ENGINEERING CHANGE NOTICE

Page 1 of 2

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13a. Description of Change Full replacement of Revision 0 with Revision 1 document. NOTE: ACCORDING TO SECTION WP-6.7, REV. 0 OF WHC-CM-6-32, SAFETY ANALYSIS AND NUCLEAR ENGINEERING WORK PROCEDURES, CALCULATION NOTES ARE USED TO DOCUMENT THE ORIGINATOR'S ANALYSIS BUT ARE NOT TO BE USED TO AUTHORIZE ACTIVITIES OR JUSTIFY FACILITY MODIFICATIONS, OR CHANGES TO AN AUTHORIZATION BASIS, SAFETY BASIS, OR DESIGN BASIS.			
14a. Justification (mark one) Criteria Change ☒ Design Improvement ☐ Environmental ☐ Facility Deactivation ☐ As-Found ☐ Facilitate Const ☐ Const. Error/Omission ☐ Design Error/Omission ☐			
14b. Justification Details Minor errors discovered in the Revision 0 document have been corrected in the Revision 1 issue.			
15. Distribution (include name, MSIN, and no. of copies) See attached distribution list.

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Calculation Notes That Support Accident Scenario and Consequence Development for the Leak from a Railcar/Tank Trailer at the 204-AR Waste Unloading Facility

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Abstract: This document supports the development and presentation of the following accident scenario in the TWRS Final Safety Analysis Report:

Leak from Railcar/Tank Trailer.

The calculations needed to quantify the risk associated with this accident scenario are included within.

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**CALCULATION NOTES THAT SUPPORT ACCIDENT SCENARIO AND CONSEQUENCE DEVELOPMENT
FOR THE LEAK FROM A RAILCAR/TANK TRAILER AT THE 204-AR WASTE UNLOADING
FACILITY**

REVISION 1

**Tank Waste Remediation System Final Safety Analysis Report Project
Safety Analysis & Nuclear Engineering**

September 1996

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LIST OF TERMS

CFR	Code of Federal Regulations
DST	double shell tanks
FSAR	final safety analysis report
HEDOP	Hanford Environmental Dose Overview Panel
LSA	low specific activity
SARP	safety analysis report for packaging
Sv	sievert
TWRS	Tank Waste Remediation System
ULD	unit liter dose
UNH	uranyl nitrate hexahydrate
WHC	Westinghouse Hanford Company

LIST OF TRADEMARKS

BREMCALC	BREMCALC a computer program for calculating electron and positron Bremsstrahlung
MICROSHIELD	MICROSHIELD is a registered trademark of Grove Engineering, Inc., Rockville, MD
MICROSKYSHINE	MICROSKYSHINE is a registered trademark of Grove Engineering, Inc., Rockville, MD

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**CALCULATION NOTES THAT SUPPORT ACCIDENT SCENARIO AND CONSEQUENCE DEVELOPMENT
FOR THE LEAK FROM A RAILCAR/TANK TRAILER AT THE 204-AR WASTE UNLOADING
FACILITY**

1.0 INTRODUCTION AND PURPOSE

This document supports the development and presentation of the following accident scenario in the TWRS Final Safety Analysis Report:

Leak from Railcar/Tank Trailer.

The calculations needed to quantify the risk associated with this accident scenario are included in the following sections to aid in the understanding of this accident scenario:

Section 2.0 - Radiological Source Term Discussion and Development

Section 3.0 - Airborne Material Released From Accident Scenario

Section 4.0 - Calculated Radiological Doses

Section 5.0 - Calculated Toxicological Exposures

Each section provides information concerning methodology, assumptions, input data, calculations, results, conclusions, and references, if applicable.

The release to the maximum onsite receptor is assumed to last for 12 hours. The release to the maximum offsite receptor is assumed to last for 24 hours.

Information validation forms citing assumptions that were approved for use specifically in this analysis are included in Section 6.0. These forms are also on file with TWRS Project Files.

1.1 ACCIDENT SCENARIO DESCRIPTION

The hazards analysis performed for the TWRS FSAR identified hazardous conditions at the 204-AR Waste Unloading Facility that could involve the release of airborne and waterborne radiation contamination in the unloading room and directly outside the facility at the entrance to the unloading bay due to a catastrophic leak from a large liquid source such as a rail car or tank trailer.

Catastrophic leaks from both rail cars and tank trailers were evaluated to determine a bounding accident scenario at the 204-AR Waste Unloading Facility. It was found that the rail car maximizes the volume and severity of a catastrophic leak. This is because the waste transported by rail cars contains higher concentrations of radionuclides than the waste transported by tank trailers and a rail car can potentially carry 56,775 L (15,000 gal) more than a tank trailer. A maximum shipment of 75,700 L (20,000 gal) can be carried by the rail car whereas only 18,925 L (5,000 gal) can be transported in a tank trailer.

In order to maximize the material released to the environment, a catastrophic leak or spill inside the facility was found not to be bounding since the facility contains a passive barrier that would prevent the total volume of a catastrophic leak from leaving the facility. The passive barrier inside the 204-AR Waste Unloading Facility unloading bay consists of a subsurface catch tank and the pit that contains it. The catch tank and its pit have a combined holding capacity of approximately 17,030 L (4,500 gal).

Considering the discussion provided, the accident scenario that will be analyzed involves a rail car breach as the rail car is about to enter the 204-AR Waste Unloading Facility. The rail car is derailed, overturned, and breached, and its entire liquid waste contents are spilled outside onto the ground. The spilled liquid waste is assumed to pool on the surface near the 204-AR Waste Unloading Facility for a period of time and then the pool is absorbed by the sandy soil.

Other variations of this scenario were considered and found not to be credible. These included collision with another rail car containing liquid waste and a derailment and breach that involves a fire. There is currently only one rail car that is available to transfer waste to the facility and the waste is not flammable and is thermally stable (WHC 1995).

1.2 ACCIDENT FREQUENCY DEVELOPMENT

An annual frequency category of *unlikely* (10^{-2} to 10^{-4} per year) was qualitatively assigned to the railcar accident scenario. National and Hanford Site specific railroad accident statistics and calculations performed in WHC (1993), *Safety Analysis Report for Packaging, Railroad Liquid Waste Tank Car*, and in Kelly (1996), *204-AR Facility Rail Accident Analysis*, indicate that the frequency of railcar accidents resulting in large spills is less than 1×10^{-4} per year. Thus, the unlikely frequency category is considered appropriate and conservative for the railcar accident scenario.

The consequences associated with this accident scenario will be compared to the risk acceptance guidelines for *unlikely* accidents as provided in WHC-CM-4-46, Rev. 1.

1.3 REFERENCES

- Kelly, J. E, 1996, *204-AR Facility Rail Accident Analysis*, WHC-SD-WM-CN-021, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1991, *Safety Analysis Manual*, WHC-CM-4-46, Section 4.0, Rev. 1, November 15, 1991, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1993, *Safety Analysis Report for Packaging, Railroad Liquid Waste Tank Car*, SD-RE-SAP-013, Rev. 6, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1995, *Operating Specifications for the 204-AR Waste Unloading Facility*, OSD-T-151-00008, Rev. E-2, Westinghouse Hanford Company, Richland, Washington.

2.0 RADIOLOGICAL SOURCE TERM DISCUSSION AND DEVELOPMENT

2.1 PURPOSE

The purpose of this section is to provide a discussion of the development of the radiological source term that is most appropriate for use with the chosen accident scenario at the 204-AR Waste Unloading Facility.

2.2 ASSUMPTIONS/OVERVIEW AND BACKGROUND INFORMATION

Radioactive liquid wastes from Hanford Site facilities (i.e., T Plant, 222-S Laboratories, and the 340 Facility) are delivered to the 204-AR Waste Unloading Facility by either a railroad tank car (75,700 L [20,000 gal] capacity) or a tank trailer (18,925 L [5,000 gal] capacity). Transport of liquid waste is performed only by rail car from the 340 Facility and T-Plant whereas only a tank trailer is used to transfer liquid waste from the 222-S Laboratories. The liquid wastes shipped from these facilities are considered to be newly generated although in some cases they are produced as a result of analyzing and characterizing existing tank farm wastes through sampling activities.

Since these wastes are newly generated and some facilities have limited control over the composition or amount of liquid waste entering its system, the radionuclide inventory of each waste shipment to the 204-AR Waste Unloading Facility is variable.

The maximum volume of newly generated waste shipped to the 204-AR Waste Unloading Facility by rail car is administratively limited based on the results of the rail car transport safety analysis (WHC 1993). This is necessary to ensure that a postulated rail car accident occurring in the 300 Area does not result in onsite and offsite calculated doses exceeding risk acceptance guidelines. Before each shipment to the 204-AR, a maximum allowable shipment volume is calculated. In actual practice, most waste shipments to the 204-AR are far below the maximum allowable shipment volume for that particular waste.

For tank trailers, the shipment must be verified by the generator prior to shipping to be within the bulk low specific activity (LSA) shipping limit in accordance with 49 CFR 173.425 and the proper onsite shipping papers must be prepared in accordance with the requirements in WHC-CM-2-14, *Hazardous Material Packaging and Shipping*. Additional analysis provided in Nguyen (1994) and other applicable sections of 49 CFR 173 (i.e., 49 CFR 173.403 and 49 CFR 173.435) also include requirements for tank trailer shipments.

The newly generated waste shipped to the 204-AR Waste Unloading Facility from the 222-S Laboratories contains much lower concentrations of radionuclides than the waste shipped in rail cars from the 300 Area (i.e., 340 Facility). Considering this fact and the total volumes that can be shipped by both methods of transport, it is assumed that accident scenarios at the 204-AR Waste Unloading Facility involving rail cars bound similar accidents for tank trailers. The following discussion concentrates on rail car shipments.

2.3 METHODOLOGY/INPUT DATA

In order to perform accurate accident analysis for the 204-AR Waste Unloading Facility it is necessary to determine the bounding radionuclide inventory of the waste shipped and received.

The following 3 facts are known:

1. The radionuclide inventory of each waste shipment to the 204-AR Waste Unloading Facility is variable.
2. The waste shipped to the 204-AR Waste Unloading Facility is volume-limited based on the results of the rail car transport safety analysis (WHC 1993) which requires generators shipping by rail to calculate a maximum shipment volume based on the radionuclide content of the waste to be shipped.

It is important to note that even though a rail car shipment could be as large as 75,700 L (20,000 gal), the radiological consequences of a release this large would be no worse than those from a significantly smaller waste shipment. This is because of the administrative procedure to limit the volume of each waste shipment destined for the 204-AR Unloading Facility based on the specific radionuclide content of each shipment. Simply said, a 75,700 L (20,000 gal) shipment of highly radioactive liquid waste would not be permitted. This administrative control is imposed because of requirements established in the rail car transport safety analysis (WHC 1993).

3. In order to retain inventory control for the double-shell tank farms, a liquid waste with a unit liter dose of greater than 6.1×10^5 Sv/L (DST Liquids) cannot be received and transferred through the 204-AR Waste Unloading Facility since the final destination is the double-shell tank farms (i.e., 241-AP-106). The discussion of all ULD's in this section refer to the dose per unit liter inhaled.

Using these facts and applying the methodology presented in WHC (1993), a source term for the waste received at the 204-AR Waste Unloading Facility can be developed.

2.4 CALCULATIONS

First, it is assumed that a batch of DST Liquids will be shipped. To determine the maximum authorized shipment volume of this type of liquid, the same worksheet that is presented and utilized in WHC (1993) is used. Using concentration values for DST Liquids taken from WHC-SD-WM-SARR-037 (Cowley et al. 1996) and WHC-SD-WM-SARR-016 (Van Keuren 1996) and the SARP Maximum Shipment Size Worksheet (Figure 2-1), the maximum authorized shipping volume of DST Liquids is 2,123 L (561 gal).

Postulating that the maximum authorized shipping volume of 2,123 L (561 gal) will not result in maximized dose consequences from the spilling, pooling and resuspension of wetted soil (due to the difference in spilled volume, pool size and wetted soil area), a ULD is also determined for an

unknown liquid that would correlate to the maximum authorized shipping volume of 75,700 L (20,000 gal).

Using the SARP Maximum Shipment Size Worksheet, the ULD of a liquid waste that would allow a shipment of 75,700 L (20,000 gal) can be calculated.

From the Worksheet, if $V_{\max} = 20,000 \text{ gal}$, then $n = 25,000 \text{ gal}$.

Therefore, using the same heel reduction factor ($h = 0.80$) and release fraction ($R = 0.0017^1$),

$$\text{Total AB} = 0.80 / [(25,000 \text{ gal})(0.0017)]$$

$$\text{Total AB} = 0.019/\text{gal}$$

Setting up a simple ratio between the ULD that is known and the known Total AB values gives:

$$\text{Total AB}_{\text{DST Liq.}} / \text{ULD}_{\text{DST Liq.}} = \text{Total AB}_{\text{Unknown Liq.}} / \text{ULD}_{\text{Unknown Liq.}}$$

$$(0.67/\text{gal}) / 6.1 \times 10^3 \text{ Sv/L} = (0.019/\text{gal}) / X \text{ Sv/L}$$

$$X \text{ Sv/L} = 1.7 \times 10^2 \text{ Sv/L}$$

The ULD associated with the shipment of 75,700 L (20,000 gal) is reduced by approximately 36 times when compared to the ULD for DST Liquids.

$$(6.1 \times 10^3 \text{ Sv/L}) / (1.7 \times 10^2 \text{ Sv/L}) = 35.88$$

Reducing the concentrations of each radionuclide included in the SARP Maximum Shipment Size Worksheet by the same factor gives the concentration values shown in Table 2-1. These concentrations will be used to calculate the direct radiation and skyshine doses as a result of exposure to a pool-type release.

A completed SARP worksheet with these revised values is shown in Figure 2-2.

2.5 RESULTS/CONCLUSIONS

The ULD for DST Liquids ($6.1 \times 10^3 \text{ Sv/L}$) correlating to a maximum authorized shipment of 2,123 L (561 gal) will be used to evaluate the dose consequences of a spill, pool, and resuspension release. Additionally, the ULD that was developed here ($1.7 \times 10^2 \text{ Sv/L}$) for a maximum authorized shipment of 75,700 L (20,000 gal) will also be used to evaluate a similar spill, pool, and resuspension release in Sections 3.0 and 4.0. The differences here would be the amount of the spill, the size of the pool, and the area of wetted soil as a result of the release.

¹Although no fire is postulated to occur in this accident scenario at the 204-AR Waste Unloading Facility, the release fraction with fire is utilized in the Maximum Shipment Size Worksheet to ensure a conservatively limited shipment volume.

2.6 REFERENCES

- Cowley, W. L., 1996, *Development of Radiological Concentrations and Unit Liter Doses for TWRS FSAR Radiological Consequence Calculations*, WHC-SD-WM-SARR-037, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- Nguyen, P. M., 1994, *Safety Evaluation for Packaging for the 222-S Laboratory Cargo Tank*, WHC-SD-TP-SEP-011, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- Van Keuren, J. C., 1996, WHC-SD-WM-SARR-016, Rev. 2, *Tank Waste Compositions and Atmospheric Dispersion Coefficients for use in ASA Consequence Assessments*, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1993, *Safety Analysis Report for Packaging, Railroad Liquid Waste Tank Car*, SD-RE-SAP-013, Rev. 6, Westinghouse Hanford Company, Richland, Washington.
- 49 CFR 173.403, Subpart I, "Definition," Title 49, *Code of Federal Regulations*, Part 173.403, Subpart I, as amended, U.S. Department of Transportation, Washington, D.C.
- 49 CFR 173.425, Subpart I, "Transport Requirements for Low Specific Activity (LSA) Radioactive Materials," Title 49, *Code of Federal Regulations*, Part 173.425, Subpart I, as amended, U.S. Department of Transportation, Washington, D.C.
- 49 CFR 173.435, Subpart I, "Table of A_1 and A_2 Values for Radionuclides," Title 49, *Code of Federal Regulations*, Part 173.435, Subpart I, as amended, U.S. Department of Transportation, Washington, D.C.
- WHC-CM-2-14, *Hazardous Material Packaging and Shipping Manual*, Westinghouse Hanford Company, Richland, Washington.

