FRMAC Assessment Manual

Pre-assessed Default Scenarios

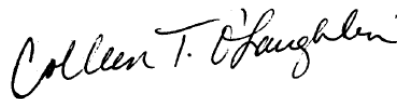# Volume 3

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FRMAC is an acronym for Federal Radiological Monitoring and Assessment Center.

PREFACE

This Federal Radiological Monitoring and Assessment Center *(FRMAC) Assessment Manual* has been extensively peer reviewed and reflects the consensus of radiological emergency response elements in the federal family. As the manual is used, we expect to receive many comments and suggestions. All comments and suggestions will be evaluated by the FRMAC Assessment Working Group and incorporated in future revisions, as appropriate.

This manual has been prepared primarily by representatives of those federal agencies which can be expected to play the major roles in a radiological emergency: the U.S. National Nuclear Security Administration (NNSA), U.S. Nuclear Regulatory Commission, and U.S. Environmental Protection Agency (EPA). All three agencies approve and authorize the release of this document. The U.S. Department of Agriculture and the U.S. Food and Drug Administration also made significant contributions regarding food safety issues. A broader audience, including representatives of the Council of Radiation Control Program Directors, was employed as a peer review group.

To ensure consistency, completeness, and the highest quality of assessed data produced by the FRMAC, assessment methods and values chosen had to be (1) scientifically defensible, (2) simple, (3) applicable to a FRMAC deployment, and (4) likely to be adopted by others. The primary purposes of this manual are to provide the user with:

- A sound scientific basis in technical and assessment processes, plus conversion values that have been agreed upon ahead of time. This assures that the correct values will be used and the results will be consistent among users from shift to shift.

- One document, for quick and easy use, that contains all technical values and assessment processes expected to be used during a radiological response. However, reference documents will be available at the FRMAC for special assessments that are not included in this manual.

This printing includes both controlled and uncontrolled copies. The controlled copies will be updated as needed. Each organization with responsibility to use this manual during a response maintains controlled copies. Only controlled copies should be used during a response.
The NNSA Nevada Operations Office has responsibility for maintaining the master copy. Please provide comments to Colleen T. O'Laughlin, U.S. National Nuclear Security Administration Nevada Operations Office, P.O. Box 98518, Las Vegas, NV 89193-8518.

The EPA is currently revising their *Protective Action Guideline* (PAG) manual (EPA92). It is anticipated that the EPA PAG manual will establish modified default assumptions and methods for performing radiological assessments which will necessitate the revision of the *FRMAC Assessment Manual*. The revision of the *FRMAC Assessment Manual* will begin once the FPA PAG manual revision is completed.

The revision of the Volume 3 of the *FRMAC Assessment Manual* is necessitated by the development of an additional pre-assessed scenario located in Section 8 *Radiological Dispersal Device Accident*. Section 8 provides information concerning the radiological impacts of a radiological dispersal device.

ACKNOWLEDGMENTS

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Volumes 1 and 2:

Tom Buhl (LANL) – resuspension approach

Harvey Clark (NSTec) – general and administrative methods

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The remainder of the manual is divided into independent sections. Listed below are

the most current revisions of each section.

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OVERVIEW/INTRODUCTION

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SECTION 1. OVERVIEW/INTRODUCTION

Purpose of This Volume

This third volume of the Federal Radiological Monitoring and Assessment Center's (FRMAC's) *FRMAC Assessment Manual,* "Pre-Assessed Default Scenarios," was constructed to be a quick-start guide to expedite response for these pre-assessed scenarios. That is, each scenario was assessed for a default case, which was selected as "typical." The volume also provides supplemental information that is unique to these scenarios. Assessment of each default scenario was conducted using the principles of the Data Quality Objective (DQO) approach. The process resulted in the definition of default Derived Response Levels (DRLs) for the key decisions associated with each of the published Protective Action Guides (PAGs) and worker protection guidelines. Special details and simplifications for each scenario are developed by phase of consideration (worker protection, Early Phase, Intermediate Phase, and Ingestion). Each of the six scenario-specific sections has the same structure. The specific objectives are to:

- Provide default DRLs for the key decisions (worker protection, evacuation/sheltering, relocation, and agricultural hold). These default DRLs will be used to assess data until sufficient information has been accumulated to eliminate assumptions used in their calculation. This gives assessors an instant start on the problem and avoids needless adjustment of DRLs.

- Identify the measurable quantities that are most effective for assessment of the scale and scope of impact. Measurement techniques are not specified. Only the quantity to be measured is specified (corresponding to DRLs). This focuses the efforts of monitoring and sampling on those measurements that have the greatest value.

- Provide guidance regarding required sensitivity and Minimum Detectable Activity (MDA). This helps the laboratories make the best use of their capabilities.

- Offer detailed information regarding each type of problem that does not fit the scope of Volumes 1 and 2.

- Highlight simplifications, rules-of-thumb, and the pitfalls unique to each scenario.

- Initiate use of the DQO process to guide prudent monitoring and assessment for radiological characterization. This sets the stage for application of a formal DQO process during recovery phase.

- Facilitate revision of DRLs by highlighting key considerations and prerequisites.

Applicability

The goal of this volume is the timely initial assessment of these "expected" scenarios. It is recognized that detailed ongoing characterization and assessment will follow. This volume is not a substitute for the methods of Volumes 1 and 2. The "typical" problem considered to assess the

scenario is not the only possible situation. There can be many variations that lead to more and less severe consequences. Use the default DRLs until there is technical justification to abandon them. This could occur very early, if the actual situation differs greatly from that anticipated. Eventually, the defaults will always be replaced by incident-specific assessment methods and DRLs. However, this will not occur until sufficient information exists to justify the change. The change will be deliberate and made in consultation with the Federal Advisory Team.

Get Oriented

- <u>Consumers of FRMAC Assessment products</u> generally do not need to be concerned with this volume. Please review Volume 1, Section 1 (Overview of Assessment).

- <u>Assessors</u>, if this is an emergency, please begin your response by reading the scenario-specific section for the problem at hand in this volume. Also, review Section 6 of Volume 1 (Administration, Internal Procedures, and Tools).

- <u>Assessment Group members</u> or all individuals routinely assigned to the Assessment Group must be conversant with the <u>complete</u> manual (all 3 volumes).

Admonitions

1. First, consult the accident-specific scenario section of Volume 3 for the problem at hand.

2. Use the default DRLs and DILs until sufficient measurements are available to eliminate assumptions used in their calculation. This may take several days.

3. Never use more than two significant figures, because the approximations, assumptions, limitations, and uncertainties will not support more.

4. <u>NEVER add additional conservatism!</u> The methodologies used in the manual are already conservative and will overestimate dose. Adding additional conservatism makes it difficult to relate assessment to risk and will create confusion.

5. Determine if DRLs and DILs can be validly applied to the entire radiological footprint. If so, proceed with use of DRLs/DILs. If not, compute doses on a location-by-location basis.

6. Document all interpretations, analyses, DRL/DIL revisions, and calculations by completing an "Assessment Product QA Cover Sheet" (as directed in V.1, Method M.6.1 and V.2 Worksheet 6.1) for each product. Document any deviations from the manual's methodology and describe why. Cite any directives to do so and concurrences.

7. <u>FRMAC does not make Protective Action Recommendations.</u> NEVER present an assessment product in a manner so that it can be construed as a recommendation.

8. Set QA flags on data ASAP. Release data and assessments as quickly as possible.

9. Establish frequent and close contacts with the FRMAC Senior Scientific Advisor and the Federal Advisory Team.

Organization

This *FRMAC Assessment Manual* is organized into three volumes. The first two are general and organized by phase. The third volume contains assessment of "default" cases for six accident scenarios. The scope of each volume is:

Volume 1, "Methods," provides: 1) an overview of Assessment, 2) detailed descriptions of generalized assessment methodologies, and 3) procedures for internal operations.

Volume 2, "Tables, Charts, Worksheets, Glossary, References," presents all supporting information necessary to employ the assessment methods provided in Volume 1, "Methods."

Volume 3, "Pre-Assessed Default Scenarios," assesses a "typical" case for six different types of accidents. A section is devoted to each generic scenario, which offers default DRLs/DILs and specially tailored methodologies. The covered scenarios include:

- Nuclear Power Plant Accident
- Nuclear Weapon Accident
- Aged Fission Product Accident
- Uranium Fuel Accident
- Radioisotopic Thermoelectric Generator Accident
- Nuclear Yield Accident

Organization of the Scenario-Specific Section

Scenario Title

    Description of the case assessed

    Scenario description

    Data Quality Objective process analysis

        Problem statement

        Decision identification

        Decision inputs

        Boundaries

        Decision rules

        Tolerance limits

        Optimization

        Default DRLs

    Worker protection

        Assessment's contribution

        Computation of turn-back guidance

    Early Phase

        Default DRLs

        Revision of DRLs

    Intermediate Phase

        Default DRLs

        Revision of DRLs

    Ingestion Phase

        Default DRLs

        Revision procedures

    Decay Corrections

    DRL Revision

    Ancillary Information and Methods

        Compilation of scenario-specific assessment methods

Default Scenarios

The scenarios below are based on events believed to most likely trigger a FRMAC. These events could be initiated by an accident or malicious activity. Therefore, some scenarios are clearly types of accidents, while others are related to types of materials. Other radiological accidents and incidents can occur, but are not likely to evolve into a FRMAC.

Nuclear Power Plant Accident

Description

Major accidents at nuclear power plants have the potential to release large amounts of radioactivity to the environment. The total inventory of radioactivity in a 1000 MW_e reactor that has been operating for one year is approximately 1.5×10^{10} Ci. Only a few hundred Ci were introduced as fuel. The remainder of the inventory is made up of fission products such as ^{131}I and ^{137}Cs, activation products such as ^{60}Co and ^{54}Mn, and radionuclides formed from neutron capture reactions such as ^{239}Pu. Fission products make up the majority of the radionuclide inventory. Some of the fission products are in gaseous form (krypton and xenon) and others are highly volatile (radioiodines), making containment particularly difficult. Control is normally maintained over the gaseous and highly volatile fission products through a series of containment features. These features include the fuel pellet ceramic matrix, fuel cladding, the reactor vessel, and the containment building. In addition, filtration (physical and chemical) and holdup systems limit the release of radioactivity from the reactor under normal operating conditions. Catastrophic accidents that pose the greatest risk are those which defeat the control measures designed to prevent the release of radioactivity. These accidents involve core damage and a prompt (within 24 hours) release and will most likely be from an unmonitored pathway. The principal nuclides which are expected to deliver the major portion of the radiation dose during the first year following a major nuclear power plant accident are ^{131}I, ^{134}Cs, ^{137}Cs, ^{103}Ru and ^{106}Ru.

Major Concerns

During plume passage, the primary concern is the thyroid dose due to inhalation of ^{131}I. The eight-day half-life of ^{131}I means that it will only be significant for the first two months following the release. Following plume passage, the major source of dose is from external exposure from radioactive deposits.

Nuclear Weapon Accident

Description

Accidents involving nuclear weapons may be categorized into three general categories. The first is an accident in which the nuclear weapon is being carried and the carrier (aircraft or vehicle)

sustains an accident resulting in a fire or explosion, but the weapon survives essentially intact, without having detonated the high explosives or undergone a nuclear yield. The second is an accident in which the high explosives in the nuclear weapon detonate with resulting dispersal of radioactive material into the environment. The third is an accident in which the weapon undergoes a significant nuclear yield. The third type of accident is extremely unlikely due to the design of U.S. nuclear weapons. U.S. weapon design is usually referred to as "one-point safe," which means there must be less than one chance in a million of producing a nuclear yield equivalent to more than four pounds of trinitrotoluene (TNT) when the high explosive is initiated and detonated at any single point. The primary nuclear weapon accident type of concern is the second type, in which a weapon undergoes a high-explosive detonation with resulting dispersal of radioactive material. The radiological hazard presented by a nuclear weapon comes from three sources: plutonium, uranium, and tritium. The most significant of these hazards is plutonium. The principal nuclide expected to deliver the major portion of the radiation dose following a nuclear weapon accident involving a high-explosive detonation without nuclear yield is ^{239}Pu.

Major Concerns

Plutonium is primarily an alpha emitter; the radiological hazard only results from internal contamination due to inhalation or ingestion. The primary uranium radionuclide present in a weapon is uranium-238 (^{238}U). The hazards are primarily toxicological rather than radiological. Ingestion or inhalation of uranium can result in a type of heavy metal poisoning with possible renal damage.

Aged Fission Product Accident

Description

Accidents related to aged fission products will most likely involve nuclear reactor wastes or fuel reprocessing materials. The severity of the incident primarily depends on the quantity and age of the material, plus the mechanism by which it is released. The scenario assessed is for spent reactor fuel, less than 10 percent ^{235}U enrichment, which has cooled for a minimum of 100 days. If the fuel has cooled for fewer than 100 days, shorter-lived radionuclides will be present and the consequences will be similar to a power reactor accident. The principal nuclides which are expected to deliver the major portion of the radiation dose following an accident involving a nuclear fuel reprocessing plant are strontium-90 (^{90}Sr), ^{137}Cs, ^{239}Pu, and americium-241 (^{241}Am). Accidents posing the greatest risk of release of radioactive material to the environment involve process explosions, fires, or a widespread loss of containment due to an external event (earthquake, terrorist attack, airplane crash). The release mechanism assumes that the entire mix of nuclides is released without fractionation of species or phase.

Major Concerns

The major concern will be external exposure to deposition and possible contamination of the skin. Inhalation of material, either in the plume or from resuspension, is not likely to be a major dose pathway. The greatest challenge will be determination of the radionuclide mix. The mix can vary greatly depending on the specific material. Moreover, the mix can change due to the chemical reactions that can occur in the environment.

Nuclear Fuel Accident

Description

A nuclear fuel accident/incident is strictly limited to unirradiated uranium fuel. Two types of accidents were assessed: 1) dispersal of a large uranium inventory as a solid (*e.g.*, metal, ceramic, powder), and 2) release of uranium as uranium hexafluoride (UF_6). It was determined that fires and explosions involving a solid form of uranium cannot lead to serious dose, exposure, or contamination. The most serious event is the massive release of UF_6, which is the default scenario assessed here. The chemical toxicity effects, which may dominate the radiological consequences, are also briefly considered.

Major Concerns

Doses are dominated by the inhalation pathway. Doses due to resuspension will be small. Therefore, plume phase is a key problem. Inhalation dose is near negligible for uranium that has recently been chemically separated, but significant for uranium old enough to come to equilibrium with the progeny. Enrichment has little effect on the consequences.

Radioisotopic Thermoelectric Generator Accident

Description

Nuclear power sources used in space can be impacted by several types of accident. These include a first stage accident at launch, orbital decay resulting in re-entry to the earth's atmosphere, and re-entry at higher than orbital velocities during a fly-by maneuver for deep space missions. Nuclear power sources used in spacecraft consist of radioisotopic thermoelectric generators (RTGs) and radioisotope heater units (RHUs). These power sources use ^{238}Pu dioxide in a ceramic form. RTGs are designed to contain their fuel under most accident conditions; however releases can occur as a result of impact with concrete or steel during a launch accident or as a result of an impact with rock following re-entry. An inadvertent re-entry represents the most severe accident environment to which RTGs could be subjected and would lead to a range of fuel end states that include intact or damaged modules, intact graphite impact shells, and fuel released at high altitude in both particulate and vapor form. The principal nuclide that is expected to deliver the radiation dose following an accident involving a nuclear-powered spacecraft is ^{238}Pu.

Major Concerns

The principal radiological hazard is due to inhalation, much as with a nuclear weapon accident. Dose conversion factors and total alpha DRLs are not greatly different from a nuclear weapon accident. The key distinction is that Pu-238 has high specific activity, which makes the event a "hot particle" problem. That is, the radioactivity of a single microscopic particle in a measurement may be large compared to the decision level. So, monitoring effectiveness may not be a matter of detection sensitivity but the challenge of probability of finding and capturing a particle.

Nuclear Yield Accident

Description

Two types of nuclear incident are considered: a nuclear criticality and a nuclear weapon explosion resulting in a significant nuclear yield. Criticality events are also possible for lower enrichment levels given sufficient mass. The primary product of a criticality event is radiation, which arises from three sources: (1) prompt gamma and neutron radiation resulting from the fission process, (2) gamma and beta radiation resulting from the decay of fission products produced by the criticality, and (3) radiation from surrounding materials that were activated by neutrons. In addition to radiation, criticality incidents may result in the dispersal of some of the fission products produced in the reaction. These consist primarily of noble gases (krypton and xenon) and radioactive radioisotopes of iodine (^{131}I, ^{135}I).

> NOTE: This section is still under development. Upon completion it will be posted on the FRMAC web site and included in the next revision of the *FRMAC Assessment Manual*.

A detonation of a nuclear weapon resulting in significant nuclear yield results in the production of blast pressure, thermal radiation, direct (nuclear) radiation, and radioactive fallout. The effects with which FRMAC deals are 1) direct nuclear radiation and 2) radioactive fallout. Direct nuclear radiation consists of prompt gamma and neutron radiation resulting from the fission process, and residual radiation resulting from the decay of fission and activation products. The direct nuclear radiation is primarily a local hazard (*i.e.*, within a few kilometers). Radioactive fallout, however, has the potential to present a hazard for much greater distances. Fallout consists of fission products condensed onto particles of inert material (*i.e.*, dirt and dust) that were vaporized in the detonation. The distance of the detonation from the earth's surface influences the amount of fallout produced. Detonations that occur at ground level produce a much greater amount of fallout than those that occur at higher altitudes. The radiological hazard of fallout is the beta and gamma radiation resulting from the decay of fission products produced in the detonation.

Major Concerns

A criticality incident has little potential for off-site consequences. However, it can be lethal to those in near proximity. The most serious hazard is from the large burst of prompt gamma and neutron radiation due to fission. If reactivity cannot be reduced and maintained well below criticality, then the potential for dangerous periodic criticality episodes exists. The fission product inventory is generally small and decays very quickly.

A nuclear detonation results in two regions of effects: 1) a near-field region sufficiently close to the detonation, such that the blast and thermal effects greatly exceed the radiological effects and 2) a far-field region which extends far downwind where only radiological hazards exist. In the near-field region radiation levels will greatly exceed decision and turn-back levels. Mortal injuries will be unavoidable. In the far-field region arrival of the fallout may be delayed sufficiently to shelter or evacuate. Contamination of crops and animals in excess of the U.S. Food and Drug Administration (FDA) Protective Action Guides (PAGs) is likely to occur over a vast area.

Radiological Dispersal Device Accident

Description

A Radiological Dispersal Device (RDD) is a device that is designed to spread radioactive material with the intent to cause panic and economic impact, and to contaminate and render large urban areas useable. Large industrial and medical sources are most likely to be used as RDD source materials. These sources include, but are not limited to cesium-137, cobalt-60, americium-241, and iridium-192.   The radioactive source material coupled with an explosive device, to disperse the material, comprise a typical RDD. An RDD poses a threat to public health and safety, and to the environment through the spread of radioactive materials. Also, the explosive device presents an immediate threat to human life, health, and property. Other means of dispersal, both passive (e.g., radioactive material randomly spread by hand) and active (e.g., aerosolized radioactive material spread by an aircraft) may be employed with an RDD. Emergency responders and decision makers should treat a passive/active dissemination of radioactive material similarly to an explosive RDD.

Major Concerns

An RDD may produce a wide range of possible consequences depending on the type and quantity of radioactive material, and the dispersal method. In addition, the location of the device, urban canyon effects, and the local meteorological conditions impact the event. Different radioactive materials have very different dispersal characteristics. The quantity of explosive and the location of the event (highly urban versus open rural areas) will dramatically impact the spread of radioactive materials. The effect of an RDD may range from a small, localized contamination area (e.g., a street, single building, or city block) to large, wide-spread contamination, conceivably several square miles. However, in most plausible scenarios, the

radioactive material would not result in acutely harmful radiation doses. The risk of the exposed individuals developing latent cancer is likely the greatest public health concern from the radioactive materials. Other hazards include fire, smoke, shock, shrapnel (from an explosion), industrial chemicals, and other chemical or biological agents that may also be present. Another concern from an RDD event is the economic impact caused by the potential loss of food crops, manufacturing capability, housing, and/or transportation infrastructure. Also, the cost of relocating the impacted populations, and decontaminating and recovering property may be excessive. Finally, large areas may be rendered unusable for extended time periods.

Caveats

- Default accident scenario cases are not worst possible cases, but "typical"
- Complexity may greatly slow assessment. Complications include changes in site conditions (on-going/multiple releases, changing release pathway, spatial mix variation, weather changes)
- FRMAC conversion factors and decision levels may differ from locally determined values
- FRMAC conversion factors may not be the newest or best available, but they are widely accepted
- Large differences should be expected between assessments based on models versus data
- DQO process is initially abbreviated
- International System of Units (SI) units are not used, but final products may be issued in SI, if required

SECTION 2. NUCLEAR POWER PLANT ACCIDENT

CHARTS

TABLES

NUCLEAR POWER PLANT ACCIDENT

2.1   Description of Generic Nuclear Power Plant Accident

This introduction defines the scenario and outlines the DQOs for the case.

2.1.1   Scenario Description

Major accidents at nuclear power plants have the potential to release large amounts of radioactivity to the environment. The total inventory of radioactivity in a 1000 MWe reactor that has been operating for one year is approximately 1.5×10^{10} Ci. Only a few hundred Ci were introduced as fuel. The remainder of the inventory is made up of fission products such as ^{131}I and ^{137}Cs, activation products such as ^{60}Co and ^{54}Mn, and radionuclides formed from neutron capture reactions such as ^{239}Pu. Fission products make up the majority of the radionuclide inventory. Some of the fission products are in gaseous form (krypton and xenon) and others are highly volatile (iodine), making containment particularly difficult. Control is normally maintained over the gaseous and highly volatile fission products through a series of containment features. These features include the fuel pellet ceramic matrix, fuel cladding, the reactor vessel, and the containment building. In addition, filtration (physical and chemical) and holdup systems limit the release of radioactivity from the reactor under normal operating conditions. Catastrophic accidents that defeat the control measures designed to prevent the release of radioactivity pose the greatest risk. These accidents involve core damage and a prompt (within 24 hours) release and will most likely be from an unmonitored pathway.

The generic nuclear power plant accident considered here involves severe core damage in a commercial light water reactor (LWR). Although there are many types of reactors, these are by far the most prevalent and contain the largest inventory of radioactive material. Although there are numerous potential sources of a radiological release from a nuclear power reactor, large-scale releases are possible only with overheating of large amounts of irradiated fuel in the core of the reactor or in the spent fuel pool.

A severe core damage accident is highly variable in both magnitude and radionuclide mix. The default case considered assumes fuel to be heated such that all noble gases and nearly all iodine are released from the fuel. The radionuclide mix released into the environment is assumed to be that released into the reactor vessel. This is generally a worst-case mix. Default assumptions must be replaced by measurements as soon as practical. Release pathway, delays in release from containment, and use of sprays/ice will have a strong effect on both the magnitude and mix of the inventory released to the environment.

The total inventory released to the environment by an accident can vary by orders of magnitude depending on the exact release scenario. It could be on the order of 1×10^7 Ci under some conditions, but could be negligible under others.

The following principal nuclides are expected to deliver the major portion of the radiation dose following a major nuclear power plant accident:

<div style="text-align: center">

Plume Phase

^{88}Kr, ^{133}Xe, ^{135}Xe

^{131}I, ^{132}I, ^{133}I, ^{135}I

^{131m}Te, ^{132}Te, ^{129}Sb

First Year

^{131}I, ^{132}Te

^{134}Cs, ^{136}Cs, ^{137}Cs

^{103}Ru, ^{106}Ru

^{140}Ba, ^{140}La

^{95}Nb, ^{95}Zr

^{141}Ce, ^{144}Ce

^{238}Pu, ^{241}Pu

</div>

This section uses the core inventory presented in NRC96 (Table C-7) along with release fractions developed from NRC95 (Tables 3.12 and 3.13). The release fractions used are the average of Pressurized-Water Reactor (PWR) and Boiling Water Reactor (BWR) values for the first three phases of an accident.

For the Intermediate Phase, dose from inhalation of resuspended material is included in calculations, but the contribution is very small compared to the external dose.

