Computed Isotopic Inventory and Dose Assessment for SRS Fuel and Target Assemblies

by

M. C. Chandler

Westinghouse Savannah River Company

Savannah River Site

Aiken, South Carolina 29808

E T Kelusky

SAIC

D C Thoman

SAIC

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**HIGH LEVEL WASTE ENGINEERING
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By

M. C. Chandler, WSRC

E. T. Ketusky, SAIC

D. C. Thoman, SAIC

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James Robert Hester
ADC/RO

Approvals

Michael C. Chandler
M. C. Chandler, Author, HLWE (Document Sponsor)

Date: 6/21/95

Gregory J. Hutchens
G. J. Hutchens, Technical Reviewer, HLWE

Date: 6/20/95

Celia I. Aponte
C. I. Aponte, Technical Reviewer, HLWE

Date: 6/20/95

R. H. Ross
R. H. Ross, Technical Reviewer, HLWE

Date: 6/20/95

James Robert Hester
J. R. Hester, Technical Reviewer, HLWE

Date: 6/20/95

J. E. Marra
J. E. Marra, Manager, HLWE Support

Date: 6/22/95

T. M. Monahan for
T. M. Monahan, Manager, HLWE

Date: 7/10/95

INTRODUCTION

Past studies have identified and evaluated important radionuclide contributors to dose from reprocessed spent fuel sent to waste for Mark 16B and 22 fuel assemblies and for Mark 31A and 31B target assemblies (References 8 - 11). Fission-product distributions after a 5- and 15-year decay time were calculated for a "representative" set of irradiation conditions (i.e., reactor power, irradiation time, and exposure) for each type of assembly (Reference 12). The numerical calculations were performed using the SHIELD/GLASS system of codes (References 2 - 6).

The sludge and supernate source terms for dose were studied separately with the significant radionuclide contributors for each identified and evaluated (References 8 - 11). Dose analysis considered both inhalation and ingestion pathways. The inhalation pathway was analyzed for both evaporative and volatile releases. Analysis of evaporative releases utilized release fractions for the individual radionuclides as defined in the ICRP-30 by DOE guidance (Reference 13). A release fraction of unity was assumed for each radionuclide under volatile-type releases, which would encompass internally initiated events (e.g., fires, explosions), process-initiated events (e.g., spills, improper transfer of materials), and externally initiated events (e.g., floods, earthquakes). Radionuclides which contributed at least 1% to the overall dose were designated as significant contributors.

The present analysis extends and complements the past analyses through considering a broader spectrum of fuel types and a wider range of irradiation conditions. The results provide for a more thorough understanding of the influences of fuel composition and irradiation parameters (i.e., reactor power, irradiation time, and exposure) on fission product distributions (at 2 years or more). Additionally, the present work allows for a more comprehensive evaluation of radionuclide contributions to dose and an estimation of the variability in the radionuclide composition of the dose source term that results from the spent fuel sent to waste encompassing a broad spectrum of fuel compositions and irradiation conditions.

SUMMARY

This report analyzes the variability of fission product distributions in waste sludge from reprocessed Savannah River Site (SRS) spent fuels and targets. Distributions are analyzed for various time periods (in the range of 2 - 20 years) following irradiation in SRS reactors. The analyses of this report utilize numerical calculations of fission product and actinide quantities in spent fuels and targets, which were performed (Reference 1) using the SHIELD/GLASS codes (References 2 - 6) over a wide spectrum of fuel compositions and irradiation conditions that are representative of the history of reactor operations at SRS (Reference 7).

Calculations which are documented in this report involve application of reprocessing separation factors to fission products and actinides to determine relative quantities of these radionuclides sent to waste. Through the application of solubility data, estimates of radionuclide composition in the sludge portion of the waste are determined (supernate radionuclide compositions are not studied). These numerical estimates of sludge composition are then compared to available tank

inventory data which provide estimates of selected sludge constituents (e.g., plutonium and uranium isotopes) based largely upon accountability records of the reprocessing facilities. Best agreement was observed when the waste in the tanks originated from fuel reprocessed in the Canyons within the last 15 to 20 years, came from the HHW stream, and contained no offsite fuel.

The impact of radionuclide variability (due to influences of variable fissile material composition, exposure, reactor power, and time since irradiation) on dose is also addressed in this report. The results of analyses showed the following list of radionuclides to be the important contributors to dose: Sr-90, Ru-106, Cs-137, Ce-144, Pm-147, Pu-238, Pu-239, Pu-240, and Pu-241.

CONCLUSIONS

Source dose calculations for waste sludge were performed for both fuel assemblies undergoing HM processing and target assemblies undergoing PUREX processing. The results confirm those of past analyses in identifying the following radionuclides, after a four-year decay time, as contributing significantly to dose (i.e., contributing at least 1% of the inhalation or ingestion dose): Sr-90, Ru-106, Cs-137, Ce-144, Pm-147, Pu-238, Pu-239, Pu-240, and Pu-241 (Am-241 which has been identified as an important contributor in the past was not analyzed due to its unavailability from the SHIELD/GLASS results used here). The analysis exceeds those of the past through the consideration of a larger variety of fuel types and a wider range of irradiation conditions.

Also, the feasibility of using the output of SHIELD/GLASS results to model radionuclide distributions in the tank farms was investigated. HM processing of spent fuel assemblies was considered since radionuclide data is more readily available for sludge sent to waste from the HM process than it is for the PUREX process. The uranium and plutonium isotopic content of monthly sludge deposits made to the individual tanks of the tank farms has been estimated and documented in Reference 16. These inventory estimates were based upon accountability records (which rely on process control samples) and estimations based upon process data and knowledge. Best agreement between the numerical predictions and the inventory data was achieved when the inventory records showed that the waste deposits occurred within the last 15 to 20 years and had no offsite fuel mixed in (waste could be considered 100% Mark 16B and 22 spent fuel) and came from the HHW stream (separation factors used in the calculations are based on that stream).

