NOTICE

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ABSTRACT

This study presents a generic methodology that can be used to evaluate the uncertainty in the calculated accidental offsite doses at the Yucca Mountain repository during the preclosure period. For demonstration purposes, this methodology is applied to two specific accident scenarios: the first involves a crane dropping an open container with consolidated fuel rods, the second involves container failure during emplacement or removal operations. The uncertainties of thirteen parameters are quantified by various types of probability distributions. The Latin Hypercube Sampling method is used to evaluate the uncertainty of the offsite dose. For the crane-drop scenario with concurrent filter failure, the doses due to the release of airborne fuel particles are calculated to be 0.019, 0.32, and 2.6 rem at confidence levels of 10%, 50%, and 90%, respectively. For the container failure scenario with concurrent filter failure, the 90% confidence-level dose is 0.21 rem.

INTRODUCTION

Several preliminary radiological-safety analyses have been performed for the prospective Yucca Mountain repository. These studies evaluated the offsite doses that could result from various postulated accidents during the preclosure period. The accidental doses, however, have a high level of uncertainty because of the wide variability of the input parameters, the lack of detailed information and data in the repository, and the limited knowledge of the phenomena involved in the accident scenarios.

In recent years, uncertainty analyses have been applied to postclosure performance assessments of the geologic disposal of nuclear waste. This paper summarizes a study that developed a methodology for evaluating the uncertainty in accidental offsite doses during the preclosure period. For demonstration purposes, this methodology is applied to two postulated accidents (one for the surface facilities and the other for the underground facilities) for the prospective Yucca Mountain repository.

METHODOLOGY AND MAJOR ASSUMPTIONS

Accident dose calculations use analytical models that predict how various input parameters affect the dose consequences. This study considers the variability and uncertainties of input parameters to determine the resulting uncertainty for the accident doses. Uncertainties in the accident scenarios and in the analytical models used to calculate the doses are not specifically addressed.

This study uses the following four basic steps to evaluate the uncertainties in the offsite radiological doses:

1. Develop accident scenarios, establish dose models, and identify parameters that have uncertainties.
2. Screen parameters to determine those with the greatest uncertainties (some judgment is used in this step).
3. Quantify uncertainties of parameters by defining ranges and types of probability distributions (experimental data and judgment are used in this step).
4. Evaluate uncertainty propagation by statistically combining input-parameter uncertainties to obtain the dose uncertainty (the Latin Hypercube Sampling Method is used to sample the input parameters; using these sample values, the offsite dose values are calculated to determine the probability distribution of the dose).

In this study, the major assumptions pertain to the analytical model used to calculate the dose. These are summarized below.
as follows:

- The particle size distribution generated from irradiated spent fuel pellets after a mechanical impact is the same as that from non-irradiated spent fuel pellets under the same impact.

- The mass fraction of respirable-size particles is a linear function of the impact energy. (This dependence is adopted from experimental data of glass specimens.6,7,8)

- Cesium is not released as gap activity from breached fuel cladding in a room temperature environment.

- Deposition of radionuclides (gravitational settling and plateout) inside the waste-handling building is neglected.

- The atmospheric dispersion factor (γ/Q) is evaluated using the Gaussian plume model with Pasquill-Gifford dispersion coefficients, Cy and Cz. In this model, γ represents the downwind concentration (Ci/cm²) and Q represents the release rate (Ci/s).

**STEP 1: DEVELOP SCENARIOS, ESTABLISH DOSE MODELS, AND IDENTIFY PARAMETERS WITH UNCERTAINTIES**

**Accident Scenarios**

The previous safety analyses1,2,3 developed numerous types of accident scenarios for both the surface facilities and the underground facilities. These accident scenarios include crane load-drop accidents, fuel-handling equipment failures, container failures, various structural failures, and collisions of transport-cask vehicles.

