PROBABILISTIC SAFETY ASSESSMENT
OF OPERATIONAL ACCIDENTS AT THE
WASTE ISOLATION PILOT PLANT

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September 2000

FOREWORD

The purpose of the New Mexico Environmental Evaluation Group (EEG) is to conduct an independent technical evaluation of the Waste Isolation Pilot Plant (WIPP) Project to ensure the protection of the public health and safety and the environment. The WIPP Project, located in southeastern New Mexico, is being constructed as a repository for the disposal of transuranic (TRU) radioactive wastes generated by the national defense programs. The EEG was established in 1978 with funds provided by the U.S. Department of Energy (DOE) to the State of New Mexico. Public Law 100-456, the National Defense Authorization Act, Fiscal Year 1989, Section 1433, assigned EEG to the New Mexico Institute of Mining and Technology and continued the original contract DE-AC04-79AL10752 through DOE contract DE-AC04-89AL58309. The National Defense Authorization Act for Fiscal Year 1994, Public Law 103-160, continues authorization.

EEG performs independent technical analyses of the suitability of the proposed site, the design of the repository, its operation, and its long-term integrity, suitability and safety of the transportation systems, suitability of the Waste Acceptance Criteria and the generator sites' compliance with them, and related subjects. These analyses include assessments of reports issued by the DOE and its contractors and other federal agencies and organizations as they relate to the potential health, safety, and environmental impacts from WIPP. Another important function of EEG is the independent environmental monitoring of radioactivity in air, water, and soil, both on-site and off-site.

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ACKNOWLEDGEMENTS

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EXECUTIVE SUMMARY

This report presents a probabilistic safety assessment of radioactive doses as consequences from accident scenarios was conducted to complement the deterministic assessment presented in the Waste Isolation Pilot Plant (WIPP) Safety Analysis Report (SAR). The International Council of Radiation Protection (ICRP) recommends both assessments be conducted to ensure that "an adequate level of safety has been achieved and that no major contributors to risk are overlooked" (ICRP 1993). To that end, the probabilistic assessment for the WIPP accident scenarios addresses the wide range of assumptions, e.g. the range of values representing the radioactive source of an accident, that could possibly have been overlooked by the SAR.

Routine releases of radionuclides from the WIPP repository to the environment during the waste emplacement operations are expected to be essentially zero. In contrast, potential accidental releases from postulated accident scenarios during waste handling and emplacement could be substantial, which necessitates the need for radiological air monitoring and confinement barriers (DOE 1999). The WIPP Safety Analysis Report (SAR) calculated doses from accidental releases to the on-site (at 100 m from the source) and off-site (at the Exclusive Use Boundary and Site Boundary) public by a deterministic approach. This approach, as demonstrated in the SAR, uses single-point values of key parameters to assess the 50-year, whole-body committed effective dose equivalent (CEDE).

The basic assumptions used in the SAR to formulate the CEDE are retained for this report's probabilistic assessment. However, for the probabilistic assessment, single-point parameter values were replaced with probability density functions (PDF) and were sampled over an expected range. Monte Carlo simulations were run, in which 10,000 iterations were performed by randomly selecting one value for each parameter and calculating the dose. Statistical information was then derived from the 10,000 iteration batch, which included 5%, 50%, and 95% dose likelihood, and the sensitivity of each assumption to the calculated doses.

As one would intuitively expect, the doses from the probabilistic assessment for most scenarios were found to be much less than the deterministic assessment. The lower dose of the

probabilistic assessment can be attributed to a "smearing" of values from the high and low end of the PDF spectrum of the various input parameters.

The analysis also found a potential weakness in the deterministic analysis used in the SAR; a detail on drum loading was not taken into consideration. Waste emplacement operations thus far have handled drums from each shipment as a single unit, i.e. drums from each shipment are kept together. Shipments typically come from a single waste stream, and therefore the curie loading of each drum can be considered nearly identical to that of its neighbor. Calculations show that if there are large numbers of drums used in the accident scenario assessment, e.g. 28 drums in the waste hoist failure scenario (CH5), then the probabilistic dose assessment calculations will diverge from the deterministically determined doses. As it is currently calculated, the deterministic dose assessment assumes one drum loaded to the maximum allowable (80 PE-Ci), and the remaining are 10% of the maximum. The effective average of drum curie content is therefore less in the deterministic assessment than the probabilistic assessment for a large number of drums.

EEG recommends that the WIPP SAR calculations be revisited and updated to include a probabilistic safety assessment.

Probabilistic Safety Assessment of Operational Accidents
at the Waste Isolation Pilot Plant

1. INTRODUCTION

The Waste Isolation Pilot Plant (WIPP) is a repository, operated by the US Department of Energy (DOE), for disposal of defense transuranic waste. It is located in southeastern New Mexico, approximately 40 kilometers (26 miles) east of Carlsbad in the Delaware Basin. The repository is a mined room and pillar construction 655 m below the surface in the Salado Formation, which consists mainly of halite (rock salt) with alternating thin layers of anhydrite with clay seams. The WIPP was certified by the US Environmental Protection Agency (EPA) in May 1998 (EPA 1998) as showing compliance with the EPA standards (EPA 1993) and criteria (EPA 1996) for long term disposal. WIPP has been receiving nonmixed transuranic waste (radioactive waste without a significant hazardous waste component) since March 1999.

A RCRA Part B permit was also issued in late November 1999. The permit allows the DOE to dispose of mixed transuranic waste. The hazardous waste constituents of the mixed waste are mainly volatile organic compounds (VOC) in addition to heavy metals, solvents, and asbestos.

As with any nuclear facility, the handling of nuclear materials can pose a risk to onsite workers and to the off-site public. One risk at WIPP involves respirable alpha-emitting radionuclide particles that become airborne as the result of an accident and are subsequently inhaled into the lungs. The consequence from the accident can be assessed by calculating the committed effective dose equivalent (CEDE), which is simply the product of several parameters including breathing rate, source, dose conversion factor, and the site-specific air dispersion factor. A detailed discussion on the calculations is given in Chapter 3.

The probabilistic method used in assessing the expected dose to an individual, described in this report, is not meant to be a predictor of dose (Apostolakis 1978). The purpose of conducting these calculations is to compare with established guidelines to gauge safety. Typically, the assessment will use parameter values that are representative of worst-case or near worst-case

conditions. For example, the DOE (DOE 1994) suggests that stable, low wind speed values be used to compute the air dispersion factor. In this respect, the dose can be maximized and give indications of the worst case scenario. However, these types of calculations are inherently conservative, and are not intended to delineate the consequence during actual accidents. One cannot truly know the meteorological conditions for some time in the future, nor what facility operations are being conducted. Thus, one limitation of using probabilistic dose calculations as predictors of consequence is due to the great uncertainty in the values needed to assess dose.

1.1 Probabilistic Safety Assessment

Although it is impossible to predict future events, past records can be used to support assumptions about the future behavior of a system. For example, the past several years of meteorological measurements at WIPP indicate that the windspeed will be between 0.134 m/s and 16.08 m/s. These measurements, and the knowledge of how often these windspeeds occur, can be used to assess the consequence for future accidents. Although not a predictor, the method of using a range of windspeeds to capture the uncertainty of measurements can give an indication of more realistic doses. This method is called a probabilistic safety assessment (PSA), and is typically used with the deterministic assessment to "ensure that an adequate level of safety has been achieved and that no major contributors to risk are overlooked" (ICRP 1993).

The use of a PSA to assess inhalation doses at WIPP is the topic of this report. In contrast to a deterministic analysis, where single-point values representing either typical or worst-case site conditions are used, a PSA can capture the entire realm of likely values and quantitatively assess the probability of each outcome. The dose from a windspeed at 2 m/s, for example, is more likely to occur than a dose from a windspeed at 0.134 m/s, and would be indicative of onsite meteorological measurements.

The drawback of using a PSA to calculate doses is that there are no guidelines or standards by which to compare consequence. The guidelines of (DOE 1994) assumes that a deterministically derived dose is less than the given guidance value. A PSA, on the other hand, presents doses on a probability scale, and the choice of probabilistic dose used in comparison with either

established guidelines or with the deterministic analysis is rather arbitrary. For this probabilistic analysis, a 95% probability of dose was used to compare with the deterministically determined dose, but was chosen as simply being reasonable and generally accepted among statisticians. A 98% probability could have easily been used. To that end, the PSA was used to simply compare probabilistic and deterministic doses with one another and to assess the reasonableness of the assumptions used in the deterministic analysis.

A deterministic analysis of possible accident consequences was presented in the WIPP Safety Analysis Report (SAR) (DOE 1999). To successfully compare the doses calculated by the PSA to doses calculated by the deterministic assessment, the same methodology of accident analysis used in the SAR was incorporated into the PSA. For example, the SAR accident assessment of a waste container spontaneously exploding, in which one drum was involved in the accident, was retained for the PSA. The only difference between the analyses is that the PSA used a range of values based on actual measurements of meteorological conditions and drum curie content and sampled from those ranges in a Monte Carlo simulation, where the SAR assumed single-point, worst-case meteorological conditions and an estimated value for drum curie content.

1.2 Normal Operational Procedures

The contact-handled (CH) TRU waste will be shipped primarily in 55 gallon drums or steel waste boxes. Much of the waste within the drums and boxes is surrounded by multiple layers of plastic bags intended to mitigate the release of respirable particles during packaging, handling, and storage. Moreover, each drum is fitted with one carbon filter and each box is fitted with two carbon filters designed to allow the release of flammable gas while preventing the release of alpha-emitting particles.

Historically, the WIPP Waste Acceptance Criteria, Revisions 0 through 4, included a specific limit of 1% by weight of respirable sized particles (10 micron) in CH-TRU waste. Moreover, particles less than 200 microns were limited to less than 15%. Revision 5 of the WAC (DOE 1996) removed the requirement, leaving an unlimited amount of respirable material as acceptable for handling and emplacement in the facility.

The WIPP Land Withdrawal Act (1996) requires shipment to WIPP in transportation containers certified by the Nuclear Regulatory Commission (NRC). NRC has certified the TRUPACT II, which is a right circular cylinder with a screw top lid. The TRUPACT II provides double containment and is non-vented and it was extensively tested to verify that the package would not be easily breached in a transportation accident. The DOE plans shipment of the TRU-PACT-II by truck, although shipment by rail is still an option.

Upon arrival at WIPP, the waste drums or boxes will be unloaded from the TRUPACT-II in the Waste Handling Building (WHB). The facility is equipped with alpha detecting continuous air monitors and in the event of a release, is also fitted with high efficiency particulate filtration. A description of the ventilation used in the WHB can be found in the WIPP Safety Analysis Report (DOE 1999). The SAR lists the accident scenarios, which could cause an uptake by workers in the WHB, underground, or in and around the above ground facilities.

For final disposal, the waste will be transported down to the mined out panels and emplaced. The mine ventilation system removes the large quantities of dust associated with the mining activity. As a result, during the normal mode of operation, air from the underground is unfiltered and returns to the surface through the exhaust shaft. The air is diverted through HEPA filters only when a release of radionuclides has been detected. Continuous air monitors are intended to provide the information to switch to HEPA filtration. Even if the monitoring system is functioning properly, it may take several minutes before a warning signal to notify the Central Monitoring Room (CMR) would result in the exhaust ventilation being switched to HEPA filtration. In such a case, the uptake of radioactivity by workers on the surface is dictated by the wind speed and stability conditions at the site.