RECOMMENDATIONS

The lack of agreement in some cases between the tank inventory data and the numerical predictions suggests areas in which the prediction methodology can be upgraded with future work. Modeling of the composition of the sludge in the waste tanks could be advanced and improved by the following steps: (1) development of a database which tracks fuel (including offsite) that has been processed in F and H Canyons on a monthly basis providing information on fuel type, quantity, and irradiation conditions (reactor power and exposure); (2) development of separation fractions for the LHW stream; (3) execution of additional SHIELD/GLASS runs if necessary to cover a broader variety of fuels and wider range of exposures; and (4) development

of a database which identifies for each batch of fuel processed and sent to waste the most appropriate SHIELD/GLASS results to use for modeling purposes. Note also that the separation factors (Reference 12) presently available and used in this report are based upon averages observed in the separation canyons in 1976 - 1977. The variability of separation factors over the lifetime of separation operations should be investigated and incorporated into the analysis, if necessary.

DISCUSSION

Background

The SHIELD/GLASS codes were used in Reference 1 to calculate the production of radionuclides in SRS reactor assemblies and their decay over a period of time. Various charge types encompassing different types of fuel and a variety of lattice arrangements were chosen for analysis to reflect a broad range of reactor operations for the purpose of assessing impact on variability in radionuclide compositions. For each charge type, calculations of radionuclide quantities in the spent fuels and targets were performed over a wide range of reactor powers, exposures and times since irradiation. This set of data is analyzed in the present work to assess fission product variability.

To determine estimates of the radionuclide composition in the sludge waste following reprocessing of the spent fuels and targets, it was necessary to apply separation fractions and solubility partitions to the radionuclide data of Reference 1.

Separation factors signify the percent loss (either intentionally or unintentionally) of the various radionuclides from spent fuel to high level waste. Two sets of separation factors are listed below for the two major separation campaigns that have occurred over the history of canyon reprocessing operations, namely, the PUREX and HM campaigns. The PUREX process is used to separate plutonium from uranium; the HM process is used to recover uranium and byproduct neptunium.

Table 1 Radionuclide Separation Fractions (Reference 12)

Element	Percent Loss to Waste	
	PUREX	HM
Tritium	28	28
Noble Gases	0	0
Uranium	0.02	0.15
Neptunium	6.0	3.5
Plutonium	0.22	100
Am, Cm, etc.,	100	100
Fission Products	100	100

Fresh waste from the canyon reprocessing operations is sent to tanks in F and H Area for storage in the form of sludge and supernate. The solubility fraction of a radionuclide represents the fraction of the total radionuclide quantity that is soluble and is thus present in the supernate with the remainder in the sludge portion. Solubility fractions (associated with the 12 radionuclides shown to have a significant impact on dose) are listed in the following table.

Table 2 Solubility Fractions (Reference 14)

Radionuclide	Solubility Fraction
Sr-90	2.5e-04
Ru-106	0.5
Cs-137	0.95
Ce-144	1.0e-04
Pm-147	1.0e-04
Pu-239	3.0e-04
Pu-240	3.0e-04
Pu-241	3.0e-04
Pu-238	3.0e-04

The application of the separation fractions and solubility fractions (actually, multiplication of the solubility complement) to the radionuclide compositions calculated numerically for spent fuel and target assemblies yielded predicted radionuclide compositions for sludge waste.

The dose conversion factors (Reference 13) convert the radionuclide composition values to dose values. Separate dose conversion factors exist for each of the various dose pathways. Inhalation and ingestion dose conversion factors for the radionuclides of concern are shown in the table below. Also shown are release fractions which are needed to adjust the inhalation dose conversion factors under evaporative conditions.

Table 3 Dose Conversion Factors (DCFs) and Release Fractions (Derived from Reference 13)

Radionuclide	Ingestion DCF [Rem/g]	Inhalation DCF [Rem/g]	Release Fraction
Sr-90	1.81e+07	1.81e+08	0.001
Ru-106	7.03e+07	1.47e+09	0.01
Cs-137	4.33e+06	2.77e+06	0.01
Ce-144	6.38e+07	1.12e+09	0.001
Pm-147	8.81e+05	3.15e+07	0.001
Pu-239	2.67e+05	3.16e+07	0.001
Pu-240	9.75e+05	1.16e+08	0.001
Pu-241	8.86e+06	1.03e+09	0.001
Pu-238	6.51e+07	7.88e+09	0.001

Results

The SHIELD/GLASS results of Reference 1 were performed for four different fuel assemblies (Mark 5E, 12A, 16B, and 22 types) and three different target assemblies (Mark 31A, 31B, and 50A types) over a broad spectrum of reactor conditions. The output from these numerical calculations served as input for a variety of radionuclide analyses that were performed and which are discussed below.

Sludge Dose Contributions

HM Processing Contribution

Past studies in the development of source terms for High Level Waste (HLW) have identified and evaluated important radionuclide contributors to dose from reprocessed spent fuel sent to waste for Mark 16B and 22 fuel assemblies and for Mark 31A and 31B target assemblies (References 8 - 11). These analyses, performed for a representative set of irradiation conditions for each type of assembly, demonstrated that only a limited number of radionuclides contributed at least 1% or more to the total dose. The analysis presented in this section extends and complements the past work through the consideration of a larger variety of fuel types and a wider range of irradiation conditions. Dose contributions from only the sludge portion of waste are examined. Doses are studied for decay periods of 4 and 20 years (recall, data are available for 2, 4, 10, and 20 years). These time frames encompass the 5- and 15-year decay periods examined in the past analyses.

Following the methodology of past work (References 8 - 11), the dose analysis will consider HM processing of spent fuel assemblies and PUREX processing of spent target assemblies. This approach is consistent with the general practices of reprocessing activities performed in the F and H Canyons.

Starting from the computed radionuclide production data [units of grams per assembly] of the irradiated fuel, four major steps were performed to determine radionuclide contributions to dose: (1) multiplication by specific activity to arrive at radionuclide activity [Curies per assembly]; (2) multiplication by separation fractions (which represent percentages of total quantities of radionuclides sent to waste); (3) multiplication by "insolubility fractions" (which represent percentages of total quantities of radionuclides deposited in the sludge portion of the waste); and (4) multiplication by dose conversion factors [units of Rem per Curie] to convert activity levels to dose levels. The tabular data needed to perform these sludge dose calculations are documented in the spreadsheet in Appendix A.

