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INTEGRATED BETA AND GAMMA RADIATION DOSE CALCULATIONS FOR THE FERROCYANIDE WASTE TANKS

EXECUTIVE SUMMARY

Some of the storage tanks at the Hanford Site contain significant amounts of ferrocyanide waste. Ferrocyanide, when combined with an oxidizer, may undergo uncontrollable exothermic reactions. Ferrocyanide was used to separate long-lived fission products from the high-level waste in the late 1950's. Because radioactive background fields tend to break down ferrocyanide compounds by radiolysis, the total integrated beta and gamma radiation dose received by the ferrocyanide waste is needed to estimate the amount of ferrocyanide compounds presently remaining in the waste tanks.

Measurements were made and calculations performed to obtain a relationship to convert the gamma count rate data to dose rates. This relationship is applied to the gamma count rate data obtained for each waste tank. Vertical gamma count rate data are routinely taken inside the liquid observation well (LOW) to monitor the interstitial liquid level of the tanks. These data are stored in the Computer Automated Surveillance System located in 200 East Area on the Hanford Site. This relationship is valid for waste tanks having the same material properties and radioactive sources as the ferrocyanide-bearing waste.

The ferrocyanide-bearing waste tank with the highest gamma radiation dose rate is BY-106. Its maximum dose rate is 3.79 ± 0.19 Gy/h and occurs at the bottom of the LOW. This tank also had the maximum dose rate of all the waste tanks containing a LOW. The uncertainty is based on the data obtained from the measurements and calculations but does not include variations in the material density of the tank waste.

An analytical spreadsheet model was developed to calculate and estimate the total integrated beta and gamma radiation doses in all the ferrocyanide-bearing waste tanks. These calculations were based on historical waste

transfer records for the waste tanks. It was found that tank TY-104 had the highest estimated total integrated beta and gamma exposure doses. The maximum estimated integrated beta radiation dose value was 2.1×10^7 Gy with a probable estimated average value of 4.9×10^6 Gy and a minimum estimated value of 1.8×10^6 Gy. The maximum estimated integrated gamma radiation dose value was 3.6×10^7 Gy, with a probable estimated average value of 8.5×10^6 Gy and a minimum estimated value of 3.1×10^6 Gy. Note: One Gray is equal to 100 rad.

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INTEGRATED BETA AND GAMMA RADIATION DOSE CALCULATIONS FOR THE FERROCYANIDE WASTE TANKS

1.0 INTRODUCTION

Defense radioactive waste is stored in underground waste tanks at the Hanford Site. The radiation dose rates inside the waste tanks are not well known. It is important to learn the radiation dose rates and total integrated radiation doses received by the wastes to predict the operating lifetime of tank instrumentation and to estimate the radiation degradation of chemical components of the waste. Of particular interest is the radiological degradation of ferrocyanide compounds found in some waste tanks because of their potential chemical reactivity. Radiation-induced chemical changes over the 40 years of waste storage have affected the waste composition and chemical reactivity. Analyses of samples from ferrocyanide-bearing waste tanks 241-C-109 and 241-C-112 showed ferrocyanide concentrations are only about 13 to 19% of the estimated analytical process flow sheet inventories (Simpson et al. 1993a, 1993b). Experiments are required to understand ferrocyanide degradation to bridge the gap between what was placed in the tanks and what now remains. Integrated beta and gamma radiation dose and dose rate values are necessary to evaluate the degradation process.

1.1 PURPOSE

The objective of these analyses is to determine correlations relating the gamma count rates to dose rates present in the liquid observation wells (LOWs) and then use them to estimate the total integrated beta and gamma radiation doses the ferrocyanide-bearing waste have received over the 40 years of storage. This information will allow for better estimates of the amount of degradation of ferrocyanide compounds in the wastes. Alpha particles also contribute to radiolysis of the ferrocyanide compounds, however, calculations of the total integrated alpha radiation dose is beyond the scope of this work. Because the waste varies from one tank to another, the total integrated radiation doses for each waste tank was calculated on an individual basis.

The total integrated beta and gamma radiation dose calculations were performed in several stages. These stages are separated into the experimental approach and analytical approach of this report. The experimental approach consisted of measuring the gamma radiation dose rate in three ferrocyanide-bearing waste tanks using thermoluminescent dosimeters (TLDs) inserted inside the LOWs. The measured TLD gamma radiation dose rates were then fitted to a 5th order equation based on the gamma count rate data routinely taken for each waste tank. This equation was then applied to the other ferrocyanide waste tanks to obtain radiation dose rate information for these tanks. The analytical approach consisted of calculating the total beta and gamma integrated radiation doses. Numerical integration of the measured and calculated radiation dose rates were performed based on historical waste tank transfer records and calculated radionuclide waste composition.

1.2 BACKGROUND

Defense radioactive waste has accumulated at the Hanford Site in underground waste tanks since the mid-1940's. During the 1950's, additional storage space was necessary to store the increased defense radioactive waste. To obtain this additional storage space while minimizing the construction of additional storage tanks, a process was developed that could remove the long-lived fission products from the liquid waste in the waste tanks. In implementing this process, approximately 140 metric tons of ferrocyanide were added to the waste. The process resulted in the long-lived radionuclides, ^{137}Cs and ^{90}Sr , settling to the bottom of the waste tanks. The liquid waste was then discharged to the trenches or evaporated. As a result of this process, the total waste volume in the storage tanks was reduced, at the same time minimizing the amount of long-lived radionuclides discharged to the environment. This treatment was employed on the U Plant waste effluent, bismuth phosphate decontamination waste, and selected storage wastes in the tanks.

Ferrocyanide is a stable compound because it does not dissociate in aqueous solutions. Ferrocyanide in the presence of oxidizing material such as sodium nitrate or sodium nitrite has been made, in the laboratory, to undergo self-sustaining violent exothermic chemical reactions by heating it to high temperatures or by applying an electrical spark of sufficient energy. Significant exothermic reactions can begin at temperatures as low as 220 °C, but the lowest propagating reaction temperature observed is approximately 285 °C, using dry and relatively pure ferrocyanide sodium nitrate/nitrite compounds. The violent exothermic reactive nature of ferrocyanide in the presence of sodium nitrate or sodium nitrite compositions has been known for decades, but the conditions under which the ferrocyanide compounds can undergo a self-sustaining exothermic reaction have not been thoroughly studied (Scheele 1993).

2.0 EXPERIMENTAL APPROACH

An experimental method of measuring the gamma radiation dose rate in terms of Grays per hour in the waste tank LOWs was developed. Using the information obtained from this experiment, an equation to convert the gamma count rate data taken routinely to dose rates was developed. Based upon the range of observed gamma count rates, three ferrocyanide-bearing waste tanks were chosen for experimental measurement of the gamma radiation dose rate. The three ferrocyanide waste tanks chosen were BY-104, BY-110, and TX-118. An equation relating the gamma count rate data taken in the LOW of these three waste tanks to Grays per hour was developed. This relationship should be applicable to the gamma count rate measurements taken in the LOWs of other waste tanks assuming that the waste density, the main source of gamma radiation and gamma ray attenuation near and in the LOW are the same for all of the waste tanks.

2.1 EXPERIMENTAL PROCEDURE

Waste tanks with a LOW are routinely scanned using a Geiger-Muller (GM) tube gamma detector. The waste tanks are monitored for changes in the interstitial liquid level of the radioactive waste. The gamma count rate information is stored in the Computer Automated Surveillance System (CASS) located in the 200 East Area of the Hanford Site in Richland, Washington (Spurling 1991). The LOWs are made from fiberglass and reinforced epoxy-polyester resins. The LOWs have an inner diameter of 7.62 cm and an outside diameter of 8.89 cm. They are sealed at the bottom end and extend to within 1 to 58 cm of the bottom waste tank steel liner. A cross-sectional view of a typical single shell ferrocyanide-bearing waste tank is shown in Figure 2-1. (Hanlon 1994).

The GM tube is of type TGM-V302HT, operating at a voltage of 900 V DC, with an upper temperature limit of 150 °C. The detector is cylindrical in shape with a height of 1.27 cm and radius of 0.9525 cm. Its center is about 3.651 cm from the bottom of the surrounding casing. The detector is surrounded by 2.103 cm of lead shielding, so that it can only detect gamma radiation through a 0.254 cm wide slit in its side. The detector also has 0.762 cm of lead shielding inside of the slit. Figure 2-2 shows a cross sectional view of the detector present in the LOW. This detector was designed to measure the gamma radiation dose rate at different levels of the waste without interference from gamma radiation originating above or below the level of the detector's slit opening. The gamma count rate measurements are taken at every 0.254 cm along the vertical height of the LOW.

