3.6×10^{-3}	5.8×10^{-3}	7.9×10^{-4}	1.2×10^{-2}
15	154	2.0×10^{-2}	2.8×10^{-3}	5.0×10^{-3}	2.1×10^{-3}	2.0×10^{-2}
19	277	7.9×10^{-3}	6.0×10^{-2}	3.3×10^{-3}	3.2×10^{-3}	1.5×10^{-2}
23	404	5.4×10^{-3}	1.2×10^{-2}	1.7×10^{-3}	5.1×10^{-3}	1.1×10^{-2}
25	467	3.8×10^{-3}	3.7×10^{-3}	5.1×10^{-3}	1.3×10^{-3}	1.8×10^{-2}
27	529	1.0×10^{-3}	7.3×10^{-4}	2.0×10^{-3}	4.7×10^{-3}	7.9×10^{-3}
31	648	(a)	(a)	(a)	(a)	(a)
33	710	(a)	(a)	(a)	(a)	(a)
35	769	2.1×10^{-3}	1.9×10^{-4}	6.5×10^{-4}	6.9×10^{-4}	7.2×10^{-3}

(a) Not available at time of publication.

TABLE A.8. IAEA Leach Rate Based on ^{144}Ce in g of Fuel/m²-d for 28,000 MWd/MTU Spent Fuel at 25°C

Series	Cumulative Days	WIPP Brine	0.015M CaCl_2 Solution	0.03M NaCl Solution	0.03M NaHCO_3 Solution	Deionized Water
1	1	1.5×10^0	8.8×10^{-1}	1.3×10^0	6.3×10^{-1}	1.5×10^0
2	2	8.2×10^{-2}	1.7×10^{-1}	9.6×10^{-2}	2.6×10^{-1}	4.6×10^{-1}
3	3	2.9×10^{-1}	7.1×10^{-2}	1.4×10^{-1}	2.6×10^{-1}	2.3×10^{-1}
4	4	1.1×10^{-1}	4.9×10^{-2}	1.1×10^{-1}	2.0×10^{-1}	1.8×10^{-1}
5	11	7.7×10^{-2}	2.5×10^{-1}	2.2×10^{-1}	1.7×10^{-2}	5.0×10^{-1}
7	25	2.6×10^{-2}	1.6×10^{-1}	1.5×10^{-1}	1.5×10^{-2}	1.2×10^{-1}
12	60	3.2×10^{-2}	4.4×10^{-2}	1.2×10^{-1}	1.7×10^{-2}	1.5×10^{-1}
13	91	7.4×10^{-3}	(a)	(a)	(a)	(a)
15	154	5.1×10^{-2}	2.1×10^{-2}	1.2×10^{-2}	1.2×10^{-3}	1.7×10^{-1}
19	277	5.6×10^{-3}	1.0×10^{-1}	2.4×10^{-2}	6.0×10^{-3}	1.5×10^{-1}
23	404	7.0×10^{-3}	1.8×10^{-2}	8.4×10^{-3}	9.7×10^{-3}	1.2×10^{-1}
25	467	9.4×10^{-3}	1.1×10^{-3}	3.5×10^{-2}	1.0×10^{-3}	1.2×10^{-1}
27	529	1.2×10^{-3}	3.6×10^{-3}	7.3×10^{-3}	7.2×10^{-3}	8.4×10^{-2}
31	648	(a)	(a)	(a)	(a)	(a)
33	710	(a)	(a)	(a)	(a)	(a)
35	769	4.4×10^{-3}	2.1×10^{-4}	1.6×10^{-3}	(a)	3.1×10^{-2}

(a) Not available at time of publication.

TABLE A.9. IAEA Leach Rate Based on ^{154}Eu in g of Fuel/m²-d for 28,000 MWd/MTU Spent Fuel at 25°C