2.1.2   Data Quality Objective Process

The seven steps of EPA's Data Quality Objective (DQO) process are applied to the nuclear power plant accident scenario in the discussion below. Application detail is minimal in the manual but will develop as incident-specific work proceeds. A complete and formal DQO treatment is not expected until the recovery phase, specifically, at the beginning of long-term monitoring.

2.1.2.1   DQO Step 1 – State the Problem

The key element of step 1 of EPA's DQO process, State the Problem, is addressed by the scenario description above. The remaining elements, which are not scenario specific, are covered in the organizational overview of the Assessment Group, Volume 1 of the *FRMAC Assessment Manual*. These elements include team composition, customer/decision maker interface, and resources. This overview is primarily presented in the Introduction. Team members and roles are described in the Introduction, and Section 6: Administration, Internal Procedures, and Tools. The specific identity of the decision maker(s) is dynamically defined by consultations between the FRMAC Director and representatives of the Lead Federal Agency (LFA) and state(s). As part of the FRMAC team, the Assessment Group is responsible for providing the assessments of measurements and predictions necessary for decision-makers to protect the public and emergency workers from excessive exposure to radioactive materials. The *FRMAC Assessment Manual* is the technical basis for these assessments. Assessment results are interpretations of

measurements in terms of published PAGs, or as otherwise directed by the LFA and Advisory Team. Should the *FRMAC Assessment Manual* not provide the information needed to address a specific issue, then technical experts outside the FRMAC will be enlisted.

2.1.2.2   DQO Step 2 – Identify the Decisions

The key element of step 2, Identify the Decisions, is the enumeration of the major protective actions and their respective "triggers." The FRMAC does not make Protective Action Recommendations (PARs), but it does identify those areas where specific actions may be technically warranted. It may also identify potential mitigating measures. The decision maker is expected to consult with the staff to develop the alternative actions for each decision.

2.1.2.2.1 POTENTIAL CONSEQUENCES

The potential consequences of a nuclear power plant accident are the result of releases from overheating of irradiated fuel. These releases mainly consist of xenon, krypton, cesium, tellurium, iodine, and lesser amounts of other radionuclides. If the fuel has been irradiated in the core within the past two months, the release could include enormous amounts of iodine. Severe damage to the core of a power reactor could result in large enough releases to cause very high thyroid doses (> 1000 rem) from inhalation, beta burns to exposed skin, plus early health effects more than ten miles from the site due to exposure to deposition. Contamination warranting food restrictions may extend great distances (>50 miles). During plume passage, the primary concern is the thyroid dose due to inhalation of ^{131}I. The eight-day half-life of ^{131}I means that it will only be significant for the first two months following the release. Following plume passage, the major source of dose is from external exposure from deposition. A release could occur over a period ranging from minutes to days. In fact, a severe release is expected to last for several days.

Damage to fuel in the spent fuel pool could result in releases of large amounts of Cs; however, it is unlikely that integrated doses in excess of the EPA PAGs would extend more than one mile from the site. Releases not involving overheating of the fuel in the core or spent fuel pool are unlikely to result in a release off-site warranting any protective actions. This would include coolant releases, filter fires, mechanical damage to fuel, or damage to dry storage casks.

2.1.2.2.2 POTENTIAL ACTIONS

The Assessment Group provides decision makers with the technical basis for protective actions (radiological assessment). The major potential actions are those necessary to reduce risk due to exposure to acceptable levels. The actions are prioritized to address the most serious and time-sensitive potential effects first.

The major protective actions include (generally prioritized):
- Expedited evacuation where potential for early health effects exists
- Evacuation of immobile populations (hospitals, prisons…)

- Evacuation of general public
- Sheltering of public and immobile populations
- Identification/treatment of potential early health effect victims (*i.e.*, identification of unevacuated areas where the population may have the potential for early health effects so that these individuals may be identified and treated, as needed)
- Relocation of unevacuated populace to avoid future risk
- Suspension of agricultural production
- Condemnation of foods

Other decisions might include:

- Exposure planning for emergency workers
- Selection of measurements and monitoring locations
- Guidelines for re-entry
- Identification and selection of mitigation options

2.1.2.3   DQO Step 3 – Inputs to the Decisions

2.1.2.3.1 INFORMATIONAL INPUTS

Because the decisions are very time-sensitive, particularly at an early time, the radiological assessment must proceed with whatever quantity and quality of data are available at the time the decision must be made. Most initial decisions will be based on predictive plume models. Initial measurements will be used to validate or renormalize the model. As quickly as possible, sufficient measurements must be acquired to replace dependence on the model. As time progresses and decisions become less time critical, the quantity and quality of data will improve. Eventually guidelines will be implemented on the collection and analysis of measurements and models will become interpolation tools.

During the Early (and much of the Intermediate) Phase of the accident, the assessment methods and reference data provided in the *FRMAC Assessment Manual* are expected to be sufficient for the radiological assessment. Default decision levels (DRLs) presented in Table V3.2.2. are to be used until sufficient data have been collected to eliminate assumptions. Revision of a DRL is acceptable only if an assumption can be eliminated. Several revisions may occur over the course of time as assumptions are eliminated.

Exposure rate serves as an adequate surrogate to identify where urgent protective actions and relocation are warranted. This is because the radionuclide mix associated with releases from irradiated fuel strongly yield gamma radiation. Therefore, following the start of a release, DRLs for gross gamma exposure rates will be used to determine where evacuation and relocation from contaminated territory are warranted in accordance with EPA PAGs. However, use of exposure rate as a surrogate is <u>not</u> warranted, if the radionuclide mix varies such that a single value for a DRL cannot be defined. In that case, assessment must proceed on a location-by-location analysis

of the radionuclide mix. Exposure rate is not appropriate to delimit areas where restriction of locally grown food may be warranted in accordance with the U.S. Dept. of Health and Human Services (HHS) Derived Intervention Levels (DILs). This is because the exposure rate associated with the limiting HHS DIL for the LWR accident default scenario is expected to be below the detection level for most detectors. Thus, exposure rates can only be used to identify the areas where contamination of locally produced food will be clearly above the HHS DILs.

An alternative surrogate is ^{137}Cs concentration. Once the ^{137}Cs deposition density has been characterized, it could be used in place of exposure rate to identify areas warranting relocation. ^{137}Cs may be preferred to exposure rates because, unlike exposure rates, the ^{137}Cs density will remain relatively constant over time following the release.

More detailed and rigorous treatments will be introduced as the recovery phase is entered and progresses.

2.1.2.3.2 MEASUREMENT AND PREDICTION INPUTS

The table below outlines the measurements and model results that will be needed in relative order of necessity based on the time urgency of the data.

Table V3.2.1. Measurements and Predictive Inputs

Predictions	Plume Passage – TEDE
	Plume Passage – Thyroid CDE
	Post Plume Passage – Exposure rate
Field Measurements	AMS quick-look survey
	Handheld exposure rate meters
	Handheld GM meters (open/closed shield at 2 elevations)
	Field gamma spectroscopy
	Air Sampling
Sample Analyses	Soil samples
	Air samples
	Crop samples
	Water samples

2.1.2.3.3 COMPLICATING FACTORS

The environmental data must be representative in order to be a valid basis for revising the DRLs. The environmental data may not be representative because:

- The mixture of the release can change with plant conditions.

- The mixture of the release will change with time due to decay and ingrowth.

- The rates of deposition of different elements (*e.g.*, iodine, cesium) can vary by a factor of 20 or more depending on the surface (*e.g.*, grass, roads, wet, dry) and atmospheric conditions.

- The deposition density can be very complex, varying a by factor of 10 or more over short distances.

- Noble gases and their daughters will be a major fraction of any release. This could interfere with and confuse air sampling and analysis conducted to determine iodine concentrations.

2.1.2.4   DQO Step 4 - Boundary of Consideration

2.1.2.4.1 PHYSICAL BOUNDARY

The area for which assessments are needed is the entire area impacted by the plume as well as sufficient area outside of that to ensure that the extent of the area affected by the plume can be defined. Initially the extent will be the area potentially subject to evacuation or relocation. After these concerns are addressed, the limits will be extended to the surrounding agricultural production area where the FDA PAGs may have been exceeded. It may also be necessary to include food-processing facilities well outside the affected region, where contaminated foods may have been transported.

Explicit exclusions include the plant site itself and the dosimetry of individuals.

2.1.2.4.2 TEMPORAL BOUNDARY

FRMAC Assessment tools provided here are sufficient for the appraisal of doses during the Early and a significant portion of the Intermediate Phase.

Detection of conditions that threaten the core or spent fuel pool will result in declaration of a Site Area or General Emergency. Radiological assessments are likely to begin using predictions before a release occurs, because models can be executed based on plant conditions. It is expected that a FRMAC will be requested upon declaration of a General Emergency or other conditions that could result in a severe release. Since a release could occur minutes to days following request for a FRMAC, the FRMAC staff could arrive before, during, or after a major release. In fact, a severe release is expected to last for several days; thus, the FRMAC is expected to be at the scene during the release. Assessment of initial measurements may begin with commencement of the release, because Radiological Assistance Program (RAP) or Consequence Management Response Team (CMRT) (FRMAC Phase 1) responders will deploy upon notification of degrading plant conditions.

Although the life cycle of the accident response continues through the Recovery Phase, the scope of the treatment in the *FRMAC Assessment Manual* is intended to be valid only until the end of the Intermediate Phase. At that time a Recovery Plan with a plan for long-term monitoring will be created. During that period only portions of this manual may remain applicable.

2.1.2.4.3 CONSTRAINTS

Some of the potential constraints on measurements include:

- Remonitoring if new releases occur

- Deposition on snow cover

- Deposition on a leaf canopy

- Delays in monitoring due to adverse weather

- Access denied by property owners

- Inaccessible terrain

2.1.2.5   DQO Step 5 - Decision Rules

The Assessment Group does not establish decision rules nor make PARs. However, the Assessment Group uses published PAGs as decision rules for the interpretation of measurements and predictions. These PAGs are implemented as DRLs. If a measurement exceeds a specific DRL by any margin, then that location fails the test at hand. If a measurement falls short of a DRL by any margin, then it passes the test at hand.

Derived Response Levels have been defined for the following:

- Emergency Worker Turn-back limits

- Evacuation based on EPA Early Phase PAG following plume passage

- Relocation (1st year), plus 2nd and 50-year long-term objectives

- Agricultural hold based on deposition

- Food condemnation (agricultural embargo) based on food concentration

- Water condemnation based on concentration

The PAGs and computational approach may be altered by the Federal Advisory Team.

2.1.2.6   DQO Step 6 - Tolerance Limits

Assessors must establish tolerable levels of uncertainty when calculating DRLs. For example, evacuation, shelter, and agricultural product holds have a higher tolerance level than re-entry, which is higher yet than relocation, and so on. It is up to the assessor to establish these tolerable levels until a more definitive uncertainty analysis can be performed.

Sensitivities of measurements must always be adequate to detect the DRL level for the question at hand. The acceptable uncertainties are listed below.

If assessments are being used for:

- Evacuation, sheltering or agricultural hold, the tolerance limit is a factor of 10.

- Re-entry considerations, the tolerance limit is a factor of 2.

- Relocation, the tolerance limit will be negotiated, but is expected to be approximately 30%.

- Return, the tolerance limit will be negotiated and will likely be much smaller.

- Condemning foods or water, the same criteria used by USDA for evaluation of non-radiological contamination will be applied (10%).

2.1.2.7   DQO Step -7- Optimal Design

FRMAC can do little initially to optimize design, which is primarily the responsibility of the EPA under its management of FRMAC during the recovery phase.

2.2   Default Derived Response Levels (DRLs)

The default DRLs for an LWR accident are given in Table V3.2.2. These values are used for an LWR accident until the release is characterized and the DRLs can be recalculated using actual data instead of assumptions (radionuclide mix, agricultural productivity, etc.)

Table V3.2.2. Default DRLs for Releases from Irradiated Reactor Fuel

Issue	Marker	DRL	Sensitivity, Uncertainty, Spatial Density, Assumptions
Worker Protection	Exp. Rate Ext. Dose	See Table V3.2.3.	
EPA Early Phase PAG (evacuation)	Exp. Rate Predicted TEDE	10 mR/hr (no radioiodines) 2 mR/hr (with radioiodines) 1 rem	Gamma exposure rate (mR/hr) indicating that evacuation or substantial shelter could be implemented in accordance with the EPA PAGs
Relocation 1st year	Exp. Rate ^{137}Cs conc.	5 mR/hr 3 µCi/m^2	Gamma exposure rate (mR/hr) from deposition indicating that the population should be relocated in accordance with EPA PAGs (see Charts V3.2.2a and V3.2.2b)
Ingestion PAG	Exp. Rate ^{137}Cs conc.	0.5 µR/hr	See Charts V3.2.3 and V3.2.4

2.3   Worker Protection

2.3.1   Discussion of Assessment in Worker Protection

As indicated in Volume 1, the major responsibility of the Assessment Group in the area of worker protection is to provide information for determining turn-back guidance for emergency workers. The guidance is based on dose limits (TEDE) but must be presented in a manner that is useful to field personnel. This typically means that it must be presented in terms of exposure rate (as measured on a hand-held instrument) or integrated dose (as measured on a direct-reading or electronic dosimeter).

Table V3.2.3 contains default turn-back guidance for accidents involving irradiated reactor fuel. The guidance is expressed in terms of integrated exposure as registered on a self-reading or electronic dosimeter. The guidance values for the various emergency activities reflect the total (integrated) dose from external exposure at which the worker should be removed from further exposure resulting from the emergency. Careful coordination must occur among emergency workers, Field Monitoring, and Health and Safety management to ensure that workers are not allowed to exceed the applicable dose limits. The administrative limits are designed to assist in this effort.

Table V3.2.3   Default Federal Emergency Worker Dose Limits and Turn-Back Guidance for Events Involving Irradiated Reactor Fuel

Dose Limit Category or Emergency Activity	Turn-Back Guidance Expressed in Terms of Integrated Exposure Readings (*e.g.*, on a Self-Reading Dosimeter)		
	No KI[1] Taken (mR)	KI Taken Before an Exposure (mR)	No Inhalation Dose Possible (Post-Plume)[2] (mR)
Administrative Limits			
Investigation Level	75	300	1,500
Administrative level	125	500	2,500
Emergency Activity			
All	250	1,000	5,000
Protecting Major Property	500	2,000	10,000
Life Saving or Protecting Large Populations	1,250	5,000	25,000
Life Saving or Protecting Large Populations [3]	>1,250	>5,000	>25,000

[1] Potassium iodide

[2] No significant inhalation hazards as indicated by 1) no core damage, 2) no airborne plume possible, or 3) effective respiratory protection provided.

[3] Only on a voluntary basis to personnel fully aware of the risks involved

Table V3.2.3 can be used as a guideline for determining turn-back levels. For example, assuming responders will be available for 7 days, accruing dose at a rate of approximately 200 mrem per day will cause them to reach their investigation level. If no inhalation dose is possible, only exposure rates are used for this evaluation. Table V3.2.3 can be used to alter these values based upon presence of iodine and any mitigative efforts. The *Radiological Emergency Response Health and Safety Manual* (DOE01) specifies a turn-back exposure rate of 600 mR/hour. In the absence of other instructions, this reading on a survey meter would indicate that personnel should move to an area of lower exposure rate and request guidance.

2.3.2   Computation of Turn-Back Guidance

Revised turn-back guidance may be calculated using Method M.2. 1 in Volume 1 of the *FRMAC Assessment Manual* or in the computer model RASCAL 3.0, Field Monitoring to Dose Option. The nuclides listed in Section 2.1.1 of this section would be expected to be major contributors to dose for this calculation.

2.4   Early (Plume) Phase

As indicated in Volume 1, the Early Phase is considered to last for about 4 days (96 hours) for the purpose of dose assessments. The EPA evacuation PAG is 1 to 5 rem, where the dose considered is the sum of the effective dose equivalent (EDE) from external sources and the committed effective dose equivalent (CEDE) incurred from significant inhalation pathways. The committed dose equivalents (CDE) to the thyroid and skin may be 5 and 50 times higher, respectively. The early protective actions expected to be taken based on plant conditions alone (declaration of a general emergency with impending core damage) is the evacuation of the population within 2 miles all around and up to 5 miles downwind (NUREG 0654, SUP 3). The emphasis for the Early Phase is in identifying those areas that have not been evacuated where the Early Phase dose may exceed the PAG.

2.4.1   Default Derived Response Level

The use of dose rates of 10 mR/hr, 5 mR/hr or 2 mR/hr is most likely grossly over conservative when used by themselves as default triggers for evacuation for a nuclear plant accident. Such trigger values are likely to result in major areas being evacuated well beyond that necessary to protect people within the guides of the EPA PAGs. This greatly raises the risk for those moved for no benefit. After the initial actions taken within the 2 miles and 5 downwind of the accident, the recommended (NUREG 0654 SUP 3) action is to continue to assess and monitor to determine whether the initial actions were adequate or what additional actions are needed. The majorparameter missing from the linear assumptions of 10, 5, or 2 mR/hr is the expected duration of the release. It is unlikely that any release will persist unabated for 4 days (If it does at these levels, additional time is available to enlarge the area of protective action.) The duration of therelease is information that will come from the utility and NRC based on source of the release and actions being taken or planned to mitigate the release.

The DRL for the evacuation PAG is given in terms of exposure rate. Given that the Early Phase lasts for 96 (or about 100) hours, and the EPA evacuation PAG is 1 rem, the default value for the evacuation DRL for an LWR accident is 10 mR/h.

This is calculated simply by dividing the 1 rem evacuation PAG by 100 hours and assuming that 1 R is approximately equal to 1 rem. Since there will be a significant number of short-lived radionuclides for this scenario, this value is conservative for considering external exposure. This approach does not consider inhalation of material in the plume as it passes. If radioiodines are present, decrease the DRL by a factor of 5 (to 2 mR/h).

2.4.2   Revision of Evacuation DRL

Instead of revising the DRL, the preferred method of identifying areas where the Early Phase PAG may be exceeded is by performing a dose projection using a computer model (such as RASCAL) or by executing Method M.3.1 in Volume 1.

2.5   Intermediate Phase – Relocation

As indicated in Volume 1, the Intermediate Phase is the period of time which begins after any releases have been brought under control and reliable measurements are available to use for determining additional protective action recommendations. It continues until the additional protective actions are terminated. The major protective actions taken during the Intermediate Phase involve relocation and restrictions on the use of contaminated food and water.

The EPA guidance (EPA92) states that relocation is warranted if the dose from gamma exposure from deposition and inhalation of resuspension is projected to be greater than 2000 mrem the first year (or beta skin dose 50 times higher). Dose reduction due to part-time occupancy and decontamination is not to be considered. Dose reduction due to decay and weathering is to be included and has been considered in the dose factors used in the manual. The gamma exposure rate 1 meter (m) above ground level (AGL) can be used as the DRL for locating areas where relocation is warranted to meet the EPA guidance following release from irradiated reactor fuel. The areal density of ^{137}Cs can also be used to develop a relocation DRL. These DRLs will change temporally (due to decay and ingrowth) and spatially and must be re-evaluated periodically.

EPA guidance also established objectives to ensure the dose in the second year does not exceed 500 mrem and the cumulative dose over 50 years does not exceed 5000 mrem. For contamination resulting from an LWR accident, meeting the 2000 mrem PAG for the first year is expected to result in the longer-term objectives being met because of the decay of short-lived radionuclides. The process for determining if these objectives are being met will be developed as part of the long-term assessment plan and is beyond the scope of this document.

2.5.1   Default Relocation Derived Response Levels

To facilitate determination of areas where the PAG may be exceeded, DRLs can be defined that correspond to the 2-rem first year dose. As mentioned above, for an LWR accident the DRL for relocation can be expressed in terms of gamma exposure rate or areal concentration of deposited radionuclides on the ground. The default gamma exposure rate relocation DRL is 5 mR/h at about 5 hours after shutdown (or as indicated on Charts V3.2.2a and b for the appropriate time), and the default deposition concentration DRL for relocation is 3 μCi/m^2 of ^{137}Cs. Charts V3.2.2a and b present the exposure rate DRL for relocation for the default LWR accident scenario. The plotted DRL values may need to be adjusted if the state in which the FRMAC is operating has a different relocation PAG than the EPA. In that case, follow Method M.V3.2.2 to make the adjustment.

2.5.2   Revision of Relocation Derived Response Levels

The DRLs are a function of the mixture of radionuclides in the deposition. Samples should be taken and analyzed to assure that the values used in the calculations are representative of the entire affected area.

2.5.2.1   Criteria

The DRLs should be re-evaluated:

- Initially, when the actual radionuclide mix of the release is known.

- Daily for the first week to account for major changes in the composition of the deposition due to decay.

- Weekly for the first month to account for further changes in the composition of the deposition.

- Monthly thereafter, until decay no longer has a major impact.

A single value for each type of DRL may be impossible to realize. A DRL is defined only for a single radionuclide mix, and the characteristics of the release and changing meteorological conditions may cause the radionuclide mix to vary from point-to-point.

2.5.2.2   Procedure

The deposition DRLs for an LWR accident are calculated using Method M.4.1, substituting Worksheet V3.2.1 for Worksheet 4.1.

2.6   Intermediate Phase – Ingestion

As indicated in Volume 1, the Intermediate Phase is the period of time that begins after any releases have been brought under control and reliable measurements are available to use for determining additional protective action recommendations. It continues until the additional protective actions are terminated. The major protective actions taken during the Intermediate Phase involve relocation and restrictions on the use of contaminated food and water. The FDA issued recommendations regarding contaminated food in 1998 (FDA98). Key points in these recommendations were the DILs, concentrations in food at which some action should be taken to limit or preclude the use of the food product.

The purpose of the default DRLs, and even the revised DRLs, is to guide where agricultural holds may be warranted. Embargoes or other intervention actions are based solely on the analysis of food (including water and milk) samples.

There are several pathways through which food may become contaminated. This section considers the simplest case—direct deposition onto produce (*e.g.*, leafy vegetables), as well as the grass-cow-milk-man pathway.

2.6.1   Default Ingestion Derived Response Levels

As with other PAGs, DRLs are used that indicate DILs may be approached or exceeded. The DRLs may be expressed in terms of many measurable quantities. One of these is areal concentration of the deposited material (*i.e.*, the radionuclide mix). The default areal deposition concentration DRLs for ingestion are determined using the method outlined in the *Savannah River Site Ingestion Pathway Methodology Manual for Airborne Radioactive Releases* (Th00). The DRLs were calculated using the default source term decayed to 24 hours after shutdown, along with yield and retention values from Reg. Guide 1.109 (NRC77). The default DRLs are presented in Table V3.2.8.

2.6.2   Revision of Ingestion Derived Response Levels

The DRLs are a function of the mixture of radionuclides in the deposition. Samples should be taken and analyzed to assure that the values used in the calculations are representative of the entire affected area. Factors other than the radionuclide mix may also affect the DRLs. For example, the ingestion deposition DRL is a function of the agricultural productivity. The agricultural productivity may vary as a factor of the crop grown, the soil type, irrigation conditions, or other factors. Actual samples of the food item (as prepared for consumption) must be analyzed to determine whether (or where) the DILs are being exceeded.

2.6.2.1   Criteria

The DRLs should be re-evaluated:

- When the actual radionuclide mix of the release is known.

- When site-specific transfer factors have been determined (*e.g.*, agricultural production, fraction of cow's intake from pasture, etc.).

- Daily for the first week to account for major changes in the composition of the deposition due to decay.

- Weekly for the first month to account for further changes in the composition of the deposition due to decay.

- Monthly thereafter, until decay no longer has a major impact.

2.6.2.2   Procedure

The areal concentration ingestion DRLs for an LWR accident are calculated using Method M.5.8.

2.7   Decay Corrections

2.7.1   Discussion of Complications Due to Decay

Due to the changes in activity caused by radioactive decay, data collected at one point in time may be difficult to correlate with data collected at another point in time. Measured or predicted data can be corrected to create a set of data related to a common point in time, allowing analysts to better identify trends and associations in various data sets.

2.7.2   Revision of DRLs for Decay

Revising DRLs for decay may be as simple as re-running a computer code or revising a spreadsheet so that the DRL for another point in time is calculated. The Charts included in this section show values of DRLs at various points in time. Revising the DRL may simply consist of reading a new value from the appropriate chart (or a similar chart constructed for the actual mix involved in a release).