Comprehensive dose analyses were performed for both the Mark 16B and 22 fuel assemblies in order to study the effects of irradiation conditions and decay time on dose contributions. Additional dose analyses for Mark 5E and 12A fuel assemblies allow for the study of the effects of fuel composition. The results are summarized in tables located in Appendix B.

Taking into consideration both inhalation and ingestion pathways, the list of important contributors to dose (as defined as contributing 1% or more to the total) is the same as identified in References 8 and 9, except that Am-241 was not considered in the present analysis due to its unavailability from the SHIELD/GLASS results. (References 8 and 9 showed the relative contribution of Am-241 to the sludge source term from HM processing of fuel assemblies to be 3% or less.) The twelve radionuclides that contribute significantly to dose are those listed in Tables 2 and 3.

A more thorough examination of the results of the present work discloses useful information and well-defined trends. The relative contribution to dose of the significant radionuclides for various Mark 16B and 22 cases can be seen from the tables in Appendix B. The Pu-238 and Sr-90 radionuclides are seen to be the two most important contributors to dose for Mark 16B and 22 fuels; for example, these two radionuclides account for 80 - 90 % of the volatile inhalation dose at 4 years (their contribution is slightly higher at 20 years). At a 20-year decay time, the relative contribution to dose of Pu-238 increases due to its larger half life (i.e., 88 years) than Sr-90 (28 years).

The table in Appendix B also contains results from Mark 5E and 12A calculations. Note that the same radionuclides constitute the list of important contributors to dose for these fuels as for Mark 16B and 22 fuels, although some significant differences are noticeable in relative contribution levels. For example, Ce-144 is a more significant contributor to dose for Mark 12A fuel than it is for the other fuels. The Sr-90, Ce-144, and Pu-238 radionuclides are observed to contribute over 80 % of the volatile inhalation dose at 4 years for Mark 12A fuel (true also at 20 years even though the contribution of Ce-144 is negligible at this time frame). For Mark 5E fuel, the plutonium isotopes of mass number 239, 240, and 241 are the dominant contributors to volatile inhalation dose, both at 4 and 20 years.

PUREX Processing Contribution

Sludge dose analysis was also performed for waste from spent target assemblies having undergone PUREX processing. The tabular data needed to perform the sludge dose calculations are documented in Appendix A.

Dose analyses were performed for Mark 31A target assemblies in order to study the effects of irradiation conditions and decay time on dose contributions. The results are summarized in Appendix C.

Taking into consideration both inhalation and ingestion pathways, the list of important contributors to dose (as defined as contributing 1% or more to the total) is the same as identified in Reference 10, except that Am-241 was not considered in the present analysis due to its unavailability from the SHIELD/GLASS results. (Reference 10 showed the relative contribution of Am-241 to the sludge source term from PUREX processing of target assemblies to be as high as 28%.)

HM Waste Tank Inventory Comparisons

The feasibility of using SHIELD/GLASS results to model radionuclide distributions in the tank farms is investigated in this section of the report. HM processing of spent fuel assemblies will be considered since radionuclide data is more readily available for the HM process than it is for the PUREX process (Reference 15).

Specifically, the SHIELD/GLASS radionuclide masses are used to predict the relative distributions of uranium and plutonium isotopes sent to individual waste tanks in sludge form as documented by accountability records (based on process control samples) and estimations based upon process data and knowledge (Reference 15). The following radionuclides are inventoried in the individual tank sludge report (Reference 16): U-233, U-234, U-235, U-236, U-238, Np-237, PU-238, Pu-239, Pu-240, Pu-241, and Pu-242.

The distribution of these radionuclides in sludge waste as a function of reactor power, exposure, and decay time was determined from the SHIELD/GLASS output. This involved adjusting the SHIELD/GLASS output of radionuclide data through the appropriate use of solubility fractions and separation fractions (HM processing) to reflect radionuclide distributions in sludge waste.

The results of calculations are presented in Appendix D, Figures D-1 through D-6. Figures D-7 through D-24 present the comparisons between inventory data and numerical predictions. The results presented in these graphs are normalized by dividing the constituent mass by the sum of the constituent masses.

Figure D-1 contains the results from a sample Mark 22 case, which shows that the distribution of the inventoried radionuclides does not change significantly over the time period of 20 years due to the long half lives of these radionuclides. Noticeable changes of small magnitude are observed in Pu-238 (half life of 88 years) and Pu-241 (15 years).

Figure D-2 shows that power level for a given exposure does not have any significant influence on the distribution of the inventoried radionuclides. Thus, for our purposes, it is not necessary to be concerned in this section with the fact that fuel sent to waste has been irradiated over a wide range of reactor power levels over the history of site operations.

The influence of exposure level on the relative quantities of the inventoried radionuclides for Mark 22 fuel is shown in Figure D-3. The most noticeable feature is the decrease in U-235 content as exposure increases, which is as expected since Mark 22 charges involve the fissioning of U-235.

Figure D-4 is an analogous graph for Mark 16B fuel showing similar behavior. Also shown on the graph are the Mark 22 results at 80 MWD/ft, which are seen to be very similar to those for Mark 16B fuel at 80 MWD/ft. This is an important fact to remember since Mark 16Bs and Mark 22s have been the only SRS fuel assemblies of significant quantity processed in the H Canyon since the late 70s (References 15 and 17). Also, the 80 MWD/ft value is representative of typical

exposures that the Mark 22s and Mark 16Bs have undergone (the analyses of Reference 12 used values of 80 MWD/ft and 95 MWD/ft for Mark 22 and Mark 16B exposures, respectively, as the basis for their calculations).

Figures D-5 and D-6 show distributions of radionuclides at various fission exposures for Mark 5E and 12A fuel assemblies. Since these two assemblies date back to the mid-to-late sixties when a wide variety of fuel assemblies were irradiated, they will not play a prominent role in the analyses that follow. For the sake of simplicity, comparisons of numerical calculations to tank inventories will largely be confined to tanks that have received waste from the H Canyon only during the past 20 years, and thus, they largely contain waste from processed Mark 16B and 22 fuel assemblies (References 15 and 17). The Mark 5E and 12A data are included for completeness and to facilitate possible future analysis.