2. ABNORMAL OPERATIONS AND ACCIDENTS

Although waste-handling personnel are trained to safely operate the equipment, hazardous operability studies (Westinghouse 1995) suggest the potential for handling accidents that could breach containers and release radioactive material. Two potential accidents at WIPP include dropping a drum from the waste facility pallet or the waste hoist failing and plummeting a substantial distance to the underground. Such accidents may result in possible worker exposure to radioactive material and that will be the focus of this report. Other accidents involving personal injury or equipment damage that do not release radioactive material will not be investigated here.

The SAR postulates several accident scenarios. Table I lists the possible accidents and the probability associated with each accident as stated in the SAR, Appendix D (DOE 1999). The frequency of occurrence is based on fault tree analysis.

Table I. Listing of Accident Scenarios in the WIPP Safety Analysis Report.

Scenario Description	Scenario Name	Accidents Probability
Spontaneous Ignition of Drum in WHB	CH1	1.3×10^{-8} /yr
Crane Failure in WHB	CH2*	9.8×10^{-3} /yr
Puncture of Waste Container in WHB	CH3	8.0×10^{-3} /yr
Waste Container Drop in WHB	CH4	1.5×10^{-2} /yr
Waste Hoist Failure	CH5**	1.9×10^{-9} /yr
Seismic Event	CH6	N.A.
Spontaneous Ignition of Drum Underground	CH7	4.8×10^{-7} /yr
Aircraft Crash	CH8	N.A.
Waste Container Drop Underground	CH9	2×10^{-2}
Tornado Event	CH10	N.A.
Underground Roof Fall	CH11	4.7×10^{-7} /yr

* EEG-74 (Greenfield and Sargent 2000) demonstrates the uncertainty associated with
   the expected annual frequency of a crane accident.
** EEG-44 (Greenfield 1990), EEG-59 (Greenfield and Sargent 1995), and EEG-65
   (Greenfield and Sargent 1998) demonstrates the uncertainty associated with the
   expected annual frequency of a waste hoist accident.

2.1 Accident Consequences And Mitigation

The consequence of each accident was calculated in the SAR by estimating the radioactive dose from potential uptake of airborne radioactivity into the body. The worker is subjected to a radioactive plume on the surface, outside of the WHB. The plume is a result of an accident above or underground and is released through the exhaust shaft or WHB ventilation. The dose calculations considered the source of each accident and how much material could be respirable, the breathing rate, and wind speed and plume dispersion at the site. Together, these terms form the committed effective dose equivalent (CEDE), which is calculated by

$$CEDE = D = Q * C/Q * BR * DCF \tag{1}$$

where D is simply the dose in rem[1], Q is the source term in plutonium-equivalent curies (PE-Ci)[2], C/Q is the site-specific air dispersion factor in s/m^3, BR is breathing rate (equal to 20 l/min as stipulated in IRCP-23 (ICRP 1974)), and DCF is the dose conversion factor for converting the source activity of Pu-239 to rem (assumed W[3] class, consistent with the SAR; $5.1x10^8$ rem/Ci).

Table II lists the consequence calculations reported in the SAR, Appendix E (DOE 1999) for each of the scenarios listed above. The table also shows the source term assumptions used in formulating the dose, and any effects of mitigation. Mitigation for the aboveground WHB facility includes continuous on-line High Efficiency Particulate (HEPA) filtration of ventilation, and underground mitigation includes switching the underground exhaust flow to HEPA filtration in the case of an accident. HEPA filtration for the underground exhaust cannot be continuously on-line since the additional pressure drop reduces the underground airflow, which falls far below levels allowed by the RCRA Subpart B permit and other MSHA standards for miners working with diesel equipment. For calculations of a mitigated release, the leakpath factor is considered

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[1] For consistency with the SAR, English units for dose (rem) and activity (Ci) are presented. The SI equivalents are Sieverts (Sv) for dose and Bequerels (Bq) for activity. 100 rem = 1 Sv. 1 Ci = $3.7x10^{10}$ Bq.
[2] All quantities of radionuclides in the waste are expressed as Pu-239 equivalent curies (PE-Ci). The PE-Ci is derived by comparing the 50-year effective whole-body dose commitment due to inhalation of various radionuclides to that of Pu-239.
[3] The W class is a simple classification for the chemical form of the radionuclide referring to the retention time in the body. W is short for weekly; others include D as in days and Y as in years.

to be 1×10^{-6} for an above ground release, i.e., a reduction of dose by 10^6 is expected. No credit for mitigation is taken for a release underground.

Table II. Dose calculations for an above ground worker from each accident scenario from the SAR, Appendix E.

Accident Scenario	Number of Drums Used in Source Term	Drum Loading in PE-Ci	Unmitigated Dose (rem) at 100 m	Mitigated Dose (rem) at 100 m
CH1	1	80	33	3.3×10^{-5}
CH2	7	1@80, 6@8	2.7	2.7×10^{-6}
CH2 *	1 (damaged)	1100	0.16	1.6×10^{-7}
CH2 **	7 (damaged)	7@80	12	1.2×10^{-5}
CH3	4	1@80, 3@8	3.8	3.8×10^{-6}
CH3 *	1 (damaged)	1100	0.15	1.5×10^{-7}
CH3 **	4 (damaged)	4 @ 80	8	8×10^{-6}
CH4	4	1 @ 80, 3 @ 8	0.86	8.6×10^{-7}
CH4 *	1 (damaged)	1100	0.0091	9.1×10^{-9}
CH4 **	4 (damaged)	4 @ 80	2.7	2.7×10^{-6}
CH5	28	1 @ 80, 27@ 8	61	6.1×10^{-5}
CH7	1	80	33	-
CH9	7	1 @ 80, 6 @ 8	2.7	-
CH9 *	1 (damaged)	1100	0.23	-
CH9 **	7 (damaged)	7 @ 80	1.2	-
CH11	21	1 @ 80, 20 @ 8	5.2	-

*   Dose consequences are calculated by estimating the effects of a damaged drum with a loading of 1100 PE-Ci. Drum loading above 80 PE-Ci are usually placed in a 85-gallon drum overpack or reprocessed for safer confinement.

** Dose consequences are calculated by assuming a worst case scenario. Drums above 80 PE-Ci are usually overpacked or reprocessed and it is postulated that the source from this drum load would represent the largest dose possible.

It should be noted that the curie content of each waste material type that will be shipped in the TRUPACT is limited by the decay heat content limit. This limit is dependent on the waste matrix and the number of plastic bags of containment. About two-thirds of the volume of CH-TRU waste will be limited to a maximum of 1.0 – 8.0 curies/drum. Only about one-third of the waste (solid inorganic material in metal cans) is allowed to contain greater than 80 Ci. Thus, several 80 PE-Ci drums in an accident scenario is a low probability event.

3. PROBABILISTIC DOSE MODELING

The use of probabilistic modeling has gained much attention in the last several years, especially in the nuclear industry with respect to radiation safety and potential exposure. The International Commission on Radiation Protection (ICRP) identified two complementary techniques to assess the potential exposure to individuals: deterministic and probabilistic assessment methods (ICRP 1993).

The more familiar of the two is the deterministic method, where an outcome is calculated based on one set of input values. This technique was used in the WIPP SAR, shown above in Table II, where the dose was calculated with one value for the source, breathing rate, dose conversion factor, and site-specific meteorological conditions. In addition, the sensitivity of certain variables was tested to evaluate the outcome to various conditions. For example, the drum loading parameter was changed to assess the doses that could be expected from the different drum curie content. The drum curie content was increased to calculate the effect of a worst case release, if there were an accident. Although the sensitivity of variables in Eq. 1 is rather intuitive, these 'What if' scenarios allow the modeler to bound the range of doses that could be expected.

Probabilistic dose modeling, also known as probabilistic safety analysis (PSA) is very similar to the deterministic modeling; the variables are still linked in the equations by addition or multiplication. However, the PSA addresses the 'What if' scenario automatically by generating a number of scenarios based on the probability of the input variables. Each input variable inherently has uncertainty in the measurement, and it is the goal of the PSA to give weight to a value of each variable by the value's probability of occurrence, which is characterized by a probability distribution.

The technique used here for assessing the probability distribution of each variable is the Monte Carlo simulation. The Monte Carlo simulation randomly selects (or samples) a value from the probability distribution of each variable to produce a multitude of scenarios (or iterations). The probability distribution is sampled in a manner that best reproduces the shape of the distribution

(Vose 1996), with the purpose of capturing the uncertainty of each variable. The greater number of iterations chosen, the higher the accuracy of that distribution.

Modeling the dose using Eq. 1 was set up using a Microsoft Excel spreadsheet and an after-market plug-in called Crystal Ball (Decisioneering 1998). Crystal Ball allows the user to specify a probability distribution for each input variable and the number of iterations to be performed. The result of modeling Eq. 1 was a forecast of doses including a confidence associated with those doses. Confidence allows one to gauge the accuracy of the results, and is usually displayed as a percentage. If, for example, the dose was calculated to be 3 rem for a set of Monte Carlo simulations with a confidence of 95%, one could be assured that 95% of the calculated doses fell at or below 3 rem. Described differently, one could be 95% certain (confident) that the dose would fall at or below 3 rem, assuming the input probability distributions are reasonable.

The precision of the PSA relies heavily on the choice of probability distribution chosen to represent the uncertainty of each variable (Vose 1996). Care must be used in formulating the function that best represents the empirical data in the model. The probability distribution function (PDF) therefore, must be adequately understood if the model results are to have any meaning.

3.1 WIPP Dose Model

The WIPP dose model, predicated on Eq. 1, was assigned PDFs for site specific meteorological conditions for the air dispersion factor (χ/Q) and source term (Q), but left the breathing rate (BR) and dose conversion factor (DCF) as constant values. The breathing rate for a worker under light activity (20 l/m or 3.34 m^3/s) was established in ICRP-23 (Hey 1994) and was used in the WIPP SAR. The DCF of 5.1×10^8 rem/Pu-239 Ci (DOE 1988), also from the SAR, was also retained for consistency.

3.2 Site-Specific Meteorological Conditions

In September 1996 DOE established well-sited meteorological towers for measuring atmospheric conditions at WIPP. The meteorological conditions will dictate the speed and direction of a contaminated, aerosolized particulate plume. For a more complete discussion, see Rucker (2000).

WIPP has two meteorological towers: a primary Meteorological Station and the WIPP Far Field (secondary meteorological monitoring station) (Westinghouse 1997). Figure 2.1 shows the proximity of the towers to WIPP boundaries. The primary station is located approximately 500 m (1640 ft) to the northeast of the exhaust shaft and houses a 50-meter (164-ft) instrument station. The secondary station is located approximately 1000 m to the northwest of the exhaust shaft and houses a 10-meter (32.8-ft) instrument station.
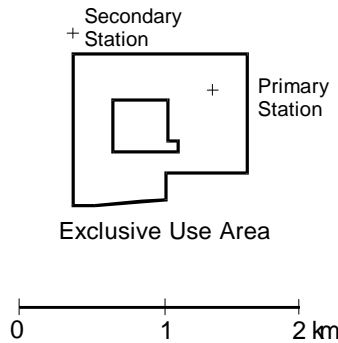

Figure 1. Location of WIPP Meteorological Towers.