The GM detector is calibrated accurately at least once a year by Battelle Pacific Northwest Laboratory using five different gamma sources, producing between 0.50 mGy/hr and 1.50 Gy/hr fields, that are traceable to the National Institute of Standards and Technology. Since the detector is routinely employed for the surveillance of the interstitial liquid and solid waste levels in the underground tanks containing a LOW, the detector's calibration is checked before usage. Tank farm operating procedures require that before usage, the GM detector be checked in two ways. The voltage setting for the counting plateau be checked and the count rate be checked against a known standard gamma source. The count rate must be within 5% of that expected when exposed to the known standard gamma source.

The gamma radiation dose rate measurement experiment consisted of inserting a string of spaced TLDs into each tank's LOW. The TLDs were of the 700 type which contains 99.993% ⁷Li. Lithium-7 is sensitive to only gamma radiation because its neutron cross section is negligible (Tsoulfanidis 1983). The TLD strings used 4 TLDs per location. The measurements were performed with 4 TLDs at each location to obtain an uncertainty in each measured dose rate of less than 5%. Each TLD had a cross sectional area of 1 cm² by 0.2 cm thick and were positioned in the center of the LOW. The TLDs were located 30.48 cm apart for the first 365.76 cm (corresponding to the waste region) and 91.44 cm apart in the following 548.64 cm (corresponding to the air region above the waste). The distances from the top of the riser flange to the inside bottom of the LOW are 1453.59, 1437.74, and 1474.93 cm for the BY-104, BY-110, and TX-118 waste tanks respectively. The bottom first TLD group was located 15.399, 13.176, and 11.906 cm above the LOW bottom for the BY-104, BY-110, and TX-118 waste tanks, respectively.

Figure 2-1. Cross Sectional View of a Ferrocyanide-Bearing Waste Tank.

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Figure 2-2. Cross Sectional View of Geiger-Muller Detector inside the Liquid Observation Well.

The TLDs were evaluated and measured by Battelle Pacific Northwest Laboratory to obtain the Grays per hour versus height data. The experimental data is reported in Appendix A. The TLDs were calibrated by exposing them to 0.01 Gy from a ^{137}Cs source #318-40, which is traceable to the National Institute of Standards and Technology. The TLD's measured doses are linear as a function of radiation exposure for the range of interest.

2.2 EXPERIMENTAL EVALUATION OF DATA

The comparison and evaluation of the experimental gamma radiation dose rate measurement with the routinely monitored gamma count rate was divided into two regions. The first region was the waste itself and the second region was the air above the waste in the tank. A more accurate linear regression calculation could be performed in the waste region without taking into consideration the data from the air region and vice versa. The GM detector does not record the correct gamma radiation dose rate above the waste surface. As mentioned before, the GM detector is designed only to measure gamma radiation from a 0.254 cm slit around its circumference. Therefore, appropriate correction factors had to be developed to determine the dose rate in the air due to radiation from the waste surface. It was also realized that the gamma radiation dose in the air region is also due to waste deposits on the tank walls and gamma ray reflections from the tank walls.

These data obtained from the GM detector for the three waste tanks of interest are presented in Figure 2-3. The gamma count rate data were taken between September and November 1993 and are stored in CASS. These data were not corrected for decay because they have decayed less than 1% of their present value, from September 1993 to February 1994. The measured experimental data obtained from the TLDs are presented in Figure 2-4. The curves in these two figures are very similar; however, they do exhibit some differences due to the different methods of detecting gamma radiation by the GM detector and the TLDs. As mentioned, the GM detector can only detect gamma radiation through its small shielded slit opening, while the unshielded TLDs can detect gamma radiation from every side and direction. As a result, the peaks in the TLDs results are larger due to gamma radiation from sources above and below the TLDs location. Furthermore, the minimum points, seen in the gamma count rate data, are not visible in the experimentally measured data because the TLDs are measuring gamma radiation from all directions, and therefore, these minimum points are washed out in the background radiation.

Both the GM detector and TLDs are very sensitive to beta radiation, however, no beta particles are expected inside the LOW. The LOW is approximately 0.508 cm thick and made of fiberglass, preventing any beta particles from traveling through the LOW material. Of greater interest are the continuous x-rays, called bremsstrahlung, produced when beta particles of high energy are slowed as they pass through the LOW material.

ISOSHLD-II calculations show that any x-rays produced from bremsstrahlung will have very low energies. Because the GM detector is surrounded by 0.762 cm of lead, these low energy x-rays will not be detected. ISOSHLD-II calculations also show that the percentage of x-rays produced from bremsstrahlung compared to the total gamma radiation being measured by the TLDs is less than 1.0% (Appendix B). This value is negligible because the measurement error is 5%.

Figure 2-3. Geiger-Muller Measured Data.

Figure 2-4. Thermoluminescent Dosimeter Measured Data.

The gamma attenuation due to absorption and scattering in both sets of data were not taken into consideration in calculating the gamma dose rates in the waste tanks. Simple calculations show that the attenuation of gamma rays traveling through the LOW walls (0.635 cm thick) is less than 5% depending on the energy of the gamma rays. If buildup factors are taken into consideration due to scattering of the gamma rays in the LOW material, then the attenuation is less than 3% of the total gamma rays passing through the wall. The low energy gamma rays are attenuated more than the high energy gamma rays.

The dose rate data obtained from the TLD measurements (Appendix A) was correlated to the routinely measured gamma count rate data stored in CASS using linear regression analysis (Strang 1986). Linear regression analysis showed that the best-fit equation was a 5th order equation for all the data obtained in the TLD measurements. The 5th order equation had the least standard error of the Y estimate. The 5th order "best-fit" equation relating the counts per second to Grays per hour is given by

$$Y = 2.46 \cdot 10^{-4} \cdot X - 1.55 \cdot 10^{-7} \cdot X^2 + 5.23 \cdot 10^{-11} \cdot X^3 - 6.14 \cdot 10^{-15} \cdot X^4 + 2.41 \cdot 10^{-19} \cdot X^5 \quad (1)$$

where

Y = Corrected gamma radiation dose rate, Gy/h
X = Dose rate obtained from GM detector, Counts/s

and this equation is independent of height in the waste region. The calculated standard deviation is ± 0.19 Gy/h. Since both the gamma count rate and the experiment measurements were affected by the same amount of gamma attenuation due to the LOW material, this amount cancels out in the linear regression analysis. This equation is valid only in the waste region of the waste tanks and does not take into consideration material density variations between the tank waste. The conversion from Gray to rad is 1 Gy = 100 rad.

The exposure rate in the air region above the waste can be accurately predicted by a first order linear equation. The air region does not follow an infinite planar radiating surface assumption because of the effects the tank walls have in the air region. The waste level in the storage tanks has increased and decreased over time leaving behind on the walls a thick waste deposit. The waste deposited on the wall are contributing to the gamma radiation dose rate in the air region. Furthermore, the tank walls are reflecting and attenuating gamma rays into the air region. The first order equation that accurately predicts the gamma radiation dose rate in the air region above the waste is

$$Y = -9.84 \cdot 10^{-5} \cdot D + Y_{\text{surface}} \quad (2)$$

where

Y = Correlated gamma radiation dose rate, Gy/h
D = Distance from the surface of the waste, cm
Y_{surface} = Dose rate at the surface of the waste, Gy/h

Equation 2 is given in the document to predict the radiation dose rates above the waste for analysis of the radiation effects on instrumentations and on materials. This equation was derived from the TLDs experiment and it calculates an upper limit value for the gamma radiation dose rate in the air region. The actual gamma radiation dose rate in the LOW may be less than this amount due to interference by the tank walls or other nearby instrumentation, tubes, etc. The calculated standard deviation is ± 0.06 Gy/h. This uncertainty also includes the gamma ray attenuation due to absorption and scattering due to the LOW walls in the air region. The dose rate at the surface of the waste (Y_{surface}) can be obtained from Equation 1.

The gamma radiation dose rate in the air region for each experiment was measured and compared to the results of Equation 2. The waste tank BY-104 has its LOW located in riser R-10C which is about 630 cm from the center of the tank. The LOW is near the tank wall causing the TLDs in the air region to receive less radiation than they would far away from the tank wall. The measured slope of the gamma radiation dose rate is -5.381×10^{-4} Gy/h-cm. The LOW for the waste tank BY-110 is located about 244 cm from the tank center in riser R-3. Its measured gamma radiation dose rate slope in the air space is -9.843×10^{-5} Gy/h-cm. The waste tank TX-118 has its LOW located about 152 cm from the tank center. However, the riser R-5, which houses the LOW, also has three other tubes with instrumentation. As a result, these other tubes affected the amount of gamma radiation the TLDs in the air space received. The calculated gamma radiation dose rate slope is -1.706×10^{-4} Gy/h-cm.

