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WIPP Conceptual Design Report

Addendum G: Accident Analysis for Waste Isolation Pilot Plant (WIPP) Conceptual Design Report, by Henry C. Shefelbine, Nuclear Waste Engineering Division 1142, and James H. Metcalf, Instrumentation, Dosimetry and NTS Projects Division 3313
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ADDENDUM G

ACCIDENT ANALYSIS FOR WASTE ISOLATION PILOT PLANT (WIPP) CONCEPTUAL DESIGN REPORT

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Introduction

There are at least four basic types of accidents or risks pertinent to the WIPP:

1. Ordinary industrial accidents
2. Accidents that expose plant personnel to radioactivity
3. Accidents during the operational phase of the facility that expose the public to radioactivity
4. Long-term risks that are important long after the facility has been decommissioned.

The industrial accident potential for the WIPP will be comparable to the potentials of similar industrial activities of mining and warehousing for which there is considerable experience. Therefore, detailed analysis of the potential for ordinary industrial accidents is not appropriate in a conceptual design study. An analysis of accidents that expose facility personnel to radioactivity requires more detailed descriptions of the facilities and the operating procedures than those provided by conceptual design. Personnel must operate inside the envelope of many of the barriers mentioned in the conceptual design report, particularly in handling transuranic (TRU) waste. Thus, safety will depend heavily on the procedures developed by the operating contractor.

While explicit analyses of industrial accidents and accidents that expose plant personnel to radioactivity are not included, this does not mean that such accidents are ignored in the conceptual design. Both are implicit in many of the design features, such as:

- Using mining machines instead of the more conventional drilling and blasting technique to excavate the underground storage vaults
- Separating the ventilation system for the areas being mined from the ventilation for the waste storage areas
- Routing the flow of ventilation in the storage rooms so that any airborne radioactivity will generally be carried away from the people working in the rooms
- Providing safety devices on the hoists
- Providing an elaborate badging and monitoring system for all the plant staff
- Providing sufficient maneuvering room for the forklifts in the TRU waste building

The above list is by no means exhaustive, nor are the above features included solely out of safety considerations. However, they illustrate that personnel safety received attention in the conceptual design effort. The Safety Analysis Report (SAR), which is part of the Title I and II effort, will specifically address plant personnel safety.

Protecting the public from radioactive release also received considerable attention. Examples of design features that protect the public include:

- Designing all structures and components essential to preventing radioactive release to stringent earthquake- and tornado-resistant standards
- Filtering all the exhausts from areas containing radioactive waste with High Efficiency Particulate Air (HEPA) filters
- Installing an extensive radiation monitoring system both within the facility and in the surrounding area to detect any release

This report gives a preliminary but by no means complete evaluation of the effectiveness of these measures in protecting the public from radioactive release during the operational phase of the facility. The main purposes of this analysis are to assess the suitability of the proposed site boundaries, to provide input to the design of the HVAC system, and to verify the conceptual design from the standpoint of safety to the general public during the operational phase of the facility. A more detailed analysis will be included in the SAR of the Title I and II effort.

Long-term risk assessment is by far the most difficult to perform because the risks continue for thousands of years after the facility has been decommissioned. Many studies are under way or being planned to provide the needed input including

- Detailed surveys to define accurately the geology and hydrology of the proposed site
- Laboratory experiments to evaluate the interactions of the waste and the salt
- Computer simulations of the effect of the heat generated by the waste on the geologic formations

In fact, the WIPP itself must be considered as part of the overall long-term risk assessment. Many of the experiments can only be performed in situ. Hence, WIPP starts out as a pilot plant with retrievability incorporated in its design.