This study considers one accident for the surface facility and another for the underground facility: (1) The surface-facility accident involves a crane dropping an open container of consolidated fuel rods. (2) The underground facility accident involves a container failure in the emplacement borehole during emplacement or removal operations.

**Dose Model**

The analytical model for calculating doses is divided into the following five parts: evaluation of radionuclide inventory, evaluation of radionuclide release, evaluation of radionuclide transport through the building/facility, evaluation of radionuclide transport through the atmosphere, and evaluation of radiation dose. A sequence of submodels is developed to describe and evaluate the numerous physical and chemical phenomena associated with each part of the model. A large number of parameters with uncertainties are identified in this sequence of submodels. The five parts of the preclosure dose model and the parameters that affect offsite radiation doses are identified in Table 1 and described below.

**Radionuclide Inventory.** The radionuclide inventory depends on the number of spent fuel assemblies involved in the accident.

<table>
<thead>
<tr>
<th>Radionuclide Inventory</th>
<th>Radionuclide Release</th>
<th>Radionuclide Transport Within Facility</th>
<th>Radionuclide Transport Through Atmosphere</th>
<th>Dose Evaluation</th>
</tr>
</thead>
<tbody>
<tr>
<td>Waste type (e.g., PWR, BWR, or high-level waste)</td>
<td>Gap radioactivity</td>
<td>Gravitational settling velocity</td>
<td>Wind speed and direction</td>
<td>Dose conversion factors</td>
</tr>
<tr>
<td>U-235 enrichment</td>
<td>Radionuclide release rate</td>
<td>Plateout constant</td>
<td>Atmospheric stability class</td>
<td>Breathing rate</td>
</tr>
<tr>
<td>Initial uranium loading in fuel assembly</td>
<td>Rod breach fraction</td>
<td>Resuspension</td>
<td>Ground surface roughness height</td>
<td>Exposure times</td>
</tr>
<tr>
<td>Energy partition factor</td>
<td>Impact energy</td>
<td>Facility dimensions</td>
<td>Airborne particle density</td>
<td>Size distribution of inhaled airborne particles</td>
</tr>
<tr>
<td>Burnup</td>
<td>Escape factor</td>
<td>Air leakage rates</td>
<td>Deposition velocity</td>
<td>Resuspension</td>
</tr>
<tr>
<td>Cooling time</td>
<td>Size distribution of fractured waste particles</td>
<td>Filtration efficiency</td>
<td>Temperature</td>
<td>Terrain topography</td>
</tr>
<tr>
<td>Size and contents of container</td>
<td>Mass fraction of respirable particles</td>
<td>Temperature</td>
<td>Distance to site boundary</td>
<td></td>
</tr>
<tr>
<td>Number of waste forms</td>
<td>Temperature</td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

**TABLE 1. PARAMETERS THAT AFFECT ACCIDENTAL OFFSITE RADIATION DOSES**
scenario and the quantities of radioisotopes in each assembly, and is given by

$$\text{radionuclide inventory} = n A_i$$

where \( n \) = number of spent fuel assemblies

\( A_i \) = radioactivity of isotope \( i \) in each assembly

Data from Roddy\(^5\) show how \( A_i \) varies with different enrichments, burnups, and cooling times for spent fuel from pressurized water reactors (PWR) and boiling water reactors (BWR). Variations in burnup correlate with variations in enrichment. Table 2 shows the U-235 enrichments used by Roddy\(^5\) for specific ranges of spent fuel burnup. For various spent fuel types, burnups (with correlated enrichments), and cooling times, values of \( A_i \) are obtained using data from Roddy\(^5\) and interpolating as necessary.