Series	Cumulative Days	WIPP Brine	0.015M CaCl_2 Solution	0.03M NaCl Solution	0.03M NaHCO_3 Solution	Deionized Water
1	1	1.4×10^0	1.3×10^0	1.6×10^0	8.7×10^{-1}	2.1×10^0
2	2	1.0×10^{-1}	2.0×10^{-1}	1.2×10^{-1}	3.6×10^{-1}	5.1×10^{-1}
3	3	3.1×10^{-1}	1.0×10^{-1}	1.8×10^{-1}	3.4×10^{-1}	2.5×10^{-1}
4	4	2.9×10^{-1}	7.8×10^{-2}	1.5×10^{-1}	2.9×10^{-1}	1.8×10^{-1}
5	11	8.0×10^{-2}	2.8×10^{-1}	2.2×10^{-1}	2.6×10^{-2}	4.9×10^{-1}
7	25	3.0×10^{-2}	2.1×10^{-1}	2.1×10^{-1}	2.3×10^{-2}	1.6×10^{-1}
12	60	3.1×10^{-2}	5.7×10^{-2}	1.1×10^{-1}	1.8×10^{-2}	1.0×10^{-1}
13	91	1.3×10^{-2}	3.9×10^{-2}	8.4×10^{-2}	(a)	1.3×10^{-1}
15	154	3.1×10^{-2}	4.9×10^{-2}	5.3×10^{-2}	2.0×10^{-3}	2.1×10^{-1}
19	277	9.5×10^{-3}	1.1×10^{-1}	2.5×10^{-2}	5.1×10^{-3}	2.0×10^{-1}
23	404	1.9×10^{-2}	7.0×10^{-3}	9.0×10^{-3}	8.2×10^{-3}	1.2×10^{-1}
25	467	2.2×10^{-2}	1.2×10^{-3}	6.9×10^{-2}	9.9×10^{-4}	1.5×10^{-1}
27	529	5.8×10^{-3}	4.3×10^{-3}	9.6×10^{-3}	7.2×10^{-3}	8.9×10^{-2}
31	648	(a)	(a)	(a)	(a)	(a)
33	710	(a)	(a)	(a)	(a)	(a)
35	769	1.2×10^{-2}	(a)	2.4×10^{-3}	1.2×10^{-3}	4.6×10^{-2}

(a) Not available at time of publication.

DISTRIBUTION

<u>No. of Copies</u>	<u>No. of Copies</u>
<u>OFFSITE</u>	J. B. Whitsett DOE Idaho Operations Office 550 2nd St. Idaho Falls, ID 38401
A. A. Churm DOE Chicago Patent Group 9800 South Cass Avenue Argonne, IL 60439	C. R. Cooley DOE Nuclear Waste Management Programs NEW, B-107, HQ Washington, DC 20545
R. Y. Lowrey DOE Albuquerque Operations Office P. O. Box 5400 Albuquerque, NM 87185	G. H. Daly DOE Nuclear Waste Management Programs NEW, B-107, HQ Washington, DC 20545
A. L. Taboas DOE Albuquerque Operations Office P. O. Box 5400 Albuquerque, NM 87185	J. E. Dieckhoner DOE Nuclear Waste Management Programs NEW, B-107, HQ Washington, DC 20545
S. A. Mann DOE Chicago Operations and Region Office Argonne, IL 60439	C. H. George DOE Nuclear Waste Management Programs NEW, B-107, HQ Washington, DC 20545
J. Neff Department of Energy Columbus Program Office 505 King Avenue Columbus, OH 43201	C. A. Heath DOE Nuclear Waste Management Programs NEW, B-107, HQ Washington, DC 20545
W. E. Mott DOE Division of Environmental Control Technology Washington, DC 20545	M. L. Lawrence DOE Nuclear Waste Management Programs NEW, B-107, HQ Washington, DC 20545
J. P. Hamric DOE Idaho Operations Office 550 2nd St. Idaho Falls, ID 38401	D. J. McGoff DOE Nuclear Waste Management Programs NEW, B-107, HQ Washington, DC 20545
J. W. Peel DOE Idaho Operations Office 550 2nd St. Idaho Falls, ID 38401	

No. of
Copies

S. Meyers/R. Romatowski
DOE Nuclear Waste Management
Programs
NEW, B-107, HQ
Washington, DC 20545

G. Oertel
DOE Nuclear Waste Management
Programs
NEW, B-107, HQ
Washington, DC 20545

A. F. Perge
DOE Nuclear Waste Management
Programs
NEW, B-107, HQ
Washington, DC 20545

R. W. Ramsey, Jr.
DOE Nuclear Waste Management
Programs
NEW, B-107, HQ
Washington, DC 20545

D. L. Vieth
DOE Nuclear Waste Management
Programs
NEW, B-107, HQ
Washington, DC 20545

R. D. Walton
DOE Nuclear Waste Management
Programs
NEW, B-107, HQ
Washington, DC 20545

S. W. Ahrends
DOE Oak Ridge Operations Office
P. O. Box E
Oak Ridge, TN 37830

D. Large
DOE Oak Ridge Operations Office
P. O. Box E
Oak Ridge, TN 37830

No. of
Copies

S. G. Harbinson
DOE San Francisco Operations
Office
1333 Broadway
Oakland, CA 94612

E. S. Goldberg
DOE Savannah River Operations
Office
P. O. Box A
Aiken, SC 29801

T. B. Hindman
DOE Savannah River Operations
Office
P. O. Box A
Aiken, SC 29801

R. P. Whitfield
DOE Savannah River Operations
Office
P. O. Box A
Aiken, SC 29801

Los Alamos Scientific Laboratory
P. O. Box 1663
Los Alamos NM 87544

J. B. Martin
Division of Waste Management
Nuclear Regulatory Commission
Washington, DC 20555

D. M. Rohrer
Division of Waste Management
Nuclear Regulatory Commission
Washington, DC 20555

R. Dale Smith
Division of Waste Management
Nuclear Regulatory Commission
Washington, DC 20555

R. E. Cunningham
Office of Nuclear Safety
Materials and Safeguards
Nuclear Regulatory Commission
Room 562, 7915 Eastern Avenue
Silver Springs, MD 20910