General Discussion and Results

Three accident scenarios are evaluated in this preliminary assessment of the possibility of exposing the public to a radioactive release during the operational phase of the facility:

1. The rupture of high-level waste (HLW) canisters
2. A fire in an underground TRU waste storage room
3. The rupture of a pipe in the liquid radwaste system

The following simplifying assumptions were made in performing the accident analyses:

- No attempt is made to estimate the probabilities of occurrence of the selected scenarios. Credible probability estimates require much more detail than is available in the conceptual design.
- The three accident scenarios are not selected from an exhaustive listing of all possible accidents.
- No attempt is made to evaluate the number of people who might be exposed to radiation. This requires consideration of the weather conditions and demography of the Carlsbad area.
- The hazards of these accidents to plant personnel are not considered.
- Except for the rupture of the pipe in the liquid radwaste system, final barriers to release (the air filtration system and the surface building structures) remain intact. Specifically, massive damage to the plant structures and any waste containers therein by earthquake, plane crash, or malevolent intent is not addressed.
- Only the effects of the activity inhaled from the radioactive cloud are considered. Off-site personnel will also be subjected to external radiation from the passing radioactive cloud. However, this effect will be much less significant than the inhaled activity.

The maximum dose accumulated in the first year after the accident, and the maximum 50-year dose commitment that would occur to a person in the off-site environs are given in Table I. These figures are based on very conservative estimates. More realistic estimates would reduce the figures by one or more orders of magnitude. Also, these figures are not representative of the typical dose that people in the Carlsbad area might experience. They only apply to a person standing at the point of maximum ground-level concentration, which is close to the point of release and depends on the prevailing weather conditions. The exposed individual is assumed to be an adult (20 years of age) at the time of intake who will live to an age of 70 years.

TABLE I

Maximum Dose for the Public in Case of Accident

	RH Facility Accident	TRU Storage Fire	Liquid Rad-waste Accident	10-CFR-100 Guidelines
Dose to organ in first year (rem)				
Total body	0.05	0.001	0.008	-
Bone	0.76	0.070	0.14	-
Lung	0.74	0.036	0.14	-
50-year dose commitment (rem)				
Total Body	1.00	0.068	0.19	25
Bone	17.0	2.8	3.25	-
Lung	0.84	0.071	0.16	-

The 10-CFR-100 guidelines apply to reactor site criteria,[13] and the whole body dose of 25 rem is a guideline dose for an individual standing at the boundary of the exclusion area for a period of 2 hours during a major accident such as a core meltdown. (The guideline also specifies a 300-rem maximum dose to the thyroid, but this will not apply to the WIPP because none of the wastes will contain appreciable amount of iodine.) This does not imply that 25 rem is an acceptable dose, but it does provide a guideline for evaluating reactor sites as to the effect of very low probability accidents. The three accident scenarios used in the analysis of the WIPP certainly qualify as low-probability accidents.

Accident Selection and Description

In selecting the accident scenarios, the approach was to identify large concentrations of radioactivity that occur in the normal flow of RH, TRU, and liquid radwaste. These areas are examined to identify accident environments that might apply and that could cause the suspension of some portion of the radioactivity as respirable particles. (Due to the nature of the waste and the types of operation performed at the WIPP, airborne release to the environment is by far the most likely route to the general public.)

RH Waste

The following RH waste accident scenarios were considered:

1. Dropping an HLW canister while unloading a rail shipping cask.
2. Dropping an HLW canister at some other point during its stay in the RH building.
3. A hoist failure drops three HLW canisters down the RH shaft.
4. An accident with the underground transporter that breaches the transporter cask shielding and ruptures an HLW canister.

Accidents 2 and 4 involving RH waste can be eliminated immediately because they involve a single canister, while 1 and 3 involve multiple canisters. Accident 1 could involve eight canisters—the dropped canister falls on the seven canisters remaining in the shipping cask. The third accident will involve a maximum of three canisters. The energy input to the canisters in the third accident is considerably greater, but the point of release (the RH level) is considerably farther from the surface than in the first scenario. Rather than perform a more detailed analysis to identify which scenario is the worse, they are assumed to be roughly equivalent. For the hoist failure, it is assumed that 4 percent of the curie inventory in each of the three canisters reaches the top of the RH shaft as respirable particles. For dropping a canister while unloading a shipping cask, it is assumed that 4 percent of the curie content of the dropped canister and 1 percent of the contents of each of the seven canisters left in the shipping cask are released as respirable particles. Thus, the total curie release for the two accident scenarios is assumed to be a maximum of 12 percent of the 800,000 curies in a single canister or 96,000 curies.