**TABLE 2. CORRELATION OF BURNUP AND ENRICHMENT**

<table>
<thead>
<tr>
<th>Spent Fuel Type</th>
<th>Burnup (GWD/MTU) (^a)</th>
<th>Enrichment (wt % U-235)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PWR</td>
<td>0-33</td>
<td>3.2</td>
</tr>
<tr>
<td>PWR</td>
<td>33-60</td>
<td>4.2</td>
</tr>
<tr>
<td>BWR</td>
<td>0-27.5</td>
<td>2.8</td>
</tr>
<tr>
<td>BWR</td>
<td>27.5-40</td>
<td>3.5</td>
</tr>
</tbody>
</table>

\(^a\) GWD/MTU = Gigawatt-days/Metric Ton of Uranium

**Radionuclide Release.** The submodel for radionuclide release accounts for releases of Kr-85 gas and respirable-size particles (diameters less than 10 microns) of fractured fuel from breached fuel rods as a result of a fuel-handling accident.

The scenarios evaluated in this study involve accidental drops of spent fuel assemblies. For accidental drops from short heights, only a fraction of the rods in the spent fuel assemblies may be breached. A fuel-rod breach fraction \( f_{br} \) is used in this submodel, which assumes that \( f_{br} \) is proportional to the drop height \( h \) for heights less than three meters, and that all fuel rods would breach for drop heights greater than three meters.

\[
f_{br} = \frac{h}{3} \text{ if } 0 < h < 3 \\
= 1 \text{ if } 3 \leq h
\]

where \( h \) = drop height (m)

A fraction of the radionuclides in a spent fuel assembly are gases or volatiles. These nuclides can be collected in the gap between the fuel pellet and the cladding. The fraction released to the gap is referred to as gap activity. The submodel includes a release fraction \( F_{Kr} \) of Kr-85 to account for the fraction of total krypton inventory that may be released from the gap of breached rods.

The respirable fraction of a simulated high-level waste (HLW) glass specimen resulting from a mechanical impact is approximately proportional to the impact energy density according to the following expression: \(^6,7\)

$$\text{PULF} = G (E/V)$$

where \( \text{PULF} = \text{respirable fraction of specimen} \)

\( G = \text{fracture coefficient, measured experimentally (cm}^2/\text{J}) \)

\( E = \text{impact energy absorbed by the material (J)} \)

\( V = \text{volume of the material absorbing the impact energy (cm}^3) \)

The above relation was adopted by MacDougall\(^1\) and is used in this submodel with the assumption that it is valid for irradiated fuel pellets in fuel rods. This assumption is based on the judgment that glass and fuel pellets would behave in a similar manner because they are both ceramic materials.

The impact energy associated with the scenarios evaluated in this study results from accidental drops of spent fuel assemblies. Hence, the impact energy can be expressed as follows:

$$E = mgh$$

where \( m = \text{mass of the waste form (kg)} \)

\( g = \text{gravitational acceleration (m/s}^2) \)

\( h = \text{drop height (m)} \)

During an accidental drop of a spent fuel assembly, a fraction of the impact energy will be absorbed by the structural materials, fuel assembly end fittings, fuel rod cladding, and other non-fuel hardware. The remaining fraction of the impact energy will be absorbed by the fuel pellets. An energy-partition factor \( (\text{EPF}) \) was therefore introduced by MacDougall\(^1\) to represent the fraction of the total impact energy absorbed by the spent fuel pellets.

The submodel for radionuclide releases also includes escape factors (ESCs) to account for the fractions of airborne particles that escape from breached confinement barriers (e.g., cladding, container, borehole/transporter case, and shipping case). Therefore, radionuclide releases are described by the following expression:

\[
\text{fraction of inventory released} = f_{br} F_{Kr} \quad \text{for krypton} \\
= f_{br} G (\text{mg}) (\text{EPF}) (\text{ESC}_{\text{clad}} (\text{ESC}_{\text{bore}})/V) \quad \text{for fractured fuel particles}
\]

The parameter \( \text{ESC}_{\text{bore}} \) is not applicable to the scenario in the packaging hot cell.