Reference 16 tracks monthly sludge deposits to individual tanks. Monthly data was extracted from that report for comparison with the numerical calculations of this report. Data which allowed for the most straightforward comparisons with the numerical calculations were extracted and investigated first. This data met the following criteria: (1) waste from the HM process; (2) waste from the high heat waste (HHW) stream (HM separation factors from Reference 12 and used in this report appear to apply to the HHW stream only); (3) waste sent to tanks within the last 20 years; and (4) the waste deposit does not contain offsite fuel (spreadsheet from Reference 15 contains a column for offsite fuel aluminum, so a nonzero entry in that column identifies the presence of offsite fuel in the waste deposit).

The inventory data shown in Figures D-7 through D-10 met this criteria and are shown with the results from the numerical calculations for Mark 16B fuel at three different exposures (20, 40 and 80 MWD/ft). In general, agreement is observed between the inventory data and the numerical predictions (typically, the results for 80 MWD/ft provide the best match with the inventory data). Note that in these graphs and in those that follow, the inventory data is plotted as column bars while the numerical predictions are shown as points connected by lines.

The second set of figures, Figures D-11 through D-16, show comparisons between numerical calculations (Mark 22 fuel) and tank contents which do not meet all four criterion points listed above. Specifically, the tank contents meet criteria (1) and (2), but not (4) which is the zero offsite fuel criterion, and not necessarily (3) which is the 20-year criterion. As expected, greater deviation is seen in general between the numerical calculations and the inventory data. For example, both significant underprediction of U-238 content and overprediction of Pu-239 content are observed for Tanks 32, 35, and 36.

In the third set of figures, Figures D-17 through D-24, the inventory data are assured to meet only criterion (1), that is, HM processing. Poor agreement between the numerical predictions and tank inventory data is observed for Tanks 13, 21, 22, and 43. These tanks contain waste predominantly from the low heat waste (LHW) stream which has significantly different separation factors associated with it which accounts for the poor results.

One last comparison was made, shown in Figure D-24, using the sludge waste deposit to Tank 15 made in December 1968. This deposit is from the HHW stream of HM processing and does not contain offsite fuel. Since the SRS fuel processed in this batch is Mark 12A (Reference 17), comparisons with numerical calculations were based on this fuel. There is good agreement between the numerical predictions for Mark 12A fuel at 40 MWD/ft exposure and the inventory data.

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Appendix A

Spreadsheet for Processing Calculations

Nuclide	Atomic Weight	Half Life [s]	Lambda	Specific Activity [Ci/g]	IRCP-30		IRCP-30		Loss to Waste Fraction	
					Inhal. DCF [rem/uCi]	Release Fraction	Inges. DCF [rem/uCi]	Solubility Fraction	HM	PUREX
H 1	1.01		0.00E+00	0.00E+00						
H 2	2.01		0.00E+00	0.00E+00						
H 3	3.02	3.87E+08	1.79E-09	9.67E+03	9.50E-05	1.00E+00	6.30E-05	1.00E+00	0.2800	0.2800
HE 3	3.02		0.00E+00	0.00E+00						
HE 4	4.00		0.00E+00	0.00E+00						
LI 6	6.02		0.00E+00	0.00E+00						
LI 7	7.02		0.00E+00	0.00E+00						
GE 72	71.92		0.00E+00	0.00E+00						
GE 73	72.92		0.00E+00	0.00E+00						
GE 74	73.92		0.00E+00	0.00E+00						
GE 76	75.92		0.00E+00	0.00E+00						
AS 75	74.92		0.00E+00	0.00E+00						
SE 77	76.92		0.00E+00	0.00E+00						
SE 78	77.92		0.00E+00	0.00E+00						
SE 79	78.92	2.05E+12	3.38E-13	6.97E-02	8.90E-03	1.00E-02	8.30E-03	1.00E-01	1.0000	1.0000
SE 80	79.92		0.00E+00	0.00E+00						
SE 82	81.92		0.00E+00	0.00E+00						
BR 81	80.92		0.00E+00	0.00E+00						
KR 82	81.91		0.00E+00	0.00E+00						
KR 83	82.91		0.00E+00	0.00E+00						
KR 84	83.91		0.00E+00	0.00E+00						
KR 85	84.91	3.39E+08	2.05E-09	3.92E+02						
KR 86	85.91		0.00E+00	0.00E+00						
RB 85	84.91		0.00E+00	0.00E+00						
RB 87	86.91	1.48E+18	4.68E-19	8.76E-08	3.30E-03	1.00E-03	4.80E-03	9.50E-01	1.0000	1.0000
SR 88	87.91		0.00E+00	0.00E+00						
SR 89	88.91	4.49E+06	1.54E-07	2.82E+04	3.70E-02	1.00E-03	8.70E-03	2.50E-04	1.0000	1.0000
SR 90	89.91	8.87E+08	7.82E-10	1.42E+02	1.30E+00	1.00E-03	1.30E-01	2.50E-04	1.0000	1.0000
Y 89	88.91		0.00E+00	0.00E+00						
Y 90	89.91	2.31E+05	3.00E-06	5.44E+05	8.20E-03	1.00E-03	1.00E-02	2.50E-04	1.0000	1.0000
Y 91	90.91	5.06E+06	1.37E-07	2.45E+04	4.40E-02	1.00E-03	8.90E-03	1.00E-04	1.0000	1.0000
ZR 90	89.90		0.00E+00	0.00E+00						
ZR 91	90.91		0.00E+00	0.00E+00						
ZR 92	91.91		0.00E+00	0.00E+00						
ZR 93	92.91	3.00E+13	2.31E-14	4.05E-03	3.20E-01	1.00E-03	1.60E-03	9.60E-06	1.0000	1.0000
ZR 94	93.91		0.00E+00	0.00E+00						
ZR 95	94.91	5.66E+06	1.22E-07	2.10E+04	1.80E-02	1.00E-03	3.40E-03	1.00E-05	1.0000	1.0000
ZR 96	95.91		0.00E+00	0.00E+00						
NB 95	94.91	3.03E+06	2.29E-07	3.92E+04	4.50E-03	1.00E-03	2.20E-03	1.00E-05	1.0000	1.0000
NB 95M	94.91	3.12E+05	2.22E-06	3.81E+05	2.20E-03	1.00E-03	2.00E-03	1.00E-05	1.0000	1.0000
MO 95	94.91		0.00E+00	0.00E+00						
MO 96	95.90		0.00E+00	0.00E+00						
MO 97	96.91		0.00E+00	0.00E+00						
MO 98	97.91		0.00E+00	0.00E+00						
MO100	99.91		0.00E+00	0.00E+00						
TC 99	99.00	6.72E+12	1.03E-13	1.70E-02	7.50E-03	1.00E-03	1.30E-03	3.00E-01	1.0000	1.0000
RU100	99.90		0.00E+00	0.00E+00						
RU101	100.91		0.00E+00	0.00E+00						