Figure 2-1. Using The Same Methodology Presented in SD-RE-SAP-013, Rev. 6, *Safety Analysis for Packaging, Railroad Liquid Waste Tank Car*, A Maximum Shipment Size of Liquid Waste with Radionuclide Concentrations Identical to Those of DST Liquids is Calculated to be 2,123 L (561 gal).

Maximum Shipment Size Worksheet

Date: March 13, 1996

Tank car designation: #10H-18581

Heel reduction factor (from Heel Reduction Factor Worksheet): $h = 0.80$ Limits based on doses to [x] onsite worker
[] member of publicVolume Unit: gallons

Nuclide	A Concentration (Ci/unit vol.)		B (D/L)	A x B
¹⁴ C	2.35 E-05	Ci/gal	NOT IN SARP	NOT IN SARP
⁶⁰ Co	7.15 E-04	Ci/gal	3.0 E-02	2.15 E-05
⁷⁹ Se	n/a	n/a	n/a	n/a
⁹⁰ Sr	4.70 E-01	Ci/gal	3.5 E-02	1.65 E-02
⁹⁰ Y	4.70 E-01	Ci/gal	INCL. IN Sr-90	INCL. IN Sr-90
⁹⁹ Tc	1.12 E-03	Ci/gal	1.4 E-03	1.57 E-06
¹⁰⁶ Ru	n/a	n/a	n/a	n/a
¹²⁵ Sb	n/a	n/a	n/a	n/a
¹²⁹ I	2.04 E-06	Ci/gal	NOT IN SARP	NOT IN SARP
¹³⁴ Cs	6.23 E-04	Ci/gal	6.3 E-03	3.93 E-06
¹³⁷ Cs	6.03 E+00	Ci/gal	4.1 E-03	2.47 E-02
¹⁴⁴ Ce	n/a	n/a	n/a	n/a
¹⁴⁷ Pm	3.68 E-03	Ci/gal	5.8 E-03	2.13 E-05
¹⁵⁴ Eu	4.29 E-03	Ci/gal	4.1 E-02	1.76 E-04
¹⁵⁵ Eu	n/a	n/a	n/a	n/a
²³⁷ Np	2.35 E-05	Ci/gal	NOT IN SARP	NOT IN SARP
²³⁸ Pu	1.84 E-04	Ci/gal	1.1 E+02	2.02 E-02
^{239/240} Pu	7.87 E-04	Ci/gal	2.4 E+02	1.89 E-01
²⁴¹ Pu	1.84 E-03	Ci/gal	2.4 E+00	4.41 E-03
²⁴¹ Am	3.47 E-03	Ci/gal	1.2 E+02	4.17 E-01
²⁴² Cm	1.12 E-09	Ci/gal	2.6 E+00	2.92 E-09
²⁴⁴ Cm	1.23 E-05	Ci/gal	6.3 E+01	7.73 E-04

(Attach additional sheet if necessary)

Total AB = 6.7 E-01.

Release fraction: $R = [x]$ with fire $\rightarrow 0.0017$
[] without fire $\rightarrow 0.00037$ [] other $\rightarrow$ _____
Specify authorizing document:Maximum shipping volume: $n = \frac{h}{R \times (\text{Total AB})} = \underline{702 \text{ gallons}}$ Authorized shipping volume: $V_{\max} = 702 / 1.25 = \underline{561 \text{ gal (2,123 L)}}$ Performed by: SAMPLE _____Checked by: SAMPLE _____

printed name signature date

See Worksheet Notes on next page.

WORKSHEET NOTES (FIGURE 2-1):

1. **Column A** in the table is the concentration of each radionuclide in the source term in units of Ci/gal. For the radionuclides where no available data exists, these are listed as n/a. These radionuclides that are listed as n/a have a negligible impact on the dose calculations due to their low activity concentrations.
2. **Column B** in the table is the dose limiting factor taken from Table 5 of the SARP (WHC 1993). The units of this column is (1/Ci) since $D = \text{rem/Ci}$ and $L = \text{rem}$. A few dose limiting factors are listed as "NOT IN SARP" since values for these radionuclides are not listed in Table 5 of the SARP (WHC 1993). The absence of these values is considered to have a negligible effect on the final results of this worksheet.
3. **Column A x B** in the table is the multiplication of Columns A and B. A sum of these values (Total AB) is calculated at the bottom of Column C. The units of this column is (1/gal).
4. Total AB is rounded to 2 significant digits.
5. The Heel Reduction Factor, h , is calculated by a worksheet also presented in the SARP (WHC 1993). A conservative heel reduction factor of 0.8 is chosen for use in this worksheet. This is the maximum heel reduction factor achievable based on a total of 16 grams of plutonium and the most conservative release fraction. This can be seen from the graph on page 57 of the SARP (WHC 1993).
6. It is interesting to note that the only radionuclides from the chosen source term (DST Liquids) that contribute significantly to the development of Total AB are the following: ^{90}Sr , ^{137}Cs , ^{238}Pu , $^{239/240}\text{Pu}$, and ^{241}Am .

Table 2-1. The Concentrations of Radionuclides Making Up a Shipment of 75,700 L (20,000 gal) with an Inhalation Unit Liter Dose of 1.7×10^2 Sv/L are Shown.

Radionuclide	Concentration (Ci/gal)
C-14	6.55 E-07
Co-60	1.99 E-05
Se-79	n/a
Sr-90	1.31 E-02
Y-90	1.31 E-02
Tc-99	3.13 E-05
Ru-106	n/a
Sb-125	n/a
I-129	5.70 E-08
Cs-134	1.74 E-05
Cs-137	1.68 E-01
Ce-144	n/a
Pm-147	1.03 E-04
Eu-154	1.20 E-04
Eu-155	n/a
Np-237	6.55 E-07
Pu-238	5.13 E-06
Pu-239/240	2.19 E-05
Pu-241	5.13 E-05
Am-241	9.68 E-05
Cm-242	3.13 E-11
Cm-244	3.42 E-07

n/a - No available data. These radionuclides have a negligible impact on the dose calculations due to their low activity concentrations.

Figure 2-2. The Unit Liter Dose Back-Calculated From a Maximum Authorized Shipment Volume of 75,700 L (20,000 gal) is Found to be 1.7×10^{-2} Sv/L Inhaled.

Maximum Shipment Size Worksheet

Date: March 13, 1996

Tank car designation: #10H-18581

Heel reduction factor (from Heel Reduction Factor Worksheet): $h = 0.80$ Limits based on doses to [x] onsite worker
[] member of publicVolume Unit: gallons

Nuclide	A Concentration (Ci/unit vol.)		B (D/L)	A x B
¹⁴ C	6.55 E-07	Ci/gal	NOT IN SARP	NOT IN SARP
⁶⁰ Co	1.99 E-05	Ci/gal	3.0 E-02	5.98 E-07
⁷⁹ Se	n/a	n/a	n/a	n/a
⁹⁰ Sr	1.31 E-02	Ci/gal	3.5 E-02	4.59 E-04
⁹⁰ Y	1.31 E-02	Ci/gal	INCL. IN Sr-90	INCL. IN Sr-90
⁹⁹ Tc	3.13 E-05	Ci/gal	1.4 E-03	4.39 E-08
¹⁰⁶ Ru	n/a	n/a	n/a	n/a
¹²⁵ Sb	n/a	n/a	n/a	n/a
¹²⁹ I	5.70 E-08	Ci/gal	NOT IN SARP	NOT IN SARP
¹³⁴ Cs	1.74 E-05	Ci/gal	6.3 E-03	1.09 E-07
¹³⁷ Cs	1.68 E-01	Ci/gal	4.1 E-03	6.89 E-04
¹⁴⁴ Ce	n/a	n/a	n/a	n/a
¹⁴⁷ Pm	1.03 E-04	Ci/gal	5.8 E-03	5.95 E-07
¹⁵⁴ Eu	1.20 E-04	Ci/gal	4.1 E-02	4.90 E-06
¹⁵⁵ Eu	n/a	n/a	n/a	n/a
²³⁷ Np	6.55 E-07	Ci/gal	NOT IN SARP	NOT IN SARP
²³⁸ Pu	5.13 E-06	Ci/gal	1.1 E+02	5.64 E-04
^{239/240} Pu	2.19 E-05	Ci/gal	2.4 E+02	5.26 E-03
²⁴¹ Pu	5.13 E-05	Ci/gal	2.4 E+00	1.23 E-04
²⁴¹ Am	9.68 E-05	Ci/gal	1.2 E+02	1.16 E-02
²⁴² Cm	3.13 E-11	Ci/gal	2.6 E+00	8.15 E-11
²⁴⁴ Cm	3.42 E-07	Ci/gal	6.3 E+01	2.15 E-05

(Attach additional sheet if necessary)

Total AB = $1.9 \text{ E-}02$.Release fraction: $R = [x]$ with fire $\rightarrow 0.0017$
[] without fire $\rightarrow 0.00037$ [] other $\rightarrow$ _____
Specify authorizing document:Maximum shipping volume: $n = \frac{h}{R \times (\text{Total AB})} = \underline{25,000 \text{ gallons}}$ Authorized shipping volume: $V_{\max} = 25,000 / 1.25 = \underline{20,000 \text{ gal (75,700 L)}}$ Performed by: SAMPLE _____Checked by: SAMPLE _____
printed name signature date

See Worksheet Notes on next page.

WORKSHEET NOTES (FIGURE 2-2):

1. Column A in the table is the concentration of each radionuclide in the source term in units of Ci/gal. For the radionuclides where no available data exists, these are listed as n/a. These radionuclides that are listed as n/a have a negligible impact on the dose calculations due to their low activity concentrations.
2. Column B in the table is the dose limiting factor taken from Table 5 of the SARP (WHC 1993). The units of this column is (1/Ci) since $D = \text{rem/Ci}$ and $L = \text{rem}$. A few dose limiting factors are listed as "NOT IN SARP" since values for these radionuclides are not listed in Table 5 of the SARP (WHC 1993). The absence of these values is considered to have a negligible effect on the final results of this worksheet.
3. Column A x B in the table is the multiplication of Columns A and B. A sum of these values (Total AB) is calculated at the bottom of Column C. The units of this column is (1/gal).
4. Total AB is rounded to 2 significant digits.
5. The Heel Reduction Factor, h, is calculated by a worksheet also presented in the SARP (WHC 1993). A conservative heel reduction factor of 0.8 is chosen for use in this worksheet. This is the maximum heel reduction factor achievable based on a total of 16 grams of plutonium and the most conservative release fraction. This can be seen from the graph on page 57 of the SARP (WHC 1993).
6. It is interesting to note that the only radionuclides from the chosen source term that contribute significantly to the development of Total AB are the following: ^{90}Sr , ^{137}Cs , ^{238}Pu , $^{239/240}\text{Pu}$, and ^{241}Am .

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3.0 AIRBORNE MATERIAL RELEASED FROM ACCIDENT SCENARIO

3.1 PURPOSE

The purpose of this section is to provide the calculations related to the respirable quantity released and resuspended from the spilling liquid, the resulting pool, and the wetted soil due to the accident scenario at the 204-AR Waste Unloading Facility.

3.2 METHODOLOGY/ASSUMPTIONS/INPUT DATA/CALCULATIONS

3.2.1 Development of the Release Amounts Using the Allowable Volume 2,123 L (561 gal) of DST Liquids with an Inhalation ULD of 6.1×10^3 Sv/L

The total material at risk in this unmitigated accident scenario is a rail car containing 2,123 L (561 gal) of liquid with an inhalation unit liter dose of 6.1×10^3 Sv/L. This material is released from the rail car after the rail car has been derailed, breached, and is laying on its side near the 204-AR Waste Unloading Facility. The breach or break in the rail car is assumed to occur at a location that allows a total release of all liquid. As the liquid escapes it experiences free-fall motion and splashing as it hits the soil surface (and then creates a pool).

Since it's unlikely that all the liquid waste would be released if the breach point was not near the bottom of the rail car (as it is laying on its side), the analysis performed uses a spill height of 1 m (3.3 ft) to ensure that dose consequences are calculated conservatively.

There are three release fractions that are considered in this scenario:

- the suspension of liquid from falling and splashing,
- the evaporation/resuspension of liquid from the surface pool created on the soil, and
- the resuspension of liquid from the wetted soil that made up the pool.

The quantity released from the event is determined by adding the fractions released from the spill immediately following the rail car breach, the evaporation/resuspension of liquid from the pool that exists for 2 hours, and the subsequent resuspension of liquid waste from the wetted soil for 10 hours or 22 hours. The 10 hour values are used to evaluate consequences to the onsite receptor and the 22 hour values are used to determine the offsite receptor consequences.

Release fractions for the free-fall spill of liquids onto a hard, unyielding surface are included in Section 3.2.3 of Mishima (1994). Values are provided for various solutions falling distances of either 1 m (3.3 ft) or 3 m (9.8 ft). The bounding release fraction that is chosen for use in this accident scenario is 2.0×10^{-5} and is based on an aqueous solution falling a distance of 1 m (3.3 ft) [Mishima (1994), Table 3-6]. This is considered to be an appropriate value since it is unlikely that liquid waste would be falling a distance of 3 m (9.8 ft); the diameter of the rail car tank is only

2.7 m (9 ft). To ensure that conservative radiological dose consequences are calculated, all of the liquid contained by the rail car is subjected to the chosen release fraction. The amount released is calculated:

$$(\text{Material At Risk}) \times (\text{Release Fraction}) = \text{Material Released Due to Spill}$$

$$(2,123 \text{ L}) \times (2.0 \times 10^{-5}) = 4.3 \times 10^{-2} \text{ L}$$

$$Q_{\text{spill}} = 4.3 \times 10^{-2} \text{ L}$$

The depth of the surface pool created by the catastrophic leak is assumed to be 2.5 cm (1 in.). Using this depth and a volume of surface run off of 2.1 m³ (561 gal.) yields a pool area of 84 m². Assuming a circular pool yields a radius of 5.2 m.

Airborne release fractions (and associated discussion) for a variety of pool releases are provided on pages 3-37, and 3-39 to 3-42 of Mishima (1994). For cases resembling a pond, even if the depth is shallow, Mishima suggests using

$$C_{\text{air}} = C_{\text{water}} (3 \times 10^{-11})(10^b)$$

Where:

- C_{air} = Concentration in the air just above the pool, g/m³
- C_{water} = Concentration in the water, g/m³
- $b = 0.098 v$
- v = Wind speed, m/s.

This equation was developed for use in cases when the fetch (i.e., the length of the pool in the direction of the wind) is infinite. For small fetch lengths, Appendix 3A shows the following:

$$Q = 2.8 \times 10^{-9}(F) \quad \text{L/m}^2\text{-hr}$$

Where:

F = Pond length assuming a square pond.

Q = Mass suspension rate per unit area of pool.

For the case where the liquid has seeped into the soil, Mishima suggests using the values from suspension from soil. The results of two studies are provided in Mishima (1994) for this type of release mechanism. One study evaluated the resuspension of uranyl nitrate hexahydrate (UNH) solutions from soil measured in wind tunnels and showed a resuspension rate of 4×10^{-6} /hr when averaged over 24 hours for 1 m/s winds (page 3-41 [Mishima 1994]). Mishima (1994), page 3-41, provides larger resuspension rates for greater wind speeds but the value of the atmospheric dispersion factor, χ/Q , decreases as the wind speed increases. As a result, the largest doses occur under low wind speeds. Appendix 3A shows the relationship for liquid pools.

The second study referenced on page 3-40 of Mishima (1994) measured the resuspension rates of powders deposited on a soil surface. Values for this experiment ranged between 4×10^{-5} /hr and 4×10^{-7} /hr. Relating this information to the resuspension of liquids, the statement is made that liquids are significantly less susceptible to entrainment than powders. However, once the liquid is absorbed by the soil, the release is easier as the soil itself is more easily resuspended. Because of this aspect and the fact that the first study contains conditions almost identical to those of the accident

scenario analyzed here, a value of 4×10^{-6} /hr over a 24-hour duration will be used in the analysis.