A comparison of the measured gamma radiation dose rate using TLDs with the monitored converted gamma count dose rate was performed. The calculated results using Equations 1 and 2 agree very well with the gamma count rate data, within their standard deviations. Figure 2-5 compares the experimentally measured with the calculated gamma radiation dose rate as a function of height for the waste tank BY-104. The only major disagreement is at the bottom of the waste tank where Equation 1 over predicts the gamma radiation dose rate. The calculated gamma radiation dose rate in the air region is also higher than the measured dose because the LOW is located near the wall. Figure 2-6 plots the experimentally measured and calculated gamma radiation dose rate for waste tank BY-110. There is very good agreement between both of them. Finally, Figure 2-7 gives the radiation dose rate for waste tank TX-118. Again, the calculated dose rate is over predicted at the bottom and top of the waste tank and in the air region. The differences are within the standard deviation uncertainties for both equations.

While the gamma activity detected by the GM agrees well with the collimated detector and the TLDs, the gamma activity from the waste composition near the LOW may not be representative of the average waste composition for the tank. There are uncertainties due to the procedure used to install the LOW into the waste. A water lance was employed to form an opening in the waste for the LOW. The material composition that now fills this space near the LOW may not be a composition representative of the entire tank (Keele et al. 1991).

Figure 2-5. BY-104 Estimated Radiation Exposure Dose Rate.

Figure 2-6. BY-110 Estimated Radiation Exposure Dose Rate.

Figure 2-7. TX-118 Estimated Radiation Exposure Dose Rate.

The relationship between count rate and radiation dose rate is highly dependent upon the energy of the radiation emitted, the type of radiation emitted, the geometric conditions, and the amount of absorbing material between the radioactive material and the receiving TLDs (Slade 1968). Equation 1 and 2 were developed based on conditions existing in ferrocyanide-bearing waste tanks. Therefore, these equations may not be applicable to other tanks having a waste composition different from the ferrocyanide waste.

2.3 EXPERIMENTAL APPLICATIONS

There are 20 waste tanks which are classified as containing significant amounts of ferrocyanide waste mixed with sodium nitrate and sodium nitrite. The waste tanks are listed in Table 2-1. The safety of the ferrocyanide-bearing waste tanks are of concern because concentrated ferrocyanide sodium nitrate/nitrite chemical combinations can undergo a violent exothermic chemical reaction (Postman et al. 1992). To estimate how much ferrocyanide compounds are present in the waste tanks, the integrated beta and gamma radiation doses for each waste tank must be known. Equation 1 is applied to all the ferrocyanide-bearing waste tanks to obtain the expected Grays per hour dose rate. However, only 10 of the ferrocyanide-bearing waste tanks contain LOWs. The location of the LOW for the ferrocyanide-bearing waste tank containing a LOW are also given in Table 2-1 (Hanlon 1994). The calculated gamma radiation dose rate for the other nonferrocyanide-bearing waste tanks containing a LOW are presented in Appendix C.

Table 2-1. Ferrocyanide-Bearing Waste Tanks'
Liquid Observation Well Location.

Tank	Port Location	Radial Port Location (cm)
BX-102		
BX-106		
BY-103	R-10A	243.8
BY-104	R-10C	632.5
BY-105	R-10A	632.5
BY-106	R-10B	632.5
BY-107	R-7	152.4
BY-108		
BY-110	R-3	243.8
BY-111	R-1	243.8
BY-112	R-15	894.6
C-108		
C-109		
C-111		
C-112		
T-107		
TX-118	R-5	894.6
TY-101		
TY-103	R-3	243.8
TY-104		

Gamma count rate data are routinely taken on the ten ferrocyanide-bearing waste tanks that contain LOWs to monitor for changes in their interstitial liquid level and are stored in CASS. Work is in progress to install LOWs in the rest of the ferrocyanide-bearing waste tanks. Gamma count rate data available for the ten ferrocyanide-bearing waste tanks has been converted to Grays per hour using Equation 1 and plotted. Figures 2-8 through 2-17 show calculated gamma radiation dose rate as a function of height in the waste tanks. The maximum gamma radiation dose rate is 3.79 Gy/h and occurs at the bottom of tank BY-106 (Figure 2-11).

Note, however, that the bottom of the LOW in tank BY-106 is about 2.3 cm above the tank's bottom surface and about 630 cm from the center of the tank. Therefore, the actual maximum gamma radiation dose rate at the bottom of tank BY-106 maybe higher than 3.79 Gy/h. The average gamma radiation dose rate for these ten ferrocyanide-bearing waste tanks is 0.91 Gy/h. The points on the graph were taken every 0.254 cm, which comes from the gamma count rate resolution.

Calculations and measurements performed on the waste tanks using a collimated detector have found BY-106 to have the highest gamma radiation activity of the ferrocyanide-bearing waste tanks. The gamma radiation dose rate from these calculations and measurements is 4.12 Gy/h (Keele et al. 1991). This measured value was corrected for decay of the ^{137}Cs and ^{90}Sr from January 1991 to February 1994. The estimated dose rate obtained using Equation 1 is 3.79 Gy/h. The estimated value of 3.79 Gy/h is within two standard deviations calculated from Equation 1.

Figure 2-8. BY-103 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure 2-9. BY-104 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure 2-10. BY-105 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure 2-11. BY-106 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure 2-12. BY-107 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure 2-13. BY-110 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure 2-14. BY-111 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure 2-15. BY-112 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure 2-16. TX-118 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure 2-17. TY-103 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

3.0 ANALYTICAL APPROACH

An analytical spreadsheet model for estimating the total integrated beta and gamma radiation doses for all the ferrocyanide-bearing waste tanks was developed. This spreadsheet model is based on the historical records that indicate the amount of waste that was added and removed from each tank. Only the waste tanks containing a LOW were accurately calculated because the analytical spreadsheet model utilizes the calculated gamma radiation dose rate for the waste (Cember 1983). The total integrated beta and gamma radiation doses for the other ferrocyanide-bearing waste tanks had to be estimated. The integrated beta and gamma radiation doses for each ferrocyanide-bearing waste tank was calculated based on the average and maximum radiation exposures that could have existed in the waste tanks.

3.1 ANALYTICAL EXPOSURE

The average and maximum gamma radiation exposure dose rates in the waste were calculated for the ferrocyanide-bearing waste tank containing LOWs. These calculations were performed by using Equation 1. The results are shown in Table 3-1. Table 3-1 assumes a constant exposure dose rate from the bottom of the LOW to the bottom of the waste tank. Differences in homogeneous waste batches added to the waste tanks over time caused vertical variations in the radiation exposure rates. The maximum exposure dose rates occurred because a homogeneous batch containing a higher concentration of ^{137}Cs and ^{90}Sr was added to the waste tank (Simpson 1993a, Simpson 1993b).

Table 3-1. Gamma Exposure Dose Rate.
(Gy/h) March 1994

Tank	Average	Maximum
BY-103	0.99	1.63
BY-104	1.26	1.67
BY-105	0.84	1.67
BY-106	1.53	3.79
BY-107	1.49	2.18
BY-110	1.13	2.08
BY-111	0.55	1.19
BY-112	0.45	1.61
TX-118	0.33	0.64
TY-103	0.52	1.43

The average and maximum exposure dose rates were used in normalizing the radioactive decay of the nuclides to give the correct gamma radiation dose rate in March 1994. This method was employed to account for radioactive nuclides not accounted for in the calculations. The total integrated beta and gamma radiation doses were calculated based on the average and maximum dose rates possible in each tank. For the ferrocyanide-bearing waste tanks without any LOWs, a best estimated minimum, average, and maximum exposure dose rates were used. These values were based on analysis of Table 3-1. The estimated minimum value is taken to be 0.33 Gy/h, 0.91 for the estimated average, and 3.79 Gy/h for the estimated maximum exposure dose rate values.

3.2 ANALYTICAL EVALUATION OF DATA

An analytical spreadsheet model was developed to calculate the total integrated beta and gamma radiation doses for each waste tank (Cember 1983, Slade 1968). It accurately accounted for the insertion and removal of the waste as a function of time. These input data were taken from historical records supplied by TRAC and confirmed by other calculations (Anderson 1990, Borsheim et al. 1991, Jungfleisch 1984). Since waste was removed and added sometimes within the same year, the final waste level at the end of the year for each tank was used. The model also kept track of the fission products and their decays. The dose rate from the beta and gamma particles are taken to be equal (Slade 1968). Finally, the main equations and theory used in the spreadsheet model are described and presented in detail in Appendix D. While some assumptions have been made in developing this model, those considered most important are discussed.

All of the ferrocyanide waste present in each tank was added at the beginning of first addition of ferrocyanide waste into that tank and has remained in the waste tank since addition. The waste tanks received ferrocyanide waste throughout the scavenging program lifetime. Ferrocyanide waste was added to each waste tank between 1953 and 1956. It has remained in storage since that time. Table 3-2 gives the starting year of operation for each waste tank and the year that ferrocyanide waste was first added to the waste tank. The ferrocyanide-bearing waste tanks remained in use after the scavenging program and were used for the receipt, storage, and processing of waste until the single-shell tanks were removed from active service in 1980.

