

SAFETY ANALYSIS APPROACHES FOR MIXED TRANSURANIC WASTE

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ABSTRACT

Argonne National Laboratory (ANL) has completed a survey of assumptions and techniques used for safety analyses at seven sites that handle or store mixed transuranic (TRU) waste operated by contractors for the U. S. Department of Energy (DOE). While approaches to estimating on-site and off-site consequences of hypothetical accidents differ, there are commonalities in all of the safety studies. This paper identifies key parameters and methods used to estimate the radiological consequences associated with release of waste forms under abnormal conditions. Specific facilities are identified by letters with their safety studies listed in a bibliography rather than as specific references so that similarities and differences are emphasized in a nonjudgmental manner. References are provided for specific parameters used to project consequences associated with compromise of barriers and dispersion of potentially hazardous materials. For all of the accidents and sites, estimated dose commitments are well below guidelines even using highly conservative assumptions. Some of the studies quantified the airborne concentrations of toxic materials; this paper only addresses these analyses briefly, as an entire paper could be dedicated to this subject.

APPROACH TO SAFETY COMPLIANCE

While all of the safety studies evaluated the potential consequences of abnormal conditions, their approaches differed markedly. Some of the facilities are used for storage only; others examine and repackage transuranic waste; yet another is undergoing decontamination. All of the studies estimate offsite radiological consequences from inhalation of TRU radionuclides. Some also project dose commitments from inhalation of uranium and fission and activation products. For radionuclides, the total effective dose equivalent (TEDE) is the parameter of interest; it includes contributions from radioactive materials taken into the body by inhalation and ingestion and exposures to penetrating radiation from sources that remain outside of the body. Since radionuclides deliver a dose rate as long as they remain within the body, inhaled and ingested materials impart a committed effective dose equivalent (CEDE). Photons given off by radionuclides outside of the body impart a deep dose equivalent (DDE) as long as the individual remains near them. The sum of the CEDE and the DDE equals the TEDE. Values of the TEDE, CEDE, and DDE are expressed in units of rem or Sieverts or fractions thereof. One Sievert equals 100 rem.

One major difference in the seven studies was the basic approach to the calculations required to determine facility safety. Some studies assumed a given amount of materials at risk within the

facility, and others began with a target TEDE to a maximally exposed offsite individual (MEOI) and worked back to determine the maximum amount of materials that could be at risk in any accident. The quantity at risk could be associated with the entire facility or with a given storage location that might become involved with a fire or with a particular operation, such as moving a pallet of drums. None of the guideline exposures for the MEOI, who is assumed to be a member of the general public, exceeded 25 rem for any accident probability in any facility. Some studies considered workers within the facility or located at nearby facilities. Sources are localized in four sites; they are distributed throughout multiple buildings at the other three sites. Determination of compliance with guidelines for the seven sites are summarized in Table I.

Table I Calculational Approach To Determine Facility Compliance With Guidelines

SITE	APPROACH
A	Find material at risk based on 1 rem and facility inventory on 25 rem to the MEOI
B	Find the TEDE to facility workers and at 0.1 and 3 km and to the MEOI at 6 km
C	Find the TEDE at the site gate (280 m away) and at the MEOI at 5 km
D	Find the TEDEs to worker at 0.1 km and MEOIs at 0.09, 0.15, and 0.24 km
E	Find TEDEs to worker at 0.3 km and to the MEOI at 1.1 km
F	Find TEDEs to the MEOI at nearest site boundary in 16 compass sectors
G	Find TEDEs to the MEOI at 1.9 and 4.0 km

For Site A, the facility inventory guideline of 25 rem is based on an incredible accident scenario. There are several facilities at Site G; the distance to the nearest MEOI varies between 1.2 to over 2 km. The distance of 1.9 km is selected as typical. If there is enough energy released, the effluent could be carried aloft; the highest offsite concentration for such an elevated release occurs at a distance of 4 km rather than closer to the facility.