When using this airborne release rate, it is assumed that none of the mass that seeped into the soil is protected from the release. The first study referenced in Mishima (1994), Mishima and Schwendiman (1973) indicates a plot of quantity collected versus time. It is seen from that plot that approximately 90% of the total amount collected is collected over 12 hours (defined as one operational shift). This results in an airborne release rate of 7.2×10^{-6} /hr. At 22 hours essentially 100% of the release is completed. This equals a value of 4.4×10^{-6} /hr. The plot on page 32 of Mishima and Schwendiman (1973) has been reproduced and detailed in Figure 3-1 to show these derived airborne release rates.

The release over two hours is conservatively calculated based on the assumption that the entire pool is present over the entire two hours with saturated soil for 22 h. The length of a square pool having the same area as a circular pool with a 5.2 m radius is 9.2 m. The release is:

$$Q_{\text{pool}} = (2.8 \times 10^{-9} [9.2]) \text{ L/m}^2\text{-h} \times (84 \text{ m}^2)(2 \text{ hr}) = 4.3 \times 10^{-6} \text{ L}$$

The release from the wetted soil over 22 hours is:

$$Q_{\text{soil}} = (4.4 \times 10^{-6}/\text{hr})(2,123 \text{ L})(22 \text{ hr}) = 0.21 \text{ L.}$$

The total airborne radiological release over 24 hours (determined for the offsite consequences) is primarily that from the spill and the wetted soil (i.e., 0.25 liters). Q_{pool} is insignificant. All of this airborne radiological release is assumed to be respirable (i.e., respirable fraction equals 1).

The release from the wetted soil over 10 hours is:

$$Q_{\text{soil}} = (7.2 \times 10^{-6}/\text{hr})(2,123 \text{ L})(10 \text{ hr}) = 0.15 \text{ L.}$$

The airborne release rate of 7.2×10^{-6} /hr is based on an assumed straight line approximation of total release per hour from 0 to 12 hr (90% of the total is released in 12 hours yielding 7.2×10^{-6} /hr, as discussed above).

The total airborne radiological release over 12 hours (determined for the onsite consequences) is primarily that from the spill and the wetted soil (i.e., 0.19 liters). Q_{pool} is also insignificant in this case. All of this airborne radiological release is assumed to be respirable (i.e., respirable fraction equals 1).

Credit is not taken for further seepage into the soil (and therefore less is available for release from the saturated soil) over the course of long term release times (up to 22 hours).

A variation of this accident scenario was also investigated. In the variation it was assumed that the accident occurred during the winter season when the soil surface would be frozen (i.e., all spilled liquid remaining in a shallow pool for duration of release). It is clearly evident from the results shown above that this variation would be grossly non-conservative since the bulk of the airborne liquid release in this type of accident scenario comes from the wetted soil.

3.2.1.1 Results. The release from the initial spill is $Q_{\text{spill}} = 4.3 \times 10^{-2}$ L.

The release from the pool over 2 hours is $Q_{\text{pool}} = 4.3 \times 10^{-6}$ L.

The release from the wetted soil over the next 10 hours is $Q_{\text{soil}} = 0.15$ L.

Additionally, the amount of waste released during the period of 22 hours after the first 2 hours is $Q_{\text{soil}} = 0.21$ L.

3.2.2 Development of the Release Amounts Using the Allowable Volume 75,700 L (20,000 gal) of Liquid Waste with an Inhalation ULD of 1.7×10^2 Sv/L

Using the same methodology presented in the Section 3.1.1, a spill volume of 75,700 L (20,000 gal) is analyzed. A spill of this magnitude creates a pool area of 3,028 m² assuming a pool depth of 0.025 m. Assuming a circular pool produces a radius of 31.0 m.

3.2.2.1 Results. The release from the initial spill is $Q_{\text{spill}} = 1.5$ L.

The release from the pool over 2 hours is $Q_{\text{pool}} = 9.3 \times 10^{-4}$ L.

The release from the wetted soil over the next 10 hours is $Q_{\text{soil}} = 5.5$ L.

Additionally, the amount of waste released during the period of 22 hours after the first 2 hours is $Q_{\text{soil}} = 7.3$ L.

3.3 CONCLUSIONS

The radiological consequences will be calculated in Section 4.0 for both pool size releases shown in this section.

Case 1 - $Q_{\text{spill}} = 4.3 \times 10^{-2}$ L.
 $Q_{\text{pool}} = 4.3 \times 10^{-6}$ L, 2 hour release.
 $Q_{\text{soil}} = 0.15$ L, 10 hour release (onsite).
 $Q_{\text{soil}} = 0.21$ L, 22 hour release (offsite).

Case 2 - $Q_{\text{spill}} = 1.5$ L.
 $Q_{\text{pool}} = 9.3 \times 10^{-4}$ L, 2 hour release.
 $Q_{\text{soil}} = 5.5$ L, 10 hour release (onsite).
 $Q_{\text{soil}} = 7.3$ L, 22 hour release (offsite).

3.4 REFERENCES

- Mishima, J., 1994, *Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, DOE-HDBK-3010-94, U.S. Department of Energy, Washington, D.C.
- Mishima, J. and L. C. Schwendiman, 1973, *Some Experimental Measurements of Airborne Uranium (Representing Plutonium) in Transportation Accidents*, BNWL-1732, August 1973, Battelle Pacific Northwest Laboratories, Richland, Washington.

Robyler & Owczarski, 1992, *Determining Wind Spray Release from Effluent Disposal Ponds Using AIRSOURC Model*, WHC-SA-1378-FP, Westinghouse Hanford Company, Richland, Washington.

Figure 3-1. The Airborne Release Rate Over 12 hours is Interpolated from the Graph to be $7.2 \times 10^{-6}/\text{hr}$.

FIGURE 14. AERODYNAMIC ENTRAINMENT OF URANIUM DIOXIDE POWDER FROM SMOOTH, SANDY SOIL AT AN AIR VELOCITY OF 2.5 MPH

Figure excerpted from page 32 of:

Mishima, J. and L. C. Schwendiman, 1973, Some Experimental Measurements of Airborne Uranium (Representing Plutonium) in Transportation Accidents, BNWL-1732, August 1973, Battelle Pacific Northwest Laboratories, Richland, Washington.

APPENDIX 3A
MASS RELEASE RATES FROM POOLS

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Mishima (1994) provides the concentration above a pool due to the transport of liquid from the surface due to action of the wind on the surface. For the case of infinite fetch:

$$C_{\text{air}} = C_{\text{water}}(3.0 \times 10^{-11})(10^b)$$

Where:

$$\begin{aligned} C_{\text{air}} &= \text{g, liquid/m}^3, \text{ air just above the pool} \\ C_{\text{water}} &= \text{g, liquid/m}^3, \text{ liquid} \\ b &= 0.098 v \\ v &= \text{wind speed, m/s.} \end{aligned}$$

For smaller pools, Roblyer and Owczarski (1992) must be used as the fetch is not infinite. The numerical method used in the reference results in a graph of mass release rate as a function of fetch and wind speed. The graph shows a linear relationship of mass release rate to fetch and an exponential relationship to wind speed.

Using the graph, the following is obtained:

$$MR = K(F)(u)$$

Where:

$$\begin{aligned} MR &= \text{mass release rate, kg/m}^2 - \text{s} \\ F &= \text{fetch, m} \\ u &= \text{wind speed m/s} \end{aligned}$$

The values of "K" are:

u	K
10 m/s	1.35 E-12
5 m/s	6 E-13
1 m/s	1 E-12

The dose rate is a function of "1/u". The dose rate for each wind speed can be approximated by K/u.

u	Dose rate, K/u
10 m/s	1.35 E-13
5 m/s	1.2 E-13
1 m/s	1 E-12

Since the atmosphere stability is not "very stable" (i.e., Pasquill F) at 10 m/s, the dose is maximized when a wind speed of 1 m/s is used. Therefore, for releases from a pool:

$$MR = 1 \times 10^{-12} (F) \frac{kg}{m^2-s}$$

$$= 3.6 \times 10^{-9} (F) \frac{kg}{m^2-h}$$

Where the units of "F" is meters. Recalling that the density of the waste is a maximum of 1.3 kg/L, the volume release rate (VR) equals,

$$VR = 2.8 \times 10^{-9} (F) \frac{L}{m^2-h}$$

It should be noted that the use of a smaller waste density will marginally increase the volume release rate. Overall dose consequence results will be unaffected since the Q_{pool} value can be regarded as insignificant when compared to the amounts released from both the spill and the wetted soil.

4.0 CALCULATED RADIOLOGICAL DOSES

4.1 PURPOSE

The purpose of this section is to provide the radiological dose calculations related to the respirable quantity released and resuspended from the two cases analyzed in Section 3.0.

4.2 METHODOLOGY

The methodology that is used to calculate radiological dose consequences is documented in WHC-SD-WM-SARR-016, Rev. 2, *Tank Waste Compositions and Atmospheric Dispersion Coefficients for use in ASA Consequence Assessments* and WHC-SD-WM-SARR-037, Rev. 0, *Development of Radiological Concentrations and Unit Liter Doses for TWRS FSAR Radiological Consequence Calculations*.

4.3 ASSUMPTIONS

No assumptions have been made in this section.

4.4 INPUT DATA

Unit Liter Doses (ULDs)

The ULDs for this analysis are taken from WHC-SD-WM-SARR-037, Rev. 0, *Development of Radiological Concentrations and Unit Liter Doses for TWRS FSAR Radiological Consequence Calculations*.

Case 1 - $ULD_H = 6.1 \times 10^3$ Sv/L (inhalation dose)
 $ULD_I = 0.068$ Sv-m³/s-L (ingestion dose)

Case 2 - $ULD_H = 1.7 \times 10^2$ Sv/L (inhalation dose)
 $ULD_I = 0.0019$ Sv-m³/s-L (ingestion dose) - reducing the ingestion dose of DST Liquids by a factor of 35.88.

Dispersion Coefficients (χ/Q' 's)

Onsite - The χ/Q' for the 2 hour release is 3.41×10^{-2} s/m³ (Van Keuren 1996a).

To determine the χ/Q' for the 10 hour release, the logarithmic interpolation procedure described in Van Keuren (1996a) is used to correctly identify the proper χ/Q' 's to be used. Using the equation from Van Keuren (1996a) yields:

$$\frac{\log(1.13 \times 10^{-2}) - \log(\chi/Q' \text{ 10 hr})}{\log(1.13 \times 10^{-2}) - \log(4.03 \times 10^{-4})} = \frac{\log(2 \text{ hr}) - \log(10 \text{ hr})}{\log(2 \text{ hr}) - \log(8,760 \text{ hr})}$$

The onsite χ/Q' (10 hr) is 5.96×10^{-3} s/m³.

Offsite - The χ/Q' for the 2 hour release is $2.83 \times 10^{-5} \text{ s/m}^3$ (Van Keuren 1996a).

To determine the χ/Q' for the 22 hour release, the logarithmic interpolation procedure described in Van Keuren (1996a) is used to correctly identify the proper χ/Q' s to be used. Using the equation from Van Keuren (1996a) yields:

$$\frac{\log(2.12 \times 10^{-5}) - \log(\chi/Q' \text{ 22 hr})}{\log(2.12 \times 10^{-5}) - \log(1.24 \times 10^{-7})} = \frac{\log(2 \text{ hr}) - \log(22 \text{ hr})}{\log(2 \text{ hr}) - \log(8,760 \text{ hr})}$$

The offsite χ/Q' (22 hr) is $4.87 \times 10^{-6} \text{ s/m}^3$.

Dispersion coefficients have not been adjusted to account for the pool size. Adjusting would result in smaller χ/Q' values.

Breathing Rate (BR)

$3.3 \times 10^{-4} \text{ m}^3/\text{s}$ light activity breathing rate is used to calculate consequences to both the onsite and offsite receptors (Van Keuren 1996a). This breathing rate is used for the offsite receptor since most of the release occurs in the first 12 hours of the accident scenario duration making the use of a 24 hour average breathing rate non-conservative.

Amount of Material Released (Q)

Case 1 - $Q_{\text{spill}} = 4.3 \times 10^{-2} \text{ L}$.
 $Q_{\text{pool}} = 4.3 \times 10^{-6} \text{ L}$, 2 hour release
 $Q_{\text{soil}} = 0.15 \text{ L}$, 10 hour release (onsite).
 $Q_{\text{soil}} = 0.21 \text{ L}$, 22 hour release (offsite).

Case 2 - $Q_{\text{spill}} = 1.5 \text{ L}$.
 $Q_{\text{pool}} = 9.3 \times 10^{-4} \text{ L}$, 2 hour release
 $Q_{\text{soil}} = 5.5 \text{ L}$, 10 hour release (onsite).
 $Q_{\text{soil}} = 7.3 \text{ L}$, 22 hour release (offsite).

4.5 CALCULATIONS

Using the formula presented on page 4-4 of Van Keuren (1996a) and modifications for a 24 hour ingestion dose to the offsite receptor from Cowley et al. (1996), the radiological dose consequences can be calculated.

Onsite Consequences:

$$D \text{ (Sv)} = Q \text{ (L)} \times \frac{X}{Q'} \text{ (s/m}^3\text{)} \times R \text{ [m}^3\text{/s]} \times \text{ULD}_H \text{ [Sv/L]}$$

CASE 1 -*Inhalation Dose from Q_{spill} :*

$$D_{\text{spill}} \text{ (Sv)} = (4.3 \times 10^{-2} \text{ L})(3.41 \times 10^{-2} \text{ s/m}^3)(3.3 \times 10^{-4} \text{ m}^3\text{/s})(6.1 \times 10^3 \text{ Sv/L})$$

$$D_{\text{spill}} \text{ (Sv)} = 2.95 \times 10^{-3} \text{ Sv}$$

Inhalation Dose from Q_{pool} :

$$D_{\text{pool}} \text{ (Sv)} = (4.3 \times 10^{-6} \text{ L})(3.41 \times 10^{-2} \text{ s/m}^3)(3.3 \times 10^{-4} \text{ m}^3\text{/s})(6.1 \times 10^3 \text{ Sv/L})$$

$$D_{\text{pool}} \text{ (Sv)} = 2.95 \times 10^{-7} \text{ Sv}$$

Inhalation Dose from Q_{soil} :

$$D_{\text{soil}} \text{ (Sv)} = (0.15 \text{ L})(5.96 \times 10^{-3} \text{ s/m}^3)(3.3 \times 10^{-4} \text{ m}^3\text{/s})(6.1 \times 10^3 \text{ Sv/L})$$

$$D_{\text{soil}} \text{ (Sv)} = 1.80 \times 10^{-3} \text{ Sv}$$

Direct Radiation and Skyshine Dose from Circular Pool Area of 84 m²:

The total direct radiation and skyshine dose over a period of 12 hours,
 $D_{\text{direct/skyshine}} = 3.79 \times 10^{-3} \text{ Sv}$.

Reference Section 7.0 for details on the calculation of $D_{\text{direct/skyshine}}$ for this case.

The total dose to the onsite receptor over a period of 12 hours is:

$$\begin{aligned} & D_{\text{spill}} + D_{\text{pool}} + D_{\text{soil}} + D_{\text{direct/skyshine}} \\ & 2.95 \times 10^{-3} \text{ Sv} + 2.95 \times 10^{-7} \text{ Sv} + 1.80 \times 10^{-3} \text{ Sv} + 3.79 \times 10^{-3} \text{ Sv} \\ & \text{Total onsite dose} = 8.54 \times 10^{-3} \text{ Sv} \end{aligned}$$

CASE 2 -*Inhalation Dose from Q_{spill} :*

$$D_{\text{spill}} \text{ (Sv)} = (1.5 \text{ L})(3.41 \times 10^{-2} \text{ s/m}^3)(3.3 \times 10^{-4} \text{ m}^3\text{/s})(1.7 \times 10^2 \text{ Sv/L})$$

$$D_{\text{spill}} \text{ (Sv)} = 2.87 \times 10^{-3} \text{ Sv}$$

Inhalation Dose from Q_{pool} :

$$D_{\text{pool}} \text{ (Sv)} = (9.3 \times 10^{-4} \text{ L})(3.41 \times 10^{-2} \text{ s/m}^3)(3.3 \times 10^{-4} \text{ m}^3/\text{s})(1.7 \times 10^2 \text{ Sv/L})$$

$$D_{\text{pool}} \text{ (Sv)} = 1.78 \times 10^{-6} \text{ Sv}$$

Inhalation Dose from Q_{soil} :

$$D_{\text{soil}} \text{ (Sv)} = (5.5 \text{ L})(5.96 \times 10^{-3} \text{ s/m}^3)(3.3 \times 10^{-4} \text{ m}^3/\text{s})(1.7 \times 10^2 \text{ Sv/L})$$

$$D_{\text{soil}} \text{ (Sv)} = 1.84 \times 10^{-3} \text{ Sv}$$

Direct Radiation and Skyshine Dose from Circular Pool Area of 3,028 m²:

The total direct radiation and skyshine dose over a period of 12 hours,
 $D_{\text{direct/skyshine}} = 5.04 \times 10^{-3} \text{ Sv}$.