Table 3-2. Waste Tank Operational Date.

=====		
Tank	Waste Received	Ferrocyanide Addition
=====		
BX-102	1956	1956
BX-106	1956	1956
BY-103	1955	1955
BY-104	1951	1955
BY-105	1951	1954
BY-106	1953	1954
BY-107	1950	1953
BY-108	1951	1953
BY-110	1952	1954
BY-111	1952	1956
BY-112	1952	1956
C-108	1947	1952
C-109	1952	1952
C-111	1946	1952
C-112	1946	1953
T-107	1945	1952
TX-118	1952	1953
TY-101	1953	1959
TY-103	1953	1953
TY-104	1953	1953

Waste associated with processing of fuel from B, D, DR, F, and H production reactors was assumed to be the majority of the waste stored in the ferrocyanide-bearing waste tanks. Using this assumption, ORIGEN2 calculations were performed to obtain the fission product inventory for the waste (Croft 1980). Only the major gamma emitting fission products were tracked. All the fission products were assumed to be gamma emitters while the correct yield percentage of beta particles was used in the total integrated radiation dose calculations. A list of the major radionuclides taken into consideration in the calculations is given in Appendix E. Nuclides were also removed from the ORIGEN2 calculations to account for chemical reprocessing. For example, reprocessing flowsheets show complete release of ^{85}Kr to the off-gas system on fuel dissolution.

Small amounts of ^{60}Co have been detected in some waste tanks (Simpson 1993). This beta and gamma emitting isotope is not taken into consideration in the total integrated beta and gamma radiation exposure dose calculations. Cobalt-60 was produced in negligible amounts from activation of the aluminum cladding impurities. Furthermore, ^{60}Co has a half life of only 5.27 yr (Browne et al. 1978).

The amount of radionuclides added to the waste tanks was proportional to the fission product inventory obtained from ORIGEN2 calculations. The concentrations of fission products were calculated for the Production Reactors operating at a higher power level than was normal. This was done to increase the calculated concentration of short-lived fission product production for a maximum value calculation. Historical records show that some of the waste added to the waste tanks was stored waste from another tank and not newly reprocessed waste. The amount of radionuclides removed from the waste tanks was proportional to the radionuclides present in the waste at the time of removal. Furthermore, the distribution of non-accounted nuclides in the waste tanks is assumed constant. These non-accounted nuclides are the activation, short-lived fission, and actinides products. To correct for the non-accounted nuclides, the total activity of the waste tanks was normalized to equal the present average or maximum gamma radiation dose rate. Activation products account for an insignificant fraction of the beta and gamma emitting nuclides and the small amount of actinides products that ended up as waste could not be accurately determined.

Finally, the composition of the waste was assumed to be homogeneous throughout the tank and the same for each tank. This was a simplifying assumption. Vertical layering of the waste due to composition differences between batches is known to exist. Furthermore, the horizontal non-uniform waste composition is unknown. Also, the waste composition varies from one tank to another due to varying batch compositions and waste transfer mixing from one tank to another. The spreadsheet calculations were performed with these assumptions incorporated and with the assumption of last waste into the tank is the first waste removed from the tank.

3.3 ANALYTICAL APPLICATIONS

The total integrated beta and gamma radiation doses for the ferrocyanide waste was calculated using the spreadsheet model with its assumptions. This spreadsheet model corrected for the time period between the addition of radioactive waste into the tanks and the addition of ferrocyanide-bearing

waste which occurred later, see Table 3-2. The integrated dose calculations were performed from the beginning of ferrocyanide waste addition to each individual tank until March 1994. The dose rate due to beta radiation was calculated to be 1.09 times greater compared to the dose rate from gamma radiation in the ferrocyanide-bearing waste (Slade 1968). For simplicity, the dose rates are assumed to be equal. The results of the calculations are presented in Table 3-3. Note that 1 Gy is equal to 100 rad.

Table 3-3. Total Integrated Beta and Gamma Radiation Dose.
(Gray) March 1994

Tank	Average		Maximum	
	Beta	Gamma	Beta	Gamma
BY-103	4.37×10^5	9.28×10^5	7.20×10^5	1.53×10^6
BY-104	3.85×10^5	8.65×10^5	5.00×10^5	1.12×10^6
BY-105	2.47×10^5	5.50×10^5	4.91×10^5	1.09×10^6
BY-106	5.22×10^5	1.02×10^6	1.29×10^6	2.52×10^6
BY-107	8.68×10^5	1.84×10^6	1.27×10^6	2.69×10^6
BY-110	4.11×10^5	9.34×10^5	7.57×10^5	1.72×10^6
BY-111	1.82×10^5	3.79×10^5	3.93×10^5	8.19×10^5
BY-112	2.56×10^5	4.91×10^5	9.17×10^5	1.76×10^6
TX-118	1.41×10^5	2.86×10^5	2.74×10^5	5.55×10^5
TY-103	5.17×10^5	9.83×10^5	1.42×10^6	2.70×10^6

Sensitivity studies were performed on the spreadsheet model and data to obtain a best guess uncertainty estimate of the total integrated beta and gamma radiation dose values. Assuming the input information obtained from historical records is correct, the best guessed uncertainty in the measurements is 11% at the 95% confidence interval based on statistical analysis of the data. This estimate is established for the ferrocyanide waste tank data containing the worst statistics, BY-112. The sensitivity study did not take into consideration the uncertainties due to the assumptions used in the analytical spreadsheet model. They could not be calculated and were not estimated.

The total integrated beta and gamma radiation doses for the other ten ferrocyanide-bearing waste tanks without a LOW were estimated. Assuming the ferrocyanide-bearing waste tanks without a LOW have a gamma radiation dose rate within the calculated gamma radiation dose rate given in Table 3-1, upper and lower bound values can be calculated. The lowest minimum gamma radiation dose rate calculated was 0.33 Gy/h while the highest maximum was 3.79 Gy/h. The average gamma radiation dose rate for all the ferrocyanide-bearing waste tanks with LOWs was 0.91 Gy/h. Taking these values for the ferrocyanide-bearing waste tanks without a LOW, upper and lower bound total integrated beta and gamma radiation dose values can be calculated using the spreadsheet model with the corresponding historical data. The calculated exposure dose values are shown in Table 3-4 and 3-5.

Table 3-4. Estimated Total Integrated Beta Radiation Dose.
(Gray) March 1994

Tank	Minimum	Average	Maximum
BX-102	6.2×10^5	1.7×10^6	7.2×10^6
BX-106	1.1×10^6	2.9×10^6	1.2×10^7
BY-108	1.5×10^5	4.2×10^5	1.8×10^6
C-108	9.4×10^5	2.6×10^6	1.1×10^7
C-109	9.9×10^5	2.9×10^6	1.2×10^7
C-111	8.1×10^5	2.3×10^6	9.4×10^6
C-112	4.5×10^5	1.2×10^5	5.1×10^6
T-107	2.9×10^5	7.8×10^5	3.3×10^6
TY-101	3.5×10^5	9.6×10^5	4.1×10^6
TY-104	1.8×10^6	4.9×10^6	2.1×10^7

Table 3-5. Estimated Total Integrated Gamma Radiation Dose.
(Gray) March 1994

Tank	Minimum	Average	Maximum
BX-102	1.2×10^6	3.4×10^6	1.4×10^7
BX-106	1.8×10^6	5.0×10^6	2.1×10^7
BY-108	3.3×10^5	8.9×10^5	3.7×10^6
C-108	1.6×10^6	4.3×10^6	1.8×10^7
C-109	1.9×10^6	5.3×10^6	2.2×10^7
C-111	1.6×10^6	4.4×10^6	1.8×10^7
C-112	8.5×10^5	2.4×10^6	9.8×10^6
T-107	6.4×10^5	1.8×10^6	7.4×10^6
TY-101	7.0×10^5	2.0×10^6	8.1×10^6
TY-104	3.1×10^6	8.5×10^6	3.6×10^7

Samples taken from the C-109 and C-112 tanks, believed to be from undisturbed ferrocyanide-bearing waste, show concentrations very close to the ORIGIN2 results. The concentration of ^{137}Cs is closed to the values used in the spreadsheet calculations; therefore, the integrated average gamma doses should be accurate. Because large amounts of ^{90}Sr were added to the waste tanks during the 1960s from the Hot Semiwork Facility, the integrated average beta doses given in Tables 3-4 and 3-5 are underestimated. In the 1960s, aluminum cladding waste from the Hot Semiworks Facility was added to the C tanks. This Hot Semiwork Facility waste contained a high concentration of ^{90}Sr because it included strontium recovery and purification waste losses. No accurate records of how much of this waste was added to the waste tanks exist. The amount of strontium addition is unknown. Therefore, the total integrated beta doses are underestimated. Calculations performed with the information provided by Simpson's reports show that the actual beta dose rates are 1.9 times higher; therefore, the integrated beta doses could be higher by a factor of 1.9 (Simpson 1993a, Simpson 1993b). This value cannot be substantiated because the actual amount of waste from the Hot Semiworks Facility added to the tanks is difficult to estimate.