Potential concentrations of toxic chemicals at selected receptors are quantified for sites B, D, and G and treated only qualitatively in A, C, E, and F. Comparisons are made to the permissible exposure limit-time weighted average (PEL-TWA) or to the threshold limit value-time weighted average (TLV-TWA) for events that are anticipated. For less probable events, higher concentrations such as the permissible exposure limit-short term exposure limit (PEL-STEL), the threshold limit value-short term exposure limit (TVL-STEL), the emergency exposure guidance level (EEGL), or a fraction of the immediately dangerous to life or health (IDLH) have served as criteria for offsite receptors. Off site receptors remain in place for the duration of the releases, but workers in the facility have limited exposure times, on the order of minutes. A detailed discussion of the treatment of toxic chemicals is beyond the scope of this paper. The criteria for evaluating non-radiological, potentially hazardous materials are clearly addressed in Appendix A of a recent Nuclear Regulatory Commission report, Reference 1.

ACCIDENT SCENARIOS

Depending on their operations and locations, studies considered the effects of fires, explosions, drops or spills, and natural phenomena, such as earthquakes. Site specific accidents with initiators within the facility such as a fire or explosion and spills of liquids or finely divided solids are analyzed in all studies. Except for Site G, accidental criticality is not considered to be credible because of the dilution of nuclear fuel, the presence of neutron absorbers, and the fact that activities are unlikely to reconfigure fissile materials into a critical mass. All seven documents estimate annual probabilities for the hypothetical events addressed. Generally, dose

commitments are found for accidents deemed to be anticipated, unlikely or highly unlikely. Some studies listed expected non-routine events and incredible accidents, but did not estimate their radiological consequences. In any hypothetical accident, only a fraction of the total inventory of TRU material is deemed to be at risk.

GENERIC RADIONUCLIDE SOURCE TERMS FOR CONTACT-HANDLED TRU WASTE

Because of the variety in radionuclides and quantities at different sites, no quantitative information is provided herein for radionuclide inventories at specific sites. However, Table II summarizes bounding radiological source terms for common TRU isotopes and isotopic mixtures relative to the Waste Isolation Pilot Plant (WIPP) Waste Acceptance Criteria (WAC), Reference 2, limits. For contact-handled (CH) TRU waste, i.e., payload containers that have a maximum contact dose rate of 200 mrem/hr, three limits are imposed by the WIPP WAC. The fissile or fissionable radionuclide content of 55-gallon drums is limited to 200 Pu-239 fissile-gram-equivalents (FGE), calculated by the methods prescribed in the TRUPACT-II Safety Analysis Report for Packaging (SARP), Reference 3. The TRUPACT-II SARP references an ANSI/ANS standard, "American National Standard for Nuclear Criticality Control of Special Actinide Elements," Reference 4, for FGE determinations. Reference 4 lists subcritical mass limits for fissile and fissionable actinide nuclides. FGE limits for various actinides are included in Table II. The second WIPP WAC-imposed limit is a Pu-239 Equivalent Ci (PE-Ci) limit. For a 55-gallon drum of untreated CH-TRU waste, the limit is 80 PE-Ci. Appendix A from Reference 2 describes the method for determining the PE-Ci for several radionuclides. A weighting factor is used, based on 50-year CEDEs determined by methodology described in the International Commission on Radiological Protection (ICRP) Publications 26 and 30, References 5 and 6. The PE-Ci for various radionuclides are included in Table II. The third WIPP WAC-imposed limit for CH-TRU waste is the alpha activity concentration from TRU radionuclides, which must be greater than 100 nCi per gram of waste matrix. For the purpose of comparison, two waste matrix mass conditions are assumed in Table II, a minimum case of 4536 grams (10 pounds) which would be possible for light waste material such as paper, and a maximum case of 362,880 grams (800 pounds) which would be possible for heavy waste material such as cemented sludge or metal. These are not WIPP WAC-imposed mass limits, but instead are assumptions to represent bounding scenarios for the purpose of summarizing potential source terms for CH-TRU waste.