2.7.3   Decay Correction of Data

Exposure rate data are anticipated to be the data most likely to require decay correction. These data points may be corrected to a common point in time by applying a decay correction factor determined from a decay curve. This curve may be created in one of two ways. The first is to make many exposure-rate readings (over time) at one location, plot the resultant curve, and determine the decay factor. The second method is to determine the nuclide mix (from sample analysis or in-situ spectroscopy and knowledge of the source), compute the decay curve, and determine the decay factor. The decay factor can then be used within the FRMAC database to

correct the exposure rate readings to a common time. The method for decay correction of exposure rate data is presented in Method M.3.11 (in Volume 1).

Results of analyses may be corrected for decay, but this will probably not be needed very frequently.

2.8   DRL Revision

Revisions to DRLs may be made when assumptions can be eliminated or when necessitated by decay or changing conditions. The frequency required to compensate for decay depends on the makeup of the release and the particular DRL under consideration. The methods contained in this section are for those DRLs specific to LWR accident scenarios; those in Volume 1 are for DRLs common to other accident scenarios. For some LWR DRLs, the methods in Volume 1 are used, but an LWR-specific form is provided in this section for ease of use.

2.9   Ancillary Information and Methods

Method M.V3.2.0   Environmental Data Assessment for Light-Water Reactor Accidents

Purpose: A flowchart of the basic (minimal) tasks and methods used for LWR accidents is provided. A summary of the pre-calculated exposure-rate DRLs for deposition from a core-melt accident is provided in TableV3.2.2.

Discussion: The LWR accidents posing the greatest risks involve core damage and a prompt (within 24 hours) release and will most likely be from an unmonitored pathway. The FRMAC or RAP personnel may arrive at the site before, during, or after a major release. The power plant operators will recommend protective actions near the plant (two-three miles) upon detection of core damage. However, these actions will not ensure that the public is protected. The major sources of dose are external exposure and inhalation from the plume. Thyroid inhalation is particularly important. Following plume passage, the principal source of dose is from external exposure from deposition. Resuspension is not a major source of dose even for the Intermediate Phase. Therefore, methods must be in place to promptly assess the external and inhalation dose in a plume and external exposure from deposition. The inhalation dose can be estimated based on exposure rates if the ratio of exposure rate to inhalation dose can be estimated. The Early Phase and Intermediate Phase dose from deposition can be estimated based on an estimate of the release mixture. For ingestion, exposure rates can be used to identify where consumption of directly contaminated food may exceed the PAGs. For the grass–cow–milk pathway, the exposure rates will be below background, and a marker radionuclide must be used to identify the area of concern.

For reactor core-damage accidents, the mixture of a major release has been estimated. Therefore, the exposure rate to inhalation dose ratios (used to assess dose in plumes) and exposure-rate DRLs (used to assess where PAGs may be exceeded) have been precalculated. These precalculated values will provide a basis for immediate action. However, the mixture of the release may be different than assumed or may also change with time as the reactor core undergoes different damage states. Consequently, assessments based on precalculated assumptions must be confirmed using sample analysis.

The following outline provides the basic environmental assessment strategy for a severe (core-damage) LWR accident.

1. Predeployment

    Before arrival in the area near the reactor (10 miles):

    - Ensure that the monitoring teams are protected by establishing turn-back guidance based on integrated exposure.

    - Ensure that an ongoing release can be promptly assessed by precalculating the ratio of exposure rate to total dose.

- Ensure that deposition can be promptly assessed based on precalculated DRLs that relate exposure rates to early health effects and PAGs.

The goal is to be prepared to promptly assess an ongoing release and deposition in order to protect the monitoring team and the public.

2. In plume (during a release)

Estimate the inhalation dose to the public and emergency workers in the plume using inhalation dose to exposure-rate ratios and DRLs. Take samples and monitor to establish the actual mixture of the release and adjust the basis (*e.g.*, DRLs) for the assessments.

3. After plume passage

- Locate and evacuate areas with high-deposition dose rates (hot spots greater than 500 mR/hr – early health effects, and 10 mR/hr – evacuation PAGs).

- Locate areas where deposition dose rates will result in doses that exceed the Intermediate Phase PAGs.

- Identify the areas where ingestion may be a concern based on deposition exposure rates and radionuclide concentrations. Confirm where ingestion is a concern based on analysis of food, water, and milk.

Steps: Perform the steps in the flowchart on the following page in the order shown. The precalculated DRLs for a core-damage accident are listed in Table V3.2.2.

Predeployment

Estimate and provide to Health and Safety the ratio of integrated exposure to Emergency Worker Limits.

Method M.V3.2.1

Develop Exposure-to-Dose Conversion Factors Used when in the plume to assess Early Phase PAGs and Early Health Effects.

Method M.3.1

Review exposure-rate DRLs for use after plume passage to assess Early Phase PAGs and Early Health Effects.

Table V3.2.2 or Method M.3.3

Review exposure-rate DRLs for LWR accidents used after plume passage to assess relocation.

Table V3.2.2 or Method M.V3.2.2

Review exposure-rate DRLs for LWR accidents used after plume passage to assess ingestion.

Table V3.2.2 or Method M.V3.2.3

In Plume

Use Exposure-to-Dose Conversion Factors to assess exposure rates to determine if Early Phase PAGs and Early Health Effects are possible based on exposure rates in the plume.

Method M.3.2

Take measurements or samples - establish mixture of the principal radionuclides for acute bone dose and thyroid dose from inhalation.

See Table V3.2.4

Confirm or adjust the ratio of integrated exposure to Emergency Worker Limits and provide to Health and Safety.

Method M.2.1

Confirm or adjust exposure-rate-to-dose relationships used to assess Early Phase PAGs and Early Health Effects.

Method M.3.2

After Plume Passage

Identify where Early Phase PAGs and Early Health Effects Thresholds are exceeded based on deposition exposure-rate DRLs.

Method M.3.3

Identify where Relocation PAGs are exceeded based on deposition exposure-rate DRLs.

Method M.V3.2.2 or M.4.1

Identify where Ingestion PAGs may be exceeded based on deposition exposure-rate DRLs.

Method M.V3.2.3

Take measurements or samples - establish mixture of the principal radionuclides in deposition.

See Table V3.2.4

Confirm or adjust time-dependent exposure-rate DRLs used to assess where Early Phase PAGs and Early Health Effects thresholds may be exceeded.

Method M.3.3

Confirm or adjust time-dependent exposure-rate DRLs used to assess where Relocation PAGs may be exceeded.

Method M.4.1

Calculate time-dependent DRLs for a marker radionuclide in deposition used to identify where ingestion PAGs may be exceeded.

Method M.5.1

Identify where ingestion PAGs may be exceeded based on deposition maker-radionuclide DRLs.

Method M.5.1

Take and analyze food and water samples to confirm area where ingestion PAGs may be exceeded.

Method M.5.3

Figure V3.2.1 Light-Water Reactor Accident Environmental Data Assessment Flowchart

Method M.V3.2.1   Integrated Exposure Emergency Worker Turn-Back Guidance for Light-
                   Water Reactor Accidents

Purpose: This method provides emergency worker turn-back guidance for core-damage accidents at a LWR. The results of this analysis will be used by Health and Safety in developing guidance for emergency workers. Turn-back guidance is the integrated exposure on a self-reading dosimeter indicating that EPA emergency worker turn-back guidance may be exceeded. This method considers the dose received from external exposure and inhalation and is based on an assumed mix. Once the isotopic mixture of the release is known or dose projections are available, use Method M.2.1 or RASCAL (NRC01) to confirm or adjust the limits.

Discussion: This method is based on the relationship between external exposure, inhalation dose CEDE, and TEDE for a range of LWR core-damage accidents that involve the release of non-noble gases (*e.g.*, iodine and cesium).

This method uses precalculated turn-back guidance from Table V3.2.3. These levels were calculated assuming an exposure-to-total-dose (exposure plus inhalation dose) ratio of 1:5. This ratio is based on the recommendation in EPA94a.

Steps: Use Table V3.2.3, Default Federal Emergency Worker Dose Limits and Turn-back Guidance for Events Involving Irradiated Reactor Fuel. This guidance should be confirmed using Method M.2.1 or RASCAL (NRC01) when dose projections or sample analysis become available.

Method M.V3.2.2   Evaluating Deposition Exposure Rates for Light-Water Reactor Accidents

Purpose: This method is used to estimate where the Relocation PAG may be exceeded from external exposure before the isotopic mix in the deposition is known. This is valid for LWR and other accidents where inhalation of resuspension is not a major source of dose. This method should be confirmed using Method M.4.1 once the radionuclide deposition levels are known or if inhalation may be important

Discussion: The method is based on exposure rates from deposition.

LWR Accident: Charts V3.2.2a and V3.2.2b provide exposure rate derived response levels (DRL$_{Rg}$) for deposition indicating where the EPA Relocation PAG (2,000 mrem in the first year) may be exceeded for an LWR accident. The curves are within a factor of two for most LWR accidents and are conservative for LWRs that have operated for less than a year.

Steps: The steps performed in this method are summarized in the box below.

Step 1   Adjust to match state PAGs if necessary.
Step 2   Determine where Relocation PAGs may be exceeded based on exposure rates.
Step 3   Confirm the basis of relocation assessment.

Step 1. Adjust to match state PAGs if necessary.

If the state does not use the EPA Relocation PAG (2,000 mrem in the first year, See Table 4.1), multiply the Y axis scale of Charts V3.2.2a and V3.2.2b as follows:

$$\frac{State \quad Relocation \quad PAG}{EPA \quad Relocation \quad PAG}$$

$$\frac{\left(\quad\quad mrem \right)}{\left(2000 \; mrem \right)}$$

Step 2. Determine where Relocation PAGs may be exceeded based on exposure rates.

Conduct monitoring to identify where the exposure rate at 1 m AGL exceeds background by the values calculated in Step 1. This indicates that the first year dose may exceed the Intermediate Phase Relocation PAG.

Step 3. Confirm the basis of relocation assessment.

Once the radionuclide mixture of the deposition is known, confirm that this provides an adequate basis by using Method M.4.2

Method M.V3.2.3   Evaluating Ingestion Derived Response Levels from Light-Water Reactor
                   Accidents

Purpose: This method is used to roughly estimate where the Ingestion PAGs may be exceeded
for LWR accidents based on exposure rate or deposition DRLs. This method is for directly
contaminated produce, or for milk produced by cows grazing on areas of deposition.

Discussion: Exposure rate and deposition concentration DRLs can be calculated for the most
limiting DIL encountered in an LWR release. For exposure rates, this results in values lower
than those that are normally detectable with portable instruments (see Charts V3.2.3 and V3.2.4).
A rule-of-thumb for an LWR accident is that an exposure rate (due to deposited material) of 0.5
µR/hr at 12 hours after shutdown indicates that FDA DILs will probably be exceeded. Similarly,
the deposition concentration DRLs are low also. A rule-of-thumb for the deposition
concentration DRL for an LWR accident is 50 nCi/m^2 (β), or about 1,000 dpm β per 100 cm^2,
again, at about 12 hours after shutdown.

In summary, if measurements indicate that contamination is present in an area, there is the
likelihood that the FDA DILs may be exceeded for produce grown or milk produced in that area.
The exception to this may be in-situ gamma spectroscopy. In-situ gamma spectroscopy may
have the capability to detect contamination levels lower than the FDA DILs.

Steps: The steps performed in this method are summarized in the box below.

Step 1   Determine where PAGs may be exceeded based on exposure rates or deposition levels.
Step 2   Confirm the basis of ingestion assessment once the deposition mix is known.

Step 1. Determine where PAGs may be exceeded based on exposure rates.

Direct environmental monitoring to identify where above-background exposure rates at 1 m AGL
from deposition are detectable and where the deposition levels exceed the rules-of-thumb
described above.

Step 2. Confirm the basis of ingestion assessment once the deposition mix is known.

Once the radionuclide mixture of the deposition is known, confirm areas where PAGs may be
exceeded using Method M.5.2 or Method M.5.3.

Table V3.2.4. Principal Radionuclides

Purpose: This table lists, in order of importance, the radionuclides that are projected to be the principal contributors (greater than 90 percent) to dose resulting from a severe core-damage accident and a major release.

Principal Dose-Contributing Radionuclides for an LWR Core Damage Accident				
External Exposure from Plume[a]	External Exposure from Deposition[a] (1st day)	Bone Marrow Plume Inhalation[a]	Ingestion at Day 7[b]	First Year from Deposition- External and Resuspension[c]
^{132}I	^{132}Te	^{132}Te	^{131}I	^{134}Cs
^{135}I	^{133}I	^{89}Sr	^{140}Ba	^{137}Cs
^{133}I	^{135}I	^{140}Ba	^{132}Te	^{140}Ba
^{88}Kr	^{132}I	^{134}Cs	^{140}La	^{131}I
^{132}Te	^{131}I	^{131}I	^{89}Sr	^{132}Te
^{131}I	^{131m}Te	^{133}I	^{144}Ce	^{95}Zr
^{129}Sb	^{140}Ba	^{137}Cs	^{90}Sr	^{140}La
^{135}Xe	^{140}La	^{135}I	^{133}I	^{136}Cs
^{131m}Te	^{129}Sb	^{136}Cs	^{129m}Te	^{103}Ru
^{134}I	^{239}Np	^{132}I	^{239}Np	^{95}Nb
^{133}Xe		^{103}Ru	^{132}I	^{106}Ru
		^{90}Sr	^{134}Cs	^{144}Ce
		^{131m}Te	^{106}Ru	^{241}Pu
		^{91}Y	^{141}Ce	^{238}Pu
		^{129m}Te	^{91}Y	^{141}Ce

[a] Source: WASH1400.

[b] Principal radionuclides contributing to ingestion dose from those remaining on day 7 after shutdown.

[c] Principal radionuclides contributing to the first-year dose from external exposure and resuspension.

Table V3.2.5. LWR Accident Exposure-to-Dose Conversion Factor for the Public

Purpose: These factors are used to estimate the thyroid dose or total dose to include inhalation based on exposure rates in a release from an LWR core-damage accident. These default values should be used with great caution and only until estimates can be made using Method M.2.1 based on plant conditions or environmental measurements.

	Core Damage – Filtered[a] Release (99 percent efficiency)		**Core Damage - Unfiltered**	
	KI[b] Taken	**KI Not Taken**	**KI Taken**	**KI Not Taken**
	mrem/mR			
$ECF_{ext, CEDE}$	1	5	5	20
$ECF_{ext, thy}$	1	50	5	100

[a] Through filter.

[b] Potassium iodide.

Table V3.2.6. Isotopic Ratios for LWR Core-Damage Accidents

Purpose: This table (based on information from NRC96) includes the radionuclides most important during the Intermediate Phase (relocation and ingestion) for LWR core-damage accidents. Decay and ingrowth are considered. The table provides the concentration of each radionuclide relative to ^{137}Cs.

Radionuclide to ^{137}Cs Ratio Time After Shutdown							
Radionuclide	1 hr	6 hr	12 hr	24 hr	3 days	7 days	15 days	30 days
^{86}Rb	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00
^{89}Sr	3.8	3.8	3.8	3.8	3.7	3.5	3.1	2.6
^{90}Sr	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.2
^{91}Sr	4.3	3.0	1.9	0.8	0.0	0.0	0.0	0.0
^{90}Y	0.0	0.0	0.0	0.0	0.1	0.1	0.1	0.2
^{91}Y	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.1
^{95}Zr	0.3	0.3	0.3	0.3	0.3	0.2	0.2	0.2
^{97}Zr	0.3	0.2	0.2	0.1	0.0	0.0	0.0	0.0
^{95}Nb	0.3	0.3	0.3	0.3	0.3	0.3	0.3	0.2
^{99}Mo	0.3	0.3	0.2	0.2	0.1	0.0	0.0	0.0
^{99m}Tc	0.2	0.3	0.2	0.2	0.1	0.1	0.0	0.0
^{103}Ru	0.2	0.2	0.2	0.2	0.2	0.2	0.1	0.1
^{105}Ru	0.1	0.1	0.0	0.0	0.0	0.0	0.0	0.0
^{106}Ru	0.04	0.04	0.04	0.04	0.04	0.04	0.04	0.04
^{105}Rh	0.1	0.1	0.1	0.1	0.0	0.0	0.0	0.0
^{127}Te	0.6	0.4	0.3	0.2	0.1	0.1	0.1	0.1
^{127m}Te	0.1	0.1	0.1	0.1	0.1	0.1	0.1	0.1
^{129}Te	2.5	0.6	0.5	0.5	0.5	0.5	0.4	0.3
^{129m}Te	0.5	0.5	0.5	0.5	0.5	0.5	0.4	0.3
^{131m}Te	1.3	1.2	1.0	0.8	0.3	0.0	0.0	0.0
^{132}Te	12.2	11.7	11.1	10.0	6.5	2.8	0.5	0.0
^{127}Sb	0.6	0.6	0.6	0.5	0.4	0.2	0.0	0.0
^{129}Sb	3.1	1.4	0.6	0.1	0.0	0.0	0.0	0.0
^{131}I	18.1	17.7	17.4	16.6	14.0	9.9	5.0	1.4
^{132}I	23.7	14.5	11.8	10.3	6.7	2.9	0.5	0.0
^{133}I	35.6	30.1	24.7	16.5	3.3	0.1	0.0	0.0
^{134}I	27.2	0.5	0.0	0.0	0.0	0.0	0.0	0.0
^{135}I	30.3	17.9	9.6	2.7	0.0	0.0	0.0	0.0
^{134}Cs	1.6	1.6	1.6	1.6	1.6	1.6	1.6	1.6
^{136}Cs	0.6	0.6	0.6	0.6	0.5	0.4	0.3	0.1

Radionuclide to ^{137}Cs Ratio Time After Shutdown								
Radionuclide	1 hr	6 hr	12 hr	24 hr	3 days	7 days	15 days	30 days
^{137}Cs	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
^{140}Ba	6.5	6.5	6.4	6.2	5.6	4.5	2.9	1.3
^{140}La	0.3	0.8	1.4	2.3	4.3	4.8	3.3	1.5
^{141}Ce	0.3	0.3	0.3	0.3	0.3	0.2	0.2	0.1
^{143}Ce	0.2	0.2	0.2	0.1	0.1	0.0	0.0	0.0
^{144}Ce	0.2	0.2	0.2	0.2	0.2	0.2	0.2	0.1
^{143}Pr	0.2	0.2	0.2	0.2	0.2	0.2	0.1	0.0
^{147}Nd	0.1	0.1	0.1	0.1	0.1	0.1	0.0	0.0
^{239}Np	3.1	2.9	2.7	2.3	1.3	0.4	0.0	0.0

Table V3.2.7   Reduction in ^{137}Cs External Gamma Doses from Decontamination for a Reactor Accident [a]

Technique	Time Applied	Percent Dose Reduction [b]	
		1st Year	50th Year
Low Impact			
Washing/vacuuming indoor surfaces	7 day	negligible	negligible
Ammonium nitrate on buildings	30 day - 1 year	1%	1%
Firehosing of buildings	1 - 7 day	2%	1%
Medium Impact			
Sweeping roads	7 day	10%	5%
Sandblasting buildings	30 day - 1 year	5%	5%
Roof replacement	30 day - 1 year	5%	5%
Grass cutting	7 day	10%	10%
Road planning	1 year		10%
High Impact			
Vacuum sweeping roads	7 day	25%	15%
Firehosing roads	1 - 7 day	25%	15%
Soil removal to 10 cm	30 day - 1 year	30%	55%
Ploughing soil to 30 cm	30 day - 1 year	35%	55%
Road planning	30 day	45%	25%

[a] Source: IAEA93 (based on Chernobyl experience)

[b] The dose reduction indicates the amount by which the total dose from urban surfaces is reduced by the technique indicated.

Table V3.2.8. Areal Concentration DRLs for Direct Deposition on Leafy Vegetables or Produce

Radionuclide Group	FDA DIL (Bq/kg)	FDA DIL (pCi/kg)	DRL (μCi/m^2)[a]	DRL (Bq/m^2)[b]
Predominant Nuclides				
^{90}Sr	160	4300	23	8.5×10^5
^{131}I	170	4600	0.04	1.6×10^3
^{134}Cs+^{137}Cs	1200	32000	10	3.7×10^5
^{238}Pu+^{239}Pu+^{241}Am	2	54	290	1.1×10^7
^{103}Ru+^{106}Ru	(Ru-103/6800)+ (Ru-106/450)<1	(Ru-103/180,000)+ (Ru-106/12,000)<1	176	6.6×10^6
Other Nuclides				
^{89}Sr	1400	38000	8	3.0×10^5
^{91}Y	1200	32000	124	4.6×10^6
^{95}Zr	4000	110000	330	1.2×10^7
^{95}Nb	12000	320000	979	3.6×10^7
^{132}Te	4400	120000	10	3.5×10^5
^{129}I	56	1500		
^{133}I	7000	190000	2	6.8×10^4
^{140}Ba	6900	190000	24	8.9×10^5
^{141}Ce	7200	190000	567	2.1×10^7
^{144}Ce	500	14000	68	2.5×10^6
^{237}Np	4	110		
^{239}Np	28000	760000	263	9.7×10^6
^{241}Pu	120	3200	408	1.5×10^7
^{242}Cm	19	510		
^{244}Cm	2	54		

[a] DRL (areal concentration of mix) = DIL/concentration in produce per 1 μCi/m^2 deposition of mix
[b] DRL (areal concentration of mix) = DIL/concentration in produce per 1 Bq/m^2 deposition of mix

WORKSHEET V3.2.1
CALCULATE LWR RELOCATION DRLs

Prepared by: _____   Date: _____
(full name)

Provide copies to: _____   Time: _____

Measured data from: _____   Count time: _____

Nuclide	Column A $C_{g,l}$ [μCi/m²]	Column B ECF_{gi} (Table 3.4) mR/h/μCi/m²	Column C DCF_{Rgi} (1st y) (Table 4.2) mrem/μCi/m²	Column D A×B mR/h	Column E A×C mrem
^{95}Nb		1.0×10^{-2}	8.5		
^{103}Ru		6.2×10^{-3}	5.9		
^{106}Ru		2.8×10^{-3}	1.3×10^{1}		
^{95}Zr		9.6×10^{-3}	1.5×10^{1}		
^{132}Te		3.0×10^{-3}	2.4×10^{-1}		
^{131}I		5.0×10^{-3}	9.8×10^{-1}		
^{132}I		2.9×10^{-2}	6.8×10^{-2}		
^{133}I		8.0×10^{-3}	1.7×10^{-1}		
^{135}I		2.5×10^{-2}	1.7×10^{-1}		
^{134}Cs		2.0×10^{-2}	1.1×10^{2}		
^{137}Cs		7.4×10^{-3}	4.5×10^{1}		
^{140}Ba		2.4×10^{-3}	7.4×10^{-1}		
^{140}La		2.9×10^{-2}	1.2		
			3	X	Y

Formula 1 (Exposure Rate Relocation DRL

$$DRL(\_\_\_\_\_mR/h) = 2000\,mrem \times \frac{X(_____mR/h)}{Y(\_\_\_\_\_mrem)}$$

Formula 2 (Cs-137 Concentration Relocation DRL)

$$DRL(_____\mu Ci/m^2) = 2000\,mrem \times \frac{C_{g,Cs\text{-}137}(_____\mu Ci/m^2)}{Y(\_\_\_\_\_mrem)}$$

Analyst Signature: _____   **Checked by:** _____

Charts V3.2.1a and V3.2.1b. Ground Exposure Rate for LWR Core-Damage Accidents

These charts show the exposure rate from deposition as a function of time after shutdown for an LWR core-damage accident. Decay and ingrowth were considered.
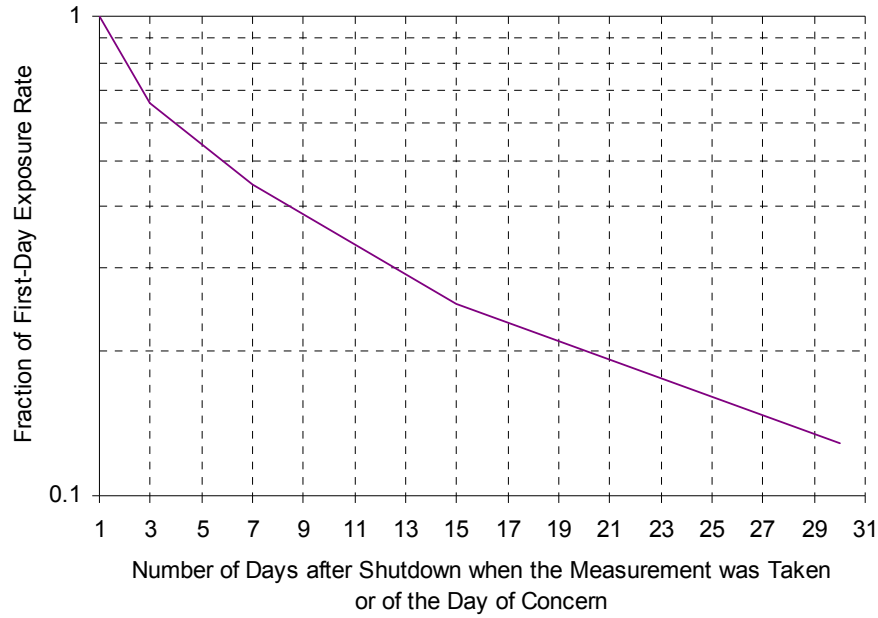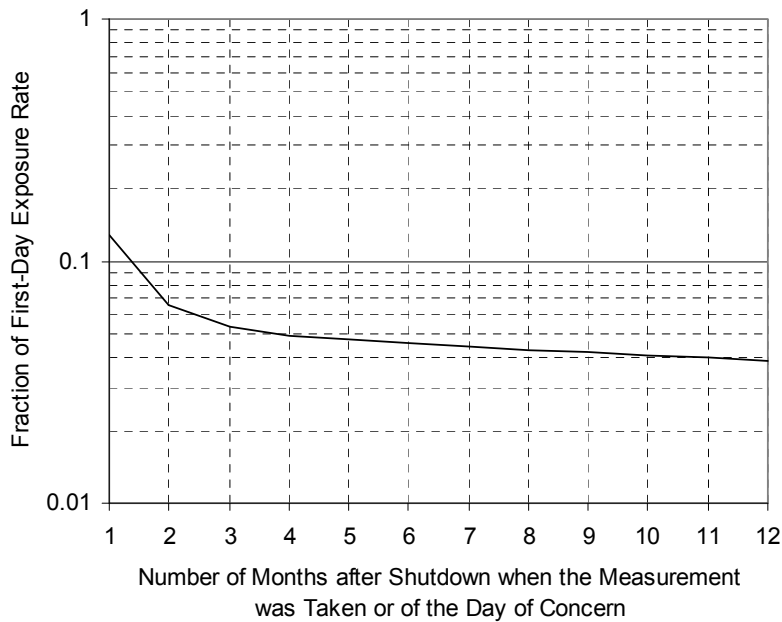
Chart V3.2.1a

Chart V3.2.1b

Charts V3.2.2a and V3.2.2b   LWR Accident Relocation Exposure Rate DRL for Hours and Days after Shutdown

These charts show the deposition resulting from an LWR core-damage accident. They show the above-background exposure rate (DRL_{Rg}) at 1 m AGL that corresponds to 2,000 mrem during the first year (EPA Relocation PAG).