The fractions of the canister inventories that are assumed to be released as respirable particles depend to a great extent on the form of the waste. The release fractions assumed above are appropriate only for waste in the form of a calcine powder. Less than 4 percent by weight of the product from a fluidized-bed calciner will be less than 100 microns in diameter.[1] Particles larger than 30 to 50 microns will not remain airborne unless excessive air velocities are assumed. Thus, the assumed 4 percent release figure is conservative for wastes calcined in a fluidized bed. The 1 percent release for seven of the canisters involved in the accident while unloading the shipping cask is based on the assumption that these seven canisters will be protected from total loss of contents by the shipping cask. There are calcining processes (namely spray calcination) whose products are almost entirely particles in the 1 to 10 micron range.[2] It was felt, however, that this form of waste would be too hazardous to either ship or accept at the WIPP. In reality, it is unlikely that calcined HLW from any process will be accepted for terminal storage without further fixation. Glass monoliths or ceramic pellets are more likely waste forms. Waste in these forms would release much less than 4 percent weight as respirable particles in the above accident scenarios.

TRU Waste

The following TRU waste accident scenarios were considered:

1. Dropping a cargo container with 70 drums of waste while unloading an ATMX car.
2. Puncturing a DOT 19A or similar plywood box with a forklift tine.
3. A hoist failure that drops 24 drums down the TRU shaft.
4. A fire in a TRU storage room that consumes all the 216 RF plywood boxes that might be stored there.

For the TRU waste, the fourth accident (burning of the 216 RF plywood boxes) involves by far the greatest inventory. It is assumed that the fire releases 4 percent of the TRU nuclide contents of each box and that each box has the maximum allowed transuranic nuclide content of 350 g (equivalent to about 150 Ci per box for ERDA TRU waste generated by Rocky Flats.)[8] This gives a source term of 3000 g or 1300 Ci of transuranic nuclides. The 4 percent release is consistent with the analysis of fires in fuel fabrication and processing plants.[3]

Liquid Radwaste

The most serious accident involving liquid radwaste appears to be a pipe rupture outside the HEPA-protected buildings that occurs during the pumping of radwaste. The problem comes in identifying the most contaminated fluid. The RH facility generates a preponderance of the contaminated liquids because the shipping cask cooling water must be disposed of and the casks and canisters undergo routine decontamination washes. Since the shipping cask cooling water will be in contact with the waste canisters much longer than the decontamination wash water, it is almost certain to be more contaminated. However, there are specific limits placed on the amount of contamination that normally can be present in the water:[9] 0.1 μCi/mL for Group I nuclides: 5μCi/mL for Group II; and 300 μCi/mL for Groups III and IV. (It is conceivable that a canister will rupture during transport, causing higher contamination levels in the cooling water. However, the compound probability of a canister rupture and a liquid radwaste pipe break is assumed to be vanishingly small.) If the contamination in the liquid is assumed to have the same concentration per nuclide as the concentration in the solid HLW, the total curie concentration in the liquid cannot exceed 1.3 μCi/mL without exceeding the 0.1 μCi/mL for Group I nuclides. It is assumed that 800 gal of shipping cask cooling water (the average daily accumulation of such) is lost through the pipe rupture. The total source term assuming maximum contamination in the water is about 4 Ci with the same nuclide composition as the HLW.

Accident Analysis

The general approach used in analyzing the accidents is outlined below. Details of the calculations are given in Appendix A.