No. of
Copies

No. of
Copies

27 DOE Technical Information Center

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

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

J. W. Bartlett
The Analytical Sciences Corp.
6 Jacob Way
Reading, MA 01867

H. Diamond
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

J. H. Kittel
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

D. Lam
Argonne National Laboratory
9700 South Cass Avenue
Argonne, IL 60439

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

A. Brandstetter
Battelle Memorial Institute
Office of Nuclear Waste
Isolation
505 King Avenue
Columbus, OH 43201

W. Carbiener
Battelle Memorial Institute
Office of Nuclear Waste
Isolation
505 King Avenue
Columbus, OH 43201

N. E. Carter
Battelle Memorial Institute
Office of Nuclear Waste
Isolation
505 King Avenue
Columbus, OH 43201

P. L. Hofmann
Battelle Memorial Institute
Office of Nuclear Waste
Isolation
505 King Avenue
Columbus, OH 43201

M. Kehnemuyi
Battelle Memorial Institute
Office of Nuclear Waste
Isolation
505 King Avenue
Columbus, OH 43201

J. Kircher
Office of Nuclear Waste
Isolation
Battelle Memorial Institute
505 King Avenue
Columbus, OH 43201

Beverly Rawles
Battelle Memorial Institute
Office of Nuclear Waste
Isolation
505 King Avenue
Columbus, OH 43201

R. Maher, Program Manager
Waste Management Programs
Savannah River Plant
E. I. Du Pont de Nemours & Co.
Aiken, SC 29801

No. of
Copies

P. Colombo
Department of Applied Science
Brookhaven National Laboratory
Uptown, NY 11973

W. H. Baker
E. I. Du Pont de Nemours & Co.
Savannah River Laboratory
Aiken, SC 29801

J. L. Crandall
E. I. Du Pont de Nemours & Co.
Savannah River Laboratory
Aiken, SC 29801

R. G. Garvin
E. I. Du Pont de Nemours & Co.
Savannah River Laboratory
Aiken, SC 29801

L. W. Meyer
E. I. Du Pont de Nemours & Co.
Savannah River Laboratory
Aiken, SC 29801

S. Mirshak
E. I. Du Pont de Nemours & Co.
Savannah River Laboratory
Aiken, SC 29801

J. A. Porter
E. I. Du Pont de Nemours & Co.
Savannah River Laboratory
Aiken, SC 29801

G. B. Levin
EG & G Idaho
P. O. Box 1625
Idaho Falls, ID 83415

R. Williams
Electric Power Research
Institute
3412 Hillview Avenue
P. O. Box 10412
Palo Alto, CA 94304

No. of
Copies

J. L. Larocca, Chairman
Engineering Research and
Development Authority
Empire State Plaza
Albany, NY 12223

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

W. F. Holcomb
Office of Radiation Programs
(A-460)
U. S. Environmental Protection
Agency
Washington, DC 20460

D. M. Rosenbaum
Office of Radiation Programs
U.S. Environmental Protection
Agency
1921 Jefferson Davis Highway
Arlington, VA 22202