Chart V3.2.2a. LWR Accident Relocation Exposure Rate DRL for Hours after Shutdown

Exposure Rate Relocation DRL
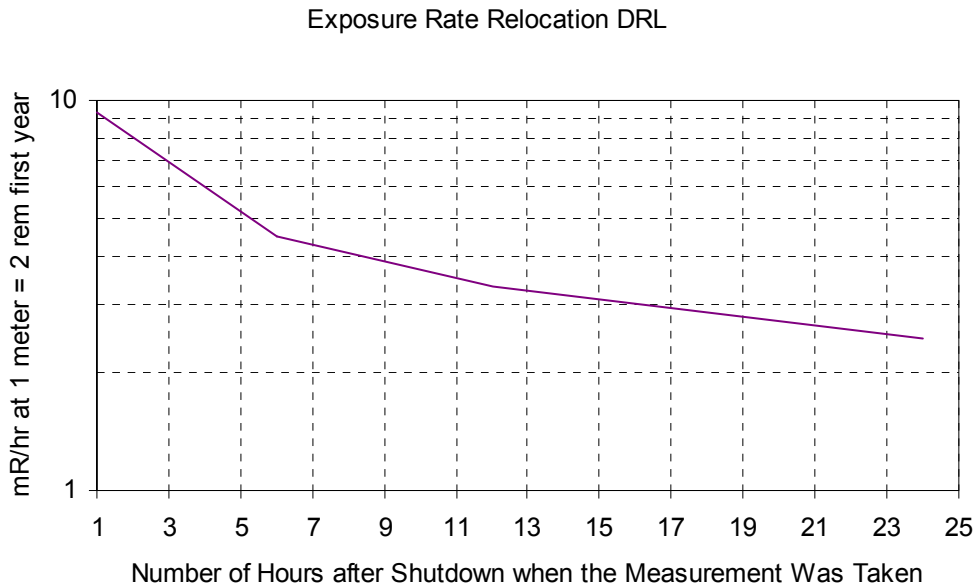
Chart V3.2.2b. LWR Accident Relocation Exposure Rate DRL for Days after Shutdown

Exposure Rate Relocation DRL

Number of Days after Shutdown when the Measurement Was Taken
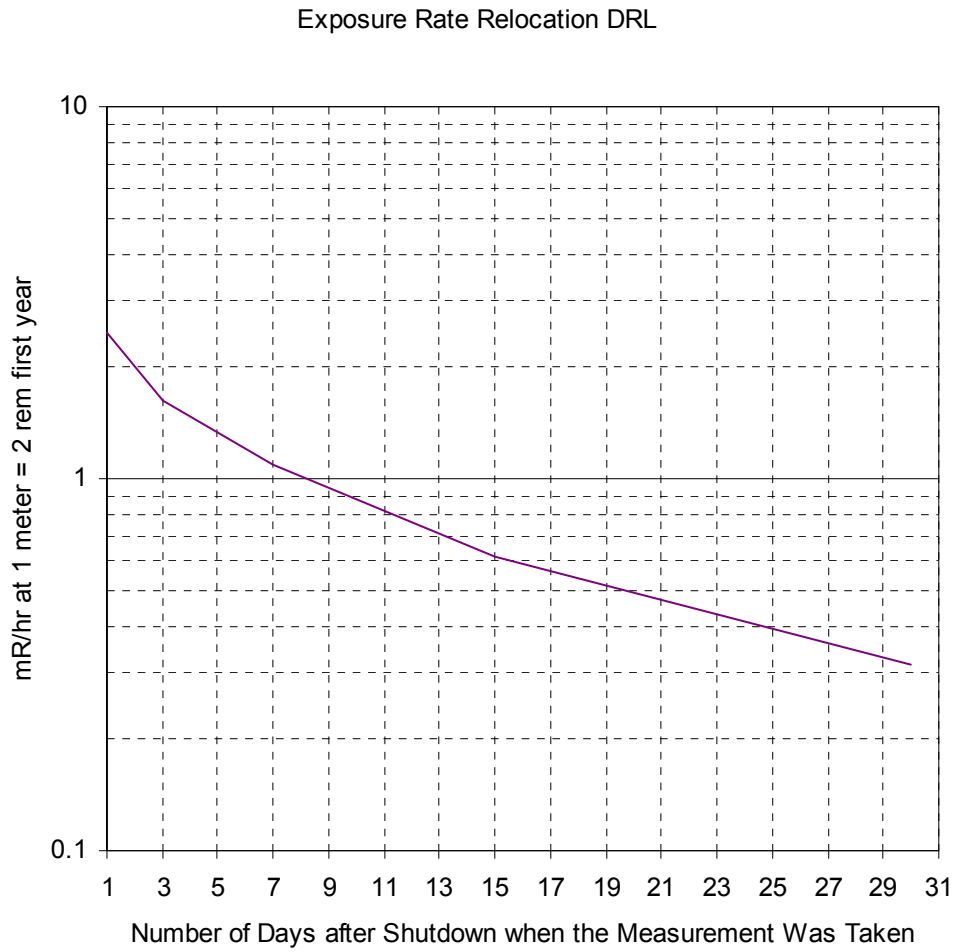
Chart V3.2.3. LWR Accident Ingestion Exposure Rate DRL for Fresh Produce

Above-background exposure rates from deposition measured at 1 m AGL that exceed the curve indicate that fresh produce grown in this area may contain concentrations above the FDA DILs. These plots are for the limiting nuclide, ^{131}I.
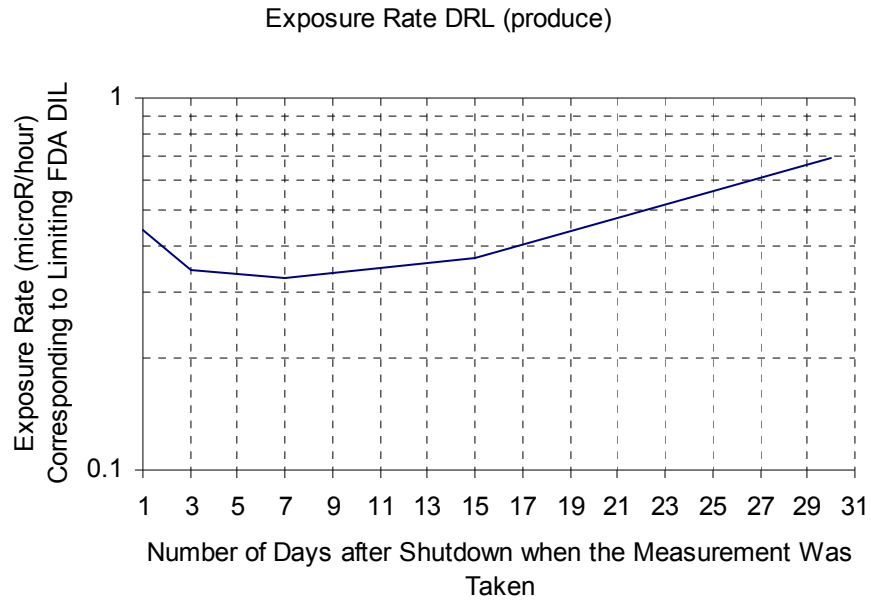
Exposure Rate DRL (produce)

Chart V3.2.4. LWR Accident Ingestion Exposure Rate DRL for Milk

Above-background exposure rates from deposition at 1 m AGL that exceed the curve indicate that milk produced by cows grazing in this area may contain radionuclide concentrations above the FDA DILs. These plots are for the limiting nuclide, [131]I.

Exposure Rate DRL (milk pathway)
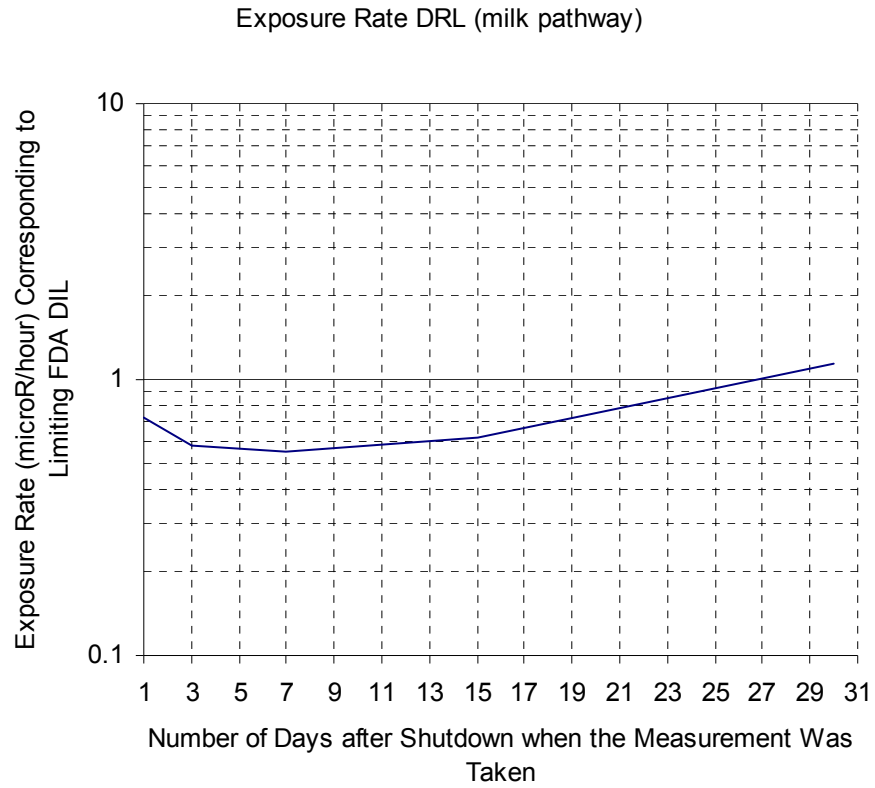
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SECTION 3. NUCLEAR WEAPON ACCIDENT

CHARTS

SECTION 3

TABLES

NUCLEAR WEAPON ACCIDENT

3.1   Description of Generic Nuclear Weapons Accident

This introduction defines nuclear weapons accident scenarios and outlines the Data Quality Objectives (DQOs) required to assess such accidents. The majority of technical information presented has been taken from the unpublished draft document, "Nuclear Emergency Response and Radiological Decision Handbook" by Gary Mansfield of the Lawrence Livermore National Laboratory. This technical information was calculated using HOTSPOT (Ho94).

Dose calculations in this section are based on the ICRP 30 Part 1 lung model, the ICRP 30 Part 1 biokinetic models, and the biokinetic models for transuranic elements present in ICRP 30, Part 4. Thus, the standard dose conversion factors should be equivalent to those presented in Federal Guidance Report # 11 (EPA88).

3.1.1   Scenario Description

This section addresses U.S. nuclear weapons accidents that disperse radionuclides, not accidents that result in nuclear yield. Weapons accidents involving nuclear yield will be addressed in Section 7.

Historically 32 nuclear weapons accidents have occurred with weapons owned by the United States. Only two of those accidents have resulted in significant dispersion of weapons-grade plutonium (WGPu) into the environment - Palomares, Spain and Thule, Greenland. Both of these accidents involved aircraft crashes and the detonation of the conventional high explosives associated with the weapon.

Unlike LWR accidents, which may take days to result in a significant release, nuclear weapons accidents, especially transportation accidents, can occur instantly. A number of scenarios can be postulated that could result in a release of WGPu or special nuclear materials (SNM). These include air or ground transportation accidents, weapons handling accidents in Department of Energy (DOE) or Department of Defense (DoD) facilities (including storage facilities or missile silos), and releases of SNM in DOE facilities. The release may not necessarily involve a complete weapon, but could result in dispersion of weapons-grade nuclear material that would require a similar response. For the purposes of this section, a transportation accident with a complete nuclear weapon will be addressed. A DOE custody transportation accident is specifically addressed.

Dispersion from a transportation accident may be the result of a fire, explosion, or both. The accident most likely will involve more than one nuclear weapon. A worst case could be postulated for a transportation accident resulting in an explosion that destroys the integrity of a weapon and releases WGPu to the environment. In terms of magnitude, the accident may release kilogram quantities of WGPu from several weapons or incur limited structural damage that results

in a tritium release. If enough fuel is present and the conventional explosives have not detonated, a fire may burn for several hours before emergency response personnel can extinguish it.

The source term may include kilogram quantities of WGPu. Tritium, uranium, and other radioactive materials may also be released. Greater consequences will result from a release of WGPu than the other radionuclides. Severity will depend on amount of material involved, the mix of materials, and the isotopic composition. Tables V3.3.1 through V3.3.3 list the nuclear data for radioactive materials that may be involved in a weapons accident (plutonium, uranium and tritium). Included are the plutonium radionuclides that makeup WGPu. The largest component is ^{239}Pu. The actual ratios of radionuclides in WGPu will vary depending on the production reactor, the chemical separation process used, and the age of the material. One radionuclide present, ^{241}Pu, is a pure beta emitter that decays to ^{241}Am with a half-life of 14.4 years. The amount of ^{241}Am will increase with the age of the WGPu mix; older material will contain a larger fraction. The remaining radionuclide components are alpha and low-energy x-ray and gamma-ray emitters. The energy range of L x-rays emitted by the plutonium radionuclides is from 11 to 21 keV, averaging at about 17 keV. ^{241}Am contributes an x-ray to the 17 keV peak, but also emits a 60 keV gamma ray. It is this higher energy 60 keV peak that is used as a marker for measurement of WGPu. Since all WGPu has a mixture of plutonium and americium radioisotope, with respect to internal contamination, the alpha-specific activity of the mixture and the relative activity of ^{241}Am in the mixture are of concern in evaluating the hazards. Table V3.3.4 lists the activity ratio of 15-year-old WGPu alpha (considered a reference mix) to ^{241}Am as 8.69. For every μCi/m^2 of ^{241}Am there will be 8.69 μCi/m^2 of WGPu mix.

The actual ratios of isotopic components of the WGPu will not be available in the early stages of an accident. Therefore, the initial assessment of consequences can be performed using total alpha, total alpha to ^{241}Am ratio, or the plutonium to ^{241}Am ratio. The WGPu mix used is the HOTSPOT reference mix used to calibrate instrumentation. The actual assessment can be modified, as more precise information becomes available.

Table V3.3.1. Nuclear Data for Radionuclides in Weapons-Grade Plutonium

Radionuclide	Principal Radiation(s)	Energy (MeV)	Yield	Half-life (years)	Specific Activity (Ci/gram)
^{238}Pu				87.7	17.1
	alpha	5.456	0.283		
		5.499	0.716		
	gamma	0.0435	3.89×10^{-4}		
		0.0999	7.47×10^{-5}		
	x-ray	0.0136*	0.039		
		0.0171*	0.056		

Radionuclide	Principal Radiation(s)	Energy (MeV)	Yield	Half-life (years)	Specific Activity (Ci/gram)
		0.0203*	0.013		
²³⁹Pu				24,065	0.0622
	alpha	5.105	0.107		
		5.143	0.152		
		5.156	0.738		
	gamma	numerous	low-yield		
		from 0.1 to 0.45			
	x-ray	0.0136*	0.015		
		0.0171*	0.021		
		0.020*	0.005		
²⁴⁰Pu				6,537	0.228
	alpha	5.124	0.265		
		5.168	0.734		
	gamma	0.045	4.5×10^{-4}		
	x-ray	0.0136*	0.038		
		0.0171*	0.053		
		0.0203*	0.012		
²⁴¹Pu				14.4	103
	Beta	0.0052	1.00		
²⁴²Pu				3.76×10^5	3.93×10^{-3}
	alpha	4.857			
		4.901			
	x-ray	0.0136*			
		0.0171*			
		0.0203*			

Radionuclide	Principal Radiation(s)	Energy (MeV)	Yield	Half-life (years)	Specific Activity (Ci/gram)
²⁴¹Am				432.2	3.44
	alpha	5.388	0.014		
		5.443	0.148		
		5.486	0.852		
		~5.5	0.005		
	gamma	0.0595	0.357		
	x-ray	0.0139*	0.13		
		0.0175*	0.20		

Nuclear data from ICRP Publication 38.

*Denotes average energy

Table V3.3.2. Nuclear Data for Selected Uranium Radionuclides

Radionuclide	Principal Radiation(s)	Energy (MeV)	Yield	Half-life (years)	Specific Activity (Ci/gram)
²³²U				72.0	21.4
	Alpha	5.137	0.003		
		5.264	0.312		
		5.320	0.685		
	Gamma	0.058	0.002		
	Low-energy x-rays	0.013- 0.019	~0.12		
²³³U				1.59×10^5	9.63×10^{-3}
	Alpha	4.729	0.016		
		4.783	0.132		
		4.825	0.844		
	Gamma	Numerous	0.1 to 0.36		
	Low-energy x-rays	0.013- 0.016	~0.1		

Radionuclide	Principal Radiation(s)	Energy (MeV)	Yield	Half-life (years)	Specific Activity (Ci/gram)
^{234}U				2.45×10^5	6.25×10^{-3}
	Alpha	4.721	0.274		
		4.773	0.723		
	Gamma	0.053	1.2×10^{-3}		
	low-energy x-rays	0.013 - 0.019	~0.1		
^{235}U				7.04×10^8	2.13×10^{-6}
	Alpha	4.366	0.176		
		4.398	0.560		
		4.417	0.022		
		4.505	0.018		
		4.558	0.044		
		4.599	0.052		
	Gamma	0.144	0.105		
		0.186	0.540		
		0.205	0.047		
	low-energy x-rays	0.013- 0.019	~0.1		
^{238}U				4.47×10^9	3.37×10^{-7}
	alpha	4.149	0.229		
		4.198	0.768		
	gamma	none, except from ^{234}Th & ^{234m}Pa	63 and 93 keV		
	low-energy x-rays	0.013- 0.019	~0.1		

Nuclear data from ICRP Publication 38.

Table V3.3.3. Radiological Properties of Tritium

Property	Value	Units
Physical Half-life:	12.35	Years
Decay Constant:	1.78×10^{-9}	sec^{-1}
Maximum Beta Energy:	18.6	keV
Average Beta Energy:	5.68	keV
Maximum Beta Range in Air:	6	mm
Maximum Penetration:	0.6	mg/cm^2
T^2 Gas (at STP):	2.6	Ci/cc
T^2 Gas (at STP):	$5.8 \times 10^{+4}$	Ci/mol
1curie =	0.385	cc
1mole =	$2.24 \times 10^{+4}$	cc
Specific Activity:		
HTO	1,450	Ci/g
T_2O	2,630	Ci/g
HT	7,240	Ci/g
T_2	9,650	Ci/g
DT	5,800	Ci/g
Dose (CEDE) Conversion Factor:	**6.4×10^{-5}**	rem /μCi
or	1.7×10^{-11}	Sv/Bq
Dose Rate in 1 Ci/m^3 **HTO** Vapor:	1.9	rem/minute
ALI (**HTO**):	$3 \times 10^{+9}$	Bq
or	81,000	μCi
DAC (**HTO**):	$8 \times 10^{+5}$	Bq/m^3
or	2.2×10^{-5}	μCi/cc
DAC (HT or T_2):	$2 \times 10^{+10}$	Bq/m^3
or	5.4×10^{-1}	μCi/cc
EPA Drinking Water Standard:	20,000	pCi/l
Biological Half-time:	10	days

Property	Value	Units
Effective Half-time:	10	days
Source Organ:	Body Water	
Source Organ Mass:	42,000	g
Target Organ:	Soft Tissues	
Target Organ Mass:	63,000	g
Specific Effective Energy:	9.0×10^{-8}	MeV/g-dis
Quality Factor:	1	
ICRP Lung Clearance Class:	D	

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Table V3.3.4. Radionuclide Ratios (Mix) for Typical Weapons-Grade Plutonium

Purpose: This table shows a typical isotopic content of weapons-grade plutonium when first produced (initial) and at 15 years. The ratio of isotopic activity to ^{241}Am activity is also shown. ^{241}Am will most likely be used as the marker radionuclide for locating deposition. The ratio of alpha activity to ^{241}Am activity is also provided. Alpha activity could be used also to locate deposition.

For 15-year-old weapons-grade plutonium

- Total alpha activity in one-gram weapons-grade plutonium = 0.09 Ci/g

- $(^{238}$Pu^{d}+239^{d}+240^{d})/(^{241}Am) activity ratio   = 7.69

- Total (alpha activity)/(^{241}Am activity) ratio   = 8.69

Radionuclide	Fraction by Initial Weight at Year 0	Isotopic Specific Activity (Ci/g)	Initial Activity at Year 0[a] (Ci/g)	Activity at Year 15[b] (Ci/g)	Mix or Radionuclide: Am-241 Activity Ratio at Year 15[c]
^{238}Pu^{d}	4.0×10^{-4}	1.7×10^{1}	6.8×10^{-3}	6.1×10^{-3}	6.0×10^{-1}
^{239}Pu^{d}	9.3×10^{-1}	6.2×10^{-2}	5.8×10^{-2}	5.8×10^{-2}	5.7
^{240}Pu^{d}	6.0×10^{-2}	2.3×10^{-1}	1.4×10^{-2}	1.4×10^{-2}	1.4
^{241}Pu^{e}	5.8×10^{-3}	1.0×10^{2}	$6.0- 10^{-1}$	2.9×10^{-1}	2.9×10^{1}
^{242}Pu	4.0×10^{-4}	3.93×10^{-3}	1.6×10^{-6}	1.6×10^{-6}	1.6×10^{-4}
^{241}Am^{d}	0.0	3.5	0.0	1.0×10^{-2}	1.0
Total	**1**		**6.7×10^{-1}**	**3.8×10^{-1}**	

a Initial activity (Ci/g) for typical weapons-grade plutonium (also available from HOTSPOT - FIDLER Calibration (Ho94).

b Activity (Ci/g) for typical weapons-grade plutonium at 15 years.

c The mix of the radionuclides for typical weapons-grade plutonium at 15 years. The mix is shown as the ratio of the activity to ^{241}Am activity for ease of calculation.

d Important radionuclides in terms of dose.

e Beta activity only

3.1.2   Data Quality Objective Process

The seven steps of EPA's DQO process are applied to the nuclear weapons accident scenario in the discussion below. Application detail is minimal in the manual, but will develop as incident-specific work proceeds. A complete and formal DQO treatment is not expected until the recovery phase, specifically, at the beginning of long-term monitoring.