RH Facility Accident

The 96-kCi source term for the postulated dropped canister is passed through two stages of HEPA filters, each with an assumed efficiency of 99.5 percent. Thus, 2.4 Ci of activity are released to the atmosphere. (The conceptual design includes three stages of HEPA filters, and the efficiency of the filters is usually taken as 99.95.[3,4,14] Therefore, the efficiency of the filtration system used in the accident calculations is a factor of 200,000 lower than the nominal efficiency.) Assuming that the release occurs over a period of an hour and that the wind speed at the time is 1 m/s, the maximum ground-level concentration will be 0.14 $\mu Ci/m^3$. Depending on the meteorological conditions, the location of the maximum concentration can range from 200 to 2000 meters from the RH facility stack.[5,6] A person standing in the location of maximum concentration will inhale a maximum of 0.17 μCi. The composition of the HLW is assumed to be comparable to that of 10-year-old HLW generated in the processing of spent commercial reactor fuel.[7,10] Using this waste composition; and assuming the inhalation of a total activity of 0.17 μCi, the dose for the first year of exposure and the total dose commitment (50-year exposure) are calculated for the skeleton, the total body, and the lungs. These doses are calculated using dose conversion factors obtained from the INREM code[11], which is based on exposure models used by the International Commission on Radiological Protection (ICRP). The 50-year dose commitments due to the inhalation of 0.17 μCi are 1 rem to the total body, 17 rem to the bone, and 0.8 rem to the lungs, of which 0.05 rem, 0.76 rem, and 0.74 rem are accumulated by the respective organs in the first year.

TRU Waste Fire

Calculation of the inhaled quantity of transuranics parallels the RH accident calculations. After two stages of HEPA filtration, only 75 mg or 32 mCi of transuranics are released to the atmosphere. The maximum ground-level concentration is 4.2×10^{-9} g/m^3 (1.8 mCi/m^3) and the maximum inhaled quantity is 5.2×10^{-9}g (2.2 nCi). Assuming the activity inhaled has the isotopic composition of typical TRU waste generated by Rocky Flats,[8] the dose for the first year after exposure and the total dose commitments are calculated using the dose conversion factors from the INREM code.[14] Dose commitments due to the inhalation of 2.2 nCi are calculated to be 0.07 rem to the total body, 2.8 rem to the bone, and 0.07 rem to the lungs, of which 0.001 rem, 0.070 rem, and 0.036 rem are accumulated by the respective organs in the first year.

Liquid Radwaste Accident

The rupture of the pipe containing liquid radwaste is assumed to result in a pool of contaminated water on the surface of the ground. As the pipe will be buried in a closed piping run, it is unlikely that any of the liquid will reach the surface. Even if all 800 gal form a pool on the surface, it is very unlikely that the pool will extend beyond the site boundary, because the closest approach between the pipe in question and the site perimeter is more than 100 meters. Therefore, the activity in the radwaste must first become airborne before it presents a hazard to the public. The resuspension of surface contamination has been measured for a wide variety of situations. Resuspension factors (activity in the air in Ci/m^3 over activity on the surface in Ci/m^2) can range in value from 1×10^{-3} to 1×10^{-11}. For this accident, where the surface is at least damp, a 1×10^{-6} resuspension factor is assumed. The surface activity is calculated to be 0.013 Ci/m^2 by assuming that the pool is 1 cm deep, and that all the nuclides in the centimeter of water contribute to the surface activity. Thus, the airborne activity is 13 nCi/m^3. No atmospheric dilution is assumed because the individual could conceivably be standing close to the edge of the pool. However, it was assumed that the exposure would only last for 2 hours, which is consistent with the guidelines in 10-CFR-100.[13] The pool would certainly take considerably longer than 2 hours to evaporate or to be cleaned up. However, the plant staff will be notified of the accident almost immediately by either visual observation or the area radiation monitors. The first action would be to warn any off-site personnel who might be endangered. Exposures to people farther from the spill who might not receive a warning within 2 hours will be reduced by atmospheric dilution (more than a factor of 10^3 for distances greater than a km).[3] The activity inhaled by an individual standing near the edge of the pool of contaminated liquid for 2 hours is assumed to be 32 nCi. The isotopic content of the liquid radwaste is assumed to be that of the HLW. The total dose commitments due to the inhalation of 32 nCi are 0.19 rem to the total body, 3.25 rem to the bones, and 0.18 rem to the lungs of which 0.008 rem, 0.14 rem, and 0.14 rem are accumulated in the first year by the respective organs.

APPENDIX A

Accident Calculations

RH Facility Accidents

The source term for the design basis accident is easily calculated from the inventory in the canisters involved and the assumed percentage released.