Reference Section 7.0 for details on the calculation of $D_{\text{direct/skyshine}}$ for this case.

The total dose to the onsite receptor over a period of 12 hours is:

$$\begin{aligned} & D_{\text{spill}} + D_{\text{pool}} + D_{\text{soil}} + D_{\text{direct/skyshine}} \\ & 2.87 \times 10^{-3} \text{ Sv} + 1.78 \times 10^{-6} \text{ Sv} + 1.84 \times 10^{-3} \text{ Sv} + 5.04 \times 10^{-3} \text{ Sv} \\ & \text{Total onsite dose} = 9.75 \times 10^{-3} \text{ Sv} \end{aligned}$$

Offsite Consequences:

$$D \text{ (Sv)} = Q \text{ (L)} \times \frac{X}{Q'} \text{ (s/m}^3) \times ((R \text{ [m}^3/\text{s}] \times \text{ULD}_h \text{ [Sv/L]}) + \text{ULD}_l \text{ [Sv-m}^3/\text{s-L]})$$

CASE 1 -

Inhalation and Ingestion Dose from Q_{spill} :

$$D_{\text{spill}} \text{ (Sv)} = (4.3 \times 10^{-2} \text{ L})(2.83 \times 10^{-5} \text{ s/m}^3)[(3.3 \times 10^{-4} \text{ m}^3/\text{s})(6.1 \times 10^3 \text{ Sv/L}) + 0.068 \text{ Sv-m}^3/\text{s-L}]$$

$$D_{\text{spill}} \text{ (Sv)} = 2.53 \times 10^{-6} \text{ Sv}$$

Inhalation and Ingestion Dose from Q_{pool} :

$$D_{\text{pool}} \text{ (Sv)} = (4.3 \times 10^{-6} \text{ L})(2.83 \times 10^{-5} \text{ s/m}^3)[(3.3 \times 10^{-4} \text{ m}^3/\text{s})(6.1 \times 10^3 \text{ Sv/L}) + 0.068 \text{ Sv-m}^3/\text{s-L}]$$

$$D_{\text{pool}} \text{ (Sv)} = 2.53 \times 10^{-10} \text{ Sv}$$

Inhalation and Ingestion Dose from Q_{soil} :

$$D_{\text{soil}} (\text{Sv}) = (0.21 \text{ L})(4.87 \times 10^{-6} \text{ s/m}^3)[(3.3 \times 10^{-4} \text{ m}^3/\text{s})(6.1 \times 10^3 \text{ Sv/L}) + 0.068 \text{ Sv-m}^3/\text{s-L}]$$

$$D_{\text{soil}} (\text{Sv}) = 2.13 \times 10^{-6} \text{ Sv}$$

There is insignificant dose contribution from direct radiation and skyshine.

The total dose to the offsite receptor over a period of 24 hours is:

$$D_{\text{spill}} + D_{\text{pool}} + D_{\text{soil}} \\ 2.53 \times 10^{-6} \text{ Sv} + 2.53 \times 10^{-10} \text{ Sv} + 2.13 \times 10^{-6} \text{ Sv}$$

$$\text{Total offsite dose} = 4.66 \times 10^{-6} \text{ Sv}$$

CASE 2 -

Inhalation and Ingestion Dose from Q_{spill} :

$$D_{\text{spill}} (\text{Sv}) = (1.5 \text{ L})(2.83 \times 10^{-5} \text{ s/m}^3)[(3.3 \times 10^{-4} \text{ m}^3/\text{s})(1.7 \times 10^2 \text{ Sv/L}) + 0.0019 \text{ Sv-m}^3/\text{s-L}]$$

$$D_{\text{spill}} (\text{Sv}) = 2.46 \times 10^{-6} \text{ Sv}$$

Inhalation and Ingestion Dose from Q_{pool} :

$$D_{\text{pool}} (\text{Sv}) = (9.3 \times 10^{-4} \text{ L})(2.83 \times 10^{-5} \text{ s/m}^3)[(3.3 \times 10^{-4} \text{ m}^3/\text{s})(1.7 \times 10^2 \text{ Sv/L}) + 0.0019 \text{ Sv-m}^3/\text{s-L}]$$

$$D_{\text{pool}} (\text{Sv}) = 1.53 \times 10^{-9} \text{ Sv}$$

Inhalation and Ingestion Dose from Q_{soil} :

$$D_{\text{soil}} (\text{Sv}) = (7.3 \text{ L})(4.87 \times 10^{-6} \text{ s/m}^3)[(3.3 \times 10^{-4} \text{ m}^3/\text{s})(1.7 \times 10^2 \text{ Sv/L}) + 0.0019 \text{ Sv-m}^3/\text{s-L}]$$

$$D_{\text{soil}} (\text{Sv}) = 2.06 \times 10^{-6} \text{ Sv}$$

There is insignificant dose contribution from direct radiation and skyshine.

The total dose to the offsite receptor over a period of 24 hours is:

$$D_{\text{spill}} + D_{\text{pool}} + D_{\text{soil}} \\ 2.46 \times 10^{-6} \text{ Sv} + 1.53 \times 10^{-9} \text{ Sv} + 2.06 \times 10^{-6} \text{ Sv}$$

$$\text{Total offsite dose} = 4.52 \times 10^{-6} \text{ Sv}$$

4.6 RESULTS

Table 4-1. Radiological Dose Consequence Results Shown For Both Cases.

CASE	SOURCE TERM (inhalation ULD as developed in Section 2.0)	TOTAL DOSE TO ONSITE RECEPTOR (Sv)	TOTAL DOSE TO OFFSITE RECEPTOR (Sv)
Case 1	6.1 E+03 (i.e., DST Liquids)	8.54 E-03	4.66 E-06
Case 2	1.7 E+02	9.75 E-03	4.52 E-06

The results shown above for the two different source terms are very close to being equal. The difference in values is due to the round-off error introduced in the calculation of the Q_{spill} , Q_{pool} , and Q_{soil} values and the size difference in the pools created from the waste leak.

This analysis confirms that the Maximum Shipment Size Worksheet as presented in WHC (1993) limits the shipment based on radionuclide content of the waste. In short, a highly radioactive waste shipment will be more volume-limited than a less radioactive liquid shipment and the inhalation dose consequences of each shipment would be equivalent.

4.7 CONCLUSIONS

Since the dose consequences shown in Section 4.6 are considered equal, the parameters and dose consequences associated with Case 1 will be presented in the TWRS FSAR. The radiological dose consequences calculated for both the onsite and offsite receptors are below the levels established in WHC-CM-4-46, Rev. 1, for *unlikely* accidents (0.05 Sv).

Consistent with the source term used in Case 1, the toxicological constituents of DST Liquids, as presented in Van Keuren (1996b), will be used to evaluate the toxicological consequences of this accident scenario. The analysis is presented in Section 5.0.

4.8 REFERENCES

- Cowley, W. L., 1996, *Development of Radiological Concentrations and Unit Liter Doses for TWRS FSAR Radiological Consequence Calculations*, WHC-SD-WM-SARR-037, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- Van Keuren, J. C., 1996a, WHC-SD-WM-SARR-016, Rev. 2, *Tank Waste Compositions and Atmospheric Dispersion Coefficients for use in ASA Consequence Assessments*, Westinghouse Hanford Company, Richland, Washington.
- Van Keuren, J. C., 1996b, WHC-SD-WM-SARR-011, Rev. 2, *Toxicological Chemical Considerations for Tank Farm Releases*, Westinghouse Hanford Company, Richland, Washington.

WHC, 1993, *Safety Analysis Report for Packaging, Railroad Liquid Waste Tank Cars*, WHC-SD-RE-SAP-013, Rev. 6, Westinghouse Hanford Company, Richland, Washington.

WHC, 1991, *Safety Analysis Manual*, WHC-CM-4-46, Section 4.0, "Risk", Rev. 1, November 15, 1995, Westinghouse Hanford Company, Richland, Washington.

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5.0 CALCULATED TOXICOLOGICAL EXPOSURES

5.1 PURPOSE

The purpose of this section is to provide the toxicological exposure calculations related to the Case 1 releases analyzed in Section 3.0.

5.2 METHODOLOGY

The methodology that is used to calculate toxicological exposure consequences is documented in WHC-SD-WM-SARR-011, Rev. 2, *Toxic Chemical Considerations for Tank Farm Releases*.

5.3 ASSUMPTIONS

The assumption made in this section is that the composition of the waste delivered to the 204-AR Waste Unloading Facility is equivalent to the composition of DST Liquids as shown in Van Keuren (1996).

This is considered a conservative assumption. Of the facilities that transfer waste to the 204-AR Waste Unloading Facility, T-Plant was considered to possess the highest potential for sending high concentrations of chemicals. The draft interim safety basis for T-Plant shows that the highest concentration of the most limiting chemical compound (sodium hydroxide [NaOH]) is maximally 160 g/L. The concentration of sodium hydroxide in DST Liquids is listed as 210 g/L.

5.4 INPUT DATA

Knowing that the probability of this unmitigated accident scenario is unlikely (see Section 1.0), the waste has the same constituents as DST Liquids, and the liquid release is continuous for the onsite receptor and a puff-type release to the offsite receptor, sum-of-fraction values can be extracted from Van Keuren (1996) to determine the toxicological consequences.

Sum-of-Fraction Values from Van Keuren (1996)

The sum-of-fraction value for the onsite receptor is 7.5×10^2 s/L.

The sum-of-fraction value for the offsite receptor is 3.4×10^{-2} /L.

Waste Release Rate for a Continuous Release

The peak release rate of waste in this scenario is found during the spill portion of the scenario. The amount of waste released during the spill was calculated previously to be 4.3×10^{-2} L. Using recommendations provided in SARR-011, a release duration of 1 minute (60 seconds) is assumed. The maximum release rate is calculated to be:

$$(4.3 \times 10^{-2} \text{ L})/60 \text{ s} = 7.2 \times 10^{-4} \text{ L/s.}$$

This value is used for the onsite receptor. A value of 4.3×10^{-2} L (amount released during spill portion of the scenario) is used for the puff release to the offsite receptor since the release time is greater than 3.5 seconds but less than 250 seconds (Van Keuren 1996).

5.5 CALCULATIONS/RESULTS

By directly multiplying the sum-of-fraction value by the waste release rate (onsite) or total release quantity (offsite), the toxicological consequences can be calculated for both the onsite and offsite receptors.

Onsite - Calculation: $(7.5 \times 10^2 \text{ L/s})(7.2 \times 10^{-4} \text{ L/s}) = 5.4 \times 10^{-1}$.

Offsite - Calculation: $(3.4 \times 10^{-2}/\text{L})(4.3 \times 10^{-2} \text{ L}) = 1.5 \times 10^{-3}$.

5.6 CONCLUSIONS

As a result of this accident the exposure to the offsite receptor is calculated to be 1.5×10^{-3} as a fraction of the risk guidelines. Similarly, the exposure to the onsite receptor is calculated to be 5.4×10^{-1} , as a fraction of the risk guidelines. Since both of these numbers are less than 1, the toxicological consequences are below the risk guidelines.

5.7 REFERENCES

Van Keuren, J. C., 1996, WHC-SD-WM-SARR-011, Rev. 2, *Toxicological Chemical Considerations for Tank Farm Releases*, Westinghouse Hanford Company, Richland, Washington.

6.0 INFORMATION VALIDATION FORMS

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Information Validation Form

Tracking # IVF-Chapter 3-01

Name of Originator	1	Organization or Team	2	Date	3
Grant W. Ryan (376-5114)		Chapter 3- Accident Analysis- 204-AR Waste Unloading Facility		March 11, 1996	

Statement of Problem

The analysis for the 204-AR Waste Unloading Facility will consider the following assumptions. EXPLICITLY concur with or deny (by including appropriate documentation) the information provided in each assumption included below.

ASSUMPTION #1 - Flammability/Thermal Stability/Organic Content of Waste Received at the 204-AR Waste Unloading Facility

It is assumed that the liquid waste received at and transferred through the 204-AR Waste Unloading Facility is not flammable and is thermally stable (WHC 1995). A separable organic phase is not allowed in waste transfers to the DST farms because of the potential for an organic fire or explosion in the double-shell underground waste receiver tank, so there is also a requirement placed on the waste generators that no separable organic layer be present in waste sent to the 204-AR Waste Unloading Facility (WHC 1995).

ASSUMPTION #2 - Generators That Send Waste to the 204-AR Facility

The ONLY waste generators that ship waste to the 204-AR Waste Unloading Facility are the 340 Facility, T-Plant, and the 222-S Laboratory.

ASSUMPTION #3 - Waste Transportation Methods

Waste from the 340-Facility and T-Plant is shipped by single-shell railcar 60H-#10-18581. The maximum holding capacity of this railcar is 75,700 liters (20,000 gallons). There are no other railcars available to transport waste to the 204-AR Facility.

Waste from the 222-S Laboratory is shipped to the 204-AR Waste Unloading Facility by Tank Trailer (pulled by semi-tractor). The Tank Trailer is described on drawing number SD-4644 on pages A1-3 and A1-4 of WHC-SD-TP-SEP-011, Rev. 0 (Nguyen 1993). The Tank Trailer has a maximum holding capacity of 18,925 liters (5,000 gallons).

ASSUMPTION #4 - Waste Transfer Destination from 204-AR Facility

The destination of waste transferred from the 204-AR Waste Unloading Facility is the double-shell underground waste storage tanks. ~~Waste is no longer transferred to any double-shell tanks classified as aging waste storage tanks (i.e., 241-AV-101, 241-AV-102, 241-AZ-101, and 241-AZ-102).~~

REFERENCES

- Nguyen, P. M., 1993, *Safety Evaluation for Packaging for the 222-S Laboratory Cargo Tank*, WHC-SD-TP-SEP-011, Rev. 0, Westinghouse Hanford Company, Richland, Washington.
- WHC, 1995, *Operating Specifications for the 204-AR Waste Unloading Facility*, OSD-T-151-00008, Rev. E-2, Westinghouse Hanford Company, Richland, Washington.

4

Alternatives	Consequences to Alternatives
5 N/A	6 N/A

Decision Reached 7		Basis for Decision 8	
Date Requested 9 March 11, 1996		Sent To 10 J. Blackwell, T. Bohan, L. Ruffin, C. Carro, Project Files	Date Requested By 11 March 19, 1996
Response #1 Agreement with assumptions			
12			
Response #2			
13			
Attachments (List)		References (List) 15	
14			
Responder #1 Name and Signature 16 Matthew S. T. Fong 4/2/96		Responder #2 Name and Signature 17 G.W. Hanson 4/8/96	
POC: Filed:		Routed:	
Further Action Required (i.e., RML, Senior Management Attention, etc.)			
18			

Information Validation Form

Tracking # IVF-Chapter 3-02

Name of Originator	1	Organization or Team	2	Date	3
Grant W. Ryan (376-5114)		Chapter 3- Accident Analysis- 204-AR Waste Unloading Facility		March 19, 1996	

Statement of Problem

The analysis for the 204-AR Waste Unloading Facility will consider the following assumptions. EXPLICITLY concur with or deny (by including appropriate documentation) the information provided in each assumption.

ASSUMPTION #1 - SHIPPING REQUIREMENTS FOR RAILCARS AND TANK TRAILERS

The maximum volume of newly generated waste shipped to the 204-AR Waste Unloading Facility by railcar is administratively limited based on the results of the railcar transport safety analysis (WHC 1993).