3.1.2.1   DQO Step 1 – State The Problem

The details of the nuclear weapons accident assessment problem are listed in the above scenario. Aside from the generic problem of determining areas where protective action should be taken, the specific problem is to accurately determine the areal concentration of WGPu and project the dose that can be avoided. Since all assessment of measurements will initially be based on assumptions using a generic reference mix of WGPu, the true mix will need to be determined. Accurate assessment of this information is necessary to project dose to ensure protection of the public and radiation workers.

3.1.2.2   DQO Step 2 – Identify The Decisions

3.1.2.2.1 INFORMATIONAL INPUTS

Information for an assessment early in the accident response will include predicted contamination levels based on atmospheric dispersion models and a few measurements. The early measurements may be sufficient to confirm a release, but will probably make it difficult to estimate the extent and magnitude of the release without results from models. Since time is very important, the default reference information in this section will provide enough information to allow assessment to support timely and conservative decisions.

3.1.2.2.1 POTENTIAL CONSEQUENCES

The potential consequences of nuclear weapons accidents are the dispersal of radionuclides resulting from the compromise of the structural integrity of the nuclear weapon(s). Although fission yield is highly unlikely, it may be a consequence, and is discussed in Section 7, "Nuclear Yield Incident" of this volume. The release may result from fire, explosion, or impact. The material of primary concern in this dispersal is WGPu. Tritium and uranium are other radionuclides that could be dispersed, but with less radiological consequence. The inhalation hazards for Uranium vs. Plutonium are compared in Method M.V3.3.3

Tritium does not present an external hazard. The internal hazard from exposure to tritium depends on its chemical form. Tritium gas is not readily absorbed in the body either from inhalation or skin absorption. However, if it has been oxidized into tritiated water it can be

easily absorbed. It can be converted to water by either fire or explosion. Other chemical forms of tritium exist but are only a concern for workers at the accident site.

Detonation of the conventional explosives will result in a much greater release than a fire. Table V3.3.5 compares the WGPu release fractions for fires and explosions. The downwind consequences of an explosion will be proportional to the quantity of high explosive, in addition to the quantity of WGPu. Consequences from a fire will depend on the amount of fuel available. In any case, the primary pathway of exposure will be via inhalation. External exposure is of no concern, except to those individuals actually working at the accident site. Other mechanisms for receiving dose from plutonium include ingestion of material from contaminated surfaces and contamination of cuts or wounds.

After plume passage inhalation may continue from resuspension of deposited material. The committed dose accumulation rate from inhalation may be as high as 4 mrem per hour one kilometer downwind from an area contaminated to $100 \ \mu Ci/m^2$ of WGPu with a resuspension factor of 1×10^{-6}. The dose received from either the initial plume or resuspension will depend on the release fraction and particle size of the material dispersed. The inhalation pathway will probably be a problem out to 5 to 20 miles.

Inhaled material, in the 1 to 10 micron particle size range, is deposited in the pulmonary region of the lung. Material is transferred to other organs in the body with the time of transfer depending on solubility. Bone surfaces, red bone marrow, liver, and gonads are of concern for any inhalation of plutonium in addition to the lungs. The long half-life of the alpha-emitting plutonium radionuclides and the slow clearance from the body result in integrated dose over the remainder of an individual's life from inhalation of WGPu.

Contamination of crops and foods will be a very large problem. The new FDA PAGs tolerate very little transuranic contamination ($5.4 \times 10^{-5} \ \mu Ci/kg$ or $\sim 5.4 \times 10^{-4} \ \mu Ci/m^2$). Embargo of agriculture over very large areas should be expected. These areas could extend many tens of miles downwind. Moreover, monitoring these levels will be extremely difficult. It may be necessary to extrapolate from plume model predictions until results from sophisticated fixed laboratory analyses can be obtained.

The Palomares, Spain nuclear weapons accident in 1966 resulted from the collision of two aircraft refueling over the Mediterranean coast of Spain, near the village of Palomares. The collision resulted in the explosive dispersal of WGPu from two weapons on impact. Initially 630 acres were contaminated including cultivated fields and parts of the village. Resuspension of the deposited material, primarily by wind resulted in contamination of another 20 acres (DNA75).

Table V3.3.5   Airborne Release Fractions and Respirable Fractions for Various Accident Scenarios

Material	Accident Scenario	Airborne Release Fraction (ARF)	Respirable Fraction (RF)	Respirable Release (ARF · RF)	Range of ARF · RFs Observed	Comments or Reference
Pu metal fines	Fire	5×10^{-4}	1.0	5×10^{-4}		self-sustained oxidation
Pu oxide powder	large room fire	6×10^{-3}	1.0	6×10^{-3}		
Pu contamination	waste drum breach	1×10^{-3}	0.1	1×10^{-4}		building collapse, fork lift accident
Pu contamination	waste drum fire	5×10^{-4}	1.0	5×10^{-4}		
WGPu metal	fuel fire	1×10^{-2}	5×10^{-2}	5×10^{-4}	1×10^{-1} to $3 \times 10^{-1?}$	Stephens 1995, Vixen A
WGPu metal	fuel fire	5×10^{-4}	1	5×10^{-4}		LLNL HOTSPOT (default)
WGPu metal	fuel fire	1×10^{-2}	1	1×10^{-2}		LLNL ARAC (default)
WGPu metal	fuel fire	2×10^{-1}	3×10^{-3}	6×10^{-3}		AWE (U.K.)
WGPu metal	HE burn to detonation	8×10^{-1}	2×10^{-1}	1.6×10^{-1}	1.2×10^{-1} to 2×10^{-1}	Stephens 1995
WGPu metal	HE detonation	1	0.2	2×10^{-1}	1×10^{-1} to 3×10^{-1}	Stephens 1995, Roller Coaster
Depleted Uranium Munitions	wood, fuel oil fire			1×10^{-4}	5×10^{-5} to 4×10^{-3}	
Depleted Uranium Munitions	impact on targets			1×10^{-4}	5×10^{-5} to 1×10^{-3}	DOE94 (Class Y)

3.1.2.2.2 POTENTIAL ACTIONS

The Assessment Group provides decision makers with the technical basis for protective actions (radiological assessment). The major potential actions are those necessary to reduce risk due to exposure to acceptable levels. The actions are time sensitive and prioritized so as to address the most serious and time sensitive potential effects first.

The major protective actions include (in general order of priority):

- Expedited evacuation where potential for early health effects exists

- Identification/treatment of potential early health effect victims (*i.e.*, identification of unevacuated areas where the population may have the potential for early health effects so that these individuals may be identified and treated, as needed)

- Evacuation of immobile populations (hospitals, prisons…)

- Evacuation of general public

- Sheltering of public and immobile populations

- Relocation of unevacuated populace to avoid future risk

- Suspension of agricultural production over a large area

- Condemnation of foods

Other decisions might include:

- Exposure planning for emergency workers

- Selection of measurements and monitoring locations

- Guidelines for re-entry

- Identification and selection of mitigation options

3.1.2.3   DQO Step 3 – Inputs to the Decisions

3.1.2.3.1 INFORMATIONAL INPUTS

During the Early (and much of the Intermediate) Phase of the accident, the assessment methods and reference data provided in the *FRMAC Assessment Manual*, particularly this Section, are expected to be sufficient for the radiological assessment. Default decision levels are presented in Section 3.2 "Default Derived Response Levels (DRLs)." These are to be used until sufficient data have been collected to eliminate assumptions. Revision of a DRL is acceptable only if an assumption can be eliminated. Several revisions may occur over the course of time as assumptions are eliminated. More detailed and rigid treatments will be introduced as the recovery phase is entered.

3.1.2.3.2 MEASUREMENT AND PREDICTION INPUTS

The assessment will be based on deposition and resuspension. Since the time of plume passage and deposition will probably be less than a day, monitoring teams may not reach the affected area until after initial deposition is complete. Therefore, the assessment process will initially need to rely entirely on models early in the accident response. Initial measurements will be used to validate or renormalize the model. As quickly as possible, sufficient measurements must be acquired to replace dependence on the model. As time progresses and decisions become less time

critical, the quantity and quality of data will improve. Eventually guidelines will be implemented on the collection and analysis of measurements and models will become interpolation tools.

Lawrence Livermore National Laboratory's National Atmospheric Release Advisory Capability (NARAC) provides plots with inhalation dose from plume passage in addition to surface contamination levels. However, initially NARAC will provide a worst case projection based on the information they have. The worst case may involve projecting contamination and dose levels based on dispersion of several weapons, when in reality only one may be affected. Therefore, surface contamination measurements are essential to establish the actual magnitude and extent of contamination levels.

Two types of measurements are key to the Assessment Group: deposition and air concentration. Because plutonium is difficult to measure directly, surrogates should be used.

Ground surface areal concentration levels, either alpha or ^{241}Am, can be used as markers for WGPu to identify where urgent protective actions and relocation are warranted. Gross exposure rates may not be able to identify deposition levels that could result in early health effects or doses greater than the PAGs.

The first responders to make measurements may have only alpha survey meters. The alpha survey measurements must be converted to activity per unit area, (μCi/m^2), to be compared to the default DRLs (Section 3.2). Method M.V3.3.5 provides a procedure and chart for conversion from alpha survey meter results to an areal concentration of total alpha.

First responders may also use FIDLER detectors with single channel analyzers (SCAs) to measure contamination levels of ^{241}Am on ground surfaces. ^{241}Am areal concentrations can also be compared to the default DRLs (Section 3.2). If FIDLER detectors with SCAs or Violinist multi-channel analyzers (MCAs) are available, ^{241}Am areal concentrations can be used as a marker for WGPu if some assumptions are made about the isotopic mix and age of the deposited material.

Early in the accident complete information about the WGPu isotopic mixture may not be available. The actual mixture may be different than the assumed HOTSPOT 15-year reference mix used to calculate the DRL defaults. Samples will need to be collected and analyzed to confirm the default DRLs or recalculate them as soon as possible. Method M.V3.3.9 is used to determine the total alpha DRLs for Early Phase Evacuation/Sheltering (1 rem) for different ages of WGPu. The procedures only account for the age of the material using the HOTSPOT default mixture. Early in the response, DRLs will be based on default assumptions. Alpha survey meter and FIDLER results will be used to determine the locations for evacuation, relocation, or sheltering-in-place decisions made in accordance with EPA PAGs.

Following initial deposition, DRLs for markers (^{241}Am or total alpha) will be used to determine where evacuation and relocation from contaminated territory are warranted in accordance with EPA PAGs. However, field measurements of markers may not be appropriate to identify areas where restriction of locally grown food may be warranted in accordance with the HHS DILs. This is because the marker measurements associated with the limiting HHS DIL for the nuclear

weapons accident default scenario are expected to be below the detection level for the field instruments. In-situ gamma-ray spectroscopy and analysis of samples by the mobile laboratories may need to be used to identify the areas where contamination of locally produced food will be clearly well above the HHS DILs. But sophisticated, fixed laboratories must be used to measure concentrations at or below the DIL.

The table below outlines the measurements and model results that will be needed in relative order of necessity at the time the decision must be made.

Table V3.3.6. Measurements and Predictive Inputs

Predictions	Plume Passage - CEDE
	Plume Passage - Areal concentration levels
	Post Plume Passage - CEDE from resuspension
Field Measurements	AMS survey
	Alpha survey meters
	Single Channel Analyzers with FIDLERs
	Violinists with FIDLERs
	In-situ gamma spectroscopy
Sample Analyses	Air samples
	Soil samples
	Crop samples
	Water samples

3.1.2.3.3 COMPLICATING FACTORS

A number of problems could invalidate measurements or make them difficult to assess. Some of those problems are identified below.

- The WGPu isotopic mixture is unknown.
- The age of the WGPu is unknown.
- The variation in natural background can vary by up to 5 times.
  - The deposition density can be very complex, varying by a factor of 10 or more over short distances.
- Incorrectly calibrated SCA FIDLERs or Violinist (must correct or discard data).
- Invalid measurements due to improper alpha survey meter monitoring technique.
- Attenuation due to ground cover such as moisture or grass that degrades alpha or low energy emissions detection.

- Redistribution of deposited material from surfaces such as tree canopy over time depending on weather conditions.

- Fixation or transport of material in soils.

3.1.2.4   DQO Step 4 - Boundary of Consideration

3.1.2.4.1 PHYSICAL BOUNDARY

Initially, the area requiring assessment will be identified as potentially contaminated by the atmospheric dispersion models. The boundary may become larger or smaller depending on confirmation of contamination by measurements. As the response progresses and more resources are applied to the field monitoring effort, the area will be increasingly better defined. Following an accident, much of the area potentially contaminated may have been evacuated or sheltered in place, based on recommendations to State or local officials by the senior LFA (DOE or DOD) official at the accident site. The assessors will need to identify other areas that may require the same protective action. Areas that could be affected by resuspension also need to be assessed so a decision can be made about relocation or evacuation. As the response progresses, the food pathway will be a concern. In addition to the area of deposition, food-processing facilities outside this area will need to be included. Especially early in the response, there will be areas excluded.

The scene will be divided into two administrative regions because of national security concerns. The on-site area (region nearest the accident scene ~1,000-foot radius) will be secured to protect classified nuclear weapon recovery operations and components that may be scattered in the area. That area will be called either the National Security Area (NSA), if DOE is the LFA or National Defense Area (NDA), if DOD is the LFA. The off-site area is everything beyond the on-site.

Responsibility for federal operations will be divided between the FRMAC, DOE's Accident Response Group (ARG), and DOE's Consequence Management teams. FRMAC is responsible for activity in the off-site area. On-site environmental monitoring and assessment will be conducted by the Consequence Management teams, while on-site Health and Safety needs will be met by ARG's Accident Site Health Group (ASHG). These groups will operate in a secured environment with classified data. However, information concerning areal contamination levels, resuspension factors, and unclassified radionuclide mix will be shared with the FRMAC assessors.

3.1.2.4.2 TEMPORAL BOUNDARY

The FRMAC Assessment tools provided here are sufficient for the appraisal of doses during the Early and a significant portion of the Intermediate Phase.

A release resulting from a nuclear weapons accident may start with the actual accident or be substantially delayed. The release will be immediate, if there is an explosion of the weapon's high explosive. A fire may not cause a release for some period of time, if at all, depending upon

the degree of damage sustained by the weapon. There also exists a small risk of potentially initiating a fire or explosion as a result of attempting recovery and safeing of the weapon. Deposition will cease immediately after plume passage. However, resuspension, particularly of radioactive material in the on-site area, could result in redistribution of contamination. This redistribution can even be to previously uncontaminated locations, which are downwind and adjacent to the contaminated area.

If an explosion occurs at the time of the accident, the FRMAC will probably not be deployed until after the plume has dispersed and deposition has occurred. Local responders will probably extinguish any fires within hours of the start of the accident. Assessment will not begin until after the fact. Even assessment using models will probably not begin until after deposition from an explosion (without a fire) has occurred.

Timely initial assessment of the situation will be performed by either the Convoy Commander in Charge (DOE custody accident) or Incident Response Force (DOD custody accident). They will provide protective action advice to the local responders based on the condition of the weapon. The advice may be to do nothing, shelter in place downwind, or evacuate close in and shelter in place downwind. Regional RAP teams and the Consequence Management Response Team – Phase I should be in the area in four to six hours. FRMAC will not arrive until after the release has been dispersed and deposited.

Although the life cycle of the accident response continues through the recovery phase, the scope of the treatment in the *FRMAC Assessment Manual* is intended to be valid only until the end of the Intermediate Phase. At that time a Recovery Plan with a plan for long-term monitoring will be created. During that period only portions of this manual may remain applicable.

3.1.2.4.3 CONSTRAINTS

Some of the potential constraints on measurements include:

$ Re-monitoring, if new releases occur or if the release is on-going

$ Deposition on snow cover

$ Deposition on a leaf canopy

$ Delays in monitoring due to adverse weather

$ Access denied by property owners

$ Access denied for National security reasons

$ Access denied for safety reasons

$ Inaccessible terrain

3.1.2.5   DQO Step 5 - Decision Rules

The Assessment Group does not establish decision rules nor make protective action recommendations. However, published PAGs are used as decision rules for the interpretation of measurements and predictions. These PAGs are implemented as DRLs. If a measurement exceeds a specific DRL by any margin, then that location fails the test at hand. If a measurement falls short of a DRL by any margin, then it passes the test at hand.

$ DRLs have been defined for the following:

$ Emergency Worker Turn-back limits

$ Evacuation based on EPA Early Phase PAG following plume passage

$ Relocation (1st year), plus 2nd and 50-year long-term objectives

$ Agricultural hold based on deposition

$ Food condemnation (agricultural embargo) based on food concentration

$ Water condemnation based on concentration

The Federal Advisory Team has the authority to alter the PAGs and the computational approach.

3.1.2.6   DQO Step 6-Tolerance Limits

Assessors must establish tolerable levels of uncertainty when calculating DRLs. For example, evacuation, shelter, and agricultural product holds have a higher tolerance level than re-entry, which is higher yet than relocation, and so on. It is up to the assessor to establish these tolerable levels until a more definitive uncertainty analysis can be performed.

If assessments are being used for:

- Evacuation, sheltering, or agricultural hold considerations, the tolerance limit is a factor of 10.

- Re-entry considerations, the tolerance limit is a factor of 2.

- Relocation, then the tolerance limit will be negotiated, but is expected to be approximately 30%.

$ Return, the limit will be negotiated and will likely be much smaller.

- Condemning foods or water, the same criteria used by USDA for evaluation of non-radiological contamination will be applied (10%).

3.1.2.7   DQO Step 7- Optimal Design

There is little FRMAC can do initially to optimize design. Optimization is primarily the responsibility of the EPA under its management of FRMAC during the recovery phase.

3.2   Default Derived Response Levels (DRLs)

The default DRLs for a nuclear weapons accident (not resulting in nuclear yield) are presented in Table V3.3.7

Table V3.3.7. Default DRLs for Releases from a Nuclear Weapons Accident

Issue	Marker	DRL	Sensitivity, Uncertainty, Spatial Density, Assumptions
Worker Protection	^{241}Am, Total alpha areal conc. ^{241}Am air conc.	See Table V3.3.10	The inhalation rate, resuspension factor, and WGPu isotopic ratios are assumed.
EPA Early Phase PAG (evacuation)	^{241}Am areal conc.	3.7 μCi/m^2	Areal concentrations in μCi/m^2 indicating that evacuation or substantial shelter could be implemented in accordance with EPA PAGs.
	Total alpha areal conc.	32 μCi/m^2	If conditions are arid, windy, or involve mechanical activities, consider higher resuspension factor (up to 1×10^{-4}). Do not use exposure rate measurements.
Relocation 1st year	^{241}Am areal conc.	2.8 μCi/m^2	Areal concentrations in μCi/m^2 indicating that relocation is warranted.
	Total alpha areal conc.	25 μCi/m^2	If conditions are arid, a higher resuspension factor may be required. Monitor resuspension factor as a function of time, then adjust resuspension model. Do not use exposure rate measurements.
Ingestion PAG Fresh produce	^{241}Am areal conc.	7.3×10^{-5} μCi/m^2	Areal DRLs for assessment of deposition.
	Total alpha areal conc.		
	^{241}Am food conc.	6.4×10^{-4} μCi/m^2	Food concentration DRLs for assessment of food samples.
	Total alpha food conc.	7.3×10^{-6} μCi/kg	
		6.4×10^{-5} μCi/kg	

3.3   Worker Protection

3.3.1   Discussion of Assessment in Worker Protection

Initially, two distinct work areas will exist early in the response to a nuclear weapons accident:

- On-site, inside the NSA/NDA, where the emphasis is on recovering the damaged weapon(s).

- Off-site, outside the NSA/NDA, where the emphasis is on preventing additional risk to the public.

Eventually, the NSA/NDA will be rescinded and the two will become one with no organizational or jurisdictional distinction.

The working conditions resulting from a nuclear weapons accident are not the routine occupational environment. The health of the public in potentially contaminated areas may be at great risk. Therefore, during the Early Phase of the accident, the regulations in EPA Guidance (EPA92) for dose limits for workers performing emergency services apply. Attempting to apply the normal occupational limits to early phases of the accident may result in putting the health of the public at much greater risk. Once the threat of additional dispersion has been eliminated and the additional risk to the public has been reduced, the normal occupational limits should apply.

3.3.1.1   On-Site - Inside the NSA/NDA

FRMAC will not have responsibility for worker protection on-site. That responsibility belongs to the ARG ASHG for the health and safety of workers from all agencies involved (*i.e.*, DOE and DoD). After the on-site area collapses and becomes the off-site, FRMAC will inherit this responsibility, but key personnel and assets of the ASHG will fold into FRMAC to assist. External exposure on-site is of only moderate concern. The primary concern in a nuclear weapons accident is internal exposure via inhalation. Real-time air monitoring in the form of Alpha-Environmental Constant Air Monitor (Alpha e-cam), and routine air sampling, as well as ground surface measurements will be employed by ASHG and FRMAC Health and Safety (H&S) to determine requirements for worker protection. Very early in the response, ground contamination levels are used to determine initial worker protection requirements until air sample results are available. Protection inside the NSA includes respirators (particulate or Self-Contained Breathing Apparatus [SCBA]) and anti-contamination (anti-C) clothing. A fixative will be applied as soon as possible to lower the chance of resuspending radioactive material. Contamination control procedures are practiced including thorough monitoring and removal of anti-C clothing prior to leaving the weapons recovery area. NSA/NDA is considered a DOE facility, therefore, civilian workers inside the NSA/NDA are treated as DOE certified Radiation Workers and, therefore, must comply with 10 CFR 835. The dose limits for emergency workers are listed in Table V3.3.8

Table V3.3.8   EPA/DOE Dose Limits for Workers Performing Emergency Services

TEDE Dose Limit (rem)	Eye Dose Limit (rem)	Organ, Thyroid, Skin Dose Limit (rem)	Activity	EPA/DOE Dose Limits for Workers Performing Emergency Services
5	15	50	All	
10	30	100	Protecting major/valuable property	Where lower dose limit is not practicable
25	75	250	Lifesaving or protection of large populations	Where lower dose limit is not practicable
>25	>75	250	Lifesaving or protection of large populations	Only on a voluntary basis by persons fully aware of the risks involved.

(sources: 10 CFR 835 and EPA92)

Note that, in application of these limits, Section 1302 of 10 CFR 835 requires that:

1. The risk of injury to those individuals involved in rescue and recovery operations shall be minimized.

2. Operating management shall weigh actual and potential risks to rescue and recovery personnel against the benefits to be gained.

3. Rescue action that might involve substantial personal risk shall be performed by volunteers.

4. Each individual selected (for emergency dose limits) shall be appropriately trained and briefed beforehand of the known or anticipated hazards to which the individual will be subjected.

Stay times for exposure (inhalation) to resuspended weapons-grade plutonium may be established using Methods M.V3.3.4, while Method M.V3.3.7 provides dose accumulation rates.

Due to the nature of the work during weapons recovery activity, a resuspension factor of 1×10^{-4} should be used to estimate stay times, as used by Method M.V3.3.4 Under normal conditions, a resuspension factor of 1×10^{-6} might be used to estimate the level of airborne contamination; however, the micro-climate created by the worker's activities may significantly increase the suspension of activity from contaminated surfaces.

3.3.1.1   Off-Site - Outside the NSA/NDA

Workers monitoring outside the NSA/NDA in public areas may be from a number of state, local, or Federal agencies. As indicated in Volume 1, the major responsibility of the Assessment Group in worker protection is to provide information to FRMAC H&S for determining the turn-back guidance for emergency workers. The area outside the NSA/NDA will have considerably lower maximum contamination levels than close to the accident site. However, workers outside the NSA/NDA, in an effort not to alarm the public, may not use the extensive personal protective equipment (PPE) that is required inside the NSA/NDA.