The most significant use of WIPP meteorological data, other than establishing ambient conditions, is modeling a plume release from the exhaust shaft during an underground accident and release of aerosolized radioactive particles. The models used in estimating the concentration of a contaminant at a given point typically employ the Gaussian straight-line continuous plume transport equation for air dispersion. The equation assumes the site-specific, relative concentration factor (χ/Q) in [T/L^3],

$$\frac{\chi}{Q} = \frac{f(y)\,g(z)}{2\pi\sigma_y\sigma_z u} \tag{2}$$

where u is the wind speed [L/T], σ_y is the lateral dispersion factor [L], σ_z is the vertical dispersion factor [L], and f(y) and g(z) are horizontal and vertical correction factors, respectively, and the equation is governed by the wind speed, wind stability class, and stack height exclusively. f(y) and g(z) are represented by

$$f(y) = \exp\left[-\frac{1}{2}\left(\frac{y}{\sigma_y}\right)^2\right] \tag{3}$$

$$g(z) = \exp\left[-\frac{1}{2}\left(\frac{h_e - z}{\sigma_z}\right)^2\right] \tag{4}$$

where h_e is the effective plume height. The effective plume height is equal to the stack height, h_s, plus any change in height, Δh, due either to plume rise, stack downwash or gravitational settling. Figure 2 shows a graphical model of Gaussian plume dispersion.

The Gaussian nature of the plume spread in either lateral or vertical direction is represented by a dispersion factor, σ. This dispersion factor accounts for plume spread by mechanical and/or chemical mixing from empirical fitting formulae based on Pasquill-Gifford-Turner curves (Hey 1994). The curves for dispersion factors σ_y and σ_z are represented by the equations below.
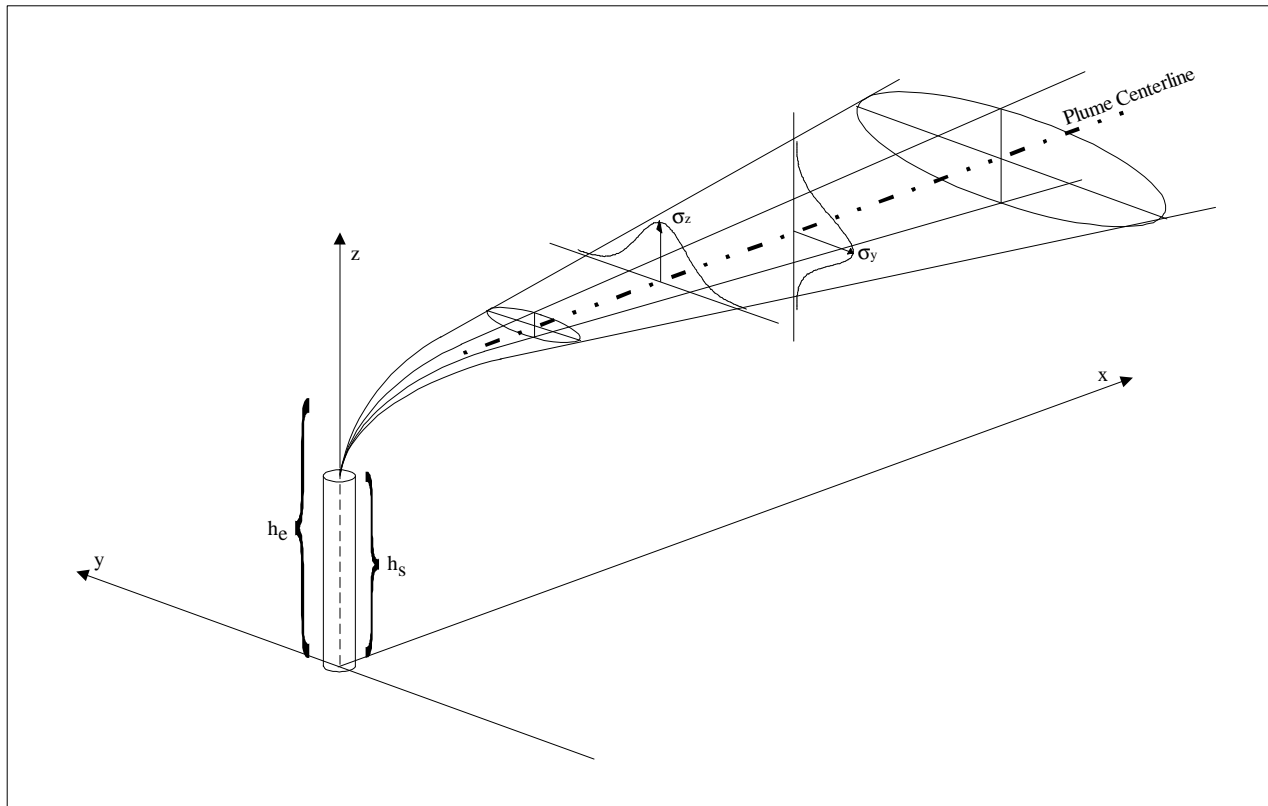
Figure 2. Gaussian Plume Dispersion Model

$$\sigma_y \; = \; A_y x^{0.9031} \tag{5}$$

$$\sigma_z \; = \; A_z x^{B_z} + C_z \tag{6}$$

where A,B,C are fitting coefficients and x is the downwind distance. The fitting coefficients are given in Table III for three distances: less than 100 m, 100 to 1000 m, and greater than 1000 m.

Table III. Fitting Coefficients for Dispersion Factors (Hey 1994)

Stability Class	A_y	A_z			B_z			C_z		
		<100	100 to 1000 m	>1000m	<100m	100 to 1000 m	>1000m	<100m	100 to 1000 m	>1000m
A	0.3658	0.192	0.00066	0.00024	0.936	1.941	2.094	0	9.27	-9.6
B	0.2751	0.156	0.0382	0.055	0.922	1.149	1.098	0	3.3	2
C	0.2089	0.116	0.113	0.133	0.905	0.911	0.911	0	0	0
D	0.1471	0.079	0.222	1.26	0.881	0.725	0.516	0	-1.7	-13
E	0.1046	0.063	0.211	6.73	0.871	0.678	0.305	0	-1.3	-34
F	0.0722	0.053	0.086	18.05	0.814	0.74	0.18	0	-0.35	-48.6
G	0.0481	0.032	0.052	10.83	0.814	0.74	0.18	0	-0.21	-29.2

Several numerical codes use these equations or similar equations (which may account for building wake effects, plume meander, etc.) to compute the concentration of a contaminant to a receptor downwind from the release point. As shown, Eqs. 3 and 4 do not incorporate plume meander or building wake effects.

The dispersion calculations conducted in the SAR for non-routine, accident release scenarios followed the format suggested in Nuclear Regulatory Guide (NRG) 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, November 1982 (NRC 1982). This a slight modification of the Gaussian model described above. The guide suggests a site-specific relative concentration (C/Q) be determined based on atmospheric conditions of the site. For neutral (D) or stable (E, F, or G) atmospheric stability conditions for windspeeds less than 6 m/s (3.1 knots) the C/Q value should follow the procedure described below.

$$\chi / Q = \frac{1}{\overline{U_{10}}(\pi\sigma_y\sigma_z + A/2)} \tag{7}$$

$$\chi / Q = \frac{1}{\overline{U_{10}}(3\pi\sigma_y\sigma_z)} \tag{8}$$

$$\chi / Q = \frac{1}{\overline{U_{10}}(\pi\Sigma_y\sigma_z)} \tag{9}$$

where,

> C/Q = relative concentration (s/m^3)
> U_{10} = windspeed (m/s) at 10 m above the ground
> σ_y = lateral plume spread (m), a function of atmospheric stability and distance
> σ_z = vertical plume spread (m), a function of atmospheric stability and distance
> Σ_y = lateral plume spread with meander and building wake effects (m), a function of
>   atmospheric stability and distance. For 800 m and less: $\Sigma_y = M\sigma_y$, where M is a

correction factor determined from a lookup chart (NRC 1982). For greater than 800 m: $\Sigma_y = (M-1)\sigma_{y800m} + \sigma_y$.

A = vertical plane cross-sectional area of release vent (m^2).

C/Q should be calculated using all three equations. The values from Eq. 7 should be compared to Eq. 8, and the largest value selected. The result is then compared to Eq. 9 and the smallest value selected. For wind stability classes of A, B, or C, only Eq. 7 and Eq. 8 need to be compared and the largest value selected.

Intuitively, it is easy to see that the conditional use of Eqs. 7-9 is not the most conservative combination; using the largest value of the three would always insure conservatism in the calculation of the site-specific air dispersion factor. However, the NRC RG 1.145 gives rationale for the selection of the most appropriate equation, citing 1) horizontal plume meander tends to dominate dispersion during light wind and stable or neutral conditions, and 2) building wake mixing becomes more effective in dispersing effluents than meander effects as the windspeed increases and the atmosphere becomes less stable (16).

The SAR reported a C/Q value of 5.11×10^{-3} s/m^3 at 100 m for conditions of stability class F, windspeed =1.5 m/s, σ_y = 4.6 m, σ_z = 2.3 m, M = 4, and A=117 m^2. At the time of the 1996 SAR update, sufficient meteorological data had not existed. The analyses relied on values of stability class and windspeed provided in DOE handbooks. However, since 1996, over three years of quality meteorological data exists and that data was used in formulating a PDF for windspeeds of the six stability classes. The next section describes the general meteorological conditions at WIPP.

3.3 General Description of Meteorological Conditions at WIPP
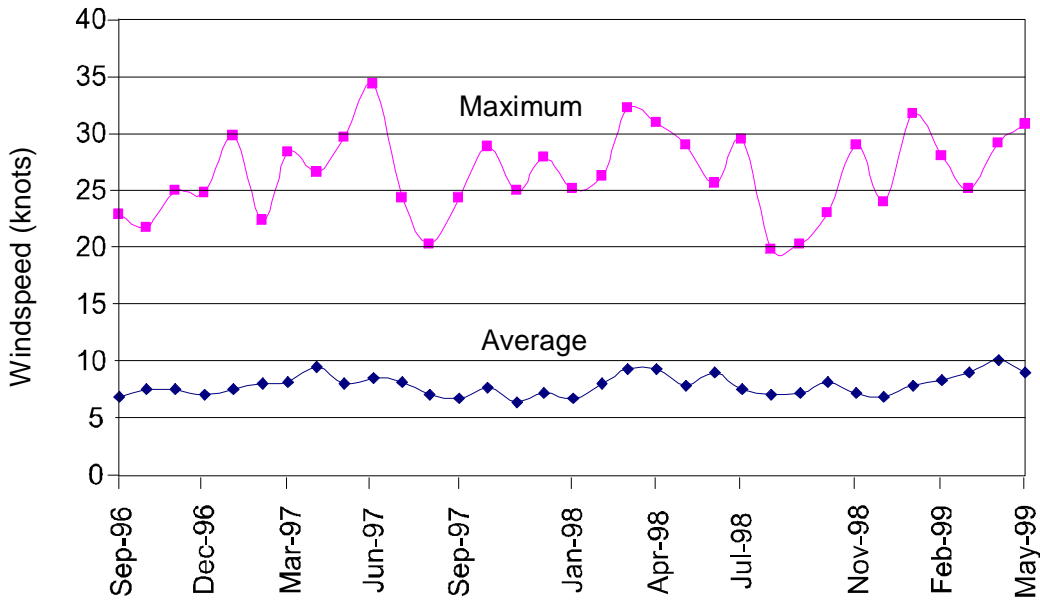The meteorological conditions at WIPP vary from day-to-day and from season-to-season.