Table II Nuclear Source Term Summary for CH-TRU Waste in 55-Gallon Drums

PARAMETER	Pu-239	Pu-238	Am-241	Pu-52 (Case 1: No Decay) [f,g]	Pu-52 (Case 2: 28.8-Yr. Decay) [f,g]	Pu-52 (Case 3: High Am-241 Conc.) [f,g]
Specific Activity [a]	6.29E-02	1.73E+01	3.47E+00	--- [i]	--- [i]	--- [i]
FGE Factor [a]	1.0	0.113	0.019	--- [j]	--- [j]	--- [j]
Maximum Isotopic Mass (g) [b]	200.0	5.1 [h]	23.1 [h]	211.2	218.0	96.3 [h]
FGE (g) of Maximum Isotopic Mass	200.0	0.58	0.43	200.0	200.0	71.4
PE-Ci Weighting	1.0	1.1	1.0	--- [k]	--- [k]	--- [k]

Factor [c]						
PE-Ci (Ci) for Maximum Isotopic Mass	12.6	80.0	80.0	17.1	38.0	80
Maximum Drum Activity (nCi) Based on Maximum Isotopic Mass	1.26E+10	8.80E+10	8.00E+10	9.04E+10	5.58E+10	1.06E+11
“Heavy” Drum Minimum Activity (nCi) [d]	3.63E+07	3.63E+07	3.63E+07	3.62E+07 [l]	3.63E+07 [l]	3.62E+07 [l]
“Heavy” Drum Minimum Isotopic Mass (g) [d]	0.58	2.10E-03	1.05E-02	0.55	0.54	0.43
“Heavy” Drum Maximum Concentration (nCi/g-waste) [d]	3.47E+04	2.42E+05	2.20E+05	2.49E+05	1.54E+05	2.91E+05
“Light” Drum Minimum Activity (nCi) [e]	4.54E+05	4.54E+05	4.54E+05	4.26E+05	4.28E+05	4.32E+05
“Light” Drum Minimum Isotopic Mass (g) [e]	7.21E-03	2.62E-05	1.31E-04	6.79E-03	6.61E-03	5.32E-03
“Light” Drum Maximum Concentration (nCi/g-waste) [e]	2.77E+06	1.94E+07	1.76E+07	1.99E+07	1.23E+07	2.33E+07

(SEE NOTES FOR TABLE II ON FOLLOWING PAGE)

NOTES FOR TABLE II

- [a] Information obtained from Reference 3.
- [b] Calculated from the FGE limit of 200g per drum, unless otherwise noted.
- [c] Information obtained from Reference 2.
- [d] "Heavy drum" scenarios assume a waste mass of 362,880 grams (800 pounds).
- [e] "Light drum" scenarios assume a waste mass of 4536 grams (10 pounds).
- [f] Three cases are presented for weapons-grade plutonium (Pu-52), with varying amounts of Am-241. For the first case, no decay is assumed (i.e., no Am-241). In the second case, 2 half-lives-worth of Pu-241 decay (28.8 years) is assumed. In the third case, a fairly high concentration of Am-241 is assumed to be mixed in with Pu-52, such as could be the case in waste originating from an americium recovery process. The mass ratio between Am-241 and Pu-239 for Cases 2 and 3 are 0.03 and 0.3, respectively.
- [g] The Pu-52 isotopic make-up (weight percent) with no decay (Case 1) consists of Pu-238 0.01%, Pu-239 93.83%, Pu-240 5.82%, Pu-241 0.34%, and Pu-242 0.02%; these were obtained from Reference 7. For Case 2 the weight percents for the same isotopes are: 0.01%, 91.32%, 5.66%, 0.08%, 0.02%, and Am-241 2.90%. for Case 3, the weight percents for the same isotopes are: 0.01%, 73.05%, 4.53%, 0.26%, 0.02%, and Am-241 22.12%. For Case 2, only decay from Pu-241 and in-growth of Am-241 were considered. For Case 3, the masses for Case 1 were used, except that Am-241 was added to provide a mass ratio between Am-241 and Pu-239 of 0.3.
- [h] The PE-Ci limit of 80 Ci per drum is more limiting than the 200 FGE limit per drum, so the values shown correspond to 80 PE-Ci.
- [i] Specific activities for other Pu-52 isotopes are: 1.73E+01 for Pu-238; 2.30E-01 for Pu-240; 1.04E+02 for Pu-241, and 3.97E-03 for Pu-242. These were obtained from Reference 3.
- [j] FGE factors for other Pu-52 isotopes are: 0.113 for Pu-238; 2.25E-02 for Pu-240; 2.25 for Pu-241, and 7.50E-03 for Pu-242. These were obtained from Reference 3.
- [k] PE-Ci Weighting Factors for other Pu-52 isotopes are: 1.1 for Pu-238; 1.0 for Pu-240; 52.0 for Pu-241, and 1.1 for Pu-242. These were obtained from Reference 2.
- [l] The activity from Pu-241 is not included since it does not meet the definition of TRU waste since the half-life is less than 20 years.