Table V3.3.8 summarizes the EPA dose limits for emergency workers. All doses (external and inhalation) received during an emergency are included in this limit. As stated in Volume 1, the difficulty is in estimating the inhalation dose. Initially doses resulting from inhalation can be estimated but must be confirmed by bioassay. Assignment of dose from bioassay can be a time-consuming process, especially for in-vitro bioassay. In an emergency, especially in the Early Phase, the time required for assignment of dose from bioassay results may not be compatible with the goal of protecting the public. In this case air-sampling results can be used as an estimate. Very early in the response, air-sampling results probably will not be available for all contaminated areas where workers are required to be. In this case, an estimate of assignment of dose can be based on the areal surface concentration and a resuspension factor.

Table V3.3.9 summarizes the default turn-back guidance. The guidance is expressed in terms of areal contamination levels for ^{241}Am and total alpha and in DAC-hours. The guidance can be used as turn-back values. Turn-back levels and the option to don PPE are at the professional discretion of each RAP team, but the DOE RAPTER course trains response teams that the turn-back concentration is 60 μCi/m^2 of WGPu. The RAP turn-back value was based on an airborne concentration of approximately ten Derived Air Concentrations (DACs), (*actually 9.375*)(assuming the resuspension factor is 1×10^{-6}). The teams are not expected to be in this concentration for more than a few hours. The committed dose accumulated in two hours at that concentration would be 50 mrem. Early in the accident, the RAP teams are more concerned with establishing that a release has occurred and grossly establishing the area that may be contaminated. They will begin monitoring before the full FRMAC is deployed.

Method M.V3.3.7 contains guidance for estimating emergency workers potential dose from inhalation during nuclear weapons accidents with dispersed WGPu. Method M.V3.3.4 can be used to determine stay times at these limits.

The guidance is based on dose limits (CEDE) but must be presented in a manner that is useful to field personnel. In the case of a nuclear weapons accident, the guidance would need to be in areal contamination concentrations. The instrument response to the turn-back concentrations will be different for each instrument based on instrument type and calibration. The conversion to the instrument response output must be applied for the instrument used. Violinist readings would be in μCi/m^2 for ^{241}Am. Using the FIDLER with a single-channel analyzer would give the total counts in the 60 keV region of interest. Alpha survey meters would give readings in counts per minute (cpm) and would have to be converted into the desired activity units.

Table V3.3.9   Default Federal Emergency Worker Dose Limits and Turn-back Guidance
for Nuclear Weapons Accidents Dispersed WGPu

(DAC = 6.4×10^{-6} µCi/m^3, Resuspension Factor assumed to be 1×10^{-6})

Dose Limit Category or Emergency Activity	^{241}Am[a] µCi/m^2 to Reach Dose Level in 2000 Hours *(total alpha/8.69)*	Total Alpha[a] µCi/m^2 to Reach Dose Level in 2000 Hours	CEDE from Inhalation Assuming No External Exposure[b] (mrem)	DAC-Hours Required to Reach Accumulated Dose Level
Administrative Limits				
Investigation Level	0.22	1.9	1500	600
Administrative Level	0.37	3.2	2500	1000
Emergency Activity				
All	0.74	6.4	5000	2000
Protecting Major Property	1.48	12.8	10000	4000
Life Saving or Protecting Large Populations	3.7	32	25000	10000
Life Saving or Protecting Large Populations (voluntary)[c]	>3.7	>32	>25000	>10000

a. Resuspension Factor of 1×10^{-6}

b. Assumes no respiratory protection; No significant external exposure; No initial plume inhalation

c. Only on a voluntary basis to personnel fully aware of the risks involved

The areal concentration limits are based on the DAC for WGPu. The dose contribution from the pure beta emitter ^{241}Pu is included. Tables V3.3.10 and V3.3.11 provide the DACs and Annual Limit of Intake (ALIs) for all potential contributing radionuclides. The values in these tables are based on ICRP 30 Part 4 models.

A DAC is the concentration of airborne activity, which, if breathed for 2000 hours a year (2400 m^3), will result in an intake of 1 ALI. The values are expressed in units of µCi/cc instead of µCi/m^3. Multiply the µCi/cc value by 10^{-6} to convert it into µCi/m^3. It is easier to convert from areal concentration in µCi/m^2 to air concentration, µCi/m^3. Caution is advised because the bone surface is the limiting value for most particle sizes and chemical forms of WGPu. The calculation has been performed for the HOTSPOT reference 15-year mix, assuming the chemical form is oxide and the particle size is 1 micron. A resuspension factor of 1×10^{-6} is assumed. Once the latest WGPu isotopic mix information is available, the values will need to be recalculated.

Table V3.3.10. Derived Air Concentrations (DACS)

Class W		f1= 1 E-03	Other
Radionuclide	0.2 microns AMAD (µCi/cc)	1.0 microns AMAD (µCi/cc)	5.0 microns AMAD (µCi/cc)
^{238}Pu:	2.4×10^{-12}	3.0×10^{-12}	2.6×10^{-12}
^{239}Pu:	2.2×10^{-12}	2.7×10^{-12}	2.4×10^{-12}
^{240}Pu:	2.2×10^{-12}	2.7×10^{-12}	2.4×10^{-12}
^{241}Pu:	1.0×10^{-10}	1.3×10^{-10}	1.1×10^{-10}
^{242}Pu:	2.4×10^{-12}	2.8×10^{-12}	2.6×10^{-12}
^{241}Am:	2.1×10^{-12}	2.6×10^{-12}	2.3×10^{-12}
WGPu (α):	2.1×10^{-12}	2.5×10^{-12}	2.2×10^{-12}
Class W		f1= 1 E-04	Nitrates
Radionuclide	0.2 microns AMAD (µCi/cc)	1.0 microns AMAD (µCi/cc)	5.0 microns AMAD (µCi/cc)
^{238}Pu:	2.4×10^{-12}	3.0×10^{-12}	2.7×10^{-12}
^{239}Pu:	2.2×10^{-12}	2.7×10^{-12}	2.4×10^{-12}
^{240}Pu:	2.2×10^{-12}	2.7×10^{-12}	2.4×10^{-12}
^{241}Pu:	1.0×10^{-10}	1.2×10^{-10}	1.1×10^{-10}
^{242}Pu:	2.4×10^{-12}	2.9×10^{-12}	2.6×10^{-12}
^{241}Am:	2.1×10^{-12}	2.6×10^{-12}	2.3×10^{-12}
WGPu (α):	2.1×10^{-12}	2.5×10^{-12}	2.3×10^{-12}
Class Y		f1= 1 E-05	Oxides
Radionuclide	0.2 microns AMAD (µCi/cc)	1.0 microns AMAD (µCi/cc)	5.0 microns AMAD (µCi/cc)
^{238}Pu:	3.8×10^{-12}	7.3×10^{-12}	1.5×10^{-11}
^{239}Pu:	3.5×10^{-12}	6.8×10^{-12}	1.4×10^{-11}
^{240}Pu:	3.5×10^{-12}	6.8×10^{-12}	1.4×10^{-11}
^{241}Pu:	1.7×10^{-10}	3.2×10^{-10}	5.8×10^{-10}
^{242}Pu:	3.7×10^{-12}	7.1×10^{-12}	1.4×10^{-11}
^{241}Am:	3.4×10^{-12}	6.6×10^{-12}	1.3×10^{-11}
WGPu (α):	3.4×10^{-12}	6.4×10^{-12}	1.3×10^{-11}

Italics indicate CDE to bone surface is limiting

Table V3.3.11. ALIs for Inhalation: Plutonium

Class W		f1= 1 E-03	Other
Radionuclide	0.2 microns AMAD (μCi)	1.0 microns AMAD (μCi)	5.0 microns AMAD (μCi)
^{238}Pu:	*0.0059*	*0.0071*	*0.0064*
^{239}Pu:	*0.0053*	*0.0064*	*0.0058*
^{240}Pu:	*0.0053*	*0.0064*	*0.0057*
^{241}Pu:	*0.24*	*0.32*	*0.26*
^{242}Pu:	*0.0057*	*0.0067*	*0.0062*
^{241}Am:	*0.0051*	*0.0062*	*0.0056*
WGPu (α):	*0.0050*	*0.0060*	*0.0054*
Class W		**f1= 1 E-04**	**Nitrates**
Radionuclide	0.2 microns AMAD (μCi)	1.0 microns AMAD (μCi)	5.0 microns AMAD (μCi)
^{238}Pu:	*0.0059*	*0.0071*	*0.0064*
^{239}Pu:	*0.0053*	*0.0064*	*0.0058*
^{240}Pu:	*0.0053*	*0.0064*	*0.0058*
^{241}Pu:	*0.24*	*0.29*	*0.26*
^{242}Pu:	*0.0057*	*0.0069*	*0.0062*
^{241}Am:	*0.0051*	*0.0062*	*0.0056*
WGPu (α):	*0.0050*	*0.0060*	*0.0054*
Class Y		**f1= 1 E-05**	**Oxides**
Radionuclide	0.2 microns AMAD (μCi)	1.0 microns AMAD (μCi)	5.0 microns AMAD (μCi)
^{238}Pu:	0.0090	0.0174	*0.0362*
^{239}Pu:	0.0085	0.0162	*0.0325*
^{240}Pu:	0.0085	0.0162	*0.0325*
^{241}Pu:	*0.40*	*0.76*	*1.39*
^{242}Pu:	0.0089	0.0171	*0.035*
^{241}Am:	0.0082	0.0158	*0.031*
WGPu (α):	0.0081	0.0154	*0.0302*

Italics Indicate CDE To Bone Surface Is Limiting

3.3.2   Computation of Turn-Back Guidance

Inhalation is the primary pathway since external exposure in a non-yield nuclear weapons accident is negligible. Air samples would allow some estimate of the internal exposure received by workers. Especially in the Early Phase, air monitoring can probably not take place at every monitoring location. Turn-back values must be based on air sample measurements or areal concentrations with an assumed resuspension factor. Because inhalation is the primary pathway for receiving dose, DAC-hours are the best approach to assigning dose until bioassay samples can be collected, analyzed, and assessed.

The isotopic mix of WGPu must either be determined from measurements or obtained from the relevant National Laboratory and the turn-back DRL revised based on this information.

Revised turn-back guidance may be calculated using Method M.2.1 in Volume 1. The nuclides listed in Table V3.3.4 would be expected to be major contributors to dose for this calculation.

3.4   Early (Plume) Phase

The Early Phase is considered to last for 96 hours for the purpose of dose assessments. The EPA evacuation PAG is 1 to 5 rem, where the dose considered is the sum of the EDE from external sources and the CEDE incurred from significant inhalation pathways. The inhalation pathway is of greatest concern in a nuclear weapons accident. It is expected that the public within $2 - 7$ miles in a sixty-degree arc downwind of the accident will either shelter in place or evacuate based on the condition of the weapon(s) involved. The release and resulting deposition will probably be over by the time the Assessment Group is involved. Therefore, the emphasis for the Early Phase is in identifying those areas that have not been evacuated where the Early Phase dose may exceed the PAG.

3.4.1   Default Derived Response Level

The default DRLs for the evacuation PAG is given in terms of areal concentration of ^{241}Am and total alpha markers in Table V3.3.7. Use these defaults until they can be revised or replaced by an alternate assessment. A complication that must be monitored carefully is resuspension that can be greatly exacerbated by dry weather, windy weather, or mechanical activity, particularly during the first few days of the event. Therefore, assessments based on frequent measurements of air concentration are preferred over those based on application of a resuspension factor to deposition. If assessments must use a resuspension factor, then it may be necessary to use a resuspension factor greater than that used by the FRMAC resuspension model (1×10^{-6}), if these aggravating conditions exist. A resuspension factor as high as 1×10^{-4} may be necessary. The FRMAC Dose Conversion Factors (DCFs) and default DRLs can be adjusted simply by scaling to the ratio of the resuspension factors as below, because the dose is almost exclusively due to inhalation of the resuspension.

$$DCF_{new} = \left(\frac{R_{new}}{1E-6} \right) \bullet DCF_{old}$$

$$DRL_{new} = \left(\frac{1E-6}{R_{new}} \right) \bullet DRL_{old}$$

3.4.2   Revision of Evacuation DRL

The DRL may need to be revised once the isotopic mix of the WGPu has been determined. Method M.3.3 in Volume 1 should be used for this purpose. If possible use base assessments on air samples rather than deposition, again using Method M.3.3 of Volume 1.

3.5   Intermediate Phase – Relocation

The areal deposition concentrations can be used as the DRL for locating areas where relocation is warranted to meet the EPA guidance following release from a nuclear weapons accident. DRLs may change with time, due to weathering and fixation of the deposited material. Resuspension may change the spatial extent, if some aggressive action has not been taken. Therefore, the DRLs must be re-evaluated periodically, which requires measurement of the resuspension factor and adjustment of the resuspension model. EPA guidance also established objectives to ensure the dose in the second year does not exceed 500 mrem and the cumulative dose over 50 years does not exceed 5000 mrem. Unlike an LWR accident, due to the long half-lives of the WGPu radionuclides, meeting these objectives will be more difficult. To meet the long-term objectives, 1) areas will need to be decontaminated or 2) continued exclusion of the public from contaminated areas may be necessary. The process for determining if these objectives are being met will be developed as part of the long-term assessment plan and is beyond the scope of this document.

3.5.1   Default Relocation Derived Response Levels

Default DRLs for the Intermediate (Relocation) Phase (first year) are presented in terms of areal concentration of [241]Am and total alpha markers in Table V3.3.7. Use these defaults until they can be revised.

3.5.2   Revision of Relocation Derived Response Levels

The DRLs are a function of the mixture of radionuclides in the deposition. Samples should be taken and analyzed to assure that the values used in the calculations are representative of the entire affected area. If multiple weapons are involved, the radionuclide mix of each could be different.

The resuspension factor in arid environments may be greater than that used by the FRMAC resuspension model (with an initial value of 10^{-6}). Monitor resuspension as a function of time and compare it with the resuspension model used by FRMAC. If adjustment is necessary, adjust only the initial resuspension value (10^{-6}), not the time dependence (1/t in days). Then the FRMAC DCFs and default DRLs can be adjusted simply by scaling to the ratio of the resuspension factors as below, because the dose is almost exclusively due to inhalation of the resuspension.

$$DCF_{new} = \left(\frac{R_{new}}{1E-6} \right) \bullet DCF_{old}$$

$$DRL_{new} = \left(\frac{1E-6}{R_{new}} \right) \bullet DRL_{old}$$

3.5.2.1   Criteria

The DRLs should be re-evaluated:

- Initially, with the best estimate of the WGPu isotopic mix.

- Daily for the first week to account for additional information gained about the WGPu isotopic composition of the deposition.

- Weekly for the first month to account for further changes in the composition of the deposition.

- Monthly thereafter, until resuspension and fixation no longer have a major impact.

- The characteristics of the release and changing meteorological conditions may dictate that a single value for each type of DRL will not be appropriate for the entire affected area.

3.6   Intermediate Phase – Ingestion

The major protective actions taken during the Intermediate Phase involve relocation and restrictions on the use of contaminated food and water. The FDA issued recommendations regarding contaminated food in 1998 (FDA98). Key points in these recommendations were the DILs, concentrations in food at which some action should be taken to limit or preclude the use of the food product.

The FDA DIL relevant to a nuclear weapon accident is in terms of the sum of the concentrations of ^{238}Pu, ^{239}Pu, and ^{241}Am in food samples. It is unlikely that other radionuclides will be of concern, unless the weapon is only comprised of uranium. FRMAC has extended the DILs to DRLs for deposition of ^{241}Am and total alpha as marker radionuclides. The DILs and DRLs are

very small and will be technically challenging to measure. Field monitoring will not suffice. Sensitive laboratory measurements are required.

Food may become contaminated through several pathways. This section considers the simplest case—direct deposition onto produce (*e.g.*, leafy vegetables). Assessments based on deposition are useful for estimation of the area where the FDA DILs may be exceeded, as described in Method M.5.2 of Volume 1. Assessment of the acceptability of food for consumption must be based on analysis of food samples using Methods M.5.3 and M.5.13 found in Volume 1.

3.6.1   Default Ingestion Derived Intervention Levels

The DILs and default DRLs are presented in terms of food concentration and areal concentration respectively. They are expressed in terms of two marker concentrations (^{241}Am and total alpha), as well as the sum of ^{238}Pu, ^{239}Pu, and ^{241}Am concentration in Table V3.3.7. Use these defaults until they can be revised.

3.6.2   Revision of Ingestion Derived Response Levels

The DRLs are a function of the mixture of radionuclides in the deposition. Samples should be taken and analyzed to assure that the values used in the calculations are representative of the entire affected area. The FDA DILs will not be adjusted, unless directed by the LFA.

3.6.2.1   Criteria

The DRLs should be re-evaluated:

- When the actual radionuclide mix of the release is known.

- When site-specific transfer factors have been determined.

3.6.2.2   Procedure

The areal concentration ingestion DRLs for a nuclear weapons accident are calculated using Method M.5.8 in Volume 1.

3.7   Decay Corrections

Radioactive decay is not a concern for the transuranic radionuclides that make up WGPu since the shortest half-life among these nuclides is that of ^{241}Pu and is14.4 years.

3.8   Ancillary Information and Methods

The following procedures and charts were essentially copied from the DRAFT, "Nuclear Emergency Response and Radiological Decision Handbook" by Gary Mansfield. This is the primary guide for the ARG's ASHG.

Method M.V3.3.1 ^{241}Am Concentration vs. Age of WGPu Mixture

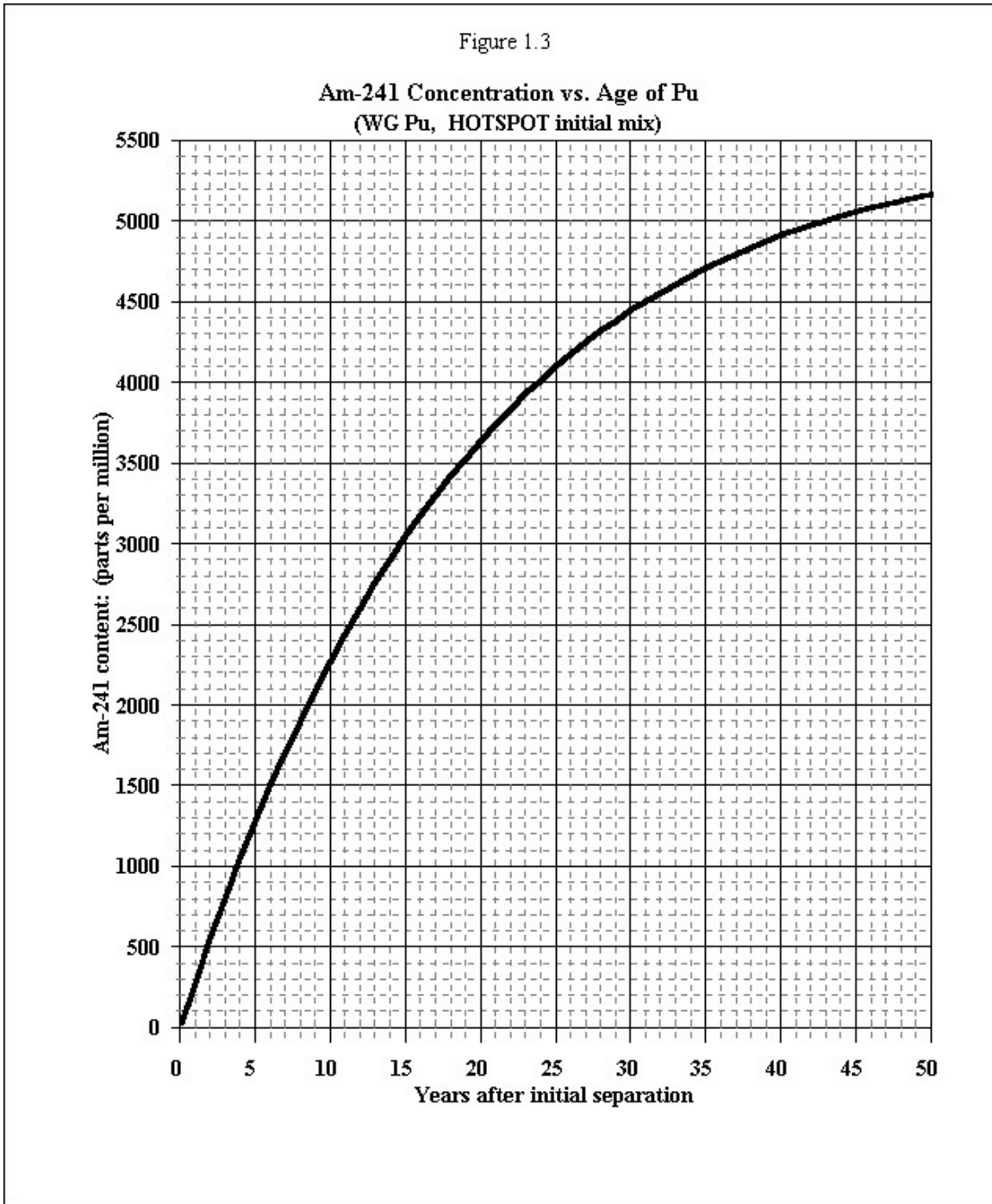**Purpose:** Estimation of the concentration of ^{241}Am in a reference mixture of weapons-grade plutonium, as a function of the time since the plutonium was chemically separated.

Discussion: Weapons-grade plutonium has no significant gamma emissions after chemical separation. However, the small amount of ^{241}Pu in WGPu decays to ^{241}Am, which is an excellent marker because it is relatively easy to detect. The curve in Chart V3.3.1 assumes the initial mixture of radionuclides is that used by HOTSPOT (Ho94).

Steps:

Use Chart V3.3.1 to estimate the concentration or parts per million (ppm) of ^{241}Am in a reference mixture of weapons-grade plutonium, as a function of the number of years that have passed since the plutonium was chemically separated.

On the X-axis (horizontal axis) move RIGHT to find the known or assumed age (years after separation) of the mixture of weapons-grade plutonium. Move UP to the curve that represents the ^{241}Am mass concentration (ppm.). Move LEFT to the Y-axis (vertical axis) to read the ^{241}Am concentration in ppm.

Chart V3.3.1. In-Growth of ^{241}Am: Weapons-Grade Plutonium

Figure 1.3

Am-241 Concentration vs. Age of Pu
(WG Pu, HOTSPOT initial mix)

Method M.V3.3.2   Alpha Activity in Weapons-Grade Pu to ^{241}Am Activity

Purpose: Estimation of the ratio of alpha activity to ^{241}Am activity in a reference mixture of weapons-grade plutonium, as a function of the time since the plutonium was chemically separated.

Discussion: Both total alpha and ^{241}Am are used as markers for WGPu. This ratio permits convenient interchange between each. As WGPu ages, the ratio of total alpha-specific activity to ^{241}Am-specific activity declines. The concentration of ^{241}Am increases, essentially zero after chemical processing, due to the decay of ^{241}Pu. The total alpha-specific activity also increases slightly, because the ^{241}Am is an α emitter, which replaces ^{241}Pu, a β emitter. The curve in Chart V3.3.2 assumes the initial mixture of radionuclides is that used by HOTSPOT (Ho94).

Steps:

Use Chart V3.3.2 to estimate the alpha activity in a mixture of weapons-grade plutonium relative to the ^{241}Am in the mixture. These values would be used to convert ^{241}Am measurements used as a "tracer" for the alpha activity in the mix.

On the X-axis (horizontal axis), move RIGHT to find the known or assumed age (years after separation) of the mixture of weapons-grade plutonium. Move UP to the curve that represents the desired quantity (either total alpha activity, Pu + Am, total Pu alpha activity, or ^{239}Pu + ^{240}Pu alpha activity). Move LEFT to the Y-axis (vertical axis) to read the ratio of the selected alpha activity to the measured ^{241}Am activity.