Similar requirements are in place to govern the transfer of liquid radioactive waste by tank trailer (Nguyen 1993) *AND APPLICABLE SECTIONS OF 49CFR 177.301-304 DOT TITLING CLOS*

ASSUMPTION #2 - WASTE SHIPMENTS BY RAILCAR BOUND SHIPMENTS BY TANK TRAILER

The newly generated waste shipped to the 204-AR Waste Unloading Facility from the 222-S Laboratories contains much lower concentrations of radionuclides than the waste shipped in railcars from the 300 Area (i.e., 340 Facility). Considering this fact and the total volumes that can be shipped by both methods of transport, it is assumed that accident scenarios at the 204-AR Waste Unloading Facility involving railcars bound similar accidents for tank trailers.

ASSUMPTION #3 - CATCH TANK AND CATCH TANK PIT VOLUME

The catch tank and its pit have a combined holding capacity of approximately 17,030 L (4,500 gal).

ASSUMPTION #4 - MAXIMUM SPECIFIC GRAVITY AND DENSITY OF LIQUID WASTE RECEIVED

The maximum specific gravity of liquid waste received at the 204-AR is expected to be no greater than 1.3. Assuming that the density of water is 1.0 g/cm³, this would make the density of liquid waste received at this facility a maximum of 1.3 g/cm³.

REFERENCES

Nguyen, P. M., 1993, *Safety Evaluation for Packaging for the 222-S Laboratory Cargo Tank*, WHC-SD-TP-SEP-011, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC, 1993, *Safety Analysis Report for Packaging, Railroad Liquid Waste Tank Car*, SD-RE-SAP-013, Rev. 6, Westinghouse Hanford Company, Richland, Washington.

Alternatives		Consequences to Alternatives	
5 N/A		6 N/A	
Decision Reached		Basis for Decision	
7		8	
Date Requested 9 March 20, 1996	Sent To 10 J. Blackwell, T. Bohan, L. Ruffin, C. Carro, Project Files	Date Requested By 11 March 29, 1996	
Response #1 <i>Agreement with Assumptions</i>			
12			
Response #2			
13			
Attachments (List)		References (List) 15	
14			
Responder #1 Name and Signature		Responder #2 Name and Signature	
16 <i>Matthew S. Tiffney</i> <i>Matthew S. Tiffney</i> 4/2/96		17 <i>W. H. Hansen</i> 4/8/96 GN HANSEN	
POC: <i>Filed:</i>		Routed:	
Further Action Required (i.e., RML, Senior Management Attention, etc.)			
18			

Information Validation Form

Tracking # IVF-Chapter 3-03

Name of Originator	1	Organization or Team	2	Date	3
Grant W. Ryan (376-5114)		Chapter 3- Accident Analysis- 204-AR Waste Unloading Facility		March 21, 1996	

Statement of Problem

The analysis for the 204-AR Waste Unloading Facility will consider the following assumptions. EXPLICITLY concur with or deny (by including appropriate documentation) the information provided in each assumption.

ASSUMPTION #1 - POOL RELEASE DURATIONS

The pool release analyzed for the accident scenario of a railcar spill at the 204-AR Waste Unloading Facility assumes a shallow pool is present for a period of 2 hours. After 2 hours, the pool is assumed to be absorbed into the soil and the remaining release is due to the resuspension of particles from the contaminated soil. The dose consequences are being analyzed for a period of 12 hours for the onsite person and a period of 24 hours for the offsite receptor. This means that for each receptor the release for the first 2 hours is from the shallow pool and the remaining release (either 10 or 22 hours, depending on receptor) is from the resuspension of particles from the contaminated soil.

ASSUMPTION #2 - POOL MODEL CONSERVATISM

The pool model assumes that all of the liquid that is released to make up the pool is also available for resuspension from the wetted soil. This is a conservative assumption since it is likely that some percentage of the liquid would seep into the soil and not be available to contribute to the dose consequences.

ASSUMPTION #3 - SURFACE TENSION OF THE LIQUID WASTE

The surface tension of the liquid waste is an important factor in determining the theoretical depth of the surface pool created by the spill. To develop the property of surface tension for inclusion in this calculation, the waste is assumed to be at a temperature of 21°C (70°F). From the CRC Handbook (Weast 1985), the surface tension of water against air is 72.75 dynes/cm @ 20°C. While it is understood that the waste delivered to this facility contains mainly water, it may also contain the inorganic compounds NaOH or NaNO₃, which could affect the surface tension. From the table, *Surface Tension of Inorganic Solutes in Water* (Weast 1985), the maximum surface tension of NaOH (@ 18°C and 35.9 wt%) is 101.05 dynes/cm. Similarly, the maximum surface tension of NaNO₃ (@ 20°C and 47.06 wt%) is 87.05 dynes/cm. Clearly a mixture of these inorganic compounds will increase the surface tension of plain water. Knowing this and realizing that neither of these compounds would be found in concentrations near the maximums shown here, a surface tension of 80.00 dynes/cm @ 20°C will be used to develop the theoretical depth of the surface pool.

REFERENCES

Weast, R. C., 1985, *CRC Handbook of Chemistry and Physics*, 66th Edition, CRC Press, Inc., Boca Raton, Florida.

Alternatives		Consequences to Alternatives	
5 N/A		6 N/A	
Decision Reached		Basis for Decision	
7		8	
Date Requested 9 March 21, 1996	Sent To 10 J. Blackwell, T. Bohan, L. Ruffin, C. Carro, Project Files	Date Requested By 11 March 29, 1996	
Response #1 <i>Agreement with Assumptions</i>			
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Response #2			
13			
Attachments (List)		References (List) 15	
14			
Responder #1 Name and Signature		Responder #2 Name and Signature	
16 <i>Matthew S Tiffany</i> <i>4/2/96</i>		17 <i>M/H</i> <i>4/9/96</i> <i>G.W. HANSON</i>	
POC: Filed:		Routed:	
Further Action Required (i.e., RML, Senior Management Attention, etc.)			
18			

Information Validation Form

Tracking # IVF-Chapter 3-04

Name of Originator	1	Organization or Team	2	Date	3
Grant W. Ryan (376-5114)		Chapter 3- Accident Analysis- 204-AR Waste Unloading Facility		March 22, 1996	

Statement of Problem

The analysis for the 204-AR Waste Unloading Facility will consider the following assumptions. EXPLICITLY concur with or deny (by including appropriate documentation) the information provided in each assumption.

ASSUMPTION #1 - TOXICOLOGICAL COMPOSITION OF WASTE RECEIVED AT THE 204-AR

The toxicological composition of the source term that will be used to evaluate accidents at the 204-AR Waste Unloading Facility is made up of the same constituents that are presented in Davis et al. (1995) for DST Liquids.

Even though limited information is known about the toxicological content of this variable content liquid waste, this assumption appears to be reasonable since the waste transferred through this facility is being added to the DST inventory.

ASSUMPTION #2 - RADIOLOGICAL COMPOSITION OF WASTE RECEIVED AT THE 204-AR

The worksheet provided in the railcar SARP (WHC 1993) will be used to develop the maximum shipment size and the radiological waste characteristics to be used in the analysis of accidents at the 204-AR Waste Unloading Facility. Two cases are considered: 1) developing the maximum shipment size using DST Liquids as the source term input and 2) determining the unit liter dose for a maximum authorized shipment of 75,700 L (20,000 gal). The details are attached.

REFERENCES

Cowley, W. L., D. L. Herting, D. A. Himes, D. A. Reynolds, T. S. Vail, J. C. Van Keuren, L. P. Leach, J. York, 1996, *Development of Radiological Concentrations and Unit Liter Doses for TWRS FSAR Radiological Consequence Calculations*, WHC-SD-WM-SARR-037, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

Davis, J. S., M. L. Dentler, and J. C. Van Keuren, 1995, WHC-SD-WM-SARR-011, Rev. 1, *Toxicological Chemical Considerations for Tank Farm Releases*, Westinghouse Hanford Company, Richland, Washington.

WHC, 1993, *Safety Analysis Report for Packaging, Railroad Liquid Waste Tank Car*, SD-RE-SAP-013, Rev. 6, Westinghouse Hanford Company, Richland, Washington.

4

Alternatives	Consequences to Alternatives
5 N/A	6 N/A
Decision Reached	Basis for Decision
7	8

Date Requested 9 March 22, 1996		Sent To 10 J. Blackwell, T. Bohan, L. Ruffin, C. Carro, Project Files		Date Requested By 11 April 1, 1996	
Response #1 <p style="text-align: center;"><i>Agree with Assumptions and methodology</i></p>					
12 Response #2					
13					
Attachments (List)			References (List) 15		
14					
Responder #1 Name and Signature 16 <i>Matthew S. Tiffney</i> <i>4/21/96</i>			Responder #2 Name and Signature 17 <i>W. H. G. Hanson</i> <i>4/9/96</i>		
POC: Filed: Routed:					
Further Action Required (i.e., RML, Senior Management Attention, etc.)					
18					

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ASSUMPTION #2 - RADIOLOGICAL COMPOSITION OF WASTE RECEIVED AT THE 204-AR

The worksheet provided in the railcar SARP (WHC 1993) will be used to develop the maximum shipment size and the radiological waste characteristics to be used in the analysis of accidents at the 204-AR Waste Unloading Facility. Two cases are considered: 1) developing the maximum shipment size using DST Liquids as the source term input and 2) determining the unit liter dose for a maximum authorized shipment of 75,700 L (20,000 gal).

In order to perform accurate accident analysis for the 204-AR Waste Unloading Facility it is necessary to determine the bounding radionuclide inventory of the waste shipped and received. The following 3 facts are known:

1. The radionuclide inventory of each waste shipment to the 204-AR Waste Unloading Facility is variable.
2. The waste shipped to the 204-AR Waste Unloading Facility is volume-limited based on the results of the railcar transport safety analysis (WHC 1993) which requires generators shipping by rail to calculate a maximum shipment volume based on the radionuclide content of the waste to be shipped.

It is important to note that even though a railcar shipment could be as large as 75,700 L (20,000 gal), the radiological consequences of a release this large would be no worse than those from a significantly smaller waste shipment. This is because of the administrative procedure to limit the volume of each waste shipment destined for the 204-AR Unloading Facility based on the specific radionuclide content of each shipment. Simply said, a 75,700 L (20,000 gal) shipment of highly radioactive liquid waste would not be permitted. This administrative control is imposed because of requirements established in the railcar transport safety analysis (WHC 1993).

3. In order to retain inventory control for the double-shell tank farms, a liquid waste with a unit liter dose of greater than 6.1×10^4 Sv/L (DST Liquids) cannot be received and transferred through the 204-AR Waste Unloading Facility since the final destination is the double-shell tank farms (i.e., 241-AP-106). The discussion of all ULD's in this document refer to the dose per unit liter inhaled.

Using these facts and applying the methodology presented in WHC (1993), a source term for the waste received at the 204-AR Waste Unloading Facility can be developed.

First, it is assumed that a batch of DST Liquids will be shipped. To determine the maximum authorized shipment volume of this type of liquid, the same worksheet that is presented and utilized in WHC (1993) is used. Using concentration values for DST Liquids taken from WHC-SD-WM-SARR-037 (Cowley et al. 1996) and the SARP Maximum Shipment Size Worksheet (Figure 1X), the maximum authorized shipping volume of DST Liquids is 2,123 L (561 gal).

Postulating that the maximum authorized shipping volume of 2,123 L (561 gal) will not result in maximized dose consequences from pooling and resuspension of a spill (due to the difference in pool size and wetted soil area), a ULD is also determined for an unknown liquid that would correlate to the maximum authorized shipping volume of 75,700 L (20,000 gal).

Using the SARP Maximum Shipment Size Worksheet, the ULD of a liquid waste that would allow a shipment of 75,700 L (20,000 gal) can be calculated.

From the Worksheet, if $V_{max} = 20,000$ gal, then $n = 25,000$ gal.

ATTACHMENT TO: IVF-Chapter 3-04

Therefore, using the same heel reduction factor ($h = 0.80$) and release fraction $\beta = 0.0017$),

$$\text{Total AB} = 0.80 / [(25,000 \text{ gal})(0.0017)]$$

$$\text{Total AB} = 0.019/\text{gal}$$

Setting up a simple ratio between the ULD that is known and the known Total AB values gives:

$$\text{Total AB}_{\text{DST Liq.}} / \text{ULD}_{\text{DST Liq.}} = \text{Total AB}_{\text{Unknown Liq.}} / \text{ULD}_{\text{Unknown Liq.}}$$

$$(0.67/\text{gal}) / 6.1 \times 10^3 \text{ Sv/L} = (0.019/\text{gal}) / X \text{ Sv/L}$$

$$X \text{ Sv/L} = 1.7 \times 10^2 \text{ Sv/L}$$

The ULD associated with the shipment of 75,700 L (20,000 gal) is reduced by approximately 36 times when compared to the ULD for DST Liquids.

$$(6.1 \times 10^3 \text{ Sv/L}) / (1.7 \times 10^2 \text{ Sv/L}) = 35.88$$

Reducing the concentrations of each radionuclide included in the SARP Maximum Shipment Size Worksheet by the same factor gives the concentration values shown in Table 2X. These concentrations will be used to calculate the direct radiation dose as a result of exposure to a pool-type release.

A completed SARP worksheet with these revised values is shown in Figure 3X.

The ULD for DST Liquids ($6.1 \times 10^3 \text{ Sv/L}$) correlating to a maximum authorized shipment of 2,123 L (561 gal) will be used to evaluate the dose consequences of a pool and resuspension release. Additionally, the ULD that was developed here ($1.7 \times 10^2 \text{ Sv/L}$) for a maximum authorized shipment of 75,700 L (20,000 gal) will also be used to evaluate a similar pool and resuspension release. The difference here would be the size of the pool and the area of wetted soil as a result of the release.

The ULD and associated maximum authorized shipment volume with the greatest dose consequences will be presented in the FSR of AP2 of the work required to reach a conclusion will be retained and published in a calculation note.

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Figure IX. Using the same methodology presented in SD-RE-SAP-013, Rev. 6, *Safety Analysis For Packaging, Railroad Liquid Waste Tank Car*, a maximum shipment size of liquid waste with radionuclide concentrations identical to those of the DST Liquids is calculated to be 2,123 L (561 gal).

Maximum Shipment Size Worksheet

Date: March 13, 1996

Tank car designation: #10H-18581

Heel reduction factor (from Heel Reduction Factor Worksheet): $h = 0.90$ Limits based on doses to [x] onsite worker
[] member of publicVolume Unit: gallons

Nuclide	A Concentration (Ci/unit vol.)		B (D/L)	A x B
¹² C	2.4E-05	Ci/gal	NOT IN SARP	NOT IN SARP
⁶⁰ Co	7.2E-04	Ci/gal	3.0E-02	2.1E-05
⁹⁰ Se	n/a	n/a	n/a	n/a
⁹⁰ Sr	4.9E-01	Ci/gal	3.5E-02	1.7E-02
⁹⁰ Y	4.9E-01	Ci/gal	INCL. IN Sr-90	INCL. IN Sr-90
¹³⁷ Cs	1.1E-03	Ci/gal	1.4E-03	1.6E-06
¹⁹² Pu	n/a	n/a	n/a	n/a
¹³⁵ Sb	n/a	n/a	n/a	n/a
¹³⁷ I	2.0E-06	Ci/gal	NOT IN SARP	NOT IN SARP
¹³⁷ Cs	1.1E-03	Ci/gal	6.3E-03	7.1E-06
¹³⁷ Cs	9.0E-00	Ci/gal	4.1E-03	2.5E-02
¹³⁴ Cs	n/a	n/a	n/a	n/a
¹³⁷ Mn	5.3E-03	Ci/gal	n/a	3.4E-05
¹³⁵ Eu	4.3E-03	Ci/gal	4.1E-02	1.8E-04
¹³⁷ Eu	n/a	n/a	n/a	n/a
¹³⁷ Mo	2.4E-05	Ci/gal	NOT IN SARP	NOT IN SARP
²³⁹ Pu	1.3E-04	Ci/gal	1.1E-02	2.0E-02
²³⁹ Pu	7.3E-04	Ci/gal	2.4E-02	1.9E-01
²⁴¹ Pu	1.3E-03	Ci/gal	2.4E-00	4.4E-03
²⁴¹ Am	3.5E-03	Ci/gal	1.2E-02	4.2E-01
²⁴¹ Cm	n/a	n/a	n/a	n/a
²⁴¹ Cm	1.2E-05	Ci/gal	6.3E-01	7.7E-04

(Attach additional sheet if necessary)

Total AB = 6.7E-01.