SITE RADIONUCLIDE INVENTORIES AND MATERIALS AT RISK

All installations consider Pu-238, Pu-239, Pu-240, Pu-241, and Am-241; some consider Cm-244 and other TRU nuclides. Six sites list specific radionuclides that are at risk. Site A estimates contributions to the TEDE from 97 nuclides; fewer nuclides are considered in the other six studies. Site E expresses the activity of all nuclides in terms of "equivalent Curies of Pu-239" as defined in the WIPP WAC, Reference 2. Fission and activation products contribute little to the CEDE because their activities are low and they are far less effective in delivering dose than the TRU nuclides per Curies (Ci) inhaled. Site D includes a facility that contains several thousand Ci of tritium; their study provides estimates of the CEDEs for this nuclide even if no TRU nuclides escape into the environment.

Of all the materials at risk, only a fraction can pass through confinement barriers and engineered safety systems into the atmosphere. All of the radioactive materials are in stored in robust containers, such as 55-gallon drums, unless they are being characterized or repackaged. Determination of the appropriate release fraction is based on engineering judgement with some guidance from regulatory agencies. Atmospheric transport, dilution, and dispersion incorporates

conservative assumptions that neglect removal by natural processes such as dry fallout and scavaging by precipitation. Concentrations of airborne materials at the various receptor locations are estimated in units of Ci or μCi per m^3 for radionuclides and μg per m^3 for toxic chemicals. Estimated concentrations of chemicals and TEDEs are compared to guidelines for three classes of accident probability: anticipated, unlikely, and extremely unlikely.

WASTE CONTAINERS

The most common waste container is the 55-gallon drum; drums may contain plastic liners to reduce the potential for leakage. Much of the material is also contained in one or more layers of plastic bags. Some sites store large items in wood or metal boxes. Drums may be stored within other layers of confinement such as overpacks or concrete culverts. Even for facilities that have high confidence about the content of each container, variations in the amount of radioactivity and toxic chemicals complicate the safety analyses. Materials at risk can be based on the container with the highest contents of actinides or an average for all of the containers at risk could be selected. A drum with the highest total radioactivity could contain only small amounts of toxic chemicals, and some nuclides are more effective than others in imparting radiation dose per unit of activity inhaled.

At three of the sites (C, D, and E), waste is removed from its primary container for sampling, characterization, or processing before it is repackaged. During these operations, only a limited quantity is at risk during any given time interval, and there is at least one barrier to prevent release to the environment. The degree of confinement depends on the quantity and form of the waste material. One site, G, considers significant amounts of un-contained materials because it has a number of buildings that are being decontaminated and decommissioned. That site has actinides in wet and dry resins, filters, solutions, and surface contaminants as well as confined TRU nuclides.