Chart V3.3.2. Alpha Activity Relative to [241]Am Activity in Weapons-Grade Plutonium

Figure 1.4

Ratio of Total Alpha / Am-241 Activity
(LLNL HOTSPOT initial mix: W.G. Pu)
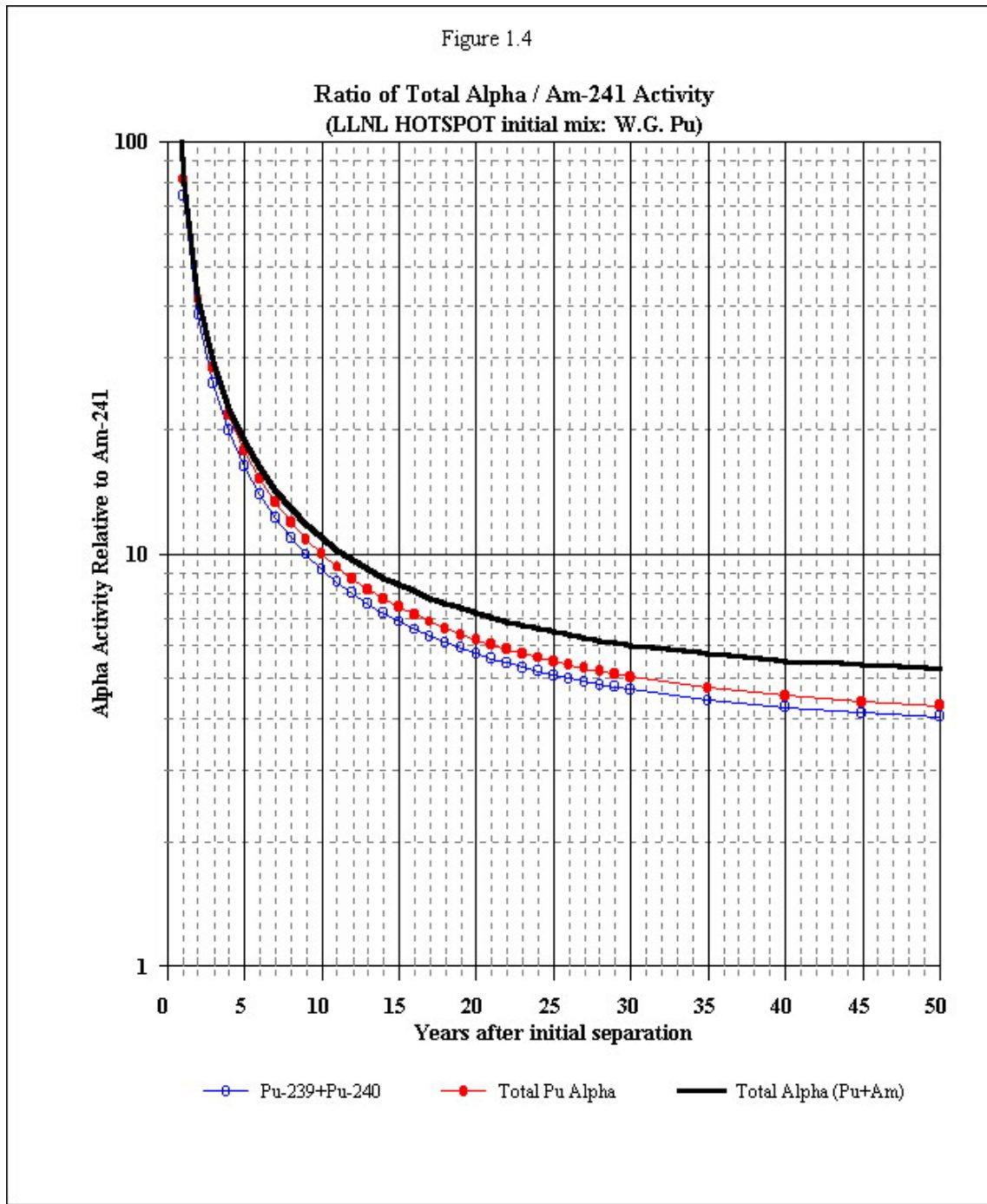
Method M.V3.3.3   Comparison of Inhalation Hazards: Uranium vs. Plutonium

Purpose: Compare the relative inhalation hazards (dose per unit mass inhaled) from uranium at various enrichments to that of weapons-grade plutonium.

Discussion: Inhalation hazards due to uranium pale in comparison to those from WGPu. However, the uranium hazard grows with increasing enrichment of ^{235}U.

From Chart V3.3.3 one may derive the following table: Clearly, the doses from the uranium are almost inconsequential compared to the doses from plutonium.

Inhaled Material	rem/µg inhaled	Ratio of Pu Dose to U Dose
Depleted Uranium	5×10^{-5}	500,000
Natural Uranium	8.5×10^{-5}	300,000
95+% Enriched Uranium	9×10^{-3}	2,800
Weapons-Grade Plutonium	25	1

Steps:

Use Chart V3.3.3 to compare the relative inhalation hazards from uranium (at various enrichments) and weapons-grade plutonium.

On the X-axis (horizontal axis), move RIGHT to find the enrichment (percent ^{235}U by mass) of the uranium mixture. Move UP to the curve that represents the dose conversion factor (DCF) for that mixture of uranium. The DCF for plutonium is, of course, constant, and is plotted for illustrative purposes. Move LEFT to the Y-axis (vertical axis) to read the DCF, in rem (50-year CEDE) per microgram inhaled.

Chart V3.3.3. Comparison of Uranium (U) and Plutonium (Pu) Inhalation Hazards

Figure 1.9

Comparison of Pu vs U Dose Hazards
INHALATION - Dose per mass inhaled
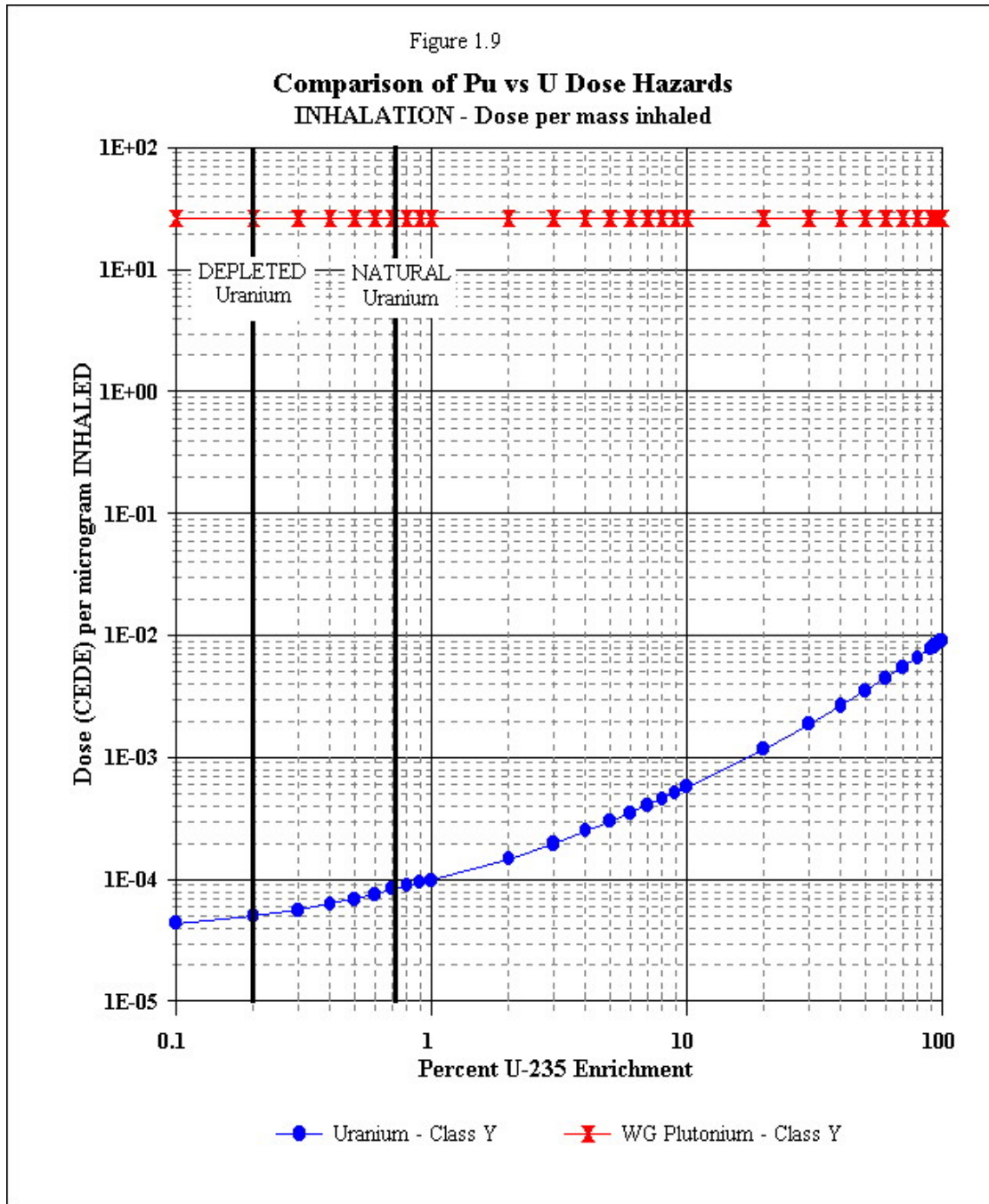
Assumptions:
- Approximate values of specific activity calculated using NRC empirical formula.
- ICRP-30 dosimetric models, Class Y, 1 micron Activity Median Aerodynamic Diameter (AMAD) material.
- Explosive dispersal characteristics (airborne release fraction and respirable fraction) assumed to be the same for uranium and plutonium.

Method M.V3.3.4   Inhalation Exposure Stay Times (Resuspension of Weapons-Grade
                   Plutonium)

Purpose: Assess deposition measurements or predictions to estimate allowable stay times for unprotected (no respiratory protection) exposure to airborne contamination that might be expected to be resuspended from contaminated surfaces.

Discussion: Inhalation of resuspended material is the primary dose pathway. Isodose curves are presented in Chart V3.3.4 as a function of deposition and exposure time. The CEDE is calculated for an unprotected individual working in an area of severe resuspension (1×10^{-4}). The chart shows the exposure time (in hours) to give the indicated dose (CEDE) from resuspension from surfaces contaminated with Class Y (oxides) of weapons-grade plutonium.

Steps:

Use Chart V3.3.4 to estimate allowable stay times for exposure resuspension.

On the X-axis (horizontal axis), move RIGHT to find the level of surface contamination (total alpha of weapons-grade plutonium) in microcuries per square meter ($\mu Ci/m^2$). Move UP to the curve representing the dose limit (CEDE) of interest. Move LEFT to the Y-axis (vertical axis) to read the exposure time (in hours) that would be expected to result in the selected dose value from resuspension of activity.

Chart V3.3.4. Emergency Worker Stay Times

Figure 3.1

Emergency Worker Stay Times
(time to give indicated CEDE)

Assumptions:

No respiratory protection,
Weapons Grade Plutonium,
Class Y, 1 micron AMAD,
Resuspension Factor = 1 E-4 (1/m),
Dose = 50-year CEDE

Exposure Hours to Give Indicated Dose

Surface Contamination (uCi/m^2)

▼ 0.1 rem   ● 1 rem   ■ 5 rem   ■ 10 rem   ✱ 25 rem
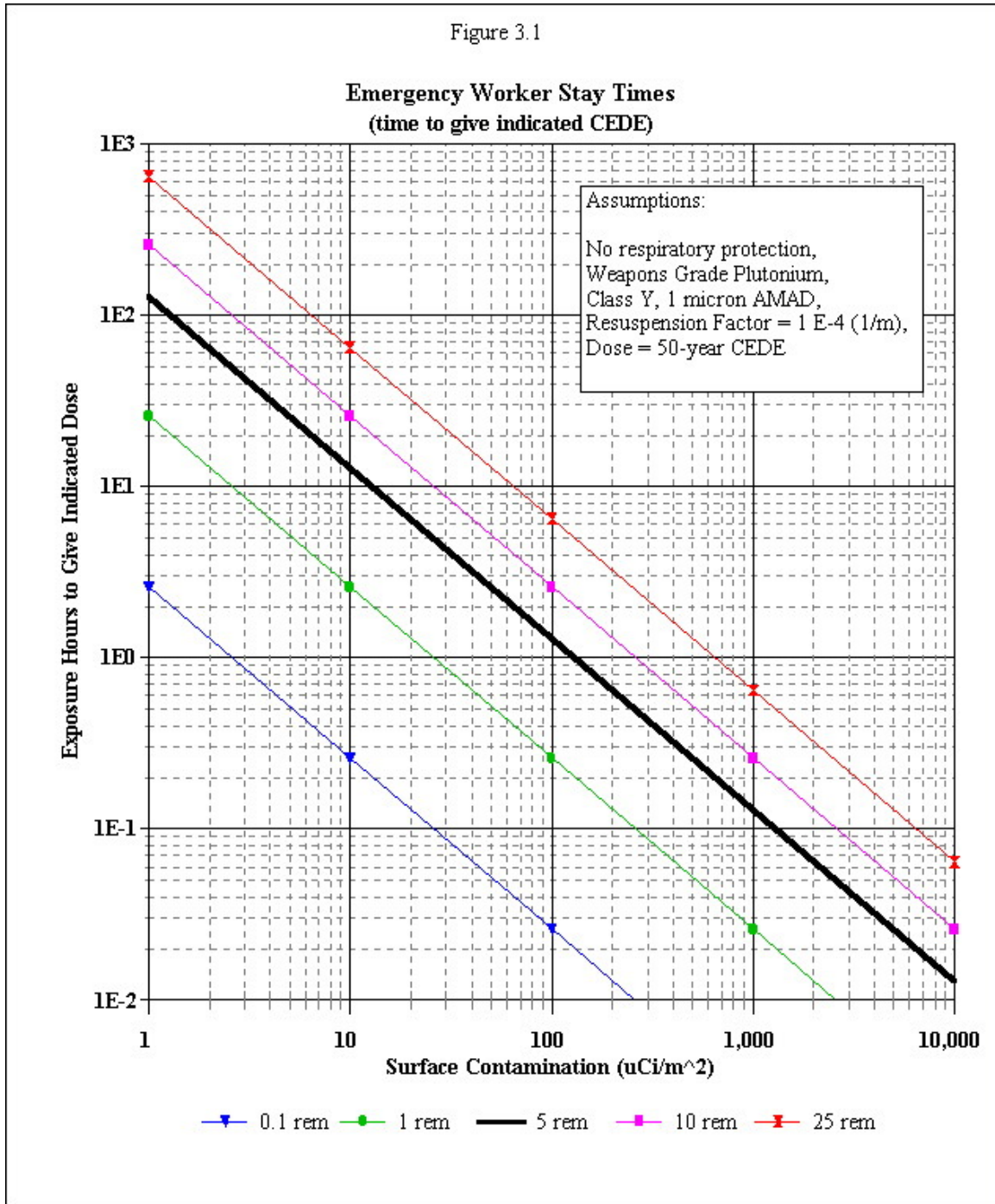
Assumes ICRP-30 Part 4 Metabolic Models and Dose Conversion Factors, breathing rate = 1.2m^3/hr.

Method M.V3.3.5   Conversion of Alpha Survey Instrument Readings (cpm) to Areal
                            Contamination (μCi/m^2)

Purpose: Convert counts per minute (cpm) read on an alpha survey instrument to surface contamination concentration in microcuries per square meter (μCi/m^2).

Discussion: Various types of survey meters differ in efficiency and detector area. The count rate measured is not the deposition concentration. Corrections must be applied.

Conversions for the following alpha survey meters are addressed by this method:

- AN/PDR-56

- AN/PDR-300

- PAC-1S

- Blue Alpha

Steps:

Use Chart V3.3.5 to convert cpm to μCi/m^2 as follows. First, on the X-axis (horizontal axis), move RIGHT to find the alpha survey instrument reading in cpm. Move UP to the line representing the alpha survey instrument used to make the measurement. Move LEFT to the Y-axis (vertical axis) to read the surface contamination concentration in μCi/m^2. Consider potential sources of error and necessary modifying factors:

- Falsely low readings from presence of moisture

- Surface self-absorption/shielding factors which should be applied for rough surfaces

- Detector more than 1 cm from surface

Chart V3.3.5   Conversion of Alpha Survey Instrument Readings (cpm) to Areal Contamination
(μCi/m^2)

Figure 3.2

**Conversion of Alpha Survey Readings
to Areal Contamination (uCi/m^2)**

Assumptions:

No losses due to moisture, dirt, etc.

INSTRUMENT	cpm/dpm	Probe area (cm^2)
AN/PDR-56	0.50	17
PAC-1S	0.50	60
AN/PDR-300	0.50	92
Blue ALpha	0.50	100
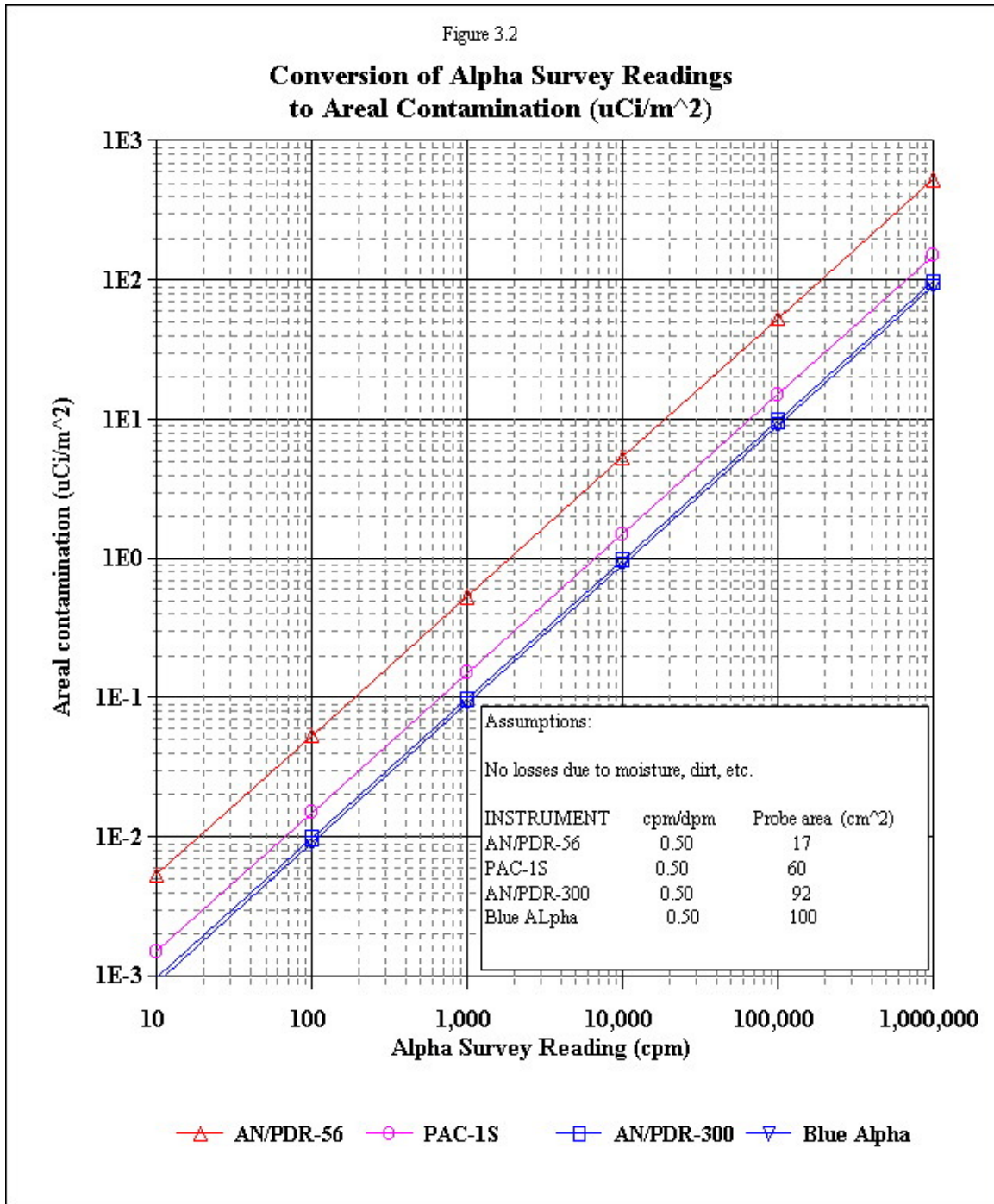
Method M.V3.3.6   Downwind Dose Rate from Resuspended WGPu (100-m Radius Source Area, 1 μCi/m^2, RF=1.0 $\times$ 10^{-4})

Purpose: Assess deposition measurements or predictions to estimate downwind dose rate due to resuspension near the accident site.

Discussion: Resuspended contamination from the very high concentrations of WGPu near the accident site (<100 m) can be carried downwind some distance creating a new inhalation hazard (plume). This can even contaminate previously uncontaminated areas.

The curves of Chart V3.3.6 are used to estimate the rate of accumulation of dose (rem of CEDE/hour of exposure) received at various distances downwind (plume centerline) from resuspension of surface contamination from a 100-m radius area contaminated to a level of 1 μCi/m^2 for a wind speed of 2 m/s.

Steps:

Use the curves of Chart V3.3.6 to estimate downwind CEDE due to inhalation. First, on the X-axis (horizontal axis), move RIGHT to find the desired downwind (plume centerline) distance (in kilometers). Move LEFT to the Y-axis (vertical axis) to read the rate of accumulation of dose(CEDE) per hour of exposure. MULTIPLY this dose by the actual surface contamination level in μCi/m^2.