Hoist failure accident:

Source = 3 canisters x 800,000 Ci/canister x 0.04 (fraction released)

= 96,000 Ci

Shipping cask accident:

Source = (1 canister x 800,000 Ci/canister x 0.04) + (7 canisters x 800,000 Ci/canister x 0.01) = 88,000 Ci

All succeeding calculations assume a release of 96,000 Ci from the canisters involved in the accident.

The amount of activity released to the atmosphere is calculated by multiplying the source term by the fraction of the activity which penetrates each stage of the HEPA filters (assume two stages of filtration, each with a removal efficiency of 99.5%):

Activity released to atmosphere = 96,000 Ci x $(0.005)^2$ = 2.4 Ci

Assuming that this activity is released over a period of 1 hour, the average activity released from the stack per unit time is

$$Q = \frac{2.4 \text{ Ci}}{3600 \text{ s}} = 6.7 \times 10^{-4} \text{ Ci/s}$$

The maximum ground-level concentration depends on the meteorological conditions at the time of the release and on the height of the release above the surface of the ground (stack height). Assuming a stack height of 30 meters, the diffusion factor ($X \bar{u}/Q$ where $\bar{u}$ is the average wind speed) is taken as 2×10^{-4} m^{-2}[5,6]. This is a maximum value based on an envelope of the curves for the various Pasquill diffusion categories. Using this diffusion factor and an assumed wind speed of 1 m/s, the ground level concentration at the assumed point of exposure is

$$X = 6.7 \times 10^{-4} \text{ Ci/s} \times 2 \times 10^{-4}/\text{m}^2 \div 1 \text{ m/s}$$

$$X = 1.34 \times 10^{-7} \text{ Ci/m}^3 = 0.134\ \mu\text{Ci/m}^3$$

The total activity inhaled by an individual standing at this location is thus determined by the product of the downwind concentration (X), the standard breathing rate ($3.47 \times 10^{-4}\ m^3/s$), and the duration of the exposure (1 hr).

$$\text{Inhaled Activity} = 0.134\ \mu Ci/m^3 \times 3.47 \times 10^{-4}\ m^3/s \times 3600\ s = 0.167\ \mu Ci$$

The HLW contains a large number of nuclides that contribute to the inhaled activity. In calculating the dose in rem, the difference of behavior of the various nuclides in the body has to be included. Table A-1 summarizes the dose calculations for the HLW accident.

<u>TRU Waste Fire</u>

The source term for the fire in the TRU storage vault is calculated by multiplying the maximum number of RF boxes in the storage vault by the maximum number of grams of transuranics allowed per box and by the assumed fraction of the transuranics released as respirable particles.

$$\text{Source term} = 216\ \text{boxes} \times 350\ g/\text{box} \times 0.04$$

$$= 3000\ g \text{ of transuranics}$$

The quantity of transuranics inhaled is calculated in the same manner as the activity of HLW inhaled.

$$\text{Released to atmosphere} = 3000\ g \times (0.005)^2 = 0.075\ g$$

$$\text{Release rate} = \frac{0.075\ g}{3600\ s} = 2.1 \times 10^{-5}\ g/s$$

$$\text{Ground-level concentration} = 2.1 \times 10^{-5}\ g/s \times 2 \times 10^{-4}\ s/m^2 \div 1\ m/s$$

$$= 4.2 \times 10^{-9}\ g/m^3$$

$$\text{Inhaled quantity} = 4.2 \times 10^{-9}\ g/m^3 \times 3.47 \times 10^{-4}\ m^3/s \times 3600\ s = 5.2 \times 10^{-9}\ g$$

Table A-2 summarizes the calculation of accumulated dose from the inhaled quantity of transuranics.

<u>Liquid Radwaste Accident</u>

In calculating the source term for the liquid radwaste accident, it is assumed that the shipping cask cooling water is contaminated to the maximum allowed by regulations[9] (0.1 µCi/mL of Group I nuclides, 5 µCi/mL for Group II nuclides, and 300 µCi/mL for nuclides in Groups III and IV), and that the nuclides in the water have the same composition as the HLW. Of the important nuclides listed in Table A-1, 47 percent of the total activity is contributed by nuclides in Group II, and 7.7 percent by nuclides in Group I. The total allowable activity of the cooling water is determined by the Group I nuclides.

$$\text{Total Activity} = \frac{0.1\ \mu Ci/mL}{0.077} = 1.3\ \mu Ci/mL$$

The source term for the accident is 800 gal of water contaminated with HLW nuclides to a total activity of 1.3 µCi/mL.