4.0 CONCLUSION

A simple procedure for calculating the expected gamma radiation dose rate in any waste tank from existing surveillance data was developed. These equations, developed for the waste and air regions are applicable to all waste tanks having the same waste composition as ferrocyanide-bearing waste and should help in estimating a more accurate gamma radiation dose rate for the design of future in-tank equipment. The calculated gamma radiation dose rate for all the waste tanks containing a LOW are presented in Appendix C. The calculations are based on all the waste tanks having the same waste composition, material density, and energy gamma spectrum.

The best estimate total integrated beta and gamma radiation doses received by the ferrocyanide-bearing wastes were obtained. Calculations were performed for the average and maximum total integrated beta and gamma radiation doses to which the ferrocyanide wastes have been exposed. The average values given in the document should be used because they were calculated conservatively. This information should help in more accurately estimating the chemical changes that have taken place in the waste over time.

The total integrated average beta and gamma radiation doses were calculated conservatively in the sense that they were underestimated. Some of the conservatism built into the calculations were that the total energy per beta and gamma particles were equal and that the dose rate from beta and gamma particles were also equal. Research has shown that beta particles have a greater effect on radiolysis of compounds than gamma particles (Tipton 1960). Calculations also show that the dose rate from beta particles is 1.09 times greater than from gamma particles in the ferrocyanide-bearing waste (Cember 1983, Slade 1968). Other conservatism included, underestimating the amount of ^{90}Sr present in the waste, assuming that the waste added to the tanks was always reprocessed waste, and not accounting for bremsstrahlung radiation. Waste from the Hot Semiworks Facility was added to some of the waste tanks in the 1960s (Simpson 1993a, Simpson 1993b). This waste had a high concentration of ^{90}Sr which could not be accurately estimated. Historical records also show that some of the waste added to the tanks was transferred from another waste tank and not necessarily waste from chemical reprocessing (Anderson 1990). Finally, bremsstrahlung radiation in the waste was not accounted for in the calculations even though it has a small affect on the dose rate (Cember 1983).

The calculations and experiments performed were compared to available data. Even though there exists limited data on the gamma radiation dose rates and waste composition distribution of nuclides in the waste tanks, the results presented in this report agree reasonably well with the data available (Keele et al. 1991, Postma et al. 1992, Simpson et al. 1993a, Simpson et al. 1993b).

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APPENDIX A

THERMOLUMINESCENT DOSIMETER EXPERIMENTAL DATA

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The exposure rate results of the experiments performed on the waste tanks BY-104, BY-110, and TX-118 on March 2 and 8, 1994 are given. Four thermoluminescent dosimeters (TLDs) located in each location were averaged to obtain an uncertainty of less than 5%. Each TLD had an exposed area of 1 cm² and were 0.2 cm thick. The positions are measured from the bottom of the liquid observation well (LOW). The LOWs are located in risers R-10C, R-3, and R-5 for the BY-104, BY-110, and TX-118, respectively.

Table A-1. BY-104.
March 1994

Position (cm)	Absorbed Dose Rate (Gy/h)
15.40	1.458 ± 0.048
45.88	1.387 ± 0.024
76.36	1.624 ± 0.017
106.84	1.694 ± 0.013
137.32	1.695 ± 0.033
167.80	1.621 ± 0.030
198.28	1.262 ± 0.028
228.76	0.982 ± 0.014
259.24	0.895 ± 0.011
289.72	0.715 ± 0.008
320.20	0.507 ± 0.018
350.68	0.508 ± 0.013
442.12	0.484 ± 0.006
533.56	0.426 ± 0.011
625.00	0.365 ± 0.010
716.44	0.316 ± 0.006
807.88	0.280 ± 0.011
899.32	0.238 ± 0.004

For calibration purposes, the TLDs were exposed to 0.01 Gy from a ¹³⁷Cs source #318-40, which is traceable to the National Institute of Standards and Technology. The relationship between radiation exposure and radiation dose reading is linear for the range of interest.

Table A-2. BY-110.
March 1994

Position (cm)	Absorbed Dose Rate (Gy/h)
13.18	2.075 ± 0.021
43.66	1.972 ± 0.048
74.14	1.962 ± 0.010
104.62	1.847 ± 0.013
135.10	1.853 ± 0.026
165.58	1.884 ± 0.023
196.06	1.613 ± 0.013
226.54	0.863 ± 0.005
257.02	0.689 ± 0.013
287.50	0.620 ± 0.020
317.98	0.495 ± 0.006
348.46	0.420 ± 0.016
439.90	0.207 ± 0.006
531.34	0.194 ± 0.006
622.78	0.205 ± 0.006
714.22	0.186 ± 0.001
805.66	0.171 ± 0.002
897.10	0.162 ± 0.005

Table A-3. TX-118.
March 1994

Position (cm)	Absorbed Dose Rate (Gy/h)
11.91	0.554 ± 0.018
42.39	0.538 ± 0.011
72.87	0.473 ± 0.004
103.35	0.422 ± 0.011
133.83	0.189 ± 0.004
164.31	0.191 ± 0.002
194.79	0.215 ± 0.003
225.27	0.160 ± 0.002
255.75	0.186 ± 0.005
286.23	0.207 ± 0.002
316.71	0.221 ± 0.003
347.19	0.223 ± 0.003
438.63	0.239 ± 0.010
530.07	0.218 ± 0.007
621.51	0.204 ± 0.003
712.95	0.191 ± 0.004
804.39	0.174 ± 0.003
895.83	0.161 ± 0.004