Release fraction: $R = [x]$ with fire $\rightarrow 0.0017$
[] without fire $\rightarrow 0.00037$ [] other $\rightarrow$ _____
Specify authorizing document:Maximum shipping volume: $n = \frac{h}{R \times (\text{Total AB})} = \frac{0.90}{0.0017 \times 6.7E-01} = 702 \text{ gallons}$ Authorized shipping volume: $V_{\text{max}} = 702 / 1.25 = 561 \text{ gal (2,123 L)}$

Performed by: _____

Checked by: _____
printed name signature date

See Worksheet Notes on next page.

ATTACHMENT T0: IVF-Chapter 3-04

WORKSHEET NOTES (FIGURE 1X):

1. Column A in the table is the concentration of each radionuclide in the source term in units of Ci/gal. For the radionuclides where no available data exists, these are listed as n/a. These radionuclides that are listed as n/a have a negligible impact on the dose calculations due to their low activity concentrations.
2. Column B in the table is the dose limiting factor taken from Table 5 of the SARP (WHC 1993). The units of this column is (1/Ci) since $D = \text{rem/Ci}$ and $L = \text{rem}$. A few dose limiting factors are listed as "NOT IN SARP" since values for these radionuclides are not listed in Table 5 of the SARP (WHC 1993). The absence of these values is considered to have a negligible effect on the final results of this worksheet.
3. Column C in the table is the multiplication of Columns A and B. A sum of these values (Total AB) is calculated at the bottom of Column C. The units of this column is (1/gal).
4. The Heel Reduction Factor, h , is calculated by a worksheet also presented in the SARP (WHC 1993). A conservative heel reduction factor of 0.8 is chosen for use in this worksheet. This is the maximum heel reduction factor achievable based on a total of 16 grams of plutonium and the most conservative release fraction. This can be seen from the graph on page 57 of the SARP (WHC 1993).
5. It is interesting to note that the only radionuclides from the chosen source term (DST Liquids) that contribute significantly to the development of Total AB are the following: ^{90}Sr , ^{137}Cs , ^{238}Pu , $^{239/240}\text{Pu}$, and ^{241}Am .

ATTACHMENT TO: IVF-Chapter 3-04

Table 2X. The Concentrations of Radionuclides Making Up a Shipment of 75,700 L (20,000 gal) with a Unit Liter Dose of 1.7×10^2 Sv/L.

Radionuclide	Concentration (Ci/gal)
C-14	6.7E-07
Co-60	2.0E-05
Se-79	n/a
Sr-90	1.4E-02
Y-90	1.4E-02
Tc-99	3.1E-05
Ru-106	n/a
Sb-125	n/a
I-129	5.6E-08
Cs-134	3.1E-05
Cs-137	1.7E-01
Ce-144	n/a
Pm-147	1.6E-04
Eu-154	1.2E-04
Eu-155	n/a
Np-237	6.7E-07
Pu-238	5.0E-06
Pu-239/240	2.2E-05
Pu-241	5.0E-05
Am-241	9.8E-05
Cm-242	n/a
Cm-244	3.3E-07

n/a -- No available data. These radionuclides have a negligible impact on the dose calculations due to their low activity concentrations.

ATTACHMENT T0: IVF-Chapter 3-04

Figure 3x. The Unit Liter Dose Back-Calculated from a Maximum Authorized Shipment Volume of 75,700 L (20,000 gal) is Found to be 1.7×10^2 Sv/L inhaled.

Maximum Shipment Size Worksheet

Date: March 13, 1996

Tank car designation: #10H-18581

Heel reduction factor (from Heel Reduction Factor Worksheet): $h = \underline{0.80}$

Limits based on doses to [x] onsite worker
[] member of public

Volume Unit: gallons

Nuclide	A Concentration (Ci/unit vol.)		B (D/L)	A x B
¹⁴ C	5.7E-07	Ci/gal	NOT IN SARP	NOT IN SARP
¹⁴⁶ Cd	2.0E-05	Ci/gal	3.0E-02	6.0E-07
¹⁵² Eu	n/a	n/a	n/a	n/a
¹⁵⁹ Gd	1.4E-02	Ci/gal	3.5E-02	4.9E-04
¹⁶⁰ Gd	1.4E-02	Ci/gal	INCL. IN Sr-90	INCL. IN Sr-90
¹⁶⁷ Gd	3.1E-05	Ci/gal	1.4E-03	4.3E-08
¹⁸⁸ Hf	n/a	n/a	n/a	n/a
¹⁸⁸ Re	n/a	n/a	n/a	n/a
¹⁸⁸ Y	5.6E-08	Ci/gal	NOT IN SARP	NOT IN SARP
¹⁹² Ir	3.1E-05	Ci/gal	6.3E-03	1.9E-07
¹⁹² Pu	1.7E-01	Ci/gal	4.1E-03	6.9E-04
¹⁹⁴ Pu	n/a	n/a	n/a	n/a
¹⁹⁷ Pm	1.3E-04	Ci/gal	n/a	9.4E-07
¹⁹⁸ Pm	1.2E-04	Ci/gal	4.1E-02	4.9E-06
¹⁹⁸ Am	n/a	n/a	n/a	n/a
²³⁸ U	5.7E-07	Ci/gal	NOT IN SARP	NOT IN SARP
²³⁸ Pu	3.0E-06	Ci/gal	1.1E-02	3.3E-04
²³⁹ Am	2.2E-05	Ci/gal	2.4E-02	5.2E-03
²⁴¹ Pu	5.0E-05	Ci/gal	2.4E-02	1.2E-04
²⁴¹ Am	9.3E-05	Ci/gal	1.2E-02	1.2E-02
²⁴² Am	n/a	n/a	n/a	n/a
²⁴² Cm	3.3E-07	Ci/gal	6.3E-01	2.1E-05

(Attach additional sheet if necessary) Total AB = 1.9E-02

Release fraction: $R = [x]$ with fire $\rightarrow 0.0017$
[] without fire $\rightarrow 0.00037$

[] other $\rightarrow$ _____
Specify authorizing document:

Maximum shipping volume: $n = \frac{h}{R \times (\text{Total AB})} = \underline{25,000 \text{ gallons}}$

Authorized shipping volume: $V_{\text{max}} = 25,000 / 1.25 = \underline{20,000 \text{ gal (75,700 L)}}$

Performed by: _____

Checked by: _____
printed name signature date

See Worksheet Notes on next page.

ATTACHMENT TO: IVF-Chapter 3-04

WORKSHEET NOTES (FIGURE 3X):

1. Column A in the table is the concentration of each radionuclide in the source term in units of Ci/gal. For the radionuclides where no available data exists, these are listed as n/a. These radionuclides that are listed as n/a have a negligible impact on the dose calculations due to their low activity concentrations.
2. Column B in the table is the dose limiting factor taken from Table 5 of the SARP (WHC 1993). The units of this column is (1/Ci) since $D = \text{rem/Ci}$ and $L = \text{rem}$. A few dose limiting factors are listed as "NOT IN SARP" since values for these radionuclides are not listed in Table 5 of the SARP (WHC 1993). The absence of these values is considered to have a negligible effect on the final results of this worksheet.
3. Column C in the table is the multiplication of Columns A and B. A sum of these values (Total AB) is calculated at the bottom of Column C. The units of this column is (1/gal).
4. The Heel Reduction Factor, h , is calculated by a worksheet also presented in the SARP (WHC 1993). A conservative heel reduction factor of 0.8 is chosen for use in this worksheet. This is the maximum heel reduction factor achievable based on a total of 16 grams of plutonium and the most conservative release fraction. This can be seen from the graph on page 57 of the SARP (WHC 1993).
5. It is interesting to note that the only radionuclides from the chosen source term that contribute significantly to the development of Total AB are the following: ^{90}Sr , ^{137}Cs , ^{238}Pu , $^{239/240}\text{Pu}$, and ^{241}Am .

7.0 DIRECT RADIATION AND SKYSHINE DOSE RATES

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DIRECT RADIATION AND SKYSHINE DOSE RATES

INTRODUCTION

For each of the cases (i.e., pool sizes and source term) discussed and developed in Section 2.0 and 3.0, the line-of-sight (direct) and skyshine dose rates are calculated. The spills are modelled as cylinders. Summarizing, the two cases considered are: 1) the pool area is 84 m² with a depth of 0.025 m, radius of 5.2 m, and the spill volume is 2,123 L (561 gal), and 2) the pool area is 3,028 m² with a depth of 0.025 m, radius of 31.0 m, and the spill volume is 75,700 L (20,000 gal).

METHODOLOGY

Line-of-sight and skyshine dose rate contributions were calculated using the MICROSHIELD (Grove 1992) and MICROSKYSHINE (Grove 1987) computer codes, respectively. WHC-SD-WM-SARR-016, Rev. 2 contains a description of the calculational methods used in the MICROSHIELD (line-of-sight contribution), MICROSKYSHINE (skyshine contribution), and BREMCALC (bremsstrahlung source term generation) computer codes. WHC-SD-WM-SARR-016, Rev. 2 recommends that the direct and skyshine doses from pools should be calculated from the edge of the pool. In this analysis, however, the doses are calculated using the center of the pool for conservatism. No controls are developed as a result of this conservatism and alternate calculations have not been performed.

RADIOLOGICAL SOURCE TERM

The source term used in this analysis is made up of the constituents and concentrations of DST liquids (Table 7-1), as derived from Cowley (1996) and Van Keuren (1996). For a spill volume of 2,123 L (561 gal) Table 7-2 is developed. For the larger spill volume with reduced radionuclide concentrations (Table 2-1), Table 7-2 is also used since the total activity in both spills is equivalent.

The results for specific radionuclides (i.e., Cs-137, Eu-154, Sr-90, and Y-90) from Table 7-2 are used to develop the dose consequences to the onsite receptor. Cs-137 is decayed by MICROSHIELD and MICROSKYSHINE to obtain the correct quantity of the Cs-137 daughter product, Ba-137m.

LINE-OF-SIGHT AND SKYSHINE DOSE RATES

For each case, two sets of MICROSHIELD and MICROSKYSHINE runs are made to calculate the total dose rate for a receptor located 100 meters from the center of the pool. The first set of runs is made to calculate the dose contribution from the important gamma emitters (^{137m}Ba and ¹⁵⁴Eu) and the second set of runs is made to calculate the dose contribution from bremsstrahlung radiation produced during the decay of ⁹⁰Sr/⁹⁰Y. Note that other beta emitters are present in the source term, however ⁹⁰Sr/⁹⁰Y are the only beta emitters with a high enough activity to be a significant dose contributor.

Table 7-3 includes output generated by BREMCALC (Rittman 1992) of the photon production rates for 1 Ci (3.7×10^{10} Bq) of $^{90}\text{Sr}/^{90}\text{Y}$, in water (and other materials). By simply ratioing the values provided in Appendix A, the photon production rates for 2.64×10^2 Ci (9.77×10^{12} Bq) of $^{90}\text{Sr}/^{90}\text{Y}$ are tabulated in Table 7-4.

Table 7-1. The concentrations of Radionuclides Making Up a Shipment of 2,123 L (561 gal) with a Unit Liter Dose of 6.1×10^3 Sv/L.

Radionuclide	Concentration (Ci/gal)
C-14	2.35 E-05
Co-60	7.15 E-04
Se-79	n/a
Sr-90	4.70 E-01
Y-90	4.70 E-01
Tc-99	1.12 E-03
Ru-106	n/a
Sb-125	n/a
I-129	2.04 E-06
Cs-134	6.23 E-04
Cs-137	6.03 E+00
Ce-144	n/a
Pm-147	3.68 E-03
Eu-154	4.29 E-03
Eu-155	n/a
Np-237	2.35 E-05
Pu-238	1.84 E-04
Pu-239/240	7.87 E-04
Pu-241	1.84 E-03
Am-241	3.47 E-03
Cm-242	1.12 E-09
Cm-244	1.23 E-05

Table 7-2. Total Pool Activities of Radionuclides Making Up Both Pool Sizes.

Radionuclide	Activity (Ci)
C-14	1.32 E-02
Co-60	4.01 E-01
Se-79	n/a
Sr-90	2.64 E+02
Y-90	2.64 E+02
Tc-99	6.31 E-01
Ru-106	n/a
Sb-125	n/a
I-129	1.15 E-03
Cs-134	3.50 E-01
Cs-137	3.38 E+03
Ce-144	n/a
Pm-147	2.06 E+00
Eu-154	2.41 E+00
Eu-155	n/a
Np-237	1.32 E-02
Pu-238	1.03 E-01
Pu-239/240	4.41 E-01
Pu-241	1.03 E+00
Am-241	1.95 E+00
Cm-242	6.31 E-07
Cm-244	6.88 E-03

Table 7-3. Photon Production Rates for Sr-90/Y-90 in Mixtures using BREMCALC (photons per second from 1 curie of each isotope).

Midpoint Energy	Water Z = 6.60	Paper Z = 6.35	Air Z = 7.36	Concrete Z = 10.99	AlZr Z = 33.38
0.015	7.83E+08	7.59E+08	8.56E+08	1.24E+09	3.72E+09
0.025	3.99E+08	3.87E+08	4.35E+08	6.26E+08	1.88E+09
0.035	2.56E+08	2.48E+08	2.79E+08	3.99E+08	1.19E+09
0.045	1.82E+08	1.77E+08	1.98E+08	2.83E+08	8.42E+08
0.055	1.38E+08	1.34E+08	1.50E+08	2.14E+08	6.34E+08
0.065	1.09E+08	1.06E+08	1.19E+08	1.69E+08	4.98E+08
0.075	8.91E+07	8.66E+07	9.67E+07	1.37E+08	4.03E+08
0.085	7.42E+07	7.21E+07	8.05E+07	1.14E+08	3.34E+08
0.095	6.29E+07	6.11E+07	6.82E+07	9.61E+07	2.81E+08
0.15	3.33E+08	3.24E+08	3.60E+08	5.03E+08	1.45E+09
0.25	1.32E+08	1.29E+08	1.42E+08	1.95E+08	5.50E+08
0.35	6.88E+07	6.71E+07	7.37E+07	9.98E+07	2.74E+08
0.475	5.50E+07	5.38E+07	5.87E+07	7.84E+07	2.10E+08
0.65	3.53E+07	3.46E+07	3.75E+07	4.89E+07	1.26E+08
0.825	1.36E+07	1.33E+07	1.43E+07	1.82E+07	4.49E+07
1	9.82E+06	9.66E+06	1.03E+07	1.28E+07	3.02E+07
1.225	5.54E+06	5.46E+06	5.75E+06	6.95E+06	1.53E+07
1.475	2.06E+06	2.03E+06	2.12E+06	2.46E+06	4.94E+06
1.7	5.34E+05	5.29E+05	5.45E+05	6.12E+05	1.11E+06
1.9	1.36E+05	1.35E+05	1.38E+05	1.51E+05	2.46E+05
2.1	1.49E+04	1.48E+04	1.50E+04	1.59E+04	2.31E+04
2.3	8.25E+01	8.23E+01	8.28E+01	8.51E+01	1.05E+02

Table 7-4. Photon Production Rates
(photons/sec) for $^{90}\text{Sr}/^{90}\text{Y}$ in Water Are
Shown.