RELEASE FRACTIONS AND RATES

Release fractions are selected based on the accident scenario and the credible pathways for escape from primary and secondary containers, and the facility into the environment. Release fractions are based on the engineering judgement of experienced analysts with guidance from sources such as Reference 8 which contains comprehensive lists of release fractions acceptable to the DOE. Another source of release fractions, Reference 9, contains empirical formulas that can be adapted to a variety of situations. These two references provide numerical guidance for estimations of the fractions of the materials at risk that can become airborne, have size distributions that are respirable, escape passive barriers (such as damaged containers), or bypass engineered safety systems (such as filter trains). Most of the TRU nuclides are in the form of nonvolatile solids; some limited amounts may be dissolved in volatile liquids. None of the TRU materials are in the form of volatile or semi-volatile solids or gases which are assigned higher release fractions that can approach unity.

Highly conservative total release fractions provided in Reference 10 for determining the hazard category of nuclear facilities may be modified to reflect specific conditions within a given facility. Values in this reference depend on the physical form of the waste only and not on the type of accident postulated. For example, waste in the form of solids, liquids, or powders is assigned a release fraction of 0.001; for tritium vapor, it is unity. Most of the studies consider detailed accident scenarios and develop appropriate release fractions for any given situation. This

approach is recognized in a recently issued standard, Reference 11, which is based on the research described in Reference 8.

Only fine particles with size distributions around 10 μm or less are likely to be respirable or to remain airborne for periods of time necessary to be transported beyond the immediate volume surrounding their point of release. References 8, 9, and 11 distinguish between the amount of material that can become airborne under a given set of conditions and the fraction of that material that is respirable. The product of these two numbers is the fraction of the material at risk that is available for transport in the atmosphere and inhalation by receptors of interest unless inhibited by barriers, such as filters, or by natural processes, such as gravitational settling.

As the number of levels of confinement increase, the release fractions decrease for any given level of thermal or mechanical stress on the containers from abnormal incidents. Hypothetical accidents appropriate to facility operations are developed that challenge the barriers to release. Most of the time, the potentially hazardous materials are secured in containers that are unlikely to be breached. Where appropriate, filtered vents allow gases and vapors to escape to prevent pressurization. Very few of the facilities studied conduct operations that are likely to provide enough kinetic or thermal energy to cause significant amounts of the TRU nuclides to become airborne for extended periods. References 8, 9, and 11 provide values for a leak path fraction for various barriers that could reduce the amount of material released. Leak path fractions are developed for specific locations within buildings at Site G. Because of the large quantity of materials formerly processed at Site G, measurements were conducted for rooms, corridors, and loading docks to an array of credible accidents. Such parameters are incorporated in safety studies for individual facilities at Site G.

Most of the TRU nuclides are in the form of a powder. None of the facilities store or operate on the wastes in an inert atmosphere, so material released is likely to be in an oxide form. In a realistic scenario, radioactive materials and chemicals would be released at a finite rate that depends on the amount and form of energy available to eject them from their normal locations within one or more levels of confinement. The magnitude of energy available to compromise barriers and to cause the potentially harmful materials to become airborne depends on the accident scenarios assumed. The energy and release fractions are the highest for explosions or fires. Development of release fractions for some sites includes estimates of the energy yield of fires and different types of explosions. For "mechanical" accidents such as dropped containers or spills, the amounts of energy are less. For the offsite receptors, the most common assumption is that the release is instantaneous; these "puff" releases eliminate the uncertainties associated with estimating appropriate release rates from the facility. Release rates are estimated for receptors that are very close to the release point because concentrations would be high for "puff" releases and realistic inhalation times would be short.