Nuclide	Atomic Weight	Half Life [s]	Lambda	Specific Activity [Ci/g]	IRCP-30		IRCP-30		Loss to Waste	
					Inhal. DCF [rem/uCi]	Release Fraction	Inges. DCF [rem/uCi]	Solubility Fraction	Fraction HM	PUREX
RU102	101.90		0.00E+00	0.00E+00						
RU103	102.91	3.42E+06	2.03E-07	3.20E+04	7.80E-03	1.00E-02	2.70E-03	5.00E-01	1.0000	1.0000
RU104	103.91		0.00E+00	0.00E+00						
RU106	105.91	3.19E+07	2.17E-08	3.34E+03	4.40E-01	1.00E-02	2.10E-02	5.00E-01	1.0000	1.0000
RH103	102.91		0.00E+00	0.00E+00						
RH103M	102.91	3.36E+03	2.06E-04	3.26E+07						
RH106	105.91	2.99E+01	2.32E-02	3.56E+09						
PD104	103.90		0.00E+00	0.00E+00						
PD105	104.91		0.00E+00	0.00E+00						
PD106	105.90		0.00E+00	0.00E+00						
PD107	106.90	2.05E+14	3.38E-15	5.15E-04	1.30E-02	1.00E-03	1.40E-04	1.00E-04	1.0000	1.0000
PD108	107.90		0.00E+00	0.00E+00						
PD110	109.91		0.00E+00	0.00E+00						
AG109	108.90		0.00E+00	0.00E+00						
AG110	109.91	2.18E+07	3.18E-08	4.71E+03						
CD110	109.90		0.00E+00	0.00E+00						
CD111	110.90		0.00E+00	0.00E+00						
CD112	111.90		0.00E+00	0.00E+00						
CD113	112.90		0.00E+00	0.00E+00						
CD114	113.90		0.00E+00	0.00E+00						
CD115M	114.90	3.85E+06	1.80E-07	2.55E+04	6.50E-02	1.00E-03	1.50E-02	5.00E-02	1.0000	1.0000
CD116	115.90		0.00E+00	0.00E+00						
IN115	114.90	1.58E+22	4.40E-23	6.23E-12	3.40E+00	1.00E-03	1.40E-01	5.00E-02	1.0000	1.0000
IN115M	114.90	1.62E+04	4.28E-05	6.06E+06						
SN116	115.90		0.00E+00	0.00E+00						
SN117	116.90		0.00E+00	0.00E+00						
SN118	117.90		0.00E+00	0.00E+00						
SN119	118.90		0.00E+00	0.00E+00						
SN120	119.90		0.00E+00	0.00E+00						
SN121M	120.90	1.58E+09	4.40E-10	5.92E+01	8.90E-03	1.00E-03	1.30E-03	4.00E-02	1.0000	1.0000
SN122	121.90		0.00E+00	0.00E+00						
SN123	122.91	1.11E+07	6.22E-08	8.24E+03	3.00E-02	1.00E-03	7.70E-03	4.00E-02	1.0000	1.0000
SN124	123.91		0.00E+00	0.00E+00						
SN126	125.91	3.15E+13	2.20E-14	2.84E-03	8.60E-02	1.00E-03	1.70E-02	4.00E-02	1.0000	1.0000
SB121	120.90		0.00E+00	0.00E+00						
SB123	122.90		0.00E+00	0.00E+00						
SB124	123.91	5.20E+06	1.33E-07	1.75E+04						
SB125	124.90	8.62E+07	8.05E-09	1.05E+03	9.80E-03	1.00E-03	2.60E-03	6.00E-03	1.0000	1.0000
SB126	125.91	1.07E+06	6.47E-07	8.36E+04						
SB126M	125.91	1.14E+03	6.08E-04	7.86E+07						
TE122	121.90		0.00E+00	0.00E+00						
TE124	123.90		0.00E+00	0.00E+00						
TE125	124.90		0.00E+00	0.00E+00						
TE125M	124.90	5.01E+06	1.38E-07	1.80E+04						
TE126	125.90		0.00E+00	0.00E+00						
TE127	126.90	3.37E+04	2.06E-05	2.64E+06						
TE127M	126.90	9.42E+06	7.36E-08	9.44E+03	1.90E-02	1.00E-02	7.90E-03	5.00E-04	1.0000	1.0000
TE128	127.90		0.00E+00	0.00E+00						