GROUP MIDPOINT ENERGY (Mev)	PRODUCTION RATE IN WATER 1 Ci Sr-90 (Table 7-3)	PRODUCTION RATE OF 2.64 E+02 Ci Sr-90 IN WATER (photons/sec)
0.015	7.8 E+08	2.1 E+11
0.025	4.0 E+08	1.1 E+11
0.035	2.6 E+08	6.9 E+10
0.045	1.8 E+08	4.8 E+10
0.055	1.4 E+08	3.7 E+10
0.065	1.1 E+08	2.9 E+10
0.075	8.9 E+07	2.3 E+10
0.085	7.4 E+07	2.0 E+10
0.095	6.3 E+07	1.7 E+10
0.150	3.3 E+08	8.7 E+10
0.250	1.3 E+08	3.4 E+10
0.350	6.9 E+07	1.8 E+10
0.475	5.5 E+07	1.5 E+10
0.650	3.5 E+07	9.2 E+09
0.825	1.4 E+07	3.7 E+09
1.000	9.8 E+06	2.6 E+09
1.225	5.5 E+06	1.5 E+09
1.475	2.1 E+06	5.5 E+08
1.700	5.3 E+05	1.4 E+08
1.900	1.4 E+05	3.7 E+07
2.100	1.5 E+04	4.0 E+06
2.300	8.3 E+01	2.2 E+04
Totals	2.7 E+09	7.3 E+11

CODE INPUT

The input parameters for Cases 1 and 2, are summarized in Tables 7-5 and 7-6, respectively.

Table 7-5. Summary of Input Parameters for Case 1 (Small Pool).

Activity	Cs-137 activity in the pool = $1.25 \text{ E}+14 \text{ Bq}$ ($3.38 \text{ E}+03 \text{ Ci} \times 3.7 \text{ E}+10 \text{ Bq/Ci}$), which results in an activity of $1.18 \text{ E}+14 \text{ Bq}$ Ba-137m (Ba-137m is the Cs-137 daughter product). Similarly, there are activities of $8.92 \text{ E}+10 \text{ Bq}$ Eu-154 and $9.77 \text{ E}+12 \text{ Bq}$ Sr-90/Y-90.
Geometry	A cylinder with a height of 0.025 m (0.082 ft) and radius of 5.2 m (17.06 ft) is created from the spilled waste.
Source Material	Water with a 1.3 g/cc (81.3 lb/ft^3) density.
Shield Material	Air with a density of 0.00122 g/cc. No credit is taken for the uncontaminated soil surrounding the pool as an immersion shield.
Receptor Location	A 1.5 m (~5 ft) high individual located 100 m from the center of the pool.
Integration Parameters	MICROSHIELD - The cylindrical source was divided into 10 radial, 10 circumferential, and 10 axial kernels or segments. MICROSKYSHINE - The cylindrical source was divided into 5 radial, 5 circumferential, and 5 axial kernels or segments.

Table 7-6. Summary of Input Parameters for Case 2 (Large Pool).

Activity	Cs-137 activity in the pool = $1.25 \text{ E}+14 \text{ Bq}$ ($3.38 \text{ E}+03 \text{ Ci} \times 3.7 \text{ E}+10 \text{ Bq/Ci}$), which results in an activity of $1.18 \text{ E}+14 \text{ Bq}$ Ba-137m (Ba-137m is the Cs-137 daughter product). Similarly, there are activities of $8.92 \text{ E}+10 \text{ Bq}$ Eu-154 and $9.77 \text{ E}+12 \text{ Bq}$ Sr-90/Y-90.
Geometry	A cylinder with a height of 0.025 m (0.082 ft) and radius of 31.0 m (101.68 ft) is created from the spilled waste.
Source Material	Water with a 1.3 g/cc (81.3 lb/ft^3) density.
Shield Material	Air with a density of 0.00122 g/cc . No credit is taken for the uncontaminated soil surrounding the pool as an immersion shield.
Receptor Location	A 1.5 m (~5 ft) high individual located 100 m from the center of the pool.
Integration Parameters	MICROSHIELD - The cylindrical source was divided into 10 radial, 10 circumferential, and 10 axial kernels or segments. MICROSKYSHINE - The cylindrical source was divided into 5 radial, 5 circumferential, and 5 axial kernels or segments.

RESULTS

Table 7-7. Summary of Dose Rates For Both Cases.

Dose Rate Component	Case 1 (rem/hr)	Case 2 (rem/hr)
Line-of-Sight	$1.08 \text{ E}-02$	$1.05 \text{ E}-02$
Line-of-Sight Bremsstrahlung	$9.49 \text{ E}-06$	$9.31 \text{ E}-06$
Skyshine	$2.08 \text{ E}-02$	$3.14 \text{ E}-02$
Skyshine Bremsstrahlung	$3.44 \text{ E}-05$	$4.78 \text{ E}-05$
Totals (rem/hr)	$3.16 \text{ E}-02$	$4.20 \text{ E}-02$
Total Dose for 12 Hours (rem)	$3.79 \text{ E}-01$	$5.04 \text{ E}-01$

REFERENCES

Cowley, W. L., 1996, *Development of Radiological Concentrations and Unit Liter Doses for TWRS FSAR Radiological Consequence Calculations*, WHC-SD-WM-SARR-037, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

- Grove, 1987, *MICROSKYSHINE*, Grove Engineering, Inc., 15125 Shady Grove Road, Rockville, Maryland.
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- Rittman, P. D., 1992, *BREMCALC - A Computer Program for Calculating Electron and Positron Bremsstrahlung*, WHC-SA-1435-FP, Westinghouse Hanford Company, Richland, Washington.
- Van Keuren, J. C., 1996, WHC-SD-WM-SARR-016, Rev. 2, *Tank Waste Compositions and Atmospheric Dispersion Coefficients for use in ASA Consequence Assessments*, Westinghouse Hanford Company, Richland, Washington.

CASE 1 OUTPUT FILES

- Line-of-Sight
- Line-of-Sight Bremsstrahlung
- Skyshine
- Skyshine Bremsstrahlung

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MicroShield 4.00 - Serial #4.00-00128

Westinghouse Hanford Company

Page : 1
 DOS File: CASE1DIR.MS4
 Run Date: July 17, 1996
 Run Time: 8:26 a.m. Wednesday
 Duration: 0:00:20

File Ref: _____
 Date: ____/____/____
 By: _____
 Checked: _____

Case Title: Case 1 Line-of-Sight Dose

GEOMETRY 7 - Cylinder Volume - Side Shields

	centimeters	feet and inches	
Dose point coordinate X:	10000.0	328.0	1.0
Dose point coordinate Y:	150.0	4.0	11.1
Dose point coordinate Z:	0.0	0.0	.0
Cylinder height:	2.5	0.0	1.0
Cylinder radius:	520.0	17.0	.7
Air Gap:	9480.0	311.0	.3

Source Volume: 2.12372e+6 cm³ 74.9983 cu ft. 129597. cu in.

MATERIAL DENSITIES (g/cm³)

Material	Source Shield	Transition Shield	Air Gap	Immersion Shield
Air		0.00122	0.00122	0.00122
Water	1.3			

BUILDUP

Method: Buildup Factor Tables
 The material reference is Transition

INTEGRATION PARAMETERS

	Quadrature Order
Radial	10
Circumferential	10
Axial (along Z)	10

SOURCE NUCLIDES

Nuclide	curies	$\mu\text{Ci}/\text{cm}^3$	Nuclide	curies	$\mu\text{Ci}/\text{cm}^3$
Ba-137m	3.1973e+003	1.5055e+003	Cs-137	3.3798e+003	1.5914e+003
Eu-154	2.4095e+000	1.1346e+000			

Page : 2
 DOS File: CASE1DIR.MS4
 Run Date: July 17, 1996
 Run Time: 8:26 a.m. Wednesday
 Title : Case 1 Line-of-Sight Dose

===== RESULTS =====					
Energy (MeV)	Activity (photons/sec)	Energy Fluence Rate (MeV/sq cm/sec)		Exposure Rate In Air (mR/hr)	
		No Buildup	With Buildup	No Buildup	With Buildup
0.1	3.607e+010	1.427e-002	3.444e-001	2.183e-005	5.269e-004
0.2	6.088e+009	8.513e-003	1.110e-001	1.502e-005	1.959e-004
0.4	6.360e+008	3.232e-003	2.159e-002	6.297e-006	4.207e-005
0.5	1.930e+008	1.488e-003	8.228e-003	2.922e-006	1.615e-005
0.6	1.065e+014	1.154e+003	5.517e+003	2.253e+000	1.077e+001
0.8	3.477e+010	6.454e-001	2.502e+000	1.228e-003	4.760e-003
1.0	2.743e+010	7.718e-001	2.593e+000	1.423e-003	4.780e-003
1.5	3.479e+010	2.076e+000	5.564e+000	3.493e-003	9.362e-003
TOTAL:	1.066e+014	1.158e+003	5.528e+003	2.259e+000	1.079e+001

MicroShield 4.00 - Serial #4.00-00128

Westinghouse Hanford Company

Page : 1
 DOS File: CASE1DBR.MS4
 Run Date: July 17, 1996
 Run Time: 9:07 a.m. Wednesday
 Duration: 0:00:34

File Ref: _____
 Date: ____/____/____
 By: _____
 Checked: _____

Case Title: Case 1 Bremsstrahlung Dose

GEOMETRY 7 - Cylinder Volume - Side Shields

	centimeters	feet and inches	
Dose point coordinate X:	10000.0	328.0	1.0
Dose point coordinate Y:	150.0	4.0	11.1
Dose point coordinate Z:	0.0	0.0	.0
Cylinder height:	2.5	0.0	1.0
Cylinder radius:	520.0	17.0	.7
Air Gap:	9480.0	311.0	.3

Source Volume: 2.12372e+6 cm³ 74.9983 cu ft. 129597. cu in.

MATERIAL DENSITIES (g/cm³)

Material	Source Shield	Transition Shield	Air Gap	Immersion Shield
Air		0.00122	0.00122	0.00122
Water	1.3			

BUILDUP

Method: Buildup Factor Tables
 The material reference is Transition

INTEGRATION PARAMETERS

	Quadrature Order
Radial	10
Circumferential	10
Axial (along Z)	10

SOURCE WAS ENTERED AS ENERGIES ONLY

Page : 2
 DOS File: CASE1DBR.MS4
 Run Date: July 17, 1996
 Run Time: 9:07 a.m. Wednesday
 Title : Case 1 Bremsstrahlung Dose

===== RESULTS =====					
Energy (MeV)	Activity (photons/sec)	Energy Fluence Rate (MeV/sq cm/sec)		Exposure Rate In Air (mR/hr)	
		No Buildup	With Buildup	No Buildup	With Buildup
0.015	2.100e+011	9.083e-012	1.557e-011	7.791e-013	1.336e-012
0.025	1.100e+011	8.954e-005	3.383e-004	1.544e-006	5.834e-006
0.035	6.900e+010	1.311e-003	1.060e-002	8.306e-006	6.712e-005
0.045	4.800e+010	2.933e-003	4.246e-002	9.753e-006	1.412e-004
0.055	3.700e+010	4.178e-003	8.578e-002	9.404e-006	1.931e-004
0.065	2.900e+010	4.900e-003	1.194e-001	8.894e-006	2.167e-004
0.075	2.300e+010	5.255e-003	1.357e-001	8.565e-006	2.212e-004
0.085	2.000e+010	5.837e-003	1.502e-001	9.068e-006	2.333e-004
0.095	1.700e+010	6.116e-003	1.515e-001	9.359e-006	2.319e-004
0.15	8.700e+010	7.202e-002	1.250e+000	1.186e-004	2.059e-003
0.25	3.400e+010	7.178e-002	7.477e-001	1.324e-004	1.379e-003
0.35	1.800e+010	7.122e-002	5.368e-001	1.374e-004	1.036e-003
0.475	1.500e+010	1.051e-001	6.060e-001	2.062e-004	1.189e-003
0.65	9.200e+009	1.159e-001	5.210e-001	2.249e-004	1.011e-003
0.825	3.700e+009	7.275e-002	2.763e-001	1.379e-004	5.236e-004
1.0	2.600e+009	7.317e-002	2.459e-001	1.349e-004	4.532e-004
1.225	1.500e+009	6.157e-002	1.837e-001	1.090e-004	3.251e-004
1.475	5.500e+008	3.182e-002	8.601e-002	5.378e-005	1.454e-004
1.7	1.400e+008	1.051e-002	2.654e-002	1.707e-005	4.312e-005
1.9	3.700e+007	3.400e-003	8.177e-003	5.343e-006	1.285e-005
2.1	4.000e+006	4.406e-004	1.015e-003	6.709e-007	1.546e-006
2.3	2.200e+004	2.854e-006	6.337e-006	4.220e-009	9.371e-009
TOTAL:	7.347e+011	7.203e-001	5.185e+000	1.343e-003	9.490e-003

MicroSkyshine

=====

(Nuclear & Radiological Safety Analysis - 1.16-007)

Page: 1

File Ref: _____

File: POOLCS1.SKY

Date: ____/____/____

Run: 7:03 a.m.

By: _____

: July 17, 1996

Checked: _____

CASE: Case 1 Skyshine Dose

GEOMETRY: Vertical cylinder area source behind a wall

DIMENSIONS (meters):

Distance between wall and detector.....	X	93.8
Depth of source behind wall.....	Y	1.5
Offset of detector.....	Z	0.
Depth of dose point.....	H	0.
Distance between center of source and wall...	R1	6.2
Thickness of cover slab.....	T1	0.
Thickness of second shield.....	T2	0.
Radius of source.....	W	5.2
Height of source.....	L	0.025

INTEGRATION PARAMETERS:

Number of Radial Segments.....	M	5
Number of Circumferential Segments.....	N	5
Number of Vertical Segments.....	C	5
Quadrature Order.....		8

MATERIAL DENSITIES (g/cc):

Ambient air: .0012

Material	Cover Slab	Lower Shield	Volume Source
-----	-----	-----	-----
Air			
Water			1.3
Concrete			
Iron			
Lead			
Zirconium			
Urania			

Buildup factor based on: AIR.

Page 2

CASE: Case 1 Skyshine Dose

SOURCE NUCLIDES:

Nuclide	Curies	Nuclide	Curies
Ba-137m	3.1973e+03	Cs-137	3.3798e+03
Eu-154	2.4095e+00		

RESULTS:

Group #	Energy (mev)	Activity (photons/sec)	Dose point rads/photon	Dose rate (mr/hr)
1	1.30	3.570e+10	4.104e-20	6.043e-03
2	1.01	2.577e+10	4.441e-20	4.720e-03
3	.84	1.637e+10	4.385e-20	2.960e-03
4	.66	1.065e+14	4.732e-20	2.078e+01
5	.48	8.702e+08	5.119e-20	1.837e-04
6	.40	1.867e+08	5.030e-20	3.872e-05
7	.24	5.886e+09	5.094e-20	1.236e-03
8	.20	2.025e+08	5.002e-20	4.177e-05
9	.12	3.607e+10	4.425e-20	6.581e-03
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
TOTALS:		1.066e+14		2.080e+01

MicroSkyshine

=====

(Nuclear & Radiological Safety Analysis - 1.16-007)

Page: 1

File Ref: _____

File: PLCS1BR.SKY

Date: ____/____/____

Run: 9:32 a.m.

By: _____

: July 17, 1996

Checked: _____

CASE: Case 1 Bremsstrahlung Skyshine Dose

GEOMETRY: Vertical cylinder area source behind a wall

DIMENSIONS (meters):

Distance between wall and detector.....	X	93.8
Depth of source behind wall.....	Y	1.5
Offset of detector.....	Z	0.
Depth of dose point.....	H	0.
Distance between center of source and wall...	R1	6.2
Thickness of cover slab.....	T1	0.
Thickness of second shield.....	T2	0.
Radius of source.....	W	5.2
Height of source.....	L	0.025

INTEGRATION PARAMETERS:

Number of Radial Segments.....	M	5
Number of Circumferential Segments.....	N	5
Number of Vertical Segments.....	C	5
Quadrature Order.....		8

MATERIAL DENSITIES (g/cc):

Ambient air: .0012

Material	Cover Slab	Lower Shield	Volume Source
-----	-----	-----	-----
Air			
Water			1.3
Concrete			
Iron			
Lead			
Zirconium			
Urania			

Buildup factor based on: AIR.

Page 2

CASE: Case 1 Bremmstrahlung Skyshine Dose

SOURCE NUCLIDES:

Source was entered by energy groups.