R. A. Brown
Exxon Nuclear Idaho
P. O. Box 2800
Idaho Falls, ID 83401

G. L. Ritter
Exxon Nuclear Idaho
P. O. Box 2800
Idaho Falls, ID 83401

T. R. Thomas
Exxon Nuclear Idaho
P. O. Box 2800
Idaho Falls, ID 83401

File Copy
Exxon Nuclear Idaho
P. O. Box 2800
Idaho Falls, ID 83401

No. of
Copies

G. E. Benedict
General Atomics Co.
P. O. Box 81608
San Diego, CA 92138

L. H. Brooks
Gulf Energy and Environmental
Systems
P. O. Box 81608
San Diego, CA 92138

D. Coles
Lawrence Livermore Laboratory
P. O. Box 808
Livermore, CA 94550

J. D. Tewhey
Lawrence Livermore Laboratory
P. O. Box 808
Livermore, CA 94550

A. E. Norris
Los Alamos Scientific Laboratory
CNC-11 Mailstop 514
Los Alamos, NM 87545

A. Ogard
Los Alamos Scientific Laboratory
CNC-11 Mailstop 514
Los Alamos, NM 87545

R. Roy
202 Materials Research
Laboratory
Pennsylvania State University
University Park, PA 16802

B. Scheetz
210 Materials Research
Laboratory
Pennsylvania State University
University Park, PA 16802

W. White
210 Materials Research
Laboratory
Pennsylvania State University
University Park, PA 16802

No. of
Copies

C. J. Kershner
Monsanto Research Corporation
Mound Laboratory
P. O. Box 32
Miamisburg, OH 45342

J. P. Duckworth
Plant Manager
Nuclear Fuels Services, Inc.
P. O. Box 124
West Valley, NY 14171

R. E. Blanco
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, TN 37830

J. O. Blomeke
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, TN 37830

D. E. Ferguson
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, TN 37830

A. L. Lotts
Oak Ridge National Laboratory
P. O. Box X
Oak Ridge, TN 37830

R. S. Lowrie
Oak Ridge National Laboratory
P. O. Box Y
Oak Ridge, TN 37830

A. B. Martin
Rockwell International
8900 DeSoto Avenue
Canoga Park, CA 91304

H. Recht
Dept. 737-710
Atomics International Division
Rockwell International
Box 309
Canoga Park, CA 91304

No. of
Copies

W. S. Bennett
Rockwell International
Rocky Flats Plant
P. O. Box 464
Golden, CO 80401

Lawrence J. Smith
Rockwell International
Rocky Flats Plant
P. O. Box 464
Golden, CO 80401

E. Vejvoda
Rockwell International
Rocky Flats Plant
P. O. Box 464
Golden, CO 80401

D. R. Anderson
Sandia Laboratories
Albuquerque, NM 87185

O. E. Jones
Sandia Laboratories
Albuquerque, NM 87185

R. G. Kepler
Sandia Laboratories
Albuquerque, NM 87185

M. A. Molecke
Sandia Laboratories
Albuquerque, NM 87185

W. Weart
Sandia Laboratories
Albuquerque, NM 87185

D. E. Harrison
Westinghouse Electric Corp.
P. O. Box 10864
Pittsburgh, PA 15236

P. Macedo
Vitreous State Laboratory
Catholic University of America
Washington, DC 20064

No. of
Copies

R. G. Post
College of Engineering
University of Arizona
Tucson, AZ 85721

L. L. Hench
Dept. of Materials Science
and Engineering
University of Florida
Gainesville, FL 32611

Hayne Palmour III
2140 Burlington Engineering
Laboratories
North Carolina State University
Raleigh, NC 27607

F. K. Pittman
3508 Sagecrest Terrace
Ft. Worth, TX 76109

ONSITE

7 DOE Richland Operations Office

E. A. Bracken
P. A. Craig
R. E. Gerton
H. E. Ransom
J. J. Schreiber
M. W. Shupe
M. J. Zamorski

11 Rockwell Hanford Operations

H. Babad
L. C. Brown
R. A. Deju
R. J. Gimera
D. R. Gustavson
E. J. Kosiancic
C. M. Manry
I. E. Reep
J. H. Roecker
D. D. Wodrich
File Copy

No. of
Copies

UNC United Nuclear Industries

T. E. Dabrowski

4 Westinghouse Hanford Company

A. G. Blasewitz

R. J. Cash

R. L. Fish

G. L. Richardson

107 Pacific Northwest Laboratory

G. L. Benson

W. J. Bjorklund

W. F. Bonner

D. J. Bradley

D. B. Cearlock

T. D. Chikalla

M. O. Cloninger

G. W. Dawson

W. J. Deutsch

C. M. Devary

F. H. Dove

J. W. Finnigan

C. R. Hann

M. S. Hanson

C. O. Harvey (2)

A. J. Haverfield

B. Hawes

M. H. Henry

O. F. Hill

J. H. Jarrett

Y. B. Katayama (25)

R. S. Kemper

D. E. Knowlton

M. R. Kreiter

K. M. Krupka

W. L. Kuhn

No. of
Copies

L. T. Lakey

J. M. Latkovich

G. B. Long (2)

R. P. Marshall

S. A. McCullough

J. L. McElroy

G. L. McVay

G. B. Mellinger

J. E. Mendel

R. D. Nelson

R. E. Nightingale

D. E. Olesen

C. R. Palmer

L. R. Pederson

P. J. Peltó

R. D. Peters

A. M. Platt

D. L. Prezbindowski (2)

D. Rai

J. F. Relyea

F. P. Roberts

W. A. Ross

J. M. Rusin

J. L. Ryan

R. J. Serne

J. W. Shade

D. H. Siemens

D. J. Silviera

R. G. Strickert

J. L. Swanson

M. T. Thomas

R. P. Turcotte

H. H. Van Tuyl

B. E. Vaughan

R. Wang

J. H. Westsik, Jr.

L. D. Williams

Technical Information (5)

Publishing Coordination Vi (2)

Water and Land Resource

Library (10)