**Radionuclide Transport in Facility.**

Airborne radioactivity released into the waste-handling building or into the underground facility may leak out into the atmosphere or may
be filtered prior to discharge. The submodel used in this study accounts for the reduction in particle releases due to filtration by including a decontamination factor (DF):

\[
\text{fraction of releases that are discharged to the atmosphere} = \text{DF}
\]

where DF = 1 for gaseous radionuclides and for scenarios in which the filtration system fails.

Other aspects of radionuclide transport within the building/facility are studied elsewhere, and future studies must address this subject. This is especially important for underground accident scenarios because airborne particle deposition in underground drifts and shafts may be significant.

**Radionuclide Transport Through Atmosphere.** Airborne radionuclides released from surface and/or underground facilities will be diluted as the radionuclides are dispersed through the atmosphere. The airborne particles also will be removed from the air stream as they are deposited on the ground. These dilution and deposition mechanisms contribute significantly to the reduction of radiation doses at the site boundary.

Atmospheric dispersion, which represents the effect of dilution, is evaluated by the Gaussian plume model as follows:

\[
\chi / Q = 1 / U_0 G_0
\]

where \(\chi / Q\) = atmospheric dispersion factor (s/m)

\[
U = \text{wind speed (m/s)}
\]

\[
G_0 = \text{horizontal dispersion coefficient (m)}
\]

\[
G_z = \text{vertical dispersion coefficient (m)}
\]

The above equation applies to ground-level releases. The values of \(G_0\) and \(G_z\) are functions of downwind distance and atmospheric stability class. MacDougall\(^1\) calculated \(G_0\) and \(G_z\) using the Pasquill-Gifford system.\(^6\)

Airborne particles transported through the atmosphere may be deposited on the ground by gravitational settling, eddy diffusion, and other mechanisms. (In this study, wet deposition is not studied because of the infrequent precipitation at the site.) This submodel includes a dry deposition factor (DEP) to represent the fraction of released particles that remain airborne during transport to the site boundary (the remaining particles are deposited on the ground and do not contribute to offsite doses).

The DEP depends on the wind speed, atmospheric stability class, release height, ground-surface roughness height, particle density, particle diameter, and downwind distance (to the site boundary) as described in Slade,\(^9\) Sehmel,\(^10\) and MacDougall.\(^1\)

According to Slade,\(^9\) DEP can be expressed in terms of a reference DEP\(^1\) as follows:

\[
\text{DEP} = \frac{v_d}{U}/v_d U
\]

The parameters \(v_d\) and \(U\) are the deposition velocity and wind speed under consideration, respectively. DEP\(^1\) is a reference deposition factor and was calculated by Slade\(^9\) for various downwind distances and atmospheric stability classes, for a reference deposition velocity of \(v_d = 0.01\) m/s, and a reference wind speed of \(U = 1\) m/s.

The deposition velocity, \(v_d\), is determined using a series of calculated curves that give \(v_d\) as a function of wind speed, particle density, and particle diameter.\(^10\)

Following MacDougall,\(^1\) the dry deposition factor (DEP) can be evaluated using the following assumptions: a distance to the site boundary of 5 km, an atmospheric stability class \(F\), a certain lognormal particle-size distribution for a fractured \(U_0\) sample,\(^6\) and a surface roughness of 1 cm for smooth desert surfaces. Values of DEP can be determined for different wind speeds and a fitted curve can approximate the results by the following equation:

\[
\text{DEP} = 0.176 \ln U + 0.1226
\]

where \(U = \text{wind speed (m/s)}\)

Also, a deposition coefficient, \(\beta\), is introduced to account for variations in DEP owing to different particle size distributions, atmospheric stability classes, ground surface roughness heights, and other effects.