Figure 3. Maximum and Average (over an entire month) Wind Speed at WIPP for the Period of September 1996 to May 1999.

Generally, however, one could expect variable wind speeds from the southeast with approximately 12 inches of precipitation per year. The region is characterized by an arid to semiarid climate, as is most of the southwestern United States.

3.3.1 Wind Speed

The windspeed around the WIPP site generally varies between 0 to 16 knots, averaging approximately 8 knots, and rising as high as 34 knots. Spring is the windiest time in the region, with April and May having the highest wind speeds, whereas summer brings about calmer winds. Figure 3 shows the average and maximum windspeeds for the period of September 1996 to May 1999.

Wind speed is important in three regards: dilution of a contaminant plume, which takes place in the direction of the plume transport; the transport time of the plume to the receptor (person or measuring device subjected to plume hazard); and buoyant plume rise – the stronger the wind the lower the plume.

3.3.2 Air Temperature

The air temperature at WIPP is temperate, rarely dropping below 20 ^{o}F (-6.7 ^{o}C) in the winter months. The maximum air temperature can reach as high as 120 ^{o}F (48.9 ^{o}C) in late summer. Figure 4 shows the minimum, average, and maximum monthly temperatures at WIPP for the months spanning September 1996 to May 1999, taken at the 10m level.
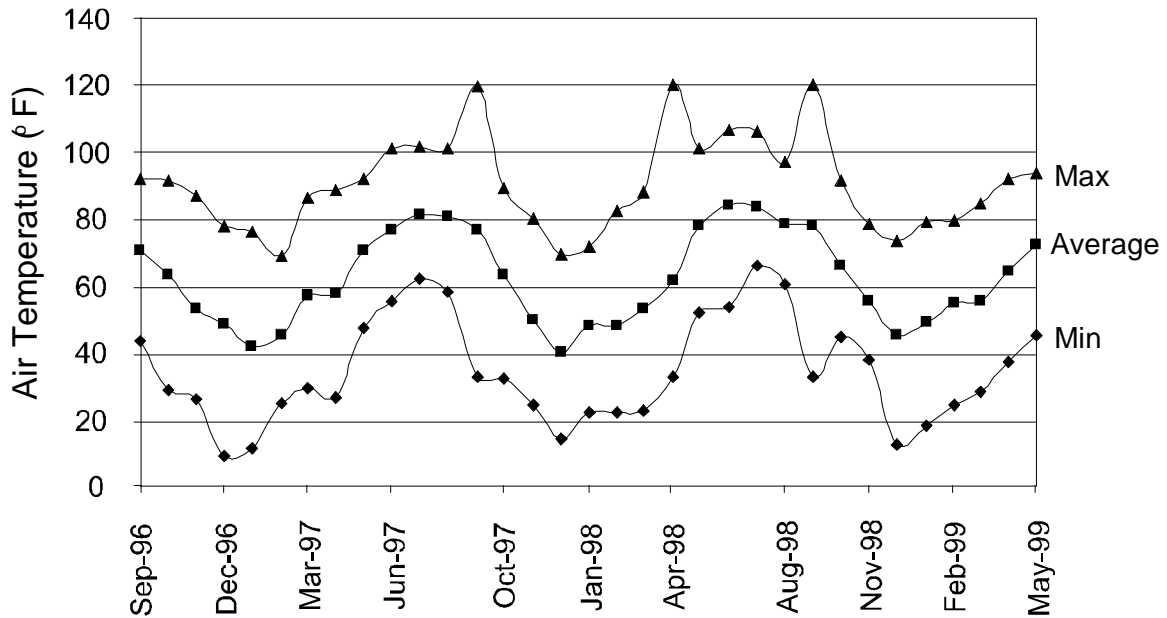

Figure 4. Monthly Minimum, Average, and Maximum Air Temperatures at WIPP for the Period of September 1996 to May 1999.

The wind stability class is a measure of resistance to vertical mixing. The wind stability is usually classified by determining the change in temperature to the change in height (IAEA 1980, EPA 1987). Usually very stable air is undesirable; pollutants are not able to dilute through mixing and transport.

The stability classes are defined in Table IV. A calculation is made, determining the change in temperature per 100 m. For the WIPP classification, the temperature difference was taken at 50 m and 10 m and the conversion is demonstrated in Eq. 10.

DT/Dz = (T$_{50}$ − T$_{10}$)*2.5   (10)

The stability class at WIPP spans the whole spectrum from A to G, but is neutral or stable about two-thirds of the time. Table V shows the percentage of time that the wind is in each category.

Table IV. Classification of Atmospheric Stability (NRC 1972)

Stability Classification	Category	DT/Dz (^{o}C/100 m)
Extremely Unstable	A	< -1.9
Moderately Unstable	B	-1.9 to -1.7
Slightly Unstable	C	-1.7 to -1.5
Neutral	D	-1.5 to -0.5
Slightly Stable	E	-0.5 to 1.5
Moderately Stable	F	1.5 to 4.0
Extremely Stable	G	> 4.0

Table V. WIPP Stability Class Designation for September 1996 to May 1999 by Percentage of Time

Stability Class	Percentage of Time in Each Category
A	22.02
B	5.21
C	4.88
D	16.35
E	18.28
F	13.47
G	19.79

The PDF for the meteorological conditions of windspeed at WIPP was determined by the data presented above. Figures 5a through 5g show the windspeed distributions for each stability class. In the spreadsheet calculations, the χ/Q for each wind stability class was weighted according to Table V.
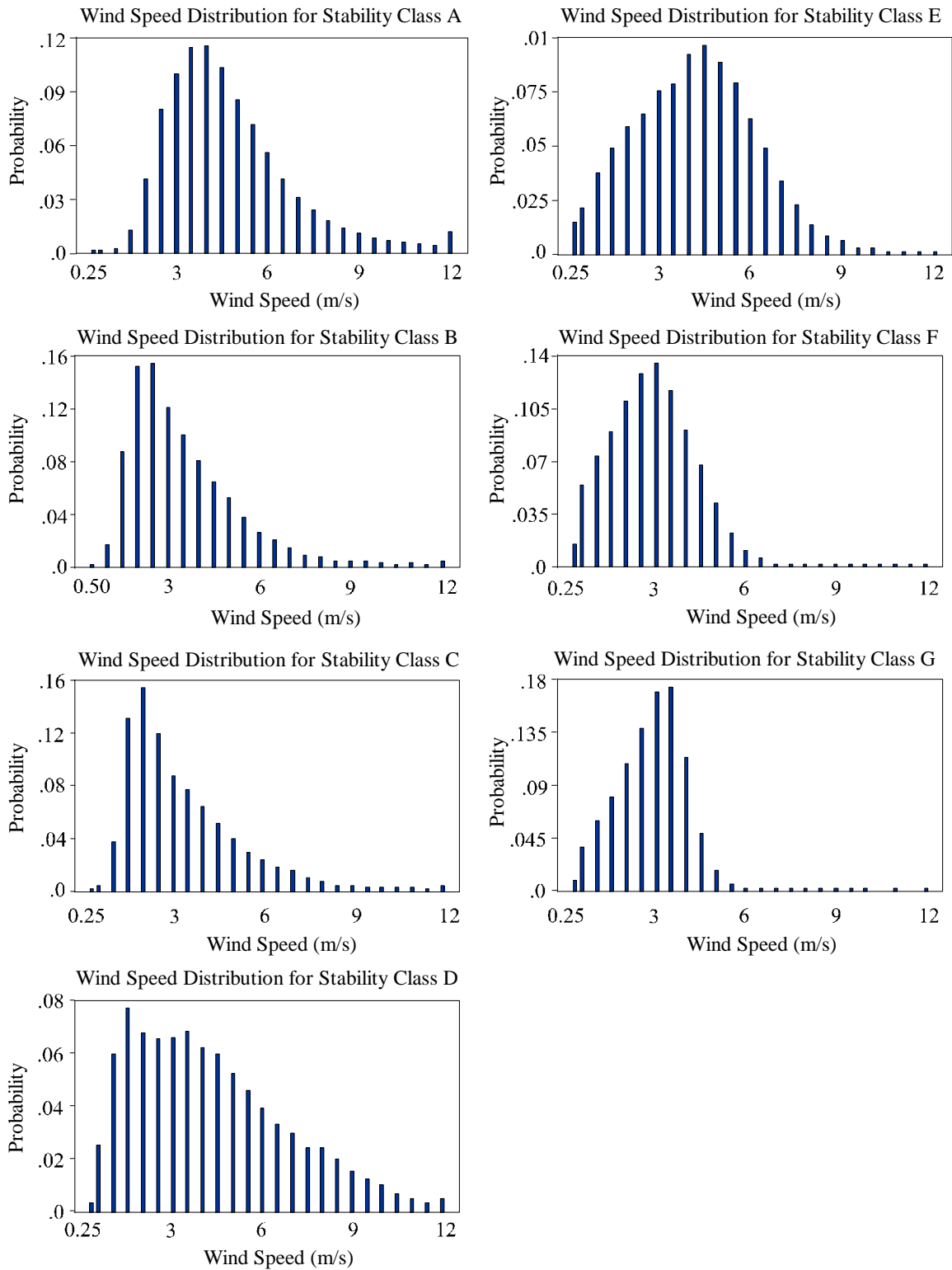
Figure 5a through 5g. Wind speed distributions separated by stability classes.

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3.3.3 Wind Direction

The wind direction is simply the direction from which the wind blows. The wind direction is usually designated into 16 categories, starting from N(orth) and going clockwise to NNE, NE… as seen in Figure 6.

The polar plot of Figure 6 shows the direction from which direction the wind blows for all data between September 1996 and May 1999. In general, the wind is from the southeast and east-southeast at the WIPP site. The spreadsheet calculations for χ/Q, however, did not consider the direction from which the wind comes.
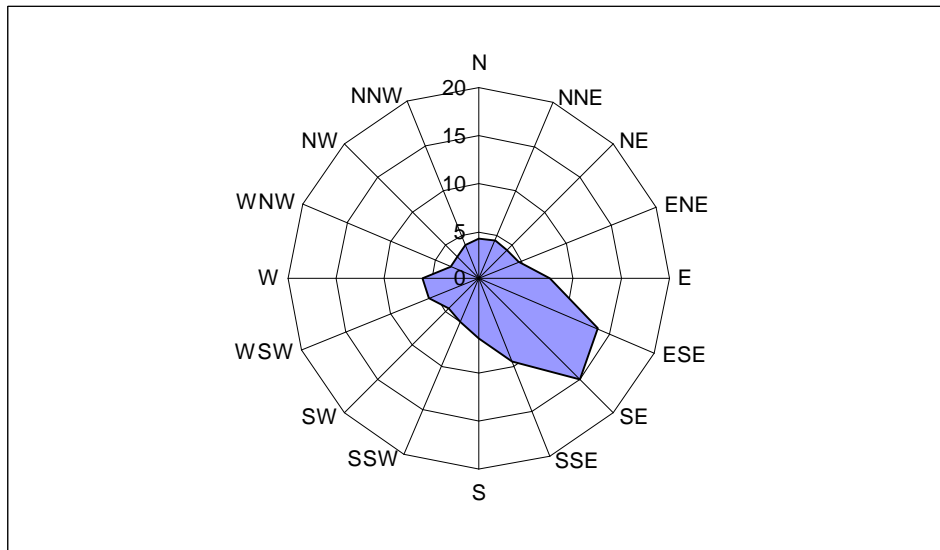

Figure 6. Percentage of Wind Direction (from Direction Indicated) for Meteorological Data of September 1996 to May 1999.