Authors of the seven studies differ in the way they present releases for the materials at risk, but the total release fraction (TRF) can be expressed as the product of three factors: the fraction that can become airborne (ARF), the respirable fraction (RF), and that fraction that can escape the facility into the atmosphere, the leak path fraction (LF). The LF is the product of a local confinement fraction (CF) from passive effects, such as plate out of active elements or removal of suspended particles by gravitational forces, and active systems (AS), such as filter trains. Tables III through XI show each factor for the seven sites for the events considered. All events that damage containers by transfer of kinetic energy rather than heat are labeled as spills in these

seven tables. Unfiltered releases show AS as 1.0. Formats in Tables III through XI differ because of the assumptions in the studies. Unless otherwise noted, the TRU waste is a solid or a powder.

Table III Release Fractions For Site A - Storage Of Packaged Waste Only

EVENT	WASTE FORM	ARF	RF	CF	AS	TRF
Fire	Volatile	0.5	1.0	0.5	1.0	0.25
Fire	Semivolatile	0.1	1.0	0.5	1.0	0.05
Fire	Other	0.01	1.0	0.5	1.0	0.005
Spill	Volatile	0.5	0.7	0.5	1.0	0.175
Spill	Semivolatile	4.0E-04	0.7	0.5	1.0	1.4E-04
Spill	Other	4.0E-05	0.7	0.5	1.0	1.4E-05

Table IV Release Fractions For Site B - Storage Of Packaged Waste Only

EVENT	WASTE FORM	ARF	RF	CF	AS	TRF
Fire	Combustible Solid (35%)	5.0E-04	1.0	1.0	1.0	5.0E-04
Fire	Noncombustible Solid (65%)	6.0E-03	1.0	1.0	1.0	6.0E-03
Fire	Aggregated Solids	2.1E-04	1.0	1.0	1.0	2.1E-04
Explosion	Any Solid	1.0E-03	1.0	1.0	1.0	1.0E-03
Spill	Any Solid	1.0E-04	1.0	1.0	1.0	1.0E-04

Table V Release Fractions For Site C - Characterization And Short Term Storage

EVENT	FILTERED	ARF	RF	CF	AS	TRF
Fire	No	0.010	0.05	0.5	1.00	2.5E-04
Fire	Yes	0.010	0.05	0.5	0.01	2.5E-06
Spill	No	0.001	0.05	0.5	1.00	2.5E-05
Spill	Yes	0.001	0.05	0.5	0.01	2.5E-07

Table VI Release Fractions For Site D - Characterization, Sampling, And Storage

EVENT	WASTE FORM	ARF	RF	CF	AS	TRF
Any	Tritium Vapor	1.0	1.0	1.0	1.0	1.0
Fire	Combustible Solid	5.0E-05	1.0	1.0	1.0	5.0E-05
Fire	Combustible Liquid	1.0E-03	1.0	1.0	1.0	1.0E-03
Spill	Any Solid Or Liquid	0	0	0	0	0

Table VII Release Fractions For Site E - Sampling And Short Term Storage

EVENT	FILTERED	ARF	RF	CF	AS	TRF
Fire	No	0.006	0.1	1.0	1.0	6.0E-04
Fire	Yes	0.006	0.1	1.0	0.001	6.0E-07
Spill	No	0.005	0.1	1.0	1.0	5.0E-04
Spill	Yes	0.005	1.0	1.0	0.001	5.0E-06

Table VIII Non-Impact Release Fractions for Site F - Storage Of Packaged Waste Only

EVENT	FILTERED	ARF	RF	CF	AS	TRF
Fire	No	5.0E-04	1.0	1.0	1.0	5.0E-04
Explosion	No	1.0	1.0	1.0	1.0	1.0
Overpressure	No	0.001	1.0	1.0	1.0	0.001
Spill	No	0.001	0.1	1.0	1.0	1.0E-04

Table IX Impact Release Fractions For Site F - Storage Of Packaged Waste Only

PHASE	SOLID WASTE FORM	ARF	RF	CF	AS	TRF
Shock	Combustible	0.001	0.1	1.0	1.0	1.0E-04
Entrainment	Combustible	1.0E-04	1.0	1.0	1.0	1.0E-04
Shock	Noncombustible	0.001	1.0	1.0	1.0	0.001
Entrainment	Noncombustible	1.0E-04	1.0	1.0	1.0	1.0E-04