The submodel applies the dry deposition factor (DEP) only to airborne particles released during accidents in which the filtration system fails (i.e., DEP is included only if DF = 1 in the previous submodel). This is because most of the airborne particles with diameter larger than 1 micron will be filtered by the high efficiency particulate air (HEPA) filter. As a result, few airborne particles released to the atmosphere after filtration will be deposited within the 5-km site boundary. Therefore, DEP can be expressed as follows:

\[
\text{DEP} = \beta (0.176 \ln U + 0.123)\text{ for particle releases with filter failure}
\]

\[= 1 \text{ for gaseous releases and for scenarios with the filtration system operating}\]

The submodel for atmospheric transport is therefore described as follows:

\[
\text{atmospheric transport factor} = \text{DEP}/U_0 G_0
\]

**Offsite Radiation Dose.** After a postulated release of radioactivity from the repository, an individual could be exposed to radiation primarily in the following two ways:
internal exposure from the inhalation of the radioactive material in the plume as it passes by, and external exposure from immersion in the plume. The calculations in MacDougall\(^1\) indicated that the inhalation doses are much larger than the immersion doses. Consequently, only the inhalation doses are included in this submodel. All dose conversion factors (DCFs) used in this report reflect 50-year dose commitments.

MacDougall\(^1\) determined that the organs receiving the largest doses are the lung (when considering Kr-85 releases) and the bone (when considering fuel particle releases). Therefore, this submodel addresses doses to these two organs only. The submodel is described as follows:

\[
\text{dose per second to organ } j \text{ per unit concentration of radioisotope } i = (R) (\text{DCF}_{ij})
\]

where \(R\) = breathing rate (m\(^3\)/s)
\(\text{DCF}_{ij}\) = inhalation dose conversion factor for radioisotope \(i\) to organ \(j\) (rem/Ci inhaled)

The submodels described above are combined to give the equations used to calculate accidental radiation doses at the site boundary for the two selected specific scenarios. The general form of these equations is similar to those specified by the Nuclear Regulatory Commission:\(^1\)

\[
\text{Dose}_{\text{lung}} = n \cdot F_{\text{fr}} \cdot F_{\text{Fr}} \left( \frac{1}{\text{DFK}_2} \right) \cdot R \cdot \text{DCF}_{\text{Kr,lung}}
\]

\[
\text{Dose}_{\text{bone}} = n \cdot F_{\text{fr}} \cdot R \cdot \text{DF} \cdot \text{ESC}_{\text{Fr}} \cdot \text{ESC}_{\text{Fr}} \cdot \text{DF} \cdot \left( \frac{1}{\text{DFK}_2} \right) \cdot R \cdot \text{DCF}_{\text{Fr, bone}}
\]

The above two equations are used to perform the uncertainty analysis.

**STEP 2: SCREEN INPUT PARAMETERS FOR UNCERTAINTY ANALYSIS**

The parameters listed in Table 1 are screened qualitatively to identify some of those that contribute most to the uncertainty of the calculated dose. Because this study is only a preliminary analysis, a limited number of parameters are selected for uncertainty analyses. It should be noted that no sensitivity analysis is performed here, and the uncertainties of some of the parameters are left for future study, even though their contributions to the uncertainty of the offsite dose have not been shown to be negligible. After the initial screening, 13 parameters with major uncertainty are identified for further analysis. These parameters are the spent-fuel type (PWR or BWR), spent-fuel burnup, spent-fuel cooling time, the Kr-85 release fraction, the dropping height of the container, the energy partition factor, the mass fracture coefficient, the cladding-escape factor, the borehole-escape factor, the filter-decontamination factor, the wind speed, the dry-deposition coefficients, and the atmospheric-stability class.

**STEP 3: QUANTIFY UNCERTAINTIES OF SELECTED PARAMETERS**

The uncertainties of the 13 parameters identified above are quantified by developing a probability distribution for each parameter. Various probability distributions are constructed for the selected parameters on the basis of either experimental data and analysis or by engineering judgment.

**Uncertainty of Spent Fuel Type**

The accident scenarios considered in this uncertainty analysis involve the drop of a container holding consolidated spent fuel in the packaging hot cell or in the underground borehole. According to Hill,\(^1\) 585 containers, each holding 6 consolidated PWR assemblies and 327 containers each holding 18 consolidated BWR assemblies will be emplaced in the repository each year. Therefore, given a crane drop accident, the probability is about 60% that the accident will involve the drop of a container holding PWR fuel and 40% that the accident will involve BWR fuel.