3.3.4 Precipitation

The amount of rainfall at the WIPP site varies significantly from month to month and year to year with 1997 being an extremely wet year at 23.9 inches (607.3 mm) and 1998 being an extremely dry year at 7.7 inches (195.6 mm). The average rainfall is about 10-12 inches (254-
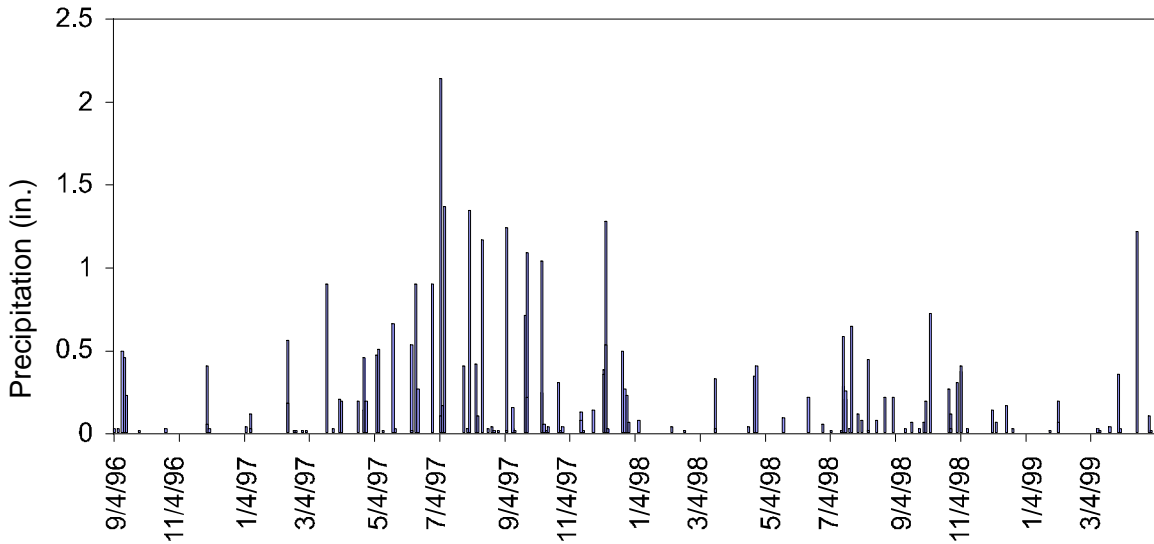
Figure 7. Daily Precipitation Record (in Inches) at WIPP from September 1996 to May 1999.

305 mm) per year. Figure 7 shows daily precipitation data for the period of September 1996 to May 1999.   Again, precipitation was not used in formulating χ/Q.

3.4 Source Term

The formulation of the source term for the PSA remained consistent with the SAR. The only change was the use of a PDF to represent the actual drum loading expected from each generator site as reported in the Baseline Inventory Report (BIR), Rev. 3 (DOE 1995). The source term, Q in Ci, was calculated by

$$Q = MAR * DR * ARF * RF * LPF \tag{10}$$

where MAR is the material at risk (Ci) and is calculated by the curie content of a drum multiplied by the number of drums involved in the accident. DR is the damage ratio and is the fraction of the MAR that is impacted by the accident. ARF is the airborne release fraction and is the fraction of radioactive material that is suspended in air resulting from the accident. RF is the respirable fraction, which relates to the fraction of particles that are in the respirable range (less

than 10 μm AMAD). LPF is the leakpath factor and is the fraction of material that is not filtered out of the air after the accident. Filtering for an above ground accident occurs in permanently installed, continuous on-line two-stage HEPA filtration system. Filtering for an underground accident is only engaged during an accident. LPF is assumed to be 1×10^{-6} for the mitigating case (filtered) and 1.0 for unmitigated release (DOE 1999). This reports assumes that all accidents for the probabilistic dose assessment are unmitigated for comparison with the SAR.

Figure 8 shows a conceptual model of assessing the source for an inhalation dose from the drum curie content. The figure is a reproduction of the one presented in Mueller et al. (1994).
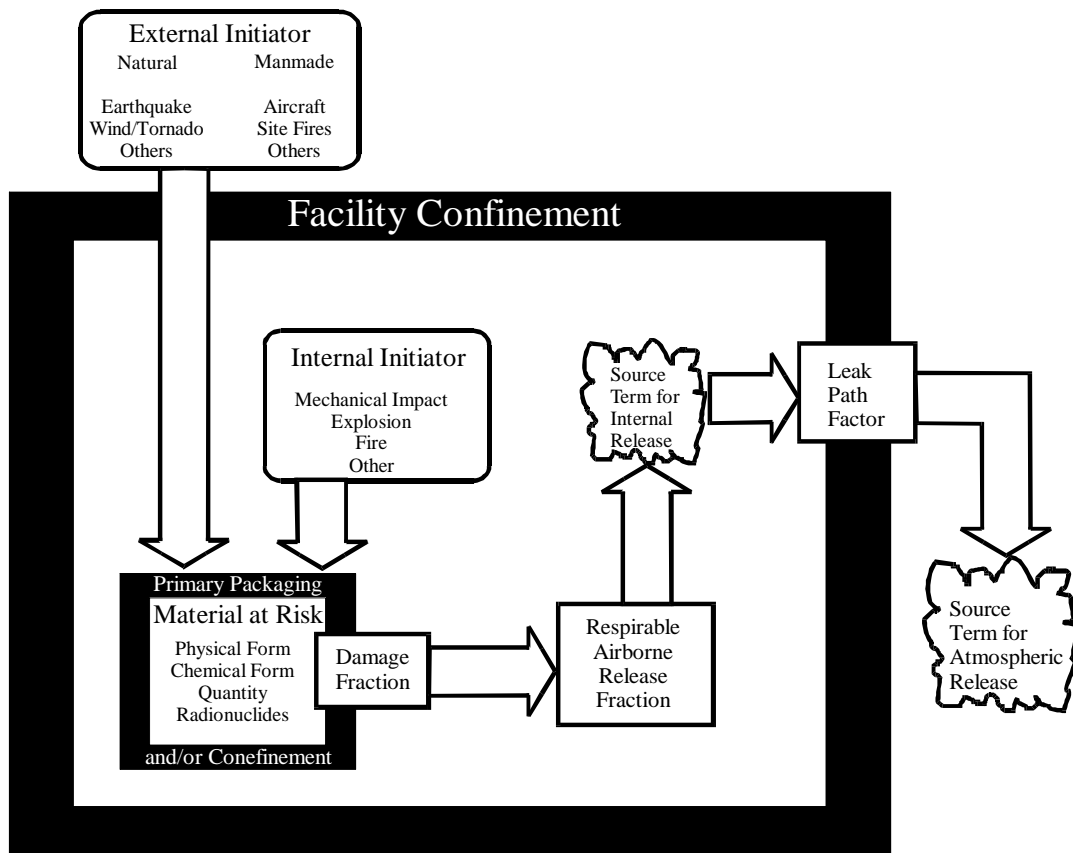

Figure 8. Conceptual model for source term development

The waste form of each drum and each type of accident dictates the values for DR, ARF, and RF and are taken from the DOE handbook on airborne releases (DOE 1994b). Table VI lists some

examples of values used in the three parameters and the product of the three. The values from
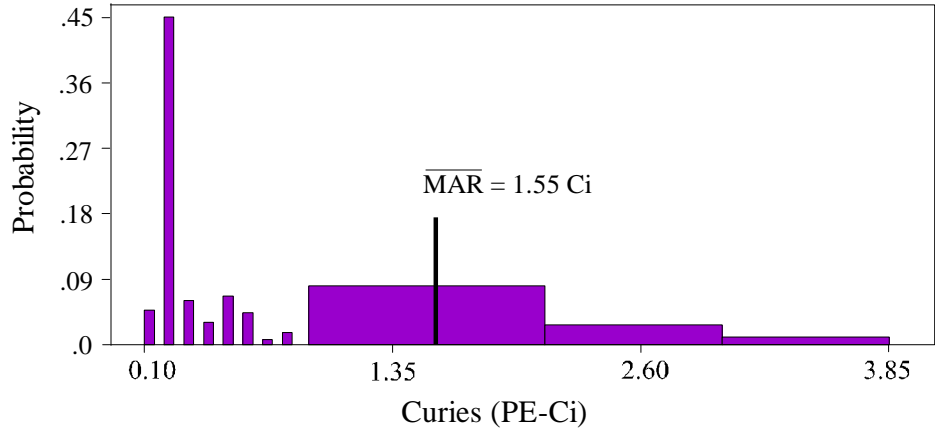Table VI were used in formulating the source term for the dose calculations.

Table VI. Example DR, ARF, and RFs used in formulating the source term.

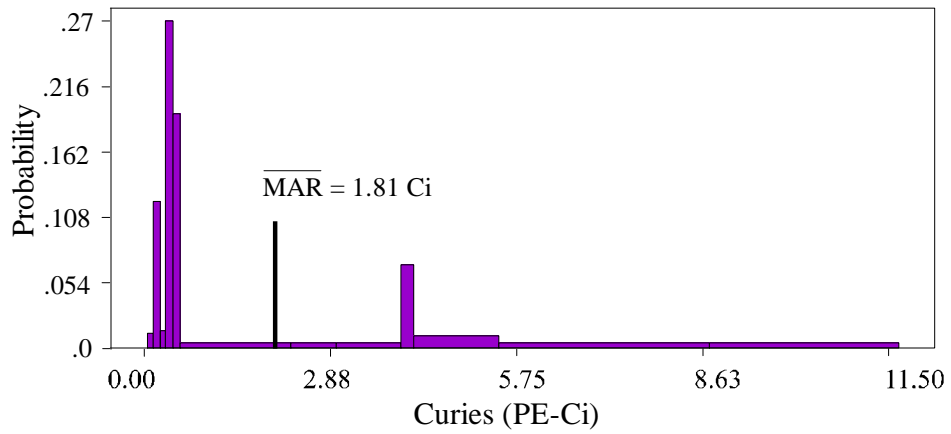| Waste Form	DR	ARF	RF	Overall Product
Combustible Solids(95%) drops less than 5 ft (drum)	1×10^{-2}	1×10^{-3}	1×10^{-1}	1×10^{-6}
Noncombustible Solids(95%) drops less than 10 ft (SWBs/overpacks)	1×10^{-2}	1×10^{-3}	1.0	1×10^{-5}
Solidified Solids, Vehicle Impact and Puncture	1×10^{-2}	2×10^{-5}	N.A.	2×10^{-7}
Noncombustible Solids(95%) drops 2000 ft (waste hoist)	2.5×10^{-1}	1×10^{-3}	1.0	2.5×10^{-4}

The curie content of each drum from all major generator sites was estimated from the BIR and a
PDF was assigned respectively. Fig. 9a-9i shows example probability density functions (PDFs)
of the nine generator sites' expected drum loading. For the PSA, a total of nine generator sites
were evaluated, including Idaho National Engineering Laboratory (INEEL), Los Alamos
National Laboratory (LANL), Lawrence Livermore National Laboratory (LLNL), Mound Site
(MD), Nevada Test Site (NTS), Oak Ridge National Laboratory (ORNL), Rocky Flats
Environmental Test Site (RFETS), Richland Site in Hanford (RL), and Savannah River Site
(SRS). The PDFs shown in Fig. 9 are for the entire expected population of drums to arrive at
WIPP as reported in the BIR (DOE 1995). Fig. 9 was created from the observed frequency
normalized to the total population.