In Table IX, the authors of the Site F study adopted a total fractional release for impact scenarios based on addition of the TRF values for both phases of the accident sequence. This yields releases of 2.0E-04 or 0.02% and 1.1E-03 or 0.11% of the materials at risk for combustible solids and noncombustible solids, respectively. The TRF for the shock phase is lower for contaminants on combustible materials because they generally have rough or porous surfaces that retain more of the material. To obtain the ARE value for the entrainment phase, a release rate of 4.0E-06 per hour suggested by Reference 8 is multiplied by 24 hours and rounded up to 1.0E-04.

Tables X and XI provide information on release fractions for the events considered for Site G that could be encountered at other sites. Values for materials such as volatile organic and aqueous solutions, wet and dry resins, and coated bulk metal that are unique to the various facilities located at this site are not included in this paper. Hydrogen explosions within containers of radioactive materials are modeled as overpressure events. Explosions that occur outside of containers are modeled as spills.

Table X Confined Material Release Fractions For Site G - Decontamination And Short Term Storage

EVENT	FILTERED	ARF	RF	CF	AS	TRF
Fire	No	5.0E-04	1.0	1.0	1.0	5.0E-04
Overpressure	No	0.001	1.0	1.0	1.0	0.001
Spill	No	0.001	0.1	1.0	1.0	1.0E-04

Table XI Unconfined Material Release Fractions For Site G - Decontamination Short Term Storage

EVENT	WASTE FORM	ARF	RF	TRF
Fire	Combustible Materials	0.05	1.0	0.05
Fire	Noncombustible Materials	0.006	0.01	6.0E-05
Free Spill	Combustible Materials	0.001	1.0	0.001
Free Spill	Noncombustible Materials	0.001	1.0	0.001

RECEPTORS

For most sites, contributions from waterborne sources are negligible. Therefore, emphasis is on estimates of the atmospheric dispersion factor, χ/Q , in s/m^3 at the receptor locations of interest. The product of the rate of release in μCi per second and χ/Q in s/m^3 gives the airborne concentration, χ , in $\mu\text{Ci/m}^3$.

All of the safety analyses projected TEDEs to an adult male receptor located at the nearest site boundary. One study (G) estimated the CEDE to a child in the same location. Several studies estimated TEDEs to workers exiting the site through a controlled point, such as a visitors reception center. Other studies estimated TEDEs to workers within the facility involved in the accident. The time spent by the receptor at the location of interest ranged from a few minutes for in-facility workers to two hours for a member of the public at the nearest site boundary. If a "puff" release is assumed, then the stay time at a given location does not enter the estimations. The maximum amount of the airborne respirable particles, vapors, and gases are inhaled.

CONVERSIONS FOR INHALED RADIONUCLIDES

The product of the breathing rate, time of inhalation, and airborne concentration estimates the amount of materials taken into the receptor. To estimate the CEDE for specific radionuclides, another factor is required that gives the rem per μCi inhaled. For TRU wastes, the TEDE is dominated by the CEDE from inhalation. Contributions to the TEDE from ingestion and from radionuclides that remain outside of the body are not significant.

Only small particles, on the order of $10\ \mu\text{m}$ or smaller, can remain airborne for the time needed to be transported to receptors located more than a few meters from their release point. As the size of the particles increase, their ability to reach the pulmonary region of the lung decreases rapidly. Particles larger than $10\ \mu\text{m}$ are unlikely to be respirable. Contributions to the TEDE from ingestion of radionuclides and from materials that remain outside of the body and from external sources are minor for the actinides.