Nuclide	Atomic Weight	Half Life [s]	Lambda	Specific Activity [Ci/g]	IRCP-30 Inhal.		IRCP-30 Inges.		Loss to Waste Fraction	
					DCF [rem/uCi]	Release Fraction	DCF [rem/uCi]	Solubility Fraction	HM	PUREX
TE130	129.91		0.00E+00	0.00E+00						
I127	126.90		0.00E+00	0.00E+00						
I129	128.90	5.01E+14	1.38E-15	1.75E-04	1.80E-01	5.00E-01	2.80E-01	1.00E+00	1.0000	1.0000
XE128	127.90		0.00E+00	0.00E+00						
XE129	128.90		0.00E+00	0.00E+00						
XE130	129.90		0.00E+00	0.00E+00						
XE131	130.91		0.00E+00	0.00E+00						
XE132	131.90		0.00E+00	0.00E+00						
XE134	133.91		0.00E+00	0.00E+00						
XE136	135.91		0.00E+00	0.00E+00						
CS133	132.91		0.00E+00	0.00E+00						
CS134	133.91	6.50E+07	1.07E-08	1.30E+03	4.70E-02	1.00E-02	7.40E-02	9.50E-01	1.0000	1.0000
CS135	134.91	7.25E+13	9.56E-15	1.15E-03	4.50E-03	1.00E-02	7.10E-03	9.50E-01	1.0000	1.0000
CS137	136.91	9.50E+08	7.30E-10	8.68E+01	3.20E-02	1.00E-02	5.00E-02	9.50E-01	1.0000	1.0000
BA134	133.90		0.00E+00	0.00E+00						
BA136	135.90		0.00E+00	0.00E+00						
BA137	136.91		0.00E+00	0.00E+00						
BA137M	136.91	1.53E+02	4.53E-03	5.39E+08						
BA138	137.91		0.00E+00	0.00E+00						
LA139	138.91		0.00E+00	0.00E+00						
CE140	139.91		0.00E+00	0.00E+00						
CE142	141.91	3.31E+18	2.09E-19	2.40E-08						
CE144	143.91	2.46E+07	2.82E-08	3.19E+03	3.50E-01	1.00E-03	2.00E-02	1.00E-04	1.0000	1.0000
PR141	140.91		0.00E+00	0.00E+00						
PR144	143.91	1.04E+03	6.69E-04	7.56E+07						
PR144M	143.91	4.32E+02	1.60E-03	1.81E+08						
ND142	141.91		0.00E+00	0.00E+00						
ND143	142.91		0.00E+00	0.00E+00						
ND144	143.91	6.62E+22	1.05E-23	1.18E-12	0.00E+00	1.00E-03	0.00E+00	1.00E-04	1.0000	1.0000
ND145	144.91		0.00E+00	0.00E+00						
ND146	145.91		0.00E+00	0.00E+00						
ND148	147.92		0.00E+00	0.00E+00						
ND150	149.92		0.00E+00	0.00E+00						
PM147	146.91	8.28E+07	8.37E-09	9.28E+02	3.40E-02	1.00E-03	9.50E-04	1.00E-04	1.0000	1.0000
PM148	147.92	4.64E+05	1.49E-06	1.64E+05	1.00E-02	1.00E-03	9.50E-03	1.00E-04	1.0000	1.0000
PM148M	147.92	3.57E+06	1.94E-07	2.14E+04	1.70E-02	1.00E-03	7.00E-03	1.00E-04	1.0000	1.0000
SM147	146.91	3.37E+18	2.05E-19	2.28E-08	7.10E+01	1.00E-03	1.80E-01	3.50E-04	1.0000	1.0000
SM148	147.91	2.52E+23	2.75E-24	3.02E-13						
SM149	148.92	3.15E+23	2.20E-24	2.40E-13						
SM150	149.92		0.00E+00	0.00E+00						
SM151	150.92	2.93E+09	2.36E-10	2.55E+01	2.90E-02	1.00E-03	3.40E-04	3.50E-04	1.0000	1.0000
SM152	151.92		0.00E+00	0.00E+00						
SM154	153.92		0.00E+00	0.00E+00						
EU151	150.92		0.00E+00	0.00E+00						
EU152	151.92	4.10E+08	1.69E-09	1.81E+02	2.20E-01	1.00E-03	6.00E-03	1.30E-04	1.0000	1.0000
EU153	152.92		0.00E+00	0.00E+00						
EU154	153.92	2.71E+08	2.56E-09	2.70E+02	2.60E-01	1.00E-03	9.10E-03	1.30E-04	1.0000	1.0000
EU155	154.92	1.51E+08	4.58E-09	4.81E+02	3.90E-02	1.00E-03	1.30E-03	1.30E-04	1.0000	1.0000

Nuclide	Atomic Weight	Half Life [s]	Lambda	Specific Activity [Ci/g]	IRCP-30		IRCP-30		Loss to Waste	
					Inhal. DCF [rem/uCi]	Release Fraction	Inges. DCF [rem/uCi]	Solubility Fraction	Fraction HM	Fraction PUREX
GD152	151.92	3.47E+21	2.00E-22	2.14E-11	2.40E+02	1.00E-03	1.50E-01	1.00E-04	1.0000	1.0000
GD154	153.92		0.00E+00	0.00E+00						
GD155	154.92		0.00E+00	0.00E+00						
GD156	155.92		0.00E+00	0.00E+00						
GD157	156.92		0.00E+00	0.00E+00						
GD158	157.92		0.00E+00	0.00E+00						
GD160	159.93		0.00E+00	0.00E+00						
TB159	158.93		0.00E+00	0.00E+00						
TB160	159.93	6.25E+06	1.11E-07	1.13E+04	2.20E-02	1.00E-03	6.40E-03	1.00E-04	1.0000	1.0000
DY160	159.93		0.00E+00	0.00E+00						
DY161	160.93		0.00E+00	0.00E+00						
DY162	161.93		0.00E+00	0.00E+00						
TH232	232.04	4.43E+17	1.56E-18	1.10E-07						
PA233	233.04	2.33E+06	2.97E-07	2.08E+04						
U233	233.04	5.02E+12	1.38E-13	9.64E-03						
U234	234.04	7.72E+12	8.98E-14	6.25E-03	1.30E+02	1.00E-03	2.60E-01	3.00E-04	0.0015	0.0002
U235	235.04	2.22E+16	3.12E-17	2.16E-06	1.20E+02	1.00E-03	2.50E-01	3.00E-04	0.0015	0.0002
U236	236.05	7.39E+14	9.38E-16	6.47E-05	1.20E+02	1.00E-03	2.50E-01	3.00E-04	0.0015	0.0002
U237	237.54	5.83E+05	1.19E-06	8.14E+04						
U238	238.05	1.41E+17	4.92E-18	3.36E-07	1.20E+02	1.00E-03	2.30E-01	3.00E-04	0.0015	0.0002
NP237	237.05	6.75E+13	1.03E-14	7.05E-04	4.90E+02	1.00E-03	3.90E+00	3.00E-04	0.0350	0.0600
PU238	238.05	2.77E+09	2.50E-10	1.71E+01	4.60E+02	1.00E-03	3.80E+00	3.00E-04	1.0000	0.0022
PU239	239.05	7.61E+11	9.11E-13	6.20E-02	5.10E+02	1.00E-03	4.30E+00	3.00E-04	1.0000	0.0022
PU240	240.05	2.07E+11	3.35E-12	2.27E-01	5.10E+02	1.00E-03	4.30E+00	3.00E-04	1.0000	0.0022
PU241	241.06	4.64E+08	1.49E-09	1.01E+02	1.00E+01	1.00E-03	8.60E-02	3.00E-04	1.0000	0.0022
PU242	242.06	1.19E+13	5.84E-14	3.93E-03	4.80E+02	1.00E-03	4.10E+00	3.00E-04	1.0000	0.0022