The source term from each generator site was multiplied by a weighting factor, which
normalized the value to the expected number of curies from each site. Table VII shows the
weighted fraction used in formulating the source term.
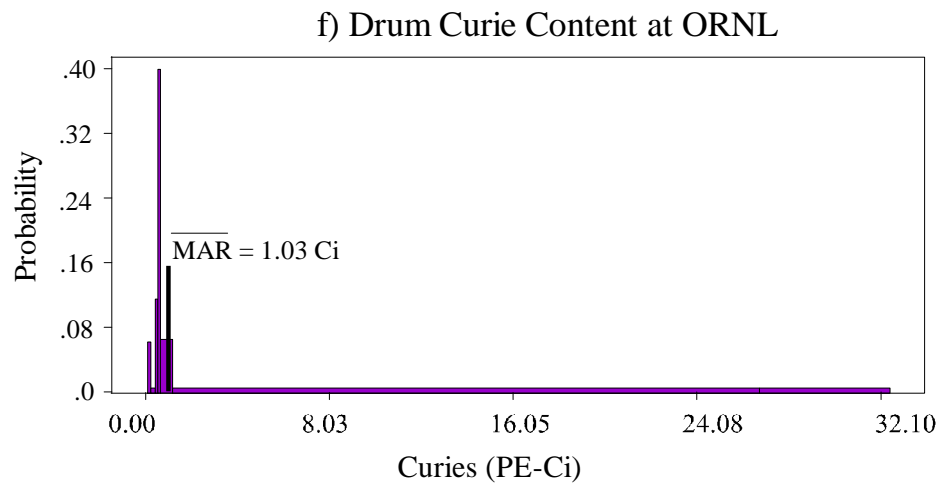
a) Drum Curie Content at INEEL

$\overline{MAR}$ = 1.55 Ci

Probability

Curies (PE-Ci)

b) Drum Curie Content at LANL

$\overline{MAR}$ = 1.81 Ci

Probability

Curies (PE-Ci)

c) Drum Curie Content at LLNL

$\overline{MAR}$ = 0.22 Ci

Probability

Curies (PE-Ci)
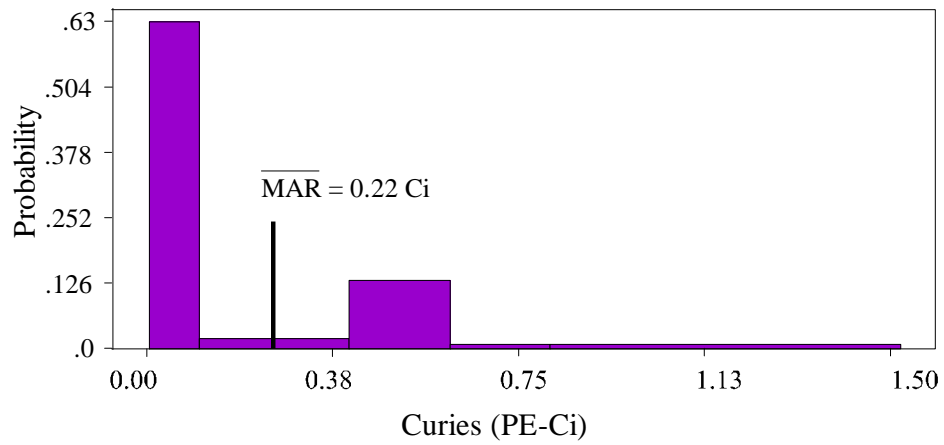
Figures 9a through 9c. PDFs of the various generator sites.

d) Drum Curie Content at MD
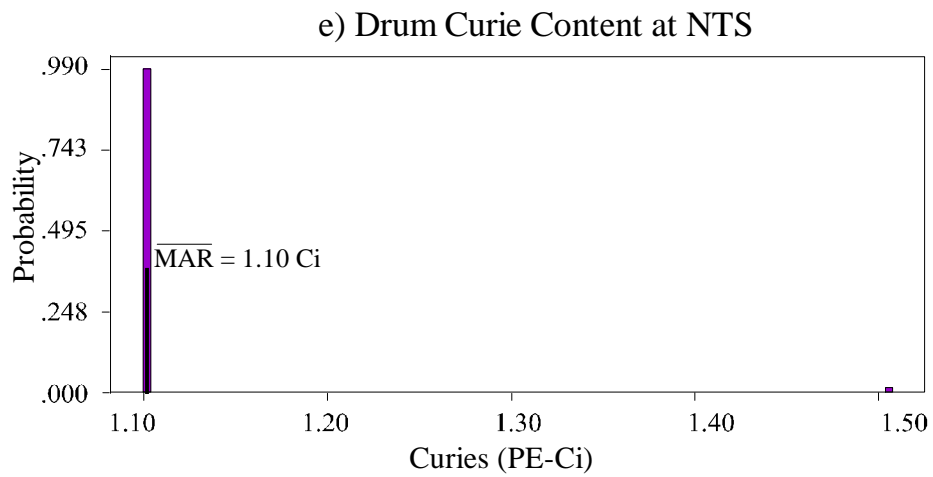

$\overline{MAR}$ = 0.85 Ci

e) Drum Curie Content at NTS

$\overline{MAR}$ = 1.10 Ci

f) Drum Curie Content at ORNL

$\overline{MAR}$ = 1.03 Ci
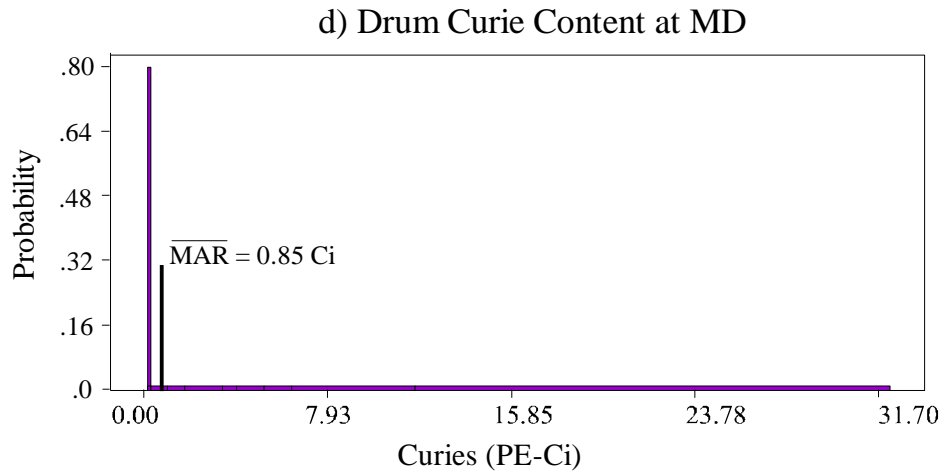
Figures 9d through 9f. PDFs of the various generator sites.
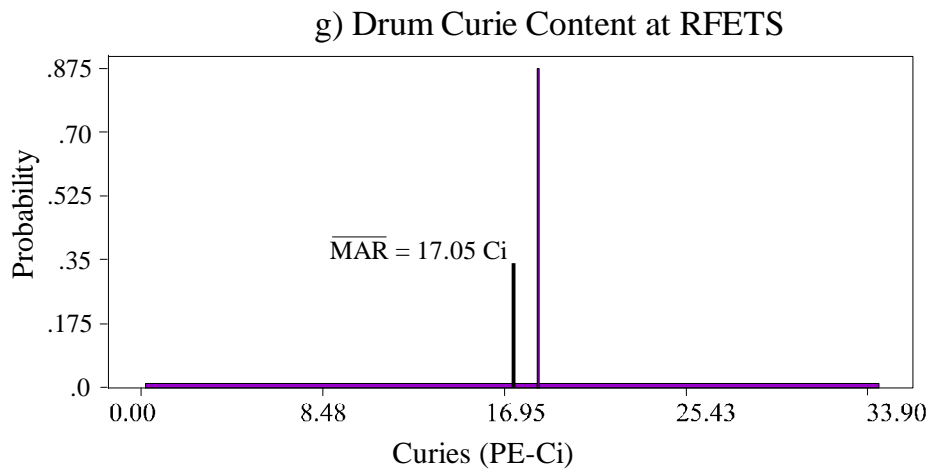
Figures 9g through 9i. PDFs of the various generator sites.

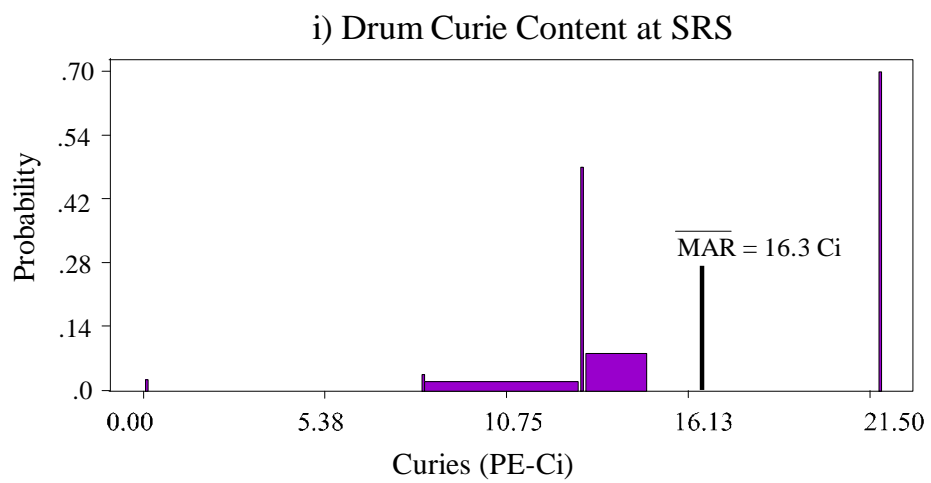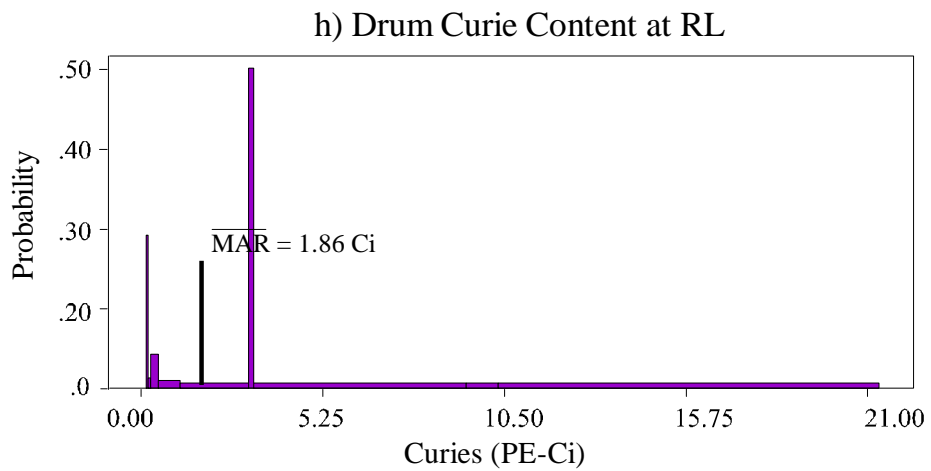Table VII. Weighted fraction used in formulating the source term.