**Uncertainty of Spent Fuel Burnup**

The variation of burnup for spent fuel assemblies is given in DOE.\(^1\) On the basis of that data, the probability distributions of PWR and BWR spent fuel burnup are plotted as histograms in Figure 1.

![figure](image)

**Figure 1. Probability Distribution for the Burnup of Spent Fuel**

**Uncertainty of Spent Fuel Cooling Time**

The probability distribution of the cooling time of a spent fuel assembly involved in a container drop accident can be obtained from the data given in DOE\(^1\) and from the conclusion that the occurrence of an accident is equally probable for every year. The results are shown in Figure 2 as a histogram. Note that the total probability expressed by the histogram is equal to 1.
interaction is a complicated phenomenon and requires further analysis. In addition, the container may fall sideways and land on its edge or the rods may buckle. In all these cases, only a fraction of the energy will be absorbed by the fuel pellets. For the purposes of this study, the value of the energy partition factor (EPF) of a consolidated fuel rod is assumed to range from 0.1 to 1.0 with a normal distribution.

Uncertainty of Cladding Escape Factor

After cladding rupture, the internal (void) gas that can be pressurised up to hundreds of psi for a PWR assembly is released. The released gas, which includes Kr, can expel a fraction of the fractured airborne-size fuel particles. A recent calculation\(^6\) for an end-on drop of a fuel rod showed that the bottom portion of the cladding and a few fuel particles at the bottom undergo plastic deformation (i.e., fuel fracture will occur where a cladding ruptures). In such a case, it is plausible that a large fraction of the airborne fuel particles will be carried out by the internal gas. However, under stress, the cladding could fail at the weakest point (i.e., thinnest, dented, or hydrated location). In case the cladding fails at some location other than the impact point (bottom portion), a smaller fraction of airborne fuel particles will be carried out by the internal gas. In the present study, the escape factor is taken to be between 0.1 and 1.0 with a normal distribution.

Uncertainty of Borehole Escape Factor

If it is assumed that the gases and airborne particles released from a breached cladding are mixed uniformly and that the gravitational settling and impact deposition of the airborne particles are neglected, then the particle escape factor from a borehole (ESC\(_{\text{bore}}\)) can be expressed as the ratio of two volumes, namely:

\[
\text{ESC}_{\text{bore}} = \frac{N V_r}{(N V_r + V_b)}
\]

where
- \(N\) = the number of fuel rods = 1,584
- \(V_r\) = the volume of gas released from one breached rod = 700 cm\(^3\)
- \(V_b\) = the volume of the void in the borehole and shielded cask, i.e., the borehole and cask volume minus the volume occupied by the rods and container = 4.7 m\(^3\)

These values result in a borehole escape factor of 0.2, which is also conservative for a container holding 18 BWR consolidated-fuel assemblies. Actually, the escape factor will be smaller because of the deposition of particles inside the borehole. Therefore, it is assumed that the escape factor is represented by a normal distribution with a minimum value of 0 and an maximum value of 0.2.
Uncertainty of Filter Decontamination Factor

Each filtration system is equipped with at least two HEPA filters in series. An average decontamination factor (DF) of 10⁻⁷ was assumed for the airborne fuel particles released from areas equipped with HEPA filtration systems that function properly. The actual value of DF will vary around this average value. It is assumed in this study that the filter DF for two HEPA filters connected in series can vary from 10⁻³ to 10⁻⁵, and any value within this range is equally probable.

Uncertainty of Dry Deposition Coefficient

The dry deposition coefficient, β, depends on certain characteristics, such as ground-surface roughness height, size distribution of particles, and atmospheric-stability class. Analyses indicate that possible variations in the particle size distribution will have a negligible effect on DEF; however, other characteristics can cause uncertainties in β. This study assumes that β can vary from 0.5 to 1.5 and that the distribution follows a normal distribution. It should be noted that in previous safety assessments, a DEF value of 5% was used. In this analysis, the values of DEF can vary from 0.03% to 80%.