RESULTS:

Group #	Energy (mev)	Activity (photons/sec)	Dose point rads/photon	Dose rate (mr/hr)
1	2.30	2.200e+04	3.317e-20	3.009e-09
2	2.10	4.000e+06	3.489e-20	5.754e-07
3	1.90	3.700e+07	3.584e-20	5.468e-06
4	1.70	1.400e+08	3.603e-20	2.080e-05
5	1.48	5.500e+08	3.626e-20	8.223e-05
6	1.23	1.500e+09	4.245e-20	2.626e-04
7	1.00	2.600e+09	4.442e-20	4.762e-04
8	.82	3.700e+09	4.369e-20	6.665e-04
9	.65	9.200e+09	4.789e-20	1.817e-03
10	.47	1.500e+10	5.119e-20	3.166e-03
11	.35	1.800e+10	4.913e-20	3.647e-03
12	.25	3.400e+10	5.090e-20	7.136e-03
13	.15	8.700e+10	4.760e-20	1.708e-02
14				
15				
16				
17				
18				
19				
20				
TOTALS:		1.717e+11		3.436e-02

CASE 2 OUTPUT FILES

- Line-of-Sight
- Line-of-Sight Bremsstrahlung
- Skyshine
- Skyshine Bremsstrahlung

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MicroShield 4.00 - Serial #4.00-00128

Westinghouse Hanford Company

Page : 1
 DOS File: CASE2DIR.MS4
 Run Date: July 17, 1996
 Run Time: 8:30 a.m. Wednesday
 Duration: 0:00:20

File Ref: _____
 Date: / /
 By: _____
 Checked: _____

Case Title: Case 2 Line-of-Sight Dose

GEOMETRY 7 - Cylinder Volume - Side Shields

	centimeters	feet and inches	
Dose point coordinate X:	10000.0	328.0	1.0
Dose point coordinate Y:	150.0	4.0	11.1
Dose point coordinate Z:	0.0	0.0	.0
Cylinder height:	2.5	0.0	1.0
Cylinder radius:	3100.0	101.0	8.5
Air Gap:	6900.0	226.0	4.5

Source Volume: 7.54768e+7 cm³ 2665.44 cu ft. 4.60587e+6 cu in.

MATERIAL DENSITIES (g/cm³)

Material	Source Shield	Transition Shield	Air Gap	Immersion Shield
Air		0.00122	0.00122	0.00122
Water	1.3			

BUILDUP

Method: Buildup Factor Tables
 The material reference is Transition

INTEGRATION PARAMETERS

	Quadrature Order
Radial	10
Circumferential	10
Axial (along Z)	10

SOURCE NUCLIDES

Nuclide	curies	$\mu\text{Ci/cm}^3$	Nuclide	curies	$\mu\text{Ci/cm}^3$
Ba-137m	3.1973e+003	4.2361e+001	Cs-137	3.3798e+003	4.4779e+001
Eu-154	2.4095e+000	3.1923e-002			

Page : 2
 DOS File: CASE2DIR.MS4
 Run Date: July 17, 1996
 Run Time: 8:30 a.m. Wednesday
 Title : Case 2 Line-of-Sight Dose

===== RESULTS =====					
Energy (MeV)	Activity (photons/sec)	Energy Fluence Rate (MeV/sq cm/sec)		Exposure Rate In Air (mR/hr)	
		No Buildup	With Buildup	No Buildup	With Buildup
0.1	3.607e+010	1.606e-002	3.360e-001	2.457e-005	5.140e-004
0.2	6.088e+009	9.122e-003	1.073e-001	1.610e-005	1.894e-004
0.4	6.360e+008	3.323e-003	2.098e-002	6.475e-006	4.089e-005
0.5	1.930e+008	1.515e-003	8.019e-003	2.973e-006	1.574e-005
0.6	1.065e+014	1.166e+003	5.388e+003	2.276e+000	1.052e+001
0.8	3.477e+010	6.459e-001	2.453e+000	1.229e-003	4.666e-003
1.0	2.743e+010	7.684e-001	2.551e+000	1.416e-003	4.701e-003
1.5	3.479e+010	2.057e+000	5.510e+000	3.461e-003	9.271e-003
TOTAL:	1.066e+014	1.169e+003	5.399e+003	2.282e+000	1.054e+001

MicroShield 4.00 - Serial #4.00-00128

Westinghouse Hanford Company

Page : 1
 DOS File: CASE2DBR.MS4
 Run Date: July 17, 1996
 Run Time: 9:08 a.m. Wednesday
 Duration: 0:00:34

File Ref: _____
 Date: / /
 By: _____
 Checked: _____

Case Title: Case 2 Bremsstrahlung Dose

GEOMETRY 7 - Cylinder Volume - Side Shields

	centimeters	feet and inches	
Dose point coordinate X:	10000.0	328.0	1.0
Dose point coordinate Y:	150.0	4.0	11.1
Dose point coordinate Z:	0.0	0.0	.0
Cylinder height:	2.5	0.0	1.0
Cylinder radius:	3100.0	101.0	8.5
Air Gap:	6900.0	226.0	4.5

Source Volume: 7.54768e+7 cm³ 2665.44 cu ft. 4.60587e+6 cu in.

MATERIAL DENSITIES (g/cm³)

Material	Source Shield	Transition Shield	Air Gap	Immersion Shield
Air		0.00122	0.00122	0.00122
Water	1.3			

BUILDUP

Method: Buildup Factor Tables
 The material reference is Transition

INTEGRATION PARAMETERS

	Quadrature Order
Radial	10
Circumferential	10
Axial (along Z)	10

SOURCE WAS ENTERED AS ENERGIES ONLY

Page : 2
 DOS File: CASE2DBR.MS4
 Run Date: July 17, 1996
 Run Time: 9:08 a.m. Wednesday
 Title : Case 2 Bremsstrahlung Dose

===== RESULTS =====					
Energy (MeV)	Activity (photons/sec)	Energy Fluence Rate (MeV/sq cm/sec)		Exposure Rate In Air (mR/hr)	
		No Buildup	With Buildup	No Buildup	With Buildup
0.015	2.100e+011	5.893e-010	9.713e-010	5.054e-011	8.331e-011
0.025	1.100e+011	1.863e-004	6.488e-004	3.213e-006	1.119e-005
0.035	6.900e+010	1.835e-003	1.333e-002	1.163e-005	8.445e-005
0.045	4.800e+010	3.687e-003	4.675e-002	1.226e-005	1.555e-004
0.055	3.700e+010	5.022e-003	8.921e-002	1.130e-005	2.008e-004
0.065	2.900e+010	5.747e-003	1.205e-001	1.043e-005	2.188e-004
0.075	2.300e+010	6.067e-003	1.348e-001	9.888e-006	2.197e-004
0.085	2.000e+010	6.660e-003	1.477e-001	1.035e-005	2.295e-004
0.095	1.700e+010	6.913e-003	1.481e-001	1.058e-005	2.267e-004
0.15	8.700e+010	7.869e-002	1.212e+000	1.296e-004	1.996e-003
0.25	3.400e+010	7.582e-002	7.237e-001	1.399e-004	1.335e-003
0.35	1.800e+010	7.376e-002	5.211e-001	1.423e-004	1.005e-003
0.475	1.500e+010	1.072e-001	5.903e-001	2.103e-004	1.158e-003
0.65	9.200e+009	1.167e-001	5.094e-001	2.266e-004	9.890e-004
0.825	3.700e+009	7.275e-002	2.709e-001	1.379e-004	5.134e-004
1.0	2.600e+009	7.285e-002	2.418e-001	1.343e-004	4.457e-004
1.225	1.500e+009	6.111e-002	1.812e-001	1.082e-004	3.208e-004
1.475	5.500e+008	3.154e-002	8.516e-002	5.330e-005	1.439e-004
1.7	1.400e+008	1.041e-002	2.635e-002	1.691e-005	4.279e-005
1.9	3.700e+007	3.369e-003	8.131e-003	5.292e-006	1.278e-005
2.1	4.000e+006	4.366e-004	1.012e-003	6.649e-007	1.541e-006
2.3	2.200e+004	2.830e-006	6.324e-006	4.185e-009	9.353e-009
TOTAL:	7.347e+011	7.407e-001	5.072e+000	1.385e-003	9.311e-003

MicroSkyshine

=====

(Nuclear & Radiological Safety Analysis - 1.16-007)

Page: 1

File Ref: _____

File: POOLCS2.SKY

Date: ____/____/____

Run: 7:01 a.m.

By: _____

: July 17, 1996

Checked: _____

CASE: Case 2 Skyshine Dose

GEOMETRY: Vertical cylinder area source behind a wall

DIMENSIONS (meters):

Distance between wall and detector.....	X	68.
Depth of source behind wall.....	Y	1.5
Offset of detector.....	Z	0.
Depth of dose point.....	H	0.
Distance between center of source and wall...	R1	32.
Thickness of cover slab.....	T1	0.
Thickness of second shield.....	T2	0.
Radius of source.....	W	31.
Height of source.....	L	0.025

INTEGRATION PARAMETERS:

Number of Radial Segments.....	M	5
Number of Circumferential Segments.....	N	5
Number of Vertical Segments.....	C	5
Quadrature Order.....		8

MATERIAL DENSITIES (g/cc):

Ambient air: .0012

Material	Cover Slab	Lower Shield	Volume Source
-----	-----	-----	-----
Air			
Water			1.3
Concrete			
Iron			
Lead			
Zirconium			
Urania			

Buildup factor based on: AIR.

Page 2

CASE: Case 2 Skyshine Dose

SOURCE NUCLIDES:

Nuclide	Curies	Nuclide	Curies
Ba-137m	3.1973e+03	Cs-137	3.3798e+03
Eu-154	2.4095e+00		

RESULTS:

Group #	Energy (mev)	Activity (photons/sec)	Dose point rads/photon	Dose rate (mr/hr)
1	1.30	3.570e+10	6.749e-20	9.937e-03
2	1.01	2.577e+10	7.040e-20	7.482e-03
3	.84	1.637e+10	6.860e-20	4.630e-03
4	.66	1.065e+14	7.144e-20	3.137e+01
5	.48	8.702e+08	7.429e-20	2.666e-04
6	.40	1.867e+08	7.213e-20	5.553e-05
7	.24	5.886e+09	7.031e-20	1.706e-03
8	.20	2.025e+08	6.815e-20	5.691e-05
9	.12	3.607e+10	5.901e-20	8.778e-03
10				
11				
12				
13				
14				
15				
16				
17				
18				
19				
20				
TOTALS:		1.066e+14		3.140e+01

MicroSkyshine

=====

(Nuclear & Radiological Safety Analysis - 1.16-007)

Page: 1

File Ref: _____

File: PLCS2BR.SKY

Date: ____/____/____

Run: 9:47 a.m.

By: _____

: July 17, 1996

Checked: _____

CASE: Case 2 Bremsstrahlung Skyshine Dose

GEOMETRY: Vertical cylinder area source behind a wall

DIMENSIONS (meters):

Distance between wall and detector.....	X	68.
Depth of source behind wall.....	Y	1.5
Offset of detector.....	Z	0.
Depth of dose point.....	H	0.
Distance between center of source and wall...	R1	32.
Thickness of cover slab.....	T1	0.
Thickness of second shield.....	T2	0.
Radius of source.....	W	31.
Height of source.....	L	0.025

INTEGRATION PARAMETERS:

Number of Radial Segments.....	M	5
Number of Circumferential Segments.....	N	5
Number of Vertical Segments.....	C	5
Quadrature Order.....		8

MATERIAL DENSITIES (g/cc):

Ambient air: .0012

Material	Cover Slab	Lower Shield	Volume Source
-----	-----	-----	-----
Air			
Water			1.3
Concrete			
Iron			
Lead			
Zirconium			
Urania			

Buildup factor based on: AIR.

Page 2

CASE: Case 2 Bremsstrahlung Skyshine Dose

SOURCE NUCLIDES:

Source was entered by energy groups.

RESULTS:

Group #	Energy (mev)	Activity (photons/sec)	Dose point rads/photon	Dose rate (mr/hr)
1	2.30	2.200e+04	5.964e-20	5.410e-09
2	2.10	4.000e+06	6.172e-20	1.018e-06
3	1.90	3.700e+07	6.260e-20	9.551e-06
4	1.70	1.400e+08	6.231e-20	3.597e-05
5	1.48	5.500e+08	6.177e-20	1.401e-04
6	1.23	1.500e+09	6.903e-20	4.269e-04
7	1.00	2.600e+09	7.038e-20	7.545e-04
8	.82	3.700e+09	6.826e-20	1.041e-03
9	.65	9.200e+09	7.201e-20	2.732e-03
10	.47	1.500e+10	7.422e-20	4.591e-03
11	.35	1.800e+10	7.006e-20	5.200e-03
12	.25	3.400e+10	7.043e-20	9.874e-03
13	.15	8.700e+10	6.413e-20	2.301e-02
14				
15				
16				
17				
18				
19				
20				
TOTALS:		1.717e+11		4.781e-02

8.0 PEER REVIEW AND HEDOP REVIEW CHECKLISTS

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CHECKLIST FOR PEER REVIEW

Document Reviewed: "Calculation Notes That Support Accident Scenario and Consequence Development for the Leak From a Railcar/Tank Trailer at the 204-AR Waste Unloading Facility," WHC-SD-WM-CN-003, Rev. 1, 9/96, by Grant Ryan.

Scope of Review: Entire Document.

Yes No NA

☒	☐	☐	Previous reviews complete and cover analysis, up to scope of this review, with no gaps.
☒	☐	☐	Problem completely defined.
☒	☐	☐	Accident scenarios developed in a clear and logical manner.
☒	☐	☐	Necessary assumptions explicitly stated and supported.
☐	☐	☒	Computer codes and data files documented.
☒	☐	☐	Data used in calculations explicitly stated in document.
☒	☐	☐	Data checked for consistency with original source information as applicable.
☒	☐	☐	Mathematical derivations checked including dimensional consistency of results.
☒	☐	☐	Models appropriate and used within range of validity or use outside range of established validity justified.
☒	☐	☐	Hand calculations checked for errors. Spreadsheet results should be treated exactly the same as hand calculations.
☐	☐	☒	Software input correct and consistent with document reviewed.
☐	☐	☒	Software output consistent with input and with results reported in document reviewed.
☒	☐	☐	Limits/criteria/guidelines applied to analysis results are appropriate and referenced. Limits/criteria/guidelines checked against references.
☒	☐	☐	Safety margins consistent with good engineering practices.
☒	☐	☐	Conclusions consistent with analytical results and applicable limits.
☒	☐	☐	Results and conclusions address all points required in the problem statement.
☐	☐	☒	Format consistent with appropriate NRC Regulatory Guide or other standards
☐	☐	☒	Review calculations, comments, and/or notes are attached.
☒	☐	☐	Document approved.

Anthony V. Savino *Anthony Savino*
 Reviewer (Printed Name and Signature)

9/18/96
 Date

HEDOP REVIEW CHECKLIST
for
Radiological and Nonradiological Release Calculations

Document reviewed (include title or description of calculation, document number, author, and date, as applicable): "Calculation Notes That Support Accident Scenario and Consequence Development for the Leak From a Railcar/Tank Trailer at the 204-AR Waste Unloading Facility," WHC-SD-WM-CN-003, Rev. 1, 9/96, by Grant Ryan.

Submitted by: Grant Ryan

Date Submitted: 9/17/96

Scope of Review: Entire Document.

YES NO* N/A

[X] [] [] [] [X] [] [] [] []	[] [] [X] [X] [] [] [X] [] []	[] [] [X] [X] [] [] [X] [] [X]	1. A detailed technical review and approval of the environmental transport and dose calculation portion of the analysis has been performed and documented. 2. Detailed technical review(s) and approval(s) of scenario and release determinations have been performed and documented. 3. HEDOP-approved code(s) were used. 4. Receptor locations were selected according to HEDOP recommendations. 5. All applicable environmental pathways and code options were included and are appropriate for the calculations. 6. Hanford site data were used. 7. Model adjustments external to the computer program were justified and performed correctly. 8. The analysis is consistent with HEDOP recommendations. 9. Supporting notes, calculations, comments, comment resolutions, or other information is attached. (Use the "Page 1 of X" page numbering format and sign and date each added page.) 10. Approval is granted on behalf of the Hanford Environmental Dose Overview Panel.

* All "NO" responses must be explained and use of nonstandard methods justified.

Anthony V. Savino *Anthony Savino*
HEDOP-Approved Reviewer (Printed Name and Signature)

9/18/96
Date

COMMENTS (add additional signed and dated pages if necessary):