Site	Curies At Each Site	Fraction of Total Curies
IN	195980	0.161
LA	104275	0.0857
LL	292	0.000240
MD	1419	0.00117
NT	3190	0.00262
OR	7805	0.00641
RF	382761	0.315
RL	109161	0.0898
SR	411191	0.3381
TOTAL	1216074	1.00

3.5 Discretization

The lateral distance from the source was incorporated into the site-specific air dispersion calculations by discretizing a 1-D grid by 50 m increments, starting at 100 m from the source to 500 m. The final results (Figure 11) will show dose versus distance.

4. DISCUSSION

Once all of the proper information was entered into the spreadsheet, including the site-specific meteorological data and the source term data, the Monte Carlo simulation was run by randomly sampling 10,000 iterations from each PDF. The spreadsheet was divided by the 6 stability classes and each stability class was discretized into the 50-m distances, starting at 100 m and ending at 500 m (9 total grid points). Within each distance column, the nine generator sites were sampled for drum loading and the source term was calculated accordingly. A total of 486 PDFs (9 generating sites x 6 stability classes x 9 spatial grid points) were sampled 10,000 times and the statistical information was retained from the sampling. The large number was chosen to more accurately represent each distribution.

The final result was a matrix of dose forecasts, showing the doses resulting from all scenarios, versus distance. A frequency histogram and a confidence interval further characterized each dose in the matrix. The histogram was compiled by keeping track of all calculated dose values and dividing them into discrete intervals. The intervals show the total count of dose calculation within that interval. The probability of dose for a specific interval can be calculated by normalizing the frequency by the total number of observations, which in this case is 10,000.

Figure 10 shows four examples of frequency distributions with their associated probabilities. The four plots show -from upper left to lower right- doses expected from scenario CH2 if the receptor is 250 m from the source, doses expected from scenario CH5 at 100 m from the source, doses expected during scenario CH7 at 400 m from the source, and doses expected from scenario CH11 at 250 m from the source. The source in this example is the above ground exhaust shaft outlet or the outside ventilation of the WHB. The example plots of Figure 10 were not chosen for any special reason. They were randomly selected to give a representation of many scenarios. Table IX lists additional statistics for the four frequency distributions.

Table IX. Statistics of Dose Calculations

Statistics	CH2 at 250m	CH5 at 100m	CH7 at 400m	CH11 at 250m
Trials	10000	10000	10000	10000
Mean	2.4	42.5	0.3	0.8
Median	2.0	33.4	0.3	0.6
Standard Deviation	4.3	92.2	1.0	1.4
Variance	18.8	8496.4	1.1	1.9
Skewness	43.74	27.51	38.41	43.74
Kurtosis	2,651.26	1,091.06	1,711.01	2,651.26
Coeff. of Variability	1.81	2.17	3.16	1.81
Dose Range Minimum	0.8	13.5	0.1	0.2
Dose Range Maximum	306.2	5034.6	53.8	96.2
Dose Range Width	305.5	5021.1	53.7	96.0

The main feature to note about the frequency distributions is that it is a lognormal distribution.
Every distribution from the probabilistic dose assessment has this shape. The Central Limit
Theorem proves this observation mathematically and states that the mean of a set of *n* variables,
where n is large, drawn independently from the same distribution will be normally distributed
(Vose 1996). The product of a large number of independent positive variables drawn from
different distributions will be approximately lognormally distributed.
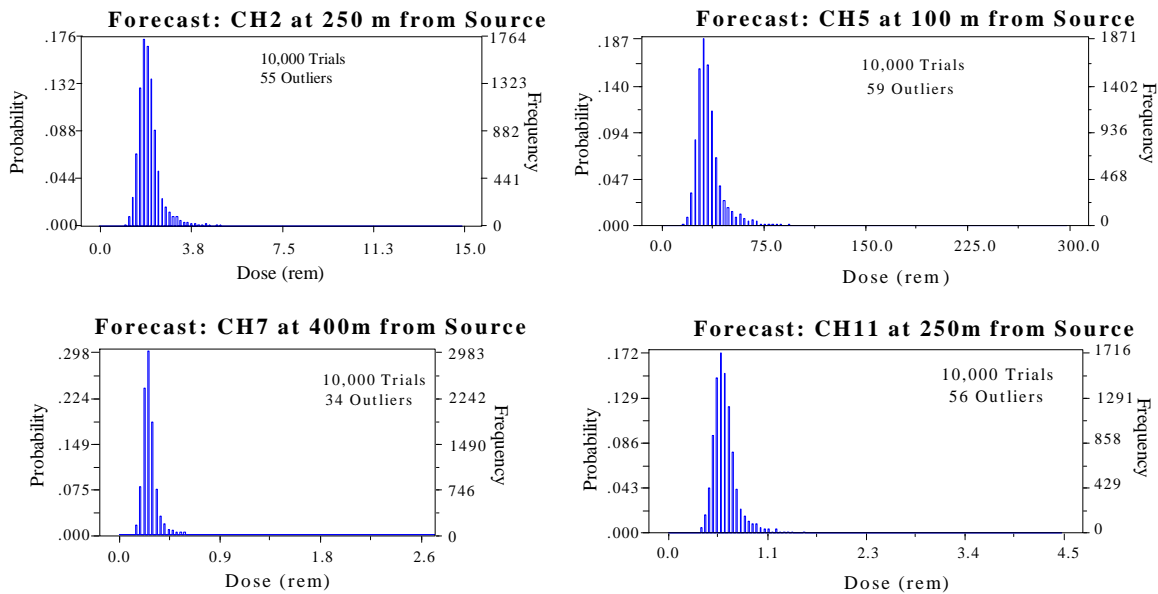

Figure 10. Inhalation dose distributions for selected scenarios.

The confidence intervals for each set of scenarios were also calculated from the compiled information. The most meaningful representation of confidence is showing the 5%, 50%, and 95% confidence limits, i.e., the expected dose that falls at or below that given confidence limit. Figure 11 shows all eight scenarios' expected dose including the confidence intervals described above. Each subplot in Figure 11 shows dose vs. distance, between 100 m and 500 m from the source at 50-m grid points. The dose is in rem and represents the 50-year whole body committed effective dose equivalent (CEDE). From Figure 11, one can see that the CH5 scenario is the most catastrophic event with a dose of 64 rem at 100 m from the source with a confidence that 95% of the doses fall at or below 64 rem. CH5 is the waste hoist scenario, where the hoist plummets 655 m with 28 drums. The WIPP SAR calculated a dose of 61 rem at 100 m (Table II) and is approximately equal to the probabilistic dose assessment.

Other scenarios, such as the CH1 and CH7 show that the SAR is conservative in its estimate of dose. The SAR's version of the CH1 scenario, where a drum spontaneously ignites in the WHB, is represented by a dose of 33 rem at 100 m (Table II). The dose calculated in the present study is 4.6 rem with a 95% confidence limit. Similarly, the dose estimate for CH7 in the SAR is approximately 7 times greater than the value shown in Figure 11.

5. CONCLUSIONS

The radioactive dose to an individual for various accident scenarios was calculated using a probability model for Eq. 1. Two of the four input parameters of Eq. 1. were assigned ranges of expected values; the remaining two were constant. A probability density function (PDF) was assigned to each data range to appropriately represent the uncertainties of the problem. A Monte Carlo simulation was employed to randomly sample from the range of values as dictated by the PDF. A large number of samples were chosen to more accurately mimic the behavior of each PDF. Hence, a large number of dose calculations were performed and statistical information was derived for the set of calculations. The results, displayed in Fig. 4., show a dose versus distance from the source of release of radioactivity for each plausible scenario in the WIPP SAR (DOE 1999), given site-specific meteorological conditions and expected radioactive content of each drum involved.

This technique was different than that presented in the WIPP SAR (DOE 1999). The SAR listed deterministically calculated dose, where single-point parameter values were used in modeling Eq. 1. The values for this type of modeling can be "best-guess" estimates, but the SAR chose to use conservative values in their calculations to bound the expected radioactive uptake and dose to a downwind receptor. The calculations for the SAR are shown in Table II, and the unmitigated dose can be compared to Figure 11.

In addition to the deterministic calculations, the SAR presented a limited sensitivity analysis of the effect of different drum activity loading. For example, Table II shows scenario CH4 with three sets of calculations: 1) the base case with 1 drum loaded at 80 Ci, and 3 drums loaded at 8 Ci, 2) 1 drum loaded at 1100 Ci, and 3) 4 drums loaded at 80 Ci. The source calculation of case 2 in this example has shown that although the drum loading is much higher than the base case, the dose is expected to be much lower. The counter-intuitive observation reflects the storage configuration for the standard 55-gallon drums containing more than 80 PE-Ci. Large curie-containing drums are overpacked into 85-gallon drums for safer handling. The product of the damage ratio, airborne release fraction, and the respirable fraction shown in Eq. 10, therefore, will be orders of magnitude lower than a drum that is not overpacked.

Comparisons of Table II and Figure 11 show that many of the dose calculations in the SAR are higher than the probabilistic dose at the 95% confidence level. Lower doses calculated in the probabilistic dose assessment are expected, since many of the input data are sampled from a relatively large range, hence a smearing of high and low values. However, higher doses seen in the probabilistic dose assessment are not expected, and may show where the deterministically determined doses may be underrepresenting the worst case conditions. For example, the probabilistic dose assessment assumes that all drums are loaded equally, and when a relatively large number of drums are involved in an accident, such as scenario CH5 (waste hoist accident – 28 drums involved), a larger dose is expected. The justification for the drums of equal loading is from observations of recent operating procedures at WIPP. A shipment of TRUPACT IIs will typically transport drums from a single waste stream, with the drum loading of each waste stream being similar. Handling of drums between unloading of the TRUPACT IIs to placement underground is typically confined to single shipments. Therefore, it is recommended that the SAR's dose consequence calculations be updated to reflect a more realistic source term based on the PDFs presented in this report. This could be accomplished by summing the 95% worst case drum curie content from each generator site, normalized to its total contribution of curies according to Table VII.