APPENDIX B

ISOSHL-D-II: BREMSSTRAHLUNG RADIATION CALCULATIONS

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Bremsstrahlung radiation calculations were performed for a detector centered in the liquid observation well. The waste is simulated ferrocyanide-bearing waste. The calculations were performed to evaluate the amount of bremsstrahlung radiation being detected by the Geiger-Muller detector and the thermoluminescent dosimeters. The following is the input to the ISOSHL-II computer code used in the calculations.

```

0          2 Ferrocyanide Tank Farm Source
NO LEAD
&INPUT NEXT=1, IGEOM=7, NSHLD=2, JBUF=1, IPRNT=0,
OPTION=1, X=300.1, T(1)=0.1, DUNIT=1, SLTH=914.4, Y=457.4,
NTHETA=11, NPSI=9, DELR=0.01, T(2)=0.0,
WEIGHT(451)=6.44E-03, WEIGHT(472)=3.52E+00,
WEIGHT(027)=2.59E-04, WEIGHT(082)=2.20E+01,
WEIGHT(084)=2.20E+01, WEIGHT(103)=6.37E-04,
WEIGHT(102)=9.30E-04, WEIGHT(117)=8.02E-01,
WEIGHT(141)=9.52E-03, WEIGHT(172)=9.22E-01,
WEIGHT(170)=9.22E-01, WEIGHT(269)=5.59E-01,
WEIGHT(290)=9.03E-05, WEIGHT(319)=1.31E-01,
WEIGHT(335)=9.12E+01, WEIGHT(336)=9.65E+01,
WEIGHT(376)=4.49E+01, WEIGHT(377)=4.49E+01,
WEIGHT(388)=0.00E+00, WEIGHT(403)=1.78E-01,
WEIGHT(415)=1.20E-01, WEIGHT(519)=2.73E-05,
WEIGHT(520)=2.46E-08, WEIGHT(476)=7.40E-06,
WEIGHT(502)=3.14E-05, WEIGHT(492)=2.11E-04,
WEIGHT(526)=1.79E-04, WEIGHT(493)=4.24E-02,
WEIGHT(494)=1.97E-03, WEIGHT(496)=1.07E-02,
WEIGHT(495)=1.80E-02, WEIGHT(499)=4.04E-05,
WEIGHT(504)=3.39E-05, WEIGHT(498)=4.06E-05,
WEIGHT(505)=5.73E-04, WEIGHT(500)=1.68E-04,
LEAD 14      11.35
1 SOURCE 1 1.2
WITH LEAD
&INPUT NEXT=1, IGEOM=7, NSHLD=2, JBUF=1, IPRNT=0,
OPTION=1, X=300.1, T(1)=0.1, DUNIT=1, SLTH=914.4, Y=457.4,
NTHETA=11, NPSI=9, DELR=0.01, T(2)=0.762,
WEIGHT(451)=6.44E-03, WEIGHT(472)=3.52E+00,
WEIGHT(027)=2.59E-04, WEIGHT(082)=2.20E+01,
WEIGHT(084)=2.20E+01, WEIGHT(103)=6.37E-04,
WEIGHT(102)=9.30E-04, WEIGHT(117)=8.02E-01,
WEIGHT(141)=9.52E-03, WEIGHT(172)=9.22E-01,
WEIGHT(170)=9.22E-01, WEIGHT(269)=5.59E-01,
WEIGHT(290)=9.03E-05, WEIGHT(319)=1.31E-01,
WEIGHT(335)=9.12E+01, WEIGHT(336)=9.65E+01,
WEIGHT(376)=4.49E+01, WEIGHT(377)=4.49E+01,
WEIGHT(388)=0.00E+00, WEIGHT(403)=1.78E-01,
WEIGHT(415)=1.20E-01, WEIGHT(519)=2.73E-05,
WEIGHT(520)=2.46E-08, WEIGHT(476)=7.40E-06,
WEIGHT(502)=3.14E-05, WEIGHT(492)=2.11E-04,
WEIGHT(526)=1.79E-04, WEIGHT(493)=4.24E-02,
WEIGHT(494)=1.97E-03, WEIGHT(496)=1.07E-02,
WEIGHT(495)=1.80E-02, WEIGHT(499)=4.04E-05,
WEIGHT(504)=3.39E-05, WEIGHT(498)=4.06E-05,
WEIGHT(505)=5.73E-04, WEIGHT(500)=1.68E-04,
LEAD 14      11.35
1 SOURCE 1 1.2
NO LEAD-NO STRONTIUM (YITTRIUM)
&INPUT NEXT=1, IGEOM=7, NSHLD=2, JBUF=1, IPRNT=0,
OPTION=1, X=300.1, T(1)=0.1, DUNIT=1, SLTH=914.4, Y=457.4,
NTHETA=11, NPSI=9, DELR=0.01, T(2)=0.0,
WEIGHT(451)=6.44E-03, WEIGHT(472)=3.52E+00,
WEIGHT(027)=2.59E-04, WEIGHT(082)=0.00E+00,
WEIGHT(084)=0.00E+00, WEIGHT(103)=6.37E-04,
WEIGHT(102)=9.30E-04, WEIGHT(117)=8.02E-01,
WEIGHT(141)=9.52E-03, WEIGHT(172)=9.22E-01,
WEIGHT(170)=9.22E-01, WEIGHT(269)=5.59E-01,
WEIGHT(290)=9.03E-05, WEIGHT(319)=1.31E-01,
WEIGHT(335)=9.12E+01, WEIGHT(336)=9.65E+01,
WEIGHT(376)=4.49E+01, WEIGHT(377)=4.49E+01,
WEIGHT(388)=0.00E+00, WEIGHT(403)=1.78E-01,
WEIGHT(415)=1.20E-01, WEIGHT(519)=2.73E-05,
WEIGHT(520)=2.46E-08, WEIGHT(476)=7.40E-06,
WEIGHT(502)=3.14E-05, WEIGHT(492)=2.11E-04,
WEIGHT(526)=1.79E-04, WEIGHT(493)=4.24E-02,

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WEIGHT(494)=1.97E-03, WEIGHT(496)=1.07E-02,
WEIGHT(495)=1.80E-02, WEIGHT(499)=4.04E-05,
WEIGHT(504)=3.39E-05, WEIGHT(498)=4.06E-05,
WEIGHT(505)=5.73E-04, WEIGHT(500)=1.68E-04,
LEAD 14 11.35
1 SOURCE 1 1.2
WITH LEAD-NO STRONTIUM (YITTRIUM)
&INPUT NEXT=1, IGEOM=7, NSHLD=2, JBUF=1, IPRNT=0,
OPTION=1, X=300.1, T(1)=0.1, DUNIT=1, SLTH=914.4, Y=457.4,
NTHETA=11, NPSI=9, DELR=0.01, T(2)=0.762,
WEIGHT(451)=6.44E-03, WEIGHT(472)=3.52E+00,
WEIGHT(027)=2.59E-04, WEIGHT(082)=0.00E+00,
WEIGHT(084)=0.00E+00, WEIGHT(103)=6.37E-04,
WEIGHT(102)=9.30E-04, WEIGHT(117)=8.02E-01,
WEIGHT(141)=9.52E-03, WEIGHT(172)=9.22E-01,
WEIGHT(170)=9.22E-01, WEIGHT(269)=5.59E-01,
WEIGHT(290)=9.03E-05, WEIGHT(319)=1.31E-01,
WEIGHT(335)=9.12E+01, WEIGHT(336)=9.65E+01,
WEIGHT(376)=4.49E+01, WEIGHT(377)=4.49E+01,
WEIGHT(388)=0.00E+00, WEIGHT(403)=1.78E-01,
WEIGHT(415)=1.20E-01, WEIGHT(519)=2.73E-05,
WEIGHT(520)=2.46E-08, WEIGHT(476)=7.40E-06,
WEIGHT(502)=3.14E-05, WEIGHT(492)=2.11E-04,
WEIGHT(526)=1.79E-04, WEIGHT(493)=4.24E-02,
WEIGHT(494)=1.97E-03, WEIGHT(496)=1.07E-02,
WEIGHT(495)=1.80E-02, WEIGHT(499)=4.04E-05,
WEIGHT(504)=3.39E-05, WEIGHT(498)=4.06E-05,
WEIGHT(505)=5.73E-04, WEIGHT(500)=1.68E-04,
LEAD 14 11.35
1 SOURCE 1 1.2
&INPUT NEXT=6, &

```

APPENDIX C

CALCULATED GAMMA RADIATION DOSE RATE

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Calculations were performed to relate the gamma count rate data taken in the tank liquid observation wells (LOWs) to dose rates for each waste tank. The most recent gamma count rate data stored in the Computer Automated Surveillance System (Sept and Nov 1993) were converted to dose rate profiles using Equations C.1 and C.2. Equation C.1 converts the counts per second gamma data obtained from the Geiger-Muller detector to Grays per hour in the waste region only of the waste tanks. The best fit equation was calculated to be

$$Y = 2.46 \cdot 10^{-4} \cdot X - 1.55 \cdot 10^{-7} \cdot X^2 + 5.23 \cdot 10^{-11} \cdot X^3 - 6.14 \cdot 10^{-15} \cdot X^4 + 2.41 \cdot 10^{-19} \cdot X^5 \quad (C.1)$$

where

Y = Corrected gamma radiation dose rate, Gooley/h
X = Dose rate obtained from Geiger-Muller detector, Counts/s

and this equation is independent of height in the waste region. The dose rate above the waste in the air region can be estimated by Equation C.2

$$Y = -9.84 \cdot 10^{-5} \cdot D + Y_{\text{surface}} \quad (C.2)$$

where

Y = Correlated gamma radiation dose rate, Gooley/h
D = Distance from the surface of the waste, cm
Y_{surface} = Dose rate at the surface of the waste, Gooley/h

Equation C.2 will calculate the maximum exposure dose rate in the center air region of the tank. The actual exposure dose rate in the air region may be less than the calculated value due to interference from the wall or nearby instrumentations. Finally as the distance above the waste increases, Equation C.2 will give a negative number for the gamma radiation dose rate. In these cases, the gamma radiation dose rate should be taken to be zero.

The calculated standard deviation for Equation C.1 is ± 0.19 Gooley/h and ± 0.06 Gooley/h for Equation C.2. Equation C.2 was not corrected for gamma attenuation. The estimated uncertainty due to gamma attenuation was included in the calculated standard deviation. Furthermore, the actual waste surface can only be calculated within ± 2.54 cm due to uncertainties in the measurement and the actual length of the LOWs. The conversion from Gray to rad is 1 Gooley = 100 rad.

Equations C.1 and C.2 were developed for waste composed of ferrocyanide compounds. The majority of the radiation source in the ferrocyanide-bearing waste is ^{137}Cs . The conversion factors developed in Equation C.1 to convert count rate to dose rate depend on the type of radiation emitted, radiation energy spectrum, geometry conditions, and waste composition. Therefore, Equations C.1 and C.2 may not be valid for all the waste tanks which have different waste compositions from the ferrocyanide-bearing waste.

Gamma count rate information exist only for the waste tanks containing a LOW. Even though double-shell waste tanks contain LOWs, no gamma count rate measurements are performed on them because gamma count rate measurements are used only to detect leakage from the single-shell waste tanks (Hanlon 1994). The single-shell waste tanks containing a LOW are given in Table C-1 and their estimated dose rate as a function of height figures follow.

Table C-1. Waste Tanks with
Liquid Observation Wells.