Uncertainty of Wind Speed

The wind speed data presented by Church²⁷,²⁸ have been used to construct the probability distribution. Their data were collected from the period from July 1982 through June 1983 for a site near the proposed repository surface facilities and are summarized by the histogram in Figure 3. Figure 3 shows that the wind speed data can be fairly well represented by a lognormal distribution with a minimum wind speed of 0.5 m/s and a maximum wind speed of 10 m/s.

![Figure 3. Probability Distribution for Wind Speed](image)

Uncertainty of Atmospheric Stability Class

The atmospheric-stability class distribution can be obtained from the field data. Because stability-class data for the repository site are not available, it is assumed that stability classes C, D, E, and F are equally distributed.

Probability distributions are constructed to quantify the uncertainties of the 13 input parameters. The characteristics of these distributions are listed in Table 3. The information provided by a probability distribution is (1) the upper and lower bounds of the range of variation and (2) the type (or shape) of the distribution. Ideally, both should be determined on the basis of data or analysis. In some cases, there are sufficient data to derive the range and type of the distribution; in other cases, where sufficient data are not available to yield the range and type, subjective judgment is used.

The results in Table 3 indicate that the ranges of variation of nine parameters are determined on the basis of data or design information. The ranges of variation of three parameters (energy partition factor, cladding escape factor, and DEF) are based on judgment. The range of variation of wind speed is determined on the basis of meteorological measurements at nearby locations. The types of distribution for four parameters are based on data and those for nine parameters are based on subjective assumptions. Five of the probability distributions are normal, four are uniform, one is lognormal, and three are described by histograms.

**STEP 4: EVALUATE UNCERTAINTY OF OFFSITE DOSE**

The Latin Hypercube Sampling (LHS) Code developed by Sandia National Laboratories¹⁹,²⁰ is employed to evaluate the propagation of uncertainties of the 13 selected input parameters as listed in Table 3. The LHS method is a stratified sampling technique that ensures coverage of the entire range of a parameter. Consequently, the number of samples (N) and hence the computer time can be reduced in comparison with the Monte Carlo method. In initial trials, N was varied from 10 to 500 and the results compared. There was a discrepancy as large as 20% between the results for N = 10, 50, and 100. However, the results for N = 100 and 500 are in close agreement, indicating convergence is reached. Therefore, the number of samples is taken to be 100, and the LHS code then generates 100 sets of input parameters. Each set contains 13 values, one for each of the 13 parameters listed in Table 3.

The 100 sets of input parameters are then used to calculate 100 values of the offsite dose, each of which is equally probable (i.e., the probability of each value is 0.01). The 100 dose values are arranged in increasing order and plotted, so that the resulting curve represents the cumulative distribution function of the
TABLE 3. PROBABILITY DISTRIBUTIONS OF PARAMETERS SELECTED FOR UNCERTAINTY ANALYSIS