Finally, although the two methods described above are different, they are not meant to be mutually exclusive, but as complementary techniques (ICRP 1993). The ICRP identified both methods for assuring an adequate level of safety has been achieved and that no major contributors to risk has been overlooked (ICRP 1993). With that, it is recommended that the WIPP SAR be updated to include a probabilistic dose assessment. The inclusion would add to the level of safety and give confidence that all possible ranges of expected conditions have been addressed. However, consistency between the results will have to be analyzed and assessed.
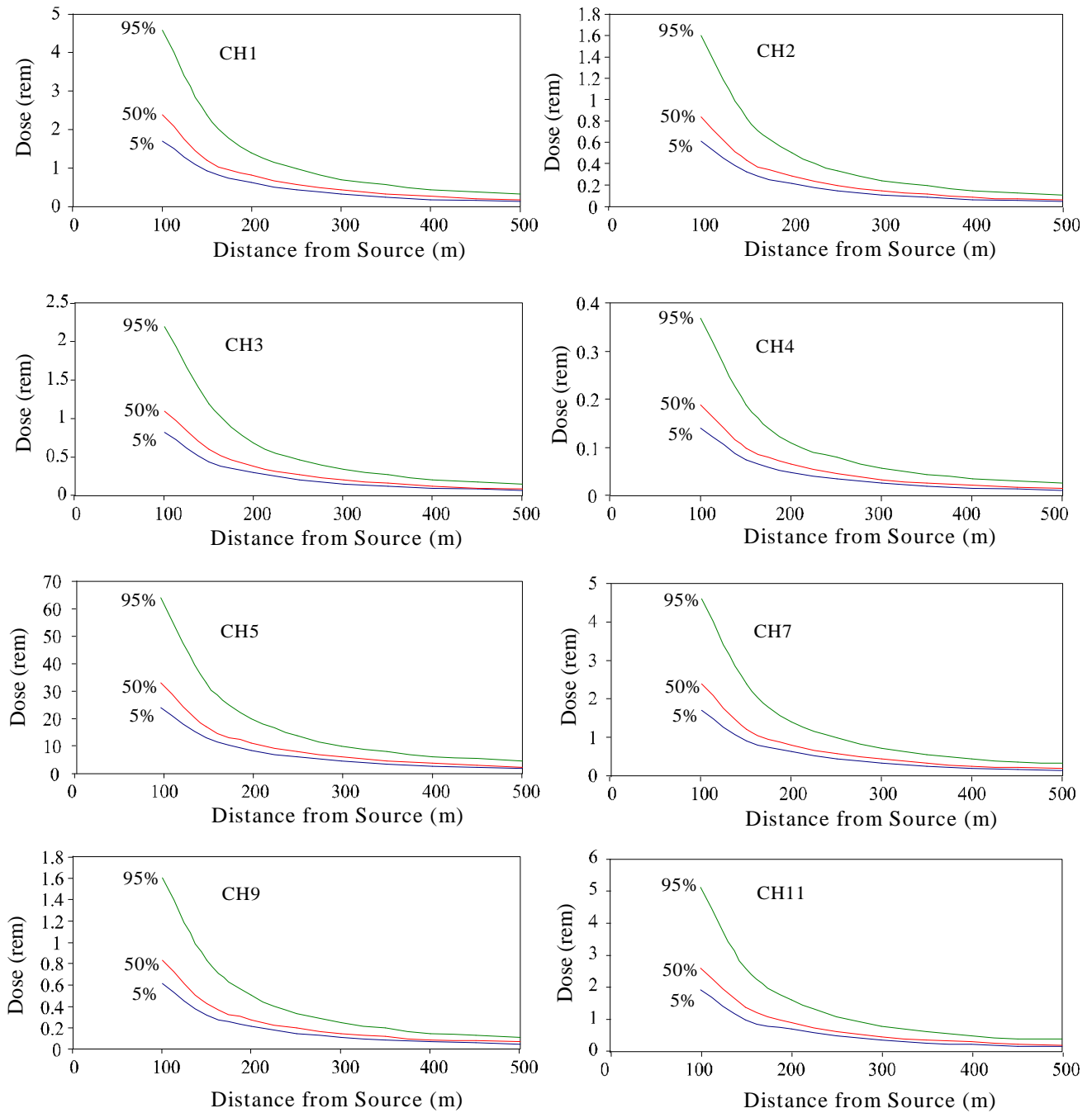
Figure 11. Probabilistic dose vs. distance for the eight accident scenarios with 5%, 50%, and 95% confidence limits.

6. REFERENCES

40 CFR Part 191. Environmental standards for the management and disposal of spent nuclear fuel, high-level and transuranic wastes. Title 40, Protection of the environment; Chapter I, Environmental Protection Agency; Code of Federal Regulations. Washington, DC: National Archives and Records Administration. July 1, 1999.

[ANSI-ANS] American National Standards Institute, American Nuclear Society. 1999. American National Standard for determining meteorological information at nuclear facilities: ANSI ANS-3.11-1999. (Draft). La Grange Park (IL): ANS.

Apostolakis G. 1978. Probability and risk assessment: The subjective viewpoint and some suggestions. Nuclear Safety 19:305-315.

Channell, JK; Rogers, JC; Neill, RH. 1986 Jun. Adequacy of TRUPACT-I design for transporting contact-handled transuranic waste to WIPP. Albuquerque (NM): Environmental Evaluation Group. EEG-33.

Decisioneering, Inc. 1998. CRYSTAL BALL [computer program]. Version 4. Denver(CO): Decisioneering, Inc.

Greenfield, MA. 1990 Jan. Probabilities of a Catastrophic Waste Hoist Accident at the Waste Isolation Pilot Plant. Albuquerque (NM): Environmental Evaluation Group. EEG-44.

Greenfield, MA; Sargent, TA. 1995 Nov. An Analysis of the Annual Probability of Failure of the Waste Hoist Brake System at the Waste Isolation Pilot Plant (WIPP). Albuquerque (NM): Environmental Evaluation Group. EEG-59.

Greenfield, MA; Sargent, TA. 1998 Jan. Probability of Failure of the Waste Hoist Brake System at the Waste Isolation Pilot Plant (WIPP). Albuquerque (NM): Environmental Evaluation Group. EEG-65.

Greenfield, MA; Sargent, TA. 2000 May. Probability of Failure of the TRUDOCK Crane System at the Waste Isolation Pilot Plant (WIPP). Albuquerque (NM): Environmental Evaluation Group. EEG-74.

Hey BE. 1994 Dec. GXQ 4.0 Program user's guide. Rev 1A. Richland (WA): Westinghouse Hanford Company. WHC-SD-GN-SWD-30002, rev 1A.

[IAEA] International Atomic Energy Agency. 1980. Atmospheric dispersion in nuclear power plant siting. Vienna (Austria): IAEA; (Safety Guide: 50-SG-S3).

[ICRP] International Commission on Radiation Protection. 1974. Report of the Task Group on Reference Man. New York: Pergamon Press; (ICRP Publication 23).

[ICRP] International Commission on Radiological Protection. 1993. Protection from potential exposure: a conceptual framework. New York: Pergamon Press; (ICRP Publication 64).

Mueller, CJ; Folga, SM; Roglans-Ribas, J; Nabelssi, B; Mishima, J. 1998 Jun. A structured methodology for waste management facility accident analysis. Nuclear Technology 122:306-17.

Rucker, DF. 2000 Aug. Air Dispersion Modeling at the Waste Isolation Pilot Plant. Albuquerque (NM): Environmental Evaluation Group. EEG-76. (in press)

[DOE] US Department of Energy. 1988 Jul. Internal dose conversion factors. Washington, DC: DOE. DOE/EH-0071.

[DOE] US Department of Energy. 1994 Dec. Airborne release fractions/rates and respirable fractions for nonreactor nuclear facilities: DOE-HDBK-3010-94. Washington, DC: DOE.

DOE] US Department of Energy. 1995 Dec. Waste Isolation Pilot Plant transuranic waste baseline inventory report (BIR). Revision 3. DOE/CAO-95-1121, rev 3.

[DOE] US Department of Energy, Carlsbad Area Office. 1996 Oct. Title 40 CFR Part 191 Compliance Certification Application for the Waste Isolation Pilot Plant. Final. 21 volumes. Carlsbad (NM): DOE. DOE/CAO-1996-2184.

[DOE] US Department of Energy, Carlsbad Area Office. 1999 Jan. WIPP safety analysis report. Rev 3. Carlsbad (NM): DOE. DOE/WIPP-95-2065, rev 3.

[EPA] US Environmental Protection Agency. 1987 (rev 1995). On-site meteorological program guidance for regulatory modeling applications. Research Triangle Park (NC): EPA. Office of Air Quality Planning and Standards. EPA-450/4-87 013.

[EPA] Environmental Protection Agency. 1993. Environmental Radiation Protection Standards for the Management and Disposal of Spent Nuclear Fuel, High-Level, and Transuranic Radioactive Wastes. 40 Code of Federal Regulations Part 191.

[EPA] Environmental Protection Agency. 1996 Feb. Criteria for the Certification and Recertification of the Waste Isolation Pilot Plant's Compliance with the 40 CFR 191 Disposal Regulations; Final Rule. Federal Register 61(28):5224-5245.

[EPA] US Environmental Protection Agency. 1998 May 18. Criteria for the certification and recertification of the Waste Isolation Pilot Plant's compliance with the 40 CFR part 191 disposal regulations: certification decision; final rule. Federal Register 63(95):27354-27406.

[NRC] US Nuclear Regulatory Commission. 1982 Nov. Atmospheric dispersion models for potential accident consequence assessments at nuclear power plants. Rev 1. Washington DC: NRC; (NRC Regulatory Guide: 1.145).

Vose D. 1996. Quantitative risk analysis: A guide to Monte Carlo simulation modeling. New York: John Wiley and Sons.

Westinghouse Electric Company, Waste Isolation Division. 1995 April. Hazard and operability study for CH-TRU waste handling system, Waste Isolation Pilot Plant, Carlsbad, NM. WCAP 14312.

Westinghouse Electric Company, Waste Isolation Division. 1997. WIPP Meteorological Quality Assurance Plan. Carlsbad (NM): Westinghouse. WP 02-EM.01.

7.0 LIST OF ACRONYMS

ANS	American Nuclear Society
ANSI	American National Standard Institute
BR	Breathing Rate
CCA	Compliance Certification Application
CDF	Cumulative Distribution Function
CEDE	Cumulative Effective Dose Equivalent
CFR	Code of Federal Regulations
CH	Contact Handled
CMR	Central Monitoring Room
DCF	Dose Conversion Factor
DOE	U.S. Department of Energy
EEG	Environmental Evaluation Group
EPA	U.S. Environmental Protection Agency
HEPA	High Efficiency Particulate
IAEA	International Atomic Energy Agency
ICRP	International Commission of Radiological Protection
LWB	Land Withdrawal Boundary
MEI	Maximally Exposed Individual
MSHA	Mining Safety and Health Administration
NCRP	National Council on Radiation Protection and Measurement
NESHAPS	National Emission Standards for Hazardous Air Pollutants
NRC	Nuclear Regulatory Commission
NRG	Nuclear Regulatory Guide
RCRA	Resource Conservation and Recovery Act
RH	Remote Handled
SAR	Safety Analysis Report
TRU	Transuranic Waste
WHB	Waste Handling Building
WIPP	Waste Isolation Pilot Plant