Tank	Location
A-101	R-19
A-103	R-19
AX-101	R-9A
B-104	R-6
B-105	R-5
BX-111	R-5
BY-101	R-9A
BY-102	R-1
BY-103	R-10A
BY-104	R-10C
BY-105	R-10B
BY-106	R-10B
BY-107	R-7
BY-109	R-12B
BY-110	R-3
BY-111	R-1
BY-112	R-15
S-101	R-2
S-102	R-5
S-103	R-2
S-105	R-14
S-106	R-4
S-108	R-16
S-109	R-8
S-110	R-2
S-111	R-16
S-112	R-8
SX-101	R-14
SX-102	R-14
SX-103	R-14
SX-104	R-16
SX-105	R-16
SX-106	R-14
T-104	R-1
T-110	R-5
T-111	R-7
TX-102	R-11A
TX-105	R-9B
TX-106	R-11A

Table C-1. Waste Tanks with
Liquid Observation Wells.
(Continued)

Tank	Location
TX-108	R-9A
TX-109	R-3
TX-110	R-13B
TX-111	R-11B
TX-112	R-12A
TX-113	R-4
TX-114	R-7
TX-115	R-9A
TX-117	R-7
TX-118	R-5
TY-103	R-3
U-102	R-2
U-103	R-19
U-105	R-19
U-106	R-9
U-107	R-19
U-108	R-19
U-109	R-9
U-111	R-19

Figure C-1. A-101 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-2. A-103 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-3. AX-101 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-4. B-104 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-5. B-105 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad.

Figure C-6. BX-111 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-7. BY-101 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-8. BY-102 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-9. BY-103 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-10. BY-104 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-11. BY-105 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-12. BY-106 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-13. BY-107 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-14. BY-109 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-15. BY-110 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-16. BY-111 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-17. BY-112 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-18. S-101 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-19. S-102 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-20. S-103 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-21. S-105 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-22. S-106 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-23. S-108 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-24. S-109 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-25. S-110 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-26. S-111 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-27. S-112 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-28. SX-101 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-29. SX-102 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-30. SX-103 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-31. SX-104 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-32. SX-105 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-33. SX-106 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-34. T-104 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-35. T-110 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-36. T-111 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-37. TX-102 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-38. TX-105 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-39. TX-106 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-40. TX-108 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-41. TX-109 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-42. TX-110 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-43. TX-111 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-44. TX-112 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-45. TX-113 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-46. TX-114 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-47. TX-115 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-48. TX-117 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-49. TX-118 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-50. TY-103 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-51. U-102 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-52. U-103 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-53. U-105 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-54. U-106 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-55. U-107 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-56. U-108 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-57. U-109 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Figure C-58. U-111 Calculated Gamma Radiation Dose Rate.

NOTE: 1 Gy = 100 rad

Analysis of the exposure dose rate in each waste tank were performed. The maximum gamma radiation dose rate from all the waste tanks evaluated is 3.79 Gy/h and is located in BY-106. All values have an uncertainty standard deviation of ± 0.19 Gy/h. This uncertainty value does not consider material density variations within each waste tank. For comparison, the average and maximum gamma radiation dose rate for each waste tank are given in Table C-2. It is assumed that Equation B.1 is applicable to all of the waste tanks.

When the maximum exposure dose rate occurs at the bottom of the LOW, it is assumed that the exposure dose rate is constant from the bottom of the LOW to the surface bottom of the waste tank. This may not be the case and the exposure dose rate at the bottom of the waste tank may be much higher than the calculated value at the bottom of the LOW. The LOWs extend to within 1 to 58 cm of the bottom waste tank steel linear.

Table C-2. Exposure Dose Rate
in Each Waste Tank.
(Gy/h) March 1994

Tank	Average	Maximum
A-101	3.01	3.45
A-103	1.82	1.91
AX-101	2.76	3.71
B-104	0.11	0.13
B-105	0.05	0.18
BX-111	1.68	1.82
BY-101	0.28	0.87
BY-102	1.23	1.63
BY-103	0.99	1.63
BY-104	1.26	1.67
BY-105	0.84	1.69
BY-106	1.53	3.79
BY-107	1.49	2.18
BY-109	1.14	1.46
BY-110	1.13	2.08
BY-111	0.54	1.19
BY-112	0.45	1.61
S-101	1.65	2.04
S-102	1.81	2.01
S-103	2.02	2.99
S-105	0.13	0.53
S-106	1.35	1.92
S-108	0.89	2.02
S-109	1.24	2.39
S-110	1.69	2.03
S-111	1.54	1.94
S-112	1.25	2.54
SX-101	1.88	2.98
SX-102	1.82	2.43
SX-103	1.27	2.09
SX-104	2.07	2.80
SX-105	1.69	2.03
SX-106	1.74	3.59

Table C-2. Exposure Dose Rate
in Each Waste Tanks.
(Continued) March 1994

Tank	Average	Maximum
T-104	1.6×10^{-4}	9.8×10^{-4}
T-110	1.4×10^{-4}	9.8×10^{-4}
T-111	3.2×10^{-3}	0.01
TX-102	1.54	1.62
TX-105	1.08	1.96
TX-106	1.10	1.53
TX-108	0.18	0.52
TX-109	1.54	1.78
TX-110	1.11	2.03
TX-111	0.80	1.62
TX-112	0.64	1.68
TX-113	0.28	0.92
TX-114	0.42	1.60
TX-115	0.25	1.04
TX-117	0.15	0.94
TX-118	0.33	0.64
TY-103	0.52	1.43
U-102	2.19	3.45
U-103	2.01	3.29
U-105	1.99	3.57
U-106	2.15	2.53
U-107	1.75	2.70
U-108	1.94	3.38
U-109	2.02	3.71
U-111	2.47	3.52

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APPENDIX D

TOTAL INTEGRATED BETA AND GAMMA RADIATION DOSE ANALYTICAL SPREADSHEET MODEL

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An analytical spreadsheet model was developed to calculate the total integrated beta and gamma radiation doses that the ferrocyanide-bearing waste has been exposed to. The principal theory and equations behind the model are presented. The theory is that the total integrated gamma radiation dose can be calculated by

$$\Phi = \int_{t_b}^{t_f} C(t) dt \quad (D.1)$$

where

- $C(t)$ = Exposure dose rate as a function of time, Gy/h
- Φ = Total integrated gamma radiation dose to ferrocyanide waste, Gy
- t_b = Year when ferrocyanide-bearing waste was added to waste tank, yr
- t_f = Year the total integrated gamma radiation dose is desired, yr

If the exposure dose rate as a function of time was known, then solving Equation D.1 would be simple. However, the exposure dose rate as a function of time is not known. It depends on when the amount of radioactive nuclides were added and removed from the waste tanks. If some assumptions are made, then Equation D.1 can be simplified and solved. The following assumptions were made:

1. All the ferrocyanide-bearing waste was assumed to be added to a waste tank at the same time and has remained there.
2. The waste is homogeneous in composition, both vertically and horizontally, within the tanks. The waste composition is assumed to be the same for all of the tanks.
3. The radionuclides are proportionally added to the waste tanks corresponding to single pass reactor fuel from the B, D, DR, F, and H production reactor fuel having a burnup of 810 MWd/MT with 100 days in the reactor and 100 days decay.
4. Only the longest-lived fission products are accounted for in the calculations. No activation or actinide products are assumed to be present in the waste.
5. The waste volume is assumed constant during any given year and all addition and removal of waste occurs at the end of the given year.
6. The dose rate due to beta and gamma particles are taken to be equal. The energy per beta and gamma emitted by each isotope is also taken to be equal. All isotopes are assumed to be 100% gamma emitters while the correct percentage of beta particle yield for each isotope is used.

Taking these assumptions into consideration, the exposure dose rate can be simplified to

$$C(t) = \beta \sum_i^{NV} \mu_i \sum_j^{ISPE} A_j e^{-K_j(t-t_b)} \quad (D.2)$$

where

- B = Conversion factor constant, Gy/Ci
- μ = Weighted average of radioactivity addition or removal, Ci
- WV = Number of times waste level changed in waste tank over time
- A = Fraction of isotope production
- K = Decay constant, yr^{-1}
- ISPE = Number of radionuclides

and where the other symbols have already been defined. Therefore, the total integrated gamma radiation dose can be rewritten as

$$\Phi = \beta \int_{t_b}^{t_f} \sum_i^{WV} \mu_i \sum_j^{ISPE} A_j e^{-K_j(t-t_b)} dt \quad (D.3)$$

Historical records show that sometimes the tanks contained radioactive waste long before the ferrocyanide-bearing waste was added into the waste tanks, ($t_{in} \leq t_b$). Since this radioactive waste contributes to the total integrated gamma radiation dose of the ferrocyanide-bearing waste, it must be accounted for. Equation D.3 can be modified to account for this radiation dose by integrating over the lifetime of the radioactive waste in the tank and subtracting the integrated radiation accumulated before the ferrocyanide-bearing waste was added to the waste tanks

$$\Phi = \beta \int_{t_{in}}^{t_f} \sum_i^{WV} \mu_i \sum_j^{ISPE} A_j e^{-K_j(t-t_b)} dt - \beta \int_{t_{in}}^{t_b} \sum_i^{WV} \mu_i \sum_j^{ISPE} A_j e^{-K_j(t-t_b)} dt \quad (D.4)$$

where

- t_{in} = Year when waste storage began in the waste tank, yr

The analytical spreadsheet model developed used these equations and assumptions to calculate the total integrated gamma radiation dose for the ferrocyanide-bearing waste. If the simplification that the dose rate from beta and gamma particles are equal, then these equations can be applied also to calculate the total integrated beta radiation dose (Slade 1968). The analytical procedure employed in the spreadsheet model was:

1. Calculate or estimate the minimum, average, and maximum radiation dose rate for each waste tank (Appendix C).
2. Estimate the radionuclides generated (Appendix E).
3. Assume a 100% recovery process of actinides and their daughter products so that fission product dose rates dominate.
4. Utilize available historical waste transfer records (Anderson 1990).
5. Correct for radionuclide decay from when the radionuclides were added to the waste tanks to when the radiation dose rate experiments were performed, March 1994.
6. Calculate the total integrated beta and gamma radiation dose using Equations D.2 and D.4.