TABLE A-1

Organ Doses Received in Accident Involving High Level Waste

Nuclide	Fraction of Activity[7,10]	Activity Inhaled (µCi)	Dose to Organ in First Year[11]: Total Body (rem)	Dose to Organ in First Year[11]: Bone (rem)	Dose to Organ in First Year[11]: Lungs (rem)	Dose Commitment to Organ[11]: Total Body (rem)	Dose Commitment to Organ[11]: Bone (rem)	Dose Commitment to Organ[11]: Lungs (rem)
^{90}Sr	0.168	0.0277	0.003	0.048	0.029	0.073	1.233	0.033
^{90}Y	0.166	0.0277	-	-	0.001	-	-	0.001
$^{106}RuRh$	0.002	0.0003	-	-	-	-	-	-
^{125}Sb	0.003	0.0005	-	-	-	-	-	-
^{125m}Te	0.001	0.0002	-	-	-	-	-	-
^{134}Cs	0.023	0.0038	-	-	0.002	-	-	0.002
^{137}Cs	0.264	0.0441	0.001	0.002	0.017	0.001	0.003	0.020
^{147}Pm	0.029	0.0048	-	-	-	-	0.001	-
^{151}Sm	0.004	0.0007	-	-	-	-	-	-
^{154}Eu	0.014	0.0023	-	0.001	0.002	-	0.007	0.002
^{155}Eu	0.001	0.0002	-	-	-	-	-	-
^{237}Np	0.000015	2.5×10^{-6}	-	-	-	-	0.008	-
^{239}Np	0.001	1.37×10^{-4}	-	-	-	-	-	-
^{238}Pu	0.00014	2.34×10^{-5}	-	0.003	0.002	0.003	0.133	0.004
^{239}Pu	0.00003	5.01×10^{-6}	-	0.001	-	0.001	0.033	0.001
^{240}Pu	0.00003	5.01×10^{-6}	-	0.001	-	0.001	0.033	0.001
^{241}Pu	0.004	6.68×10^{-4}	-	0.005	-	0.002	0.082	-
^{241}Am	0.003	5.01×10^{-4}	0.002	0.023	0.028	0.068	1.037	0.031
^{243}Am	0.00003	5.01×10^{-6}	-	-	-	0.001	0.010	-
^{242}Cm	0.001	1.67×10^{-4}	-	0.007	0.007	0.001	0.059	0.007
^{243}Cm	0.00014	2.34×10^{-5}	-	0.001	0.001	0.002	0.041	0.002
^{244}Cm	0.068	1.14×10^{-2}	0.044	0.667	0.654	0.849	14.318	0.738
		TOTAL	0.050	0.759	0.743	1.002	16.948	0.842

TABLE A-2

Organ Doses Received Following Fire in TRU Waste Storage Area

Nuclide	Curies/g[8] of Transuranics	Activity Inhaled (pCi)	Dose to Organ in First Year[11] (rem)			Dose Commitment to Organ[11] (rem)		
			Total Body	Bone	Lungs	Total Body	Bone	Lungs
^{238}Pu	0.0052	27.04	-	0.004	0.003	0.004	0.155	0.005
^{239}Pu	0.0576	299.52	0.001	0.043	0.027	0.048	1.971	0.053
^{240}Pu	0.0137	71.24	-	0.010	0.006	0.011	0.468	0.013
^{241}Pu	0.3472	1805.44	-	0.013	-	0.005	0.225	-
^{241}Am	0.0006	3.12	-	-	-	-	0.006	-
		TOTAL	0.001	0.070	0.036	0.068	2.825	0.071

$$\text{Source term} = 800 \text{ gal} \times 3785 \frac{\text{mL}}{\text{gal}} \times 1.3\ \mu\text{Ci/mL} = 4 \text{ Ci}$$