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Range of Distribution</th>
<th>Type of Distribution</th>
</tr>
</thead>
<tbody>
<tr>
<td>Spent fuel type</td>
<td>2 types: PWR and BWR</td>
<td>Data, Histogram, Data</td>
</tr>
<tr>
<td>Spent fuel burnup</td>
<td>0 GHD/MTU to 100 GHD/MTU</td>
<td>Data, Histogram, Data</td>
</tr>
<tr>
<td>Cooling time</td>
<td>12 yr to 33 yr</td>
<td>Data, Uniform, Subjective</td>
</tr>
<tr>
<td>Kr-85 release fraction</td>
<td>0.001 to 0.03</td>
<td>Data, Uniform, Subjective</td>
</tr>
<tr>
<td>Dangling height</td>
<td>0 to 10 m</td>
<td>Design Information, Uniform, Subjective</td>
</tr>
<tr>
<td>Energy partition factor</td>
<td>0.1 to 2</td>
<td>Judgment, Normal, Subjective</td>
</tr>
<tr>
<td>Mass fracture coefficient</td>
<td>1x10^-4 to 3x10^-4</td>
<td>Data, Normal, Subjective</td>
</tr>
<tr>
<td>Cladding escape factor</td>
<td>0.1 to 1</td>
<td>Judgment, Normal, Subjective</td>
</tr>
<tr>
<td>Borehole escape factor</td>
<td>0 to 0.2</td>
<td>Design Information, Normal, Subjective</td>
</tr>
<tr>
<td>Filter decontamination factor</td>
<td>10^-5 to 10^-3</td>
<td>Data, Lognormal, Data for adjacent area</td>
</tr>
<tr>
<td>Wind speed</td>
<td>0.5 m/s to 10 m/s</td>
<td>Data for adjacent area, Judgment, Normal, Subjective</td>
</tr>
<tr>
<td>Dry deposition coefficient</td>
<td>0.5 to 1.5</td>
<td>Data, Uniform, Subjective</td>
</tr>
<tr>
<td>Atmospheric stability class</td>
<td>4 classes: C, D, E, F</td>
<td>Data, Uniform, Subjective</td>
</tr>
</tbody>
</table>

offsite dose (see Figure 4). From the curve, dose values at various confidence levels can be readily determined. For example, there is 90% confidence that the dose at the site boundary will be less than 0.1 GHD.

RESULTS

During potential radiological accidents, the filtered ventilation exhaust system could either function properly or could fail. In the event of a crane-drop accident and concurrent failure of the filtration system, the critical organ doses (to the bone) resulting from the release of airborne fuel particles at confidence levels of 10%, 50%, and 90% are 1.9x10^-2, 1.2x10^-4, and 2.8 rem, respectively. For comparison, a previous analysis calculated the offsite dose to be 1.0 rem, which corresponds to a confidence level of 71%.

For the crane-drop accident in which the filtration system functions, the resulting critical organ doses (to the bone) at confidence levels of 10%, 50%, and 90% are 2.8x10^-4, 1.2x10^-6, and 2.8x10^-3 rem, respectively. The dose reported in Reference 1 for this scenario is 2x10^-3 rem, which corresponds to a confidence level of 90%.

In the event of a container-drop accident and concurrent failure of the filtered ventilation exhaust system, the critical organ doses (to the bone) at confidence levels of 10%,
50% and 90% are 6.0x10^{-4}, 2.5x10^{-2}, and 2.1x10^{-1} rem, respectively. Reference 3 reported this dose to be 1.1x10^{-1} rem, which corresponds to a confidence level of 82%. For the container failure accident in which the filtration system functions, the offsite doses at confidence levels of 10%, 50%, and 90% are 6.7x10^{-8}, 6.4x10^{-6}, and 2.4x10^{-4} rem, respectively. The dose reported in Reference 3 for this scenario is 2.1x10^{-4} rem, which corresponds to a confidence level of 95%.

The offsite lung doses attributable to Krypton-85 are about the same for both types of accidents because the release mechanisms of Krypton-85 are similar. These doses are calculated to be 1.4x10^{-7}, 1.4x10^{-6}, and 2.2x10^{-5} rem at confidence levels of 10%, 50%, and 90%, respectively. The dose reported for the same scenario in Reference 1 is 2.2x10^{-4} rem, which corresponds to a confidence level of 99%.

CONCLUSIONS

The uncertainty-analysis methodology and computer codes discussed in this paper can be applied to a broad range of radiological safety analyses for the preclosure period. This study, however, does not account for some of the uncertainties associated with accident dose calculations, such as radionuclide transport within the building/facility, dose-conversion factors, and model uncertainties. Future studies can investigate these topics and other potential accident scenarios.

REFERENCES


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