APPENDIX E

ORIGEN2: SINGLE-PASS REACTOR FUEL INPUT AND FISSION PRODUCT DATA

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The fuel was assumed to be B, D, DR, F, and H production reactor type fuel having a burnup of 810 Mwd/MT with 100 days in the reactor and with 100 days decay before being processed. A copy of the ORIGEN2 input file follows:

```

-1
-1
-1
BAS ONE METRIC TON OF SINGLE PASS FUEL AND CLADDING
RDA -1 = FRESH U FUEL
RDA -2 = INPUT COMPOSITION OF 1 MT FUEL IMPURITIES
RDA -3 = INPUT COMPOSITION OF 1 MT ALUMINUM CLADDING WITH IMPURITIES
RDA 1 = INPUT COMPOSITION OF 1 MT FUEL AND CLADDING WITH IMPURITIES
RDA 2 = DISCHARGE COMPOSITION OF 1 MT FUEL AND CLADDING AT 810 Mwd/MT
RDA 3 = COMBINED DISCHARGED ISOTOPES IN FUEL AND CLADDING DECAYED TO JAN 1994
RDA 4 = COMBINED DISCHARGED ISOTOPES IN FUEL AND CLADDING DECAYED TO JAN 1995
RDA 5 = COMBINED DISCHARGED ISOTOPES IN FUEL AND CLADDING DECAYED TO JAN 2000
RDA 6 = COMBINED DISCHARGED ISOTOPES IN FUEL AND CLADDING DECAYED TO JAN 2050
RDA 7 = COMBINED DISCHARGED ISOTOPES IN FUEL AND CLADDING DECAYED TO JAN 2100
RDA 8 = COMBINED DISCHARGED ISOTOPES IN FUEL AND CLADDING DECAYED TO JAN 2200
RDA WARNING: VECTORS ARE OFTEN CHANGED WITH RESPECT TO THEIR CONTENT.
RDA THESE CHANGES WILL BE NOTED ON RDA CARDS.
LIB 0 1 2 3 204 205 206 9 3 0 1 1
PHO 101 102 103 10
TIT INITIAL COMPOSITIONS OF UNIT AMOUNTS OF FUEL AND STRUCTURE MATERIALS
RDA READ FUEL COMPOSITION
INP -1 1 -1 -1 1 1
RDA READ FUEL COMPOSITION
INP -2 1 -1 -1 1 1
RDA READ ALUMINUM CLADDING WITH IMPURITIES (1.0 MT OF CLADDING)
INP -3 1 -1 -1 1 1
TIT IRRADIATION OF ONE METRIC TON OF DEPLETED FUEL
MOV -1 1 0 1.0
ADD -2 1 0 1.0
ADD -3 1 0 0.0502
HED 1 CHARGE
HED 2 DISCHG
HED 3 3 YR
HED 4 5 YR
HED 5 10 YR
HED 6 15 YR
HED 7 20 YR
HED 8 23 YR
HED 9 30 YR
HED 10 37 YR
HED 11 40 YR
HED 12 50 YR
BUP
IRP 20.0 8.10 1 -3 4 2 END OF THIS STEP=162 Mwd/MT
IRP 40.0 8.10 -3 -4 4 0 END OF THIS STEP=324 Mwd/MT
IRP 60.0 8.10 -4 -5 4 0 END OF THIS STEP=468 Mwd/MT
IRP 80.0 8.10 -5 -6 4 0 END OF THIS STEP=648 Mwd/MT
IRP 100.0 8.10 -6 2 4 0 END OF THIS STEP=810 Mwd/MT
BUP
OPTL 8 8 8 8 7 8 1 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTA 8 8 8 8 7 8 8 8 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
OPTF 8 8 8 8 7 8 1 8 7 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8 8
DEC 3.00 2 3 5 0 END OF THIS STEP 3 YR DECAY
DEC 5.00 3 4 5 0 END OF THIS STEP 5 YR DECAY
DEC 10.00 4 5 5 0 END OF THIS STEP 10 YR DECAY
DEC 15.00 5 6 5 0 END OF THIS STEP 15 YR DECAY
DEC 20.00 6 7 5 0 END OF THIS STEP 20 YR DECAY
DEC 23.00 7 8 5 0 END OF THIS STEP 23 YR DECAY
DEC 30.00 8 9 5 0 END OF THIS STEP 30 YR DECAY
DEC 37.00 9 10 5 0 END OF THIS STEP 37 YR DECAY
DEC 40.00 10 11 5 0 END OF THIS STEP 40 YR DECAY
DEC 50.00 11 12 5 0 END OF THIS STEP 50 YR DECAY
OUT 8 1 -1 0
STP 2
2 922340 0.0 922350 7120.0 922380 992880.0 0 0.0 FUEL ACTINIDES
0
4 010000 2.0 040000 10.0 050000 0.25 060000 650.0 FUEL IMPURITIES
4 070000 50.0 120000 25.0 130000 40.0 140000 140.0 FUEL IMPURITIES
4 240000 20.0 250000 25.0 260000 200.0 280000 50.0 FUEL IMPURITIES
4 290000 20.0 400000 10.0 480000 0.25 0 0.0 FUEL IMPURITIES

```

```

0
4 030000 80.0 050000 10.0 130000 1000000.0 140000 1700.0 CLAD IMPURITIES
4 260000 7000.0 270000 10.0 280000 13000.0 290000 1500.0 CLAD IMPURITIES
4 480000 30.0 0 0.0 CLAD IMPURITIES
0
END

```

The major gamma emitting fission product contributors were obtained from the output of the ORIGEN2 calculations. Eight isotopes make up 99% of the long-lived fission products. They were normalized and used in the calculations. Some of the nuclides were removed from the list to account for chemical reprocessing that took place. Activation products made up a very small fraction of the gamma emitting nuclides and may, therefore, be neglected. Records show that 99.9% of the actinides and their daughters were recovered and thus can be neglected (Jungfleisch 1984, Simpson 1993a, Simpson 1993b).

Table E-1. Fission Products.

Isotopes	Percentage of Total Isotope Production	Half-Life
⁹⁰ Sr	5.519	28.82 yr
¹⁰⁶ Ru	8.305	1.00 yr
¹²⁵ Sb	1.770	2.71 yr
¹³⁷ Cs	12.558	30.17 yr
¹⁴⁴ Ce	12.930	0.78 yr
¹⁴⁷ Pm	2.565	2.62 yr
¹⁵⁴ Eu	0.167	8.50 yr
¹⁵⁵ Eu	0.061	4.96 yr

The following daughter products were added to their parent fission products: ⁹⁰Sr-⁹⁰Y, ¹⁰⁶Ru-¹⁰⁶Rh, ¹²⁵Sb-^{125m}Te, ¹³⁷Cs-^{137m}Ba, and ¹⁴⁴Ce-¹⁴⁴Pr.

For simplicity, all the isotopes considered are assumed to be gamma emitters. However, the correct yield percentage of beta particles for each isotope is used in the calculations. The values assumed are shown in Table E-2 for the nine isotopes taken into consideration. These values also include the effects of their daughter products. Finally, the energy per beta and gamma particle emitted is taken to be equal (Browne et al. 1978).

Table E-2. Beta and Gamma Production.

Isotopes	Percentage of Total Beta	Percentage of Total Gamma
⁹⁰ Sr	100.0	100.0
¹⁰⁶ Ru	100.0	100.0
¹²⁵ Sb	76.9	100.0
¹³⁷ Cs	5.4	100.0
¹⁴⁴ Ce	98.6	100.0
¹⁴⁷ Pm	100.0	100.0
¹⁵⁴ Eu	100.0	100.0
¹⁵⁵ Eu	100.0	100.0

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