Appendix B

Sludge Dose Contribution From HM Processing

[illegible][illegible][illegible]

Inhalation (Volatile): Isotopic percent contribution to dose at 4 years

Fuel Exposure [MWD/ft] Power [MW/ft]	Mark 16B								
	20	20	20	40	40	40	80	80	80
	0.6	1.8	4.0	0.6	1.8	4.0	0.6	1.8	4.0
Sr-90	31%	35%	42%	19%	21%	25%	8%	11%	13%
Ru-106	0%	1%	1%	0%	0%	0%	0%	0%	0%
Cs-137	0%	0%	0%	0%	0%	0%	0%	0%	0%
Ce-144	8%	9%	11%	4%	5%	7%	0%	3%	3%
Pm-147	1%	1%	2%	1%	1%	1%	0%	0%	0%
Pu-238	53%	45%	35%	70%	66%	60%	88%	81%	78%
Pu-239	6%	6%	8%	3%	3%	4%	2%	1%	2%
Pu-240	1%	1%	1%	1%	1%	1%	1%	1%	1%
Pu-241	1%	1%	1%	2%	2%	2%	1%	3%	3%
Total	99%	99%	101%	100%	99%	99%	100%	100%	100%

Inhalation (Evaporative): Isotopic percent contribution to dose at 4 years

Fuel Exposure [MWD/ft] Power [MW/ft]	Mark 16B								
	20	20	20	40	40	40	80	80	80
	0.6	1.8	4.0	0.6	1.8	4.0	0.6	1.8	4.0
Sr-90	24%	33%	38%	18%	20%	24%	8%	11%	12%
Ru-106	6%	7%	9%	3%	4%	0%	0%	2%	3%
Cs-137	0%	0%	0%	0%	0%	0%	0%	0%	0%
Ce-144	1%	9%	11%	4%	5%	6%	0%	2%	3%
Pm-147	1%	1%	2%	1%	1%	1%	0%	0%	0%
Pu-238	50%	42%	32%	68%	63%	57%	88%	79%	76%
Pu-239	5%	6%	7%	3%	3%	4%	2%	1%	2%
Pu-240	1%	1%	1%	1%	1%	1%	1%	1%	1%
Pu-241	1%	1%	1%	2%	2%	2%	1%	3%	3%
Total	89%	100%	101%	100%	99%	95%	100%	99%	100%

Ingestion: Isotopic percent contribution to dose at 4 years

Fuel Exposure [MWD/ft] Power [MW/ft]	Mark 16B								
	20	20	20	40	40	40	80	80	80
	0	1.8	4.0	0.6	1.8	4.0	0.6	1.8	4.0
Sr-90	76%	78%	79%	69%	70%	72%	54%	57%	59%
Ru-106	1%	1%	1%	1%	1%	1%	0%	1%	1%
Cs-137	1%	1%	0%	1%	1%	0%	1%	1%	1%
Ce-144	10%	11%	12%	8%	9%	10%	0%	7%	8%
Pm-147	1%	1%	1%	1%	1%	1%	0%	0%	0%
Pu-238	10%	7%	5%	20%	17%	13%	43%	32%	28%
Pu-239	1%	1%	1%	1%	1%	1%	1%	1%	1%
Pu-240	0%	0%	0%	0%	0%	0%	0%	0%	0%
Pu-241	0%	0%	0%	0%	0%	0%	1%	1%	1%
Total	100%	100%	99%	101%	100%	99%	100%	100%	99%

Inhalation (Volatile): Isotopic percent contribution to dose at 20 years

Fuel Exposure [MWD/ft] Power [MW/ft]	Mark 16B								
	20	20	20	40	40	40	80	80	80
	0.6	1.8	4.0	0.6	1.8	4.0	0.06	1.8	4.0
Sr-90	29%	34%	42%	16%	18%	22%	10%	9%	11%
Ru-106	0%	0%	0%	0%	0%	0%	0%	0%	0%
Cs-137	0%	0%	0%	0%	0%	0%	0%	0%	0%
Ce-144	0%	0%	0%	0%	0%	0%	2%	0%	0%
Pm-147	0%	0%	0%	0%	0%	0%	0%	0%	0%
Pu-238	63%	55%	45%	78%	75%	70%	83%	87%	85%
Pu-239	5%	9%	11%	4%	4%	5%	1%	2%	2%
Pu-240	1%	1%	1%	1%	1%	1%	1%	1%	1%
Pu-241	1%	1%	1%	1%	1%	1%	2%	1%	2%
Total	98%	100%	100%	100%	99%	100%	100%	100%	101%

Inhalation (Evaporative): Isotopic percent contribution to dose at 20 years

Fuel Exposure [MWD/ft] Power [MW/ft]	Mark 16B								
	20	20	20	40	40	40	80	80	80
	0.6	1.8	4.0	0.6	1.8	4.0	0.6	1.8	4.0
Sr-90	28%	33%	41%	16%	18%	22%	10%	9%	10%
Ru-106	0%	0%	0%	0%	0%	0%	2%	0%	0%
Cs-137	0%	0%	0%	0%	0%	0%	0%	0%	0%
Ce-144	0%	0%	0%	0%	0%	0%	2%	0%	0%
Pm-147	0%	0%	0%	0%	0%	0%	0%	0%	0%
Pu-238	62%	55%	45%	78%	75%	69%	82%	87%	85%
Pu-239	7%	9%	11%	4%	4%	5%	1%	2%	2%
Pu-240	1%	1%	1%	1%	1%	1%	1%	1%	1%
Pu-241	1%	0%	1%	1%	1%	1%	2%	1%	2%
Total	100%	98%	99%	100%	99%	100%	100%	100%	100%