Assuming that the pool averages 1 cm deep, the activity per m^2 is given by

$$\text{Activity per m}^2 = 1 \text{ cm} \times \frac{10^4 \text{ cm}^2}{\text{m}^2} \times 1.3\ \mu\text{Ci/cm}^3 = 1.3 \times 10^{-2} \text{ Ci/m}^2$$

Resuspension factors (defined as the ratio of resuspended air activity in Ci/m^3 to deposited ground activity in Ci/m^2) range from 1×10^{-3} per m to 1×10^{-11} per m. For this situation, where the ground will be at least damp, a resuspension factor of 10^{-6} per m seems to be applicable. Therefore, the air activity is given by

$$\text{Air Activity} = 1.3 \times 10^{-2} \text{ Ci/m}^2 \times 1 \times 10^{-6} \text{ Ci/m} = 0.013\ \mu\text{Ci/m}^3$$

It is assumed that the exposed person is close to the edge of the pool; hence, no credit is taken for dispersion of the activity. Therefore, the activity of the inhaled air is $0.013\ \mu Ci/m^3$. It is further assumed that the exposure persists for 2 hours.

The inhaled activity is the product of the ground-level concentration, the inhalation rate, and the assumed 2-hour exposure time.

$$\text{Inhaled Activity} = 0.013\ \mu\text{Ci/m}^3 \times 3.47 \times 10^{-4} \text{ m}^3\text{/s} \times 7200 \text{ s} = 32 \text{ nCi}$$

The calculation of the dose commitment from the inhaled activity is summarized in Table A-3.

TABLE A-3

Organ Doses Received in Accident Involving Liquid Radioactive Waste

Nuclide	Fraction of Activity[7,10]	Activity Inhaled (μCi)	Dose to Organ in First Year[11] (rem)			Dose Commitment to Organ[11] (rem)		
			Total Body	Bone	Lungs	Total Body	Bone	Lungs
^{90}Sr	0.166	5.32×10^{-3}	-	0.010	0.006	0.014	0.236	0.006
^{90}Y	0.166	5.32×10^{-3}	-	-	-	-	-	-
$^{106}RuRh$	0.002	6.4×10^{-5}	-	-	-	-	-	-
^{125}Sb	0.003	9.6×10^{-5}	-	-	-	-	-	-
^{125m}Te	0.001	3.2×10^{-5}	-	-	-	-	-	-
^{134}Cs	0.023	7.36×10^{-4}	-	-	-	-	-	-
^{137}Cs	0.264	8.44×10^{-3}	-	-	0.004	-	-	0.004
^{147}Pm	0.029	9.28×10^{-4}	-	-	-	-	-	-
^{151}Sm	0.004	1.28×10^{-4}	-	-	-	-	-	-
^{154}Eu	0.014	4.48×10^{-4}	-	-	-	-	0.002	-
^{155}Eu	0.001	3.2×10^{-5}	-	-	-	-	-	-
^{237}Np	0.000015	4.8×10^{-7}	-	-	-	-	0.002	-
^{239}Np	0.001	3.2×10^{-5}	-	-	-	-	-	-
^{238}Pu	0.00014	4.48×10^{-6}	-	-	-	-	0.026	-
^{239}Pu	0.00003	9.6×10^{-7}	-	-	-	-	0.006	-
^{240}Pu	0.00003	9.6×10^{-7}	-	-	-	-	0.006	-
^{241}Pu	0.004	1.28×10^{-4}	-	-	-	-	0.016	-
^{241}Am	0.003	9.6×10^{-5}	-	0.004	0.006	0.014	0.200	0.006
^{243}Am	0.00003	9.6×10^{-7}	-	-	-	-	0.002	-
^{242}Cm	0.001	3.2×10^{-5}	-	0.002	0.002	-	0.002	0.002
^{243}Cm	0.00014	4.48×10^{-6}	-	-	-	-	0.008	-
^{244}Cm	0.068	2.18×10^{-3}	0.008	0.128	0.126	0.162	2.746	0.142
		TOTAL	0.008	0.144	0.144	0.190	3.252	0.160

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