Chart V3.3.6   Downwind Dose Rate from Resuspended WGPu (100-m Radius Source Area, 1 μCi/m^2, Resuspension Factor (RF) =1.0 x 10^{-4})
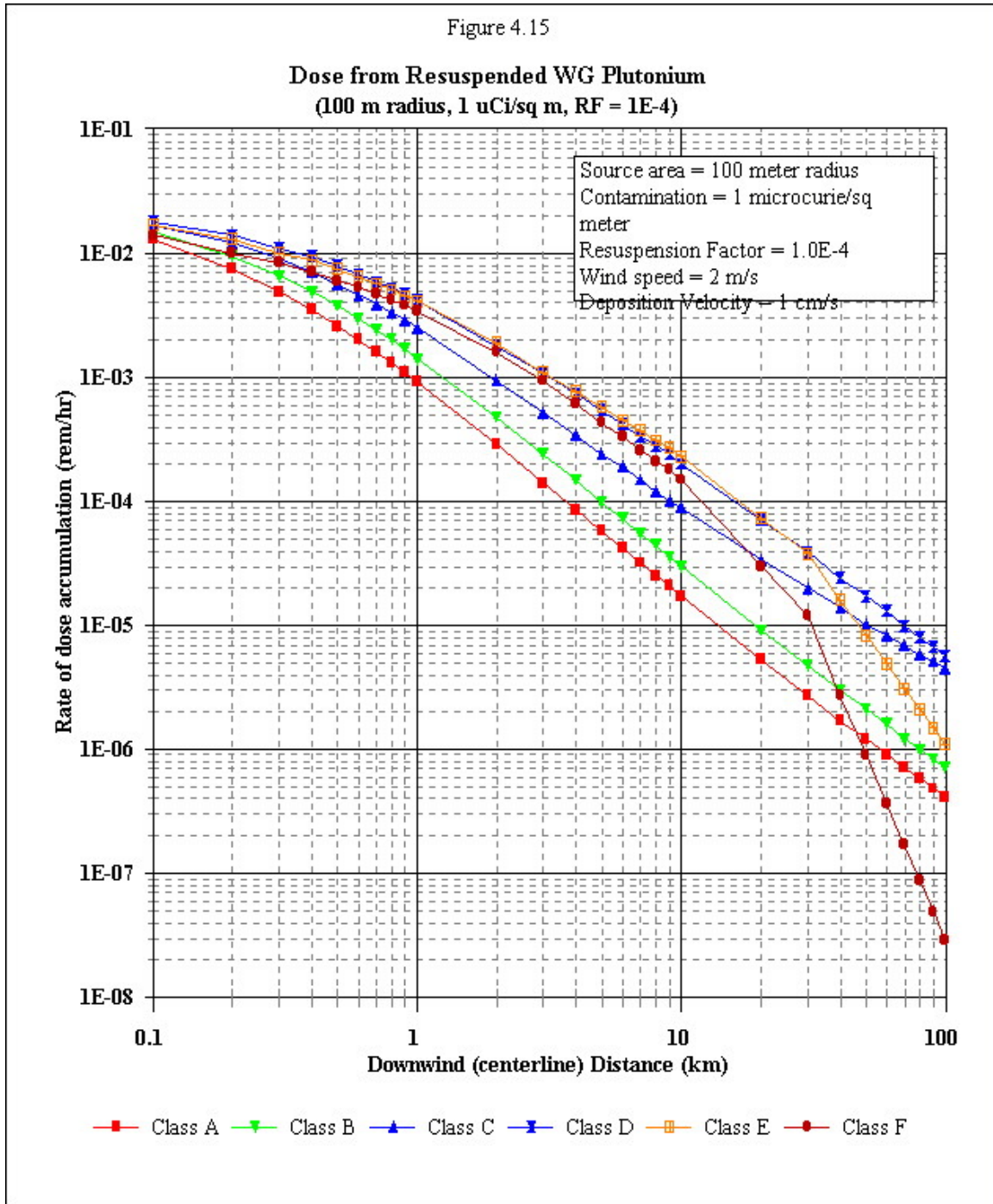
Figure 4.15

Dose from Resuspended WG Plutonium
(100 m radius, 1 uCi/sq m, RF = 1E-4)

Source area = 100-m radius, wind speed = 2 m/s, deposition velocity =1 cm/s. Assumes ICRP-30 Part 4 Metabolic Models and Dose Conversion Factors

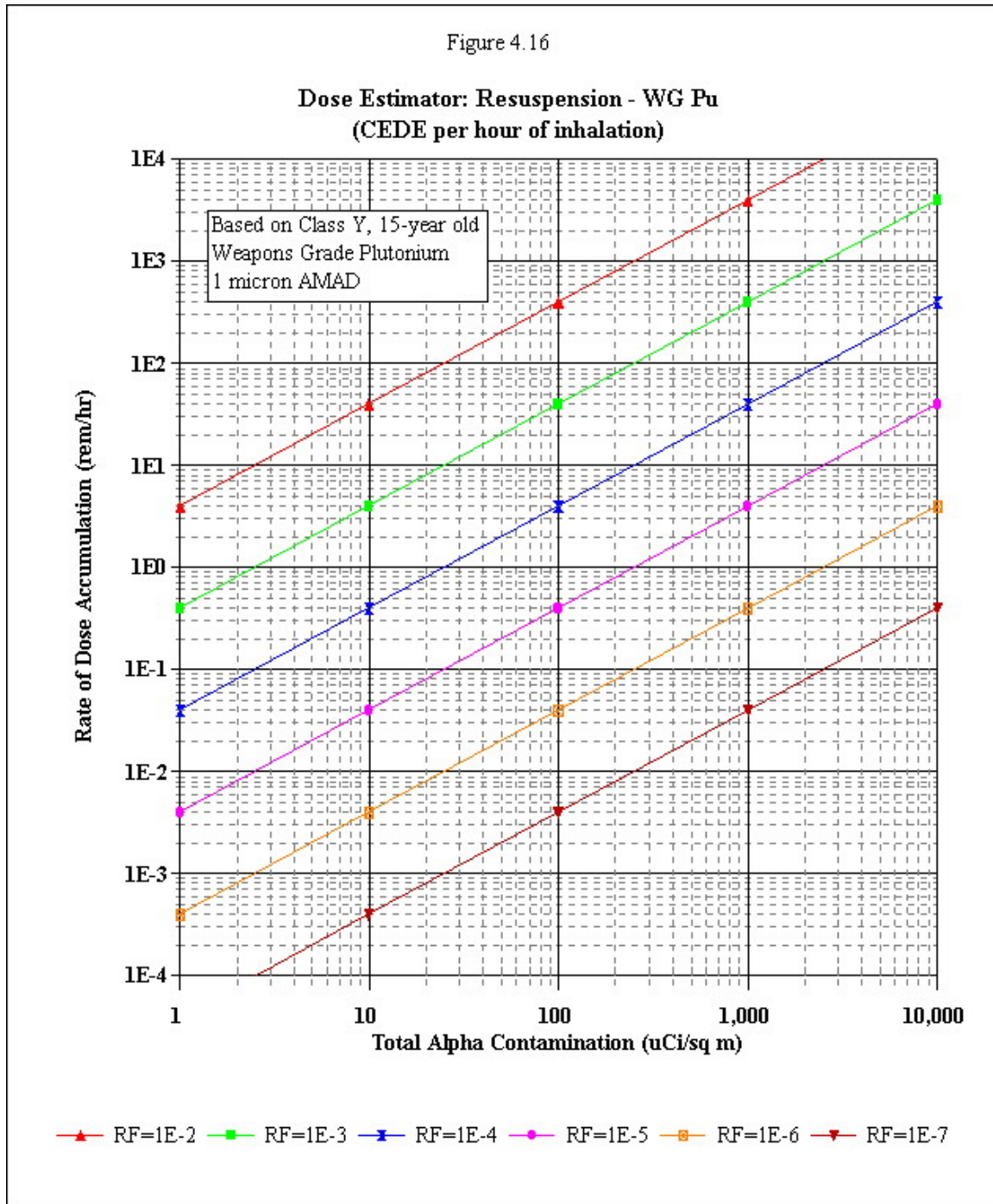Method M.V3.3.7   Rate of Dose Accumulation from Local Resuspension of Weapons-Grade
                    Plutonium, 15-year old, Class Y Pu)

Purpose: Assess deposition measurements or predictions to estimate dose rate due to resuspension of local contamination.

Discussion: Resuspension of WGPu in the immediate area creates the familiar resuspension inhalation hazard. The curves of Chart V3.3.7 may be used to make an approximate estimate of the rate of accumulation of dose (rem of CEDE per hour of exposure) in the local area of resuspended weapons-grade plutonium surface contamination for selected resuspension factors.

Steps:

Use the curves of Chart V3.3.7 to estimate the CEDE dose rate due to resuspension of contamination in the area. On the X-axis (horizontal axis), move RIGHT to find the surface contamination level (total alpha from weapons-grade Pu) in $\mu Ci/m^2$. Move UP to the curve representing the assumed resuspension factor. Move LEFT to the Y-axis (vertical axis) to read the rate of accumulation of dose (CEDE) per hour of inhalation.

Chart V3.3.7   Rate of Dose Accumulation from Local Resuspension of Weapons-Grade
Plutonium (15-year old, Class Y Pu)

Figure 4.16

Dose Estimator: Resuspension - WG Pu
(CEDE per hour of inhalation)

Based on Class Y, 15-year old
Weapons Grade Plutonium
1 micron AMAD

Source area = 100-m radius, wind speed = 2 m/s, deposition velocity =1 cm/s.
Assumes ICRP-30 Part 4 Metabolic Models and Dose Conversion Factors

Method M.V3.3.8   Dose from Breathing Contaminated Air (Weapons-Grade Plutonium and
                            Uranium)

Purpose: Assess air samples to calculate dose rate due to inhalation of air contaminated with
weapons-grade plutonium or uranium. (Note that since the curve for uranium is based on activity
inhaled it is essentially independent of the degree of enrichment of the uranium.)

Discussion: Air samples are used to determine the current dose rate. This is particularly
important for worker protection. Only the inhalation CEDE is important.

Steps:

Use Chart V3.3.8 to estimate the rate of dose accumulation from exposure to air contaminated
with either weapons-grade plutonium or uranium (any enrichment) to a specific concentration.

First, on the X-axis (horizontal axis), move RIGHT to find the known or assumed airborne
activity concentration in microcuries per cubic centimeter (μCi/cc), or the equivalent, in curies
per cubic meter (Ci/m^3). Move UP to the line representing either weapons-grade plutonium or
uranium. Move LEFT to the Y-axis (vertical axis) to read the rate of accumulation of dose at that
airborne concentration, in rem (CEDE) per *hour* of exposure.

Chart V3.3.8. Dose from Breathing Contaminated Air (WGPu and Uranium)
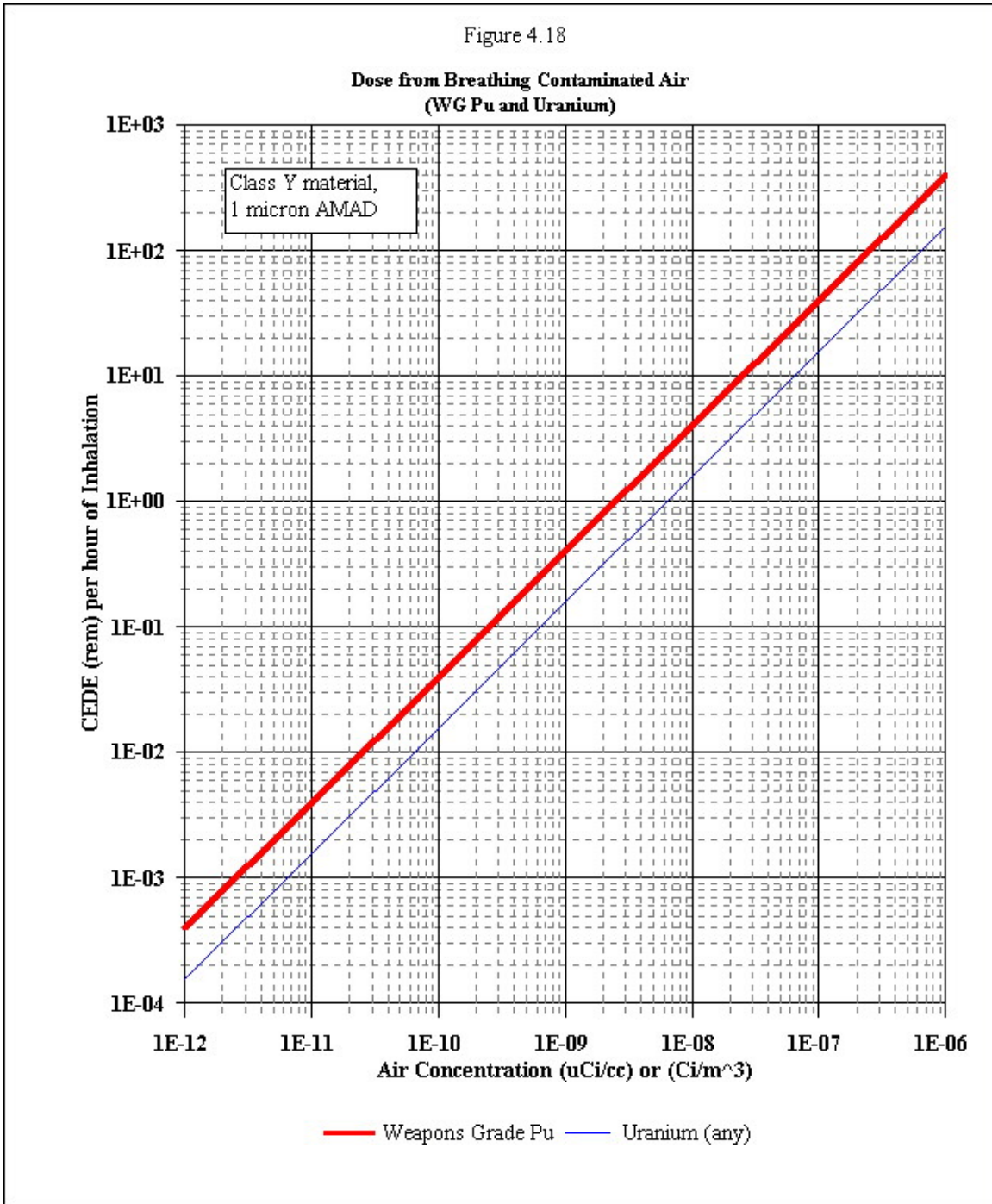
Assumes ICRP-30 models, 1 micron AMAD

Method M.V3.3.9   Total Alpha Derived Response Levels (DRLs) for Early Phase Evacuation or
                    Sheltering

Purpose: Assess deposition measurements and predictions in terms of the Early Phase PAG
using the DRL for total alpha.

Discussion: Evacuation and sheltering decisions are based on the Early Phase PAG.
Measurements and predictions of total alpha deposition can be assessed using the Early Phase
DRL for a total alpha marker ($DRL_{EP,k}$). The curves of Chart V3.3.9 show both W and Y lung
clearance classes. Class W is typically used. The chart also shows DRLs for three different
resuspension factors:

> 1E-4   "Arid Conditions"

> 1E-5

> 1E-6   "Non-Arid Conditions"

These resuspension factors are assumed to be a constant for the 96-hour exposure time. The
curves do not reflect the FRMAC time-dependent resuspension model (NCRP99). The NCRP99
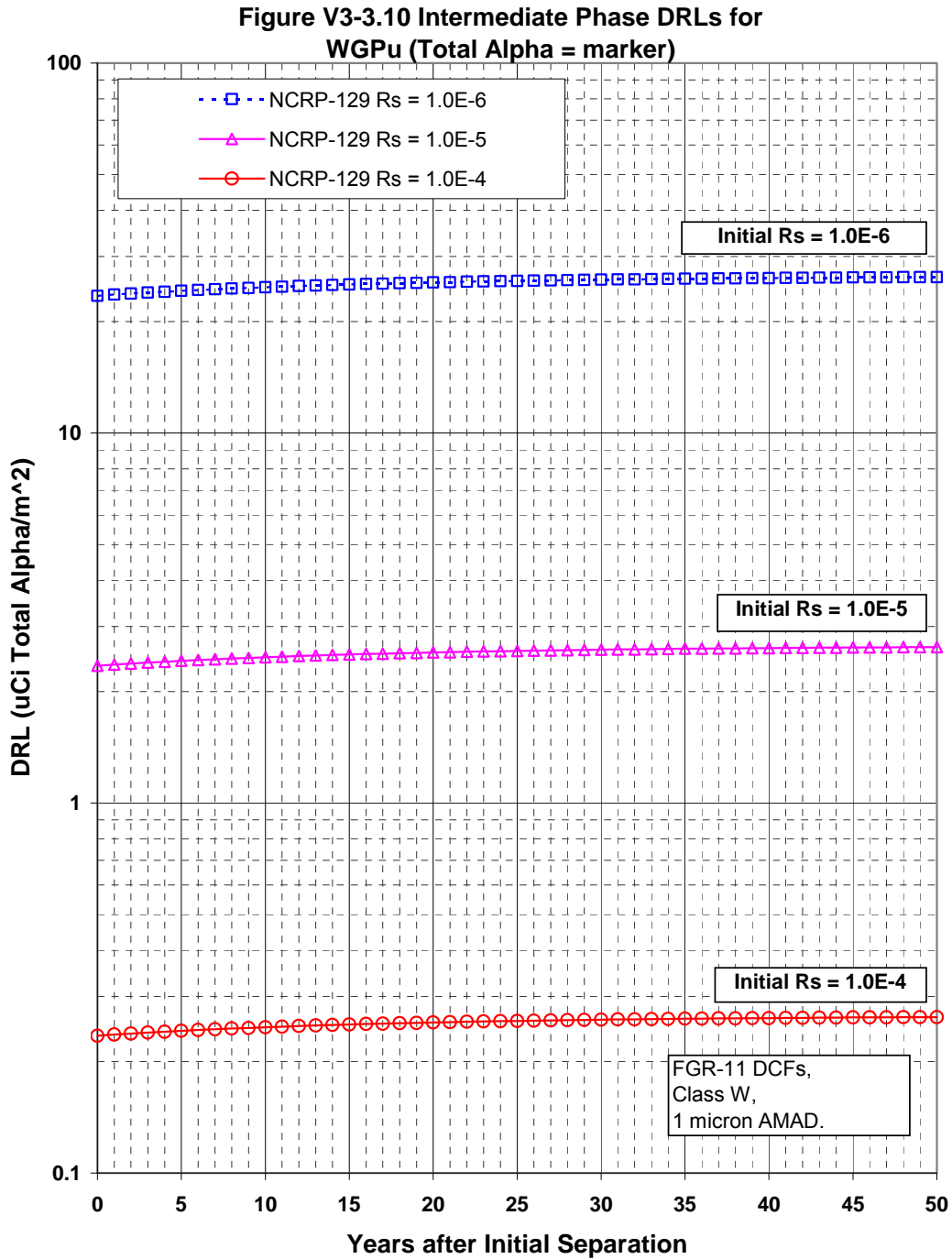time dependence reduces the DRLs reported here by a factor of 0.52.

Steps:

Use Chart V3.3.9 to calculate the surface contamination DRL of total alpha (used as a marker for
WGPu) that corresponds to the Early Phase Evacuation/Sheltering PAG of 1 rem, CEDE.

On the X-axis (horizontal axis) move RIGHT to find the known or assumed age (years after
separation) of the mixture of weapons-grade plutonium. Move UP to the line representing the
desired environmental conditions (non-Arid, Arid, or mid-way) and appropriate inhalation
clearance class (Class W or Class Y). Move LEFT to the Y-axis (vertical axis) to read the surface
contamination concentration in microcuries of total alpha per square meter ($\mu Ci/m^2$) that
corresponds to the Early Phase PAG of 1 rem.

Use this figure if the DRLs in terms of [241]Am concentration are required. Use Method M.3.3.2 to
determine the corresponding Am-Pu ratio from Chart V3.3.2.

Chart V3.3.9   Total Alpha Derived Response Levels (DRLs) for Early Phase Evacuation
or Sheltering

Figure V3-3.9 Early Phase DRLs: WG-Pu - Total Alpha
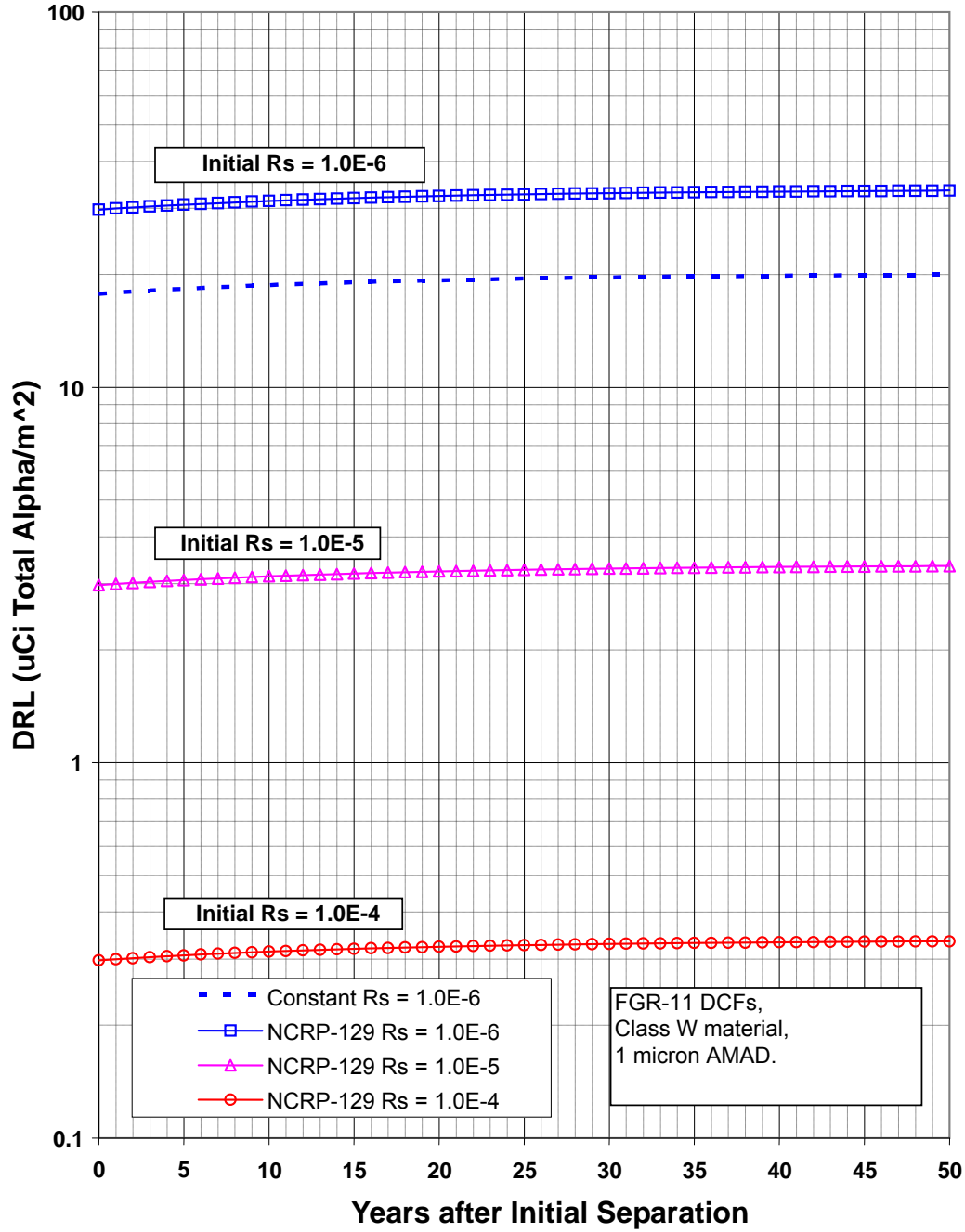
Method M.V3.3.10 Total Alpha Derived Response Levels for Intermediate Phase Relocation

Purpose: Assess surface contamination in terms of the Intermediate Phase Relocation PAG using the DRL for total alpha (Pu+Am) as a marker for WGPu.

Discussion: The initial resuspension factor is assumed to be $1.0 \times 10^{-4} \mathrm{m}^{-1}$. Exposure time period = 1 year of continuous exposure. Relocation PAG for doses from 1st year of exposure is 2 rem. In most accident scenarios, inhaled material would be expected to behave as Class Y material Note, however that considering the other inaccuracies involved in projection of doses, the differences between the Class W and Class Y curves are not significant.

Steps:

Use Chart V3.3.10 to calculate the surface contamination DRL of total alpha (used as a marker for WGPu) that corresponds to the Intermediate Phase Relocation PAG.

On the X-axis (horizontal axis), move RIGHT to find the known or assumed age (years after separation) of the mixture of weapons-grade plutonium. Move UP to the line representing the desired environmental conditions (non-Arid or Arid) and appropriate inhalation clearance class (Class W or Class Y). Move LEFT to the Y-axis (vertical axis) to read the surface contamination concentration in microcuries of total alpha per square meter (μCi/m^2) that corresponds to the 1st year relocation PAG of 2 rem.

Use this figure if the DRLs in terms of Am-241 concentration are required. Use Method M.3.3.2 to determine the corresponding Am-Pu ratio from Chart V3.3.2.

Chart V3.3.10. Total Alpha Derived Response Levels for Intermediate Phase Relocation

Figure V3-3.10 Intermediate Phase DRLs for WGPu (Total Alpha = marker)

- - □ - - NCRP-129 Rs = 1.0E-6
———△——— NCRP-129 Rs = 1.0E-5
———○——— NCRP-129 Rs = 1.0E-4

Initial Rs = 1.0E-6

Initial Rs = 1.0E-5

Initial Rs = 1.0E-4

FGR-11 DCFs,
Class W,
1 micron AMAD.

DRL (uCi Total Alpha/m^2)

Years after Initial Separation

Assumes ICRP-30 models and 1 micron AMAD

Method M.V3.3.11   Total Alpha Derived Response Levels for Long-Term Objectives

Purpose: Assess surface contamination in terms of the 2nd year and 50-year Long-Term Objectives using the DRL for total alpha (Pu+Am) as a marker for WGPu..

Discussion: The Long-Term Objectives of 500 mrem during the second year and 5,000 mrem over 50 years may be more restrictive than the Relocation PAG. Areas in which the Long-Term Objectives are exceeded, but the Relocation PAG is not exceeded, should be communicated to decision makers to highlight needs for special actions, such as mitigation measures or limitations on return. These charts use only the FRMAC resuspension model.

Steps:

Use Chart V3.3.11a (2nd year) and Chart V3.3.11b (50 year) to calculate the surface contamination DRL of total alpha (used as a marker for WGPu) that corresponds to both the 2nd year and 50-year Long-Term Objectives.

On the X-axis (horizontal axis), move RIGHT to find the known or assumed age (years after separation) of the mixture of weapons-grade plutonium. Move UP to the line representing the assumed resuspension factor for either the 2nd year or 50-year Long-Term Objective (Class W is assumed). Move LEFT to the Y-axis (vertical axis) to read the surface contamination concentration in $\mu Ci/m^2$ that corresponds to the 2nd year, or 50-year Long-Term Objective.

Use this figure if the DRLs in terms of ^{241}