Four sources of conversion factors can be used to determine the CEDE received over the 50 years following the acute inhalation. The highest CEDEs per μCi inhaled were determined using Reference 12, updated by Reference 6 for TRU nuclides. Because of their conservative predictions and their acceptance by regulatory bodies, these ICRP-30 values were the most commonly used and incorporated into programs such as Reference 13. Reference 14, ICRP-68, and Reference 15, ICRP-72, reflect some 20 years of advances in radiobiology and yielded inhalation dose commitments that were factors of two to three lower than those found using ICRP-30. The terminology also has changed; the CEDE has been replaced by the committed effective dose (CDE) in References 14 and 15. Instead of a TEDE, a total effective dose (TED)

is found by adding contributions from radionuclides that remain outside the body to the CDE. In this paper, the TEDE and CEDE terminology is retained because most of the analyses are based on conversions found in References 12 and 6. For all practical purposes, either the old or new terminology is adequate to quantify the radiological consequences associated with the release of radioactive materials into the environment.

One site (G) applies conversions for enriched and depleted uranium and weapons grade plutonium. To assure conservative results, plutonium calculations include estimates for material that has aged or decayed for 72 years after the most recent americium separation. At this decay time, the content of Am-241 reaches a maximum. Estimates for the CEDE are found using conversion factors from Reference 13 for both 50- and 70-year post-inhalation periods. The longer period is appropriate for inhalation by children; all of the other facilities include only the value for 50 years which is applicable to adults. Because of the variety of chemicals present at this site, plutonium CEDEs are given for both "W" and "Y" lung clearance rates. Contributions from uranium are estimated for all three lung clearance rates ("D", "W", and "Y").

Table XII shows the various references used for inhalation conversion factors. It should be noted that the latest inhalation conversion factors contained in Reference 15 are not selected for any of the studies reviewed. In the study for Site G, conversions embedded in Reference 16 are based on References 12 and 6.

Table XII Sources For Inhalation Conversion Factors

SITE IDENTIFIER	CONVERSION FACTOR REFERENCES
A	ICRP-68, Reference 14
B	ICRP-30, Reference 12
C	ICRP-68, Reference 14
D	Reference 13, based on References 12 and 6
E	ICRP-30, Reference 12 for Pu-239 Only
F	ICRP-30, References 12 and 6
G	Reference 16 based on References 12 and 6

RADIOLOGICAL DOSE GUIDELINES

Most of the safety studies provide guidelines for the TEDE for anticipated, unlikely, and highly unlikely accidents. Others adopted a guideline of 25,000 mrem for any credible accident. Specific TEDE guidelines vary with the facility and the maximally exposed off site receptor, a member of the general public. Some studies estimated a TEDE for the maximally exposed worker within the facility or to a worker located on site. Table XIII shows the criteria applied to the MEOI in the safety studies reviewed.

Table XIII Maximally Exposed Offsite Individual Guidelines (mrem)

SITE IDENTIFIER	ANTICIPATED ACCIDENT	UNLIKELY ACCIDENT	EXTREMELY UNLIKELY ACCIDENT
A	N/A	N/A	1,000
B	1,000	5,000	25,000
C	500	1,250	5,000
D	25,000	25,000	25,000
E	25,000	25,000	25,000
F	500	5,000	25,000
G	500	5,000	25,000

SUMMARY

As expected, there is a wide variation in the parameters selected for the seven safety analysis reports. The DOE prescribes a graded approach to safety studies that allows analysts to determine the required level of detail based on the magnitude of the hazard, the complexities of the operations, and the life cycle stage of the facility. Appropriate parameters, such as release fractions, are determined by engineering judgement and the potential for exposure of on site and off site receptors to radioactive or toxic materials. In the seven studies reviewed, the level of detail increases as the number of storage locations and activity of TRU nuclides increase. Two of the studies extensively develop the basic theory behind release fractions; the others cite applicable reference documents. Most of the studies incorporate the key safety analysis parameters into spreadsheets; two studies rely on packaged safety analysis programs for predictions of the TEDE and CEDE values at the receptors of interest. In spite of the highly conservative assumptions, the projected consequences are well within guidelines developed by the facility in conjunction with the DOE.

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