Ingestion: Isotopic percent contribution to dose at 20 years

Fuel Exposure [MWD/ft] Power [MW/ft]	Mark 16B								
	20	20	20	40	40	40	80	80	80
	0.6	1.8	4.0	0.6	1.8	4.0	0.6	1.8	4.0
Sr-90	82%	85%	88%	70%	73%	78%	56%	56%	60%
Ru-106	0%	0%	0%	0%	0%	1%	0%	0%	0%
Cs-137	2%	2%	0%	1%	1%	0%	1%	1%	1%
Ce-144	0%	0%	0%	0%	0%	0%	5%	0%	0%
Pm-147	0%	0%	0%	0%	0%	0%	0%	0%	0%
Pu-238	14%	11%	7%	26%	23%	19%	35%	41%	37%
Pu-239	2%	2%	2%	1%	1%	1%	1%	1%	1%
Pu-240	0%	0%	0%	0%	0%	0%	0%	0%	0%
Pu-241	0%	0%	0%	0%	0%	0%	1%	1%	1%
Total	100%	100%	97%	98%	98%	100%	100%	100%	100%

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Inhalation (Evaporative): Isotopic percent contribution to dose at 4 years
(continued)

Fuel	Mark 22	Mark 22	Mark 22	Mark 22	Mark 22	Mark 22	Mark 22	Mark 22
Exposure [MWD/ft]	60	60	60	60	80	80	80	80
Power [MW/ft]	0.1	0.3	0.5	0.7	0.1	0.3	0.5	0.7
Sr-90	12%	13%	13%	14%	9%	10%	10%	11%
Y-90	0%	0%	0%	0%	0%	0%	0%	0%
Ru-106	2%	3%	3%	3%	1%	2%	2%	2%
Cs-137	0%	0%	0%	0%	0%	0%	0%	0%
Ce-144	2%	3%	3%	4%	1%	2%	2%	3%
Pm-147	0%	0%	0%	0%	0%	0%	0%	0%
Pu-238	78%	75%	74%	72%	83%	80%	79%	78%
Pu-239	2%	2%	2%	2%	1%	1%	1%	1%
Pu-240	1%	1%	1%	1%	1%	1%	1%	1%
Pu-241	3%	3%	3%	3%	3%	3%	4%	4%
Total	100%	100%	100%	100%	100%	100%	100%	100%

Ingestion: Isotopic percent contribution to dose at 4 years

Fuel	Mark 22							
Exposure [MWD/ft]	20	20	20	20	40	40	40	40
Power [MW/ft]	0.1	0.3	0.5	0.7	0.1	0.3	0.5	0.7
Sr-90	75%	76%	76%	77%	67%	67%	68%	69%
Ru-106	1%	1%	1%	1%	1%	1%	1%	1%
Cs-137	1%	1%	1%	1%	1%	1%	1%	1%
Ce-144	9%	10%	11%	11%	7%	9%	9%	10%
Pm-147	1%	1%	1%	1%	1%	1%	1%	1%
Pu-238	12%	9%	8%	7%	22%	19%	18%	17%
Pu-239	1%	1%	1%	1%	1%	1%	1%	1%
Pu-240	0%	0%	0%	0%	0%	0%	0%	0%
Pu-241	0%	0%	0%	0%	1%	1%	1%	1%
Total	100%	100%	100%	100%	100%	100%	100%	100%

Ingestion: Isotopic percent contribution to dose at 4 years (continued)

Fuel	Mark 22							
Exposure [MWD/ft]	60	60	60	60	80	80	80	80
Power [MW/ft]	0.1	0.3	0.5	0.7	0.1	0.3	0.5	0.7
Sr-90	60%	61%	61%	62%	55%	55%	55%	56%
Ru-106	0%	1%	1%	1%	0%	1%	1%	1%
Cs-137	1%	1%	1%	1%	1%	1%	1%	1%
Ce-144	5%	7%	8%	8%	4%	6%	7%	7%
Pm-147	0%	0%	0%	1%	0%	0%	0%	0%
Pu-238	30%	28%	26%	25%	37%	34%	33%	32%
Pu-239	1%	1%	1%	1%	1%	1%	1%	1%
Pu-240	0%	0%	0%	0%	0%	0%	0%	0%
Pu-241	1%	1%	1%	1%	2%	2%	2%	2%
Total	100%	100%	99%	100%	100%	100%	100%	100%

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Appendix C

Sludge Dose Contribution From PUREX Processing

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Appendix D

Tank Inventory Radionuclide Comparisons, HM Processing

Figure D-1

Figure D-2

Figure D-3

Figure D-4

Figure D-5

Figure D-6

Figure D-7

Radionuclides in Tank 35H Sludge Versus Mark 16B At Various Exposures

Figure D-8

Figure D-9

Radionuclides in Tank 39H Sludge Versus Mark 16B At Various Exposures

Figure D-10

Figure D-11

Figure D-12

Figure D-13

Radionuclides in Tank 32H Sludge Versus Mark 22 At Various Exposures

Figure D-14

Figure D-15

Radionuclides in Tank 36H Sludge Versus Mark 22 At Various Exposures

Figure D-16

Radionuclides in Tank 39H Sludge Versus Mark 22 At Various Exposures

Figure D-17

Radionuclides in Tank 11H Sludge Versus Mark 22 At Various Exposures

Figure D-18

Radionuclides in Tank 12H Sludge Versus Mark 22 At Various Exposures

Figure D-19

Radionuclides in Tank 13H Sludge Versus Mark 22 At Various Exposures

Figure D-20

Radionuclides in Tank 14H Sludge Versus Mark 22 At Various Exposures

Figure D-21

Radionuclides in Tank 16H Sludge Versus Mark 22 At Various Exposures

Figure D-22

Radionuclides in Tank 21H Sludge Versus Mark 22 At Various Exposures

Figure D-23

Radionuclides in Tank 22H Sludge Versus Mark 22 At Various Exposures

Figure D-24

Radionuclides in Tank 43H Sludge Versus Mark 22 At Various Exposures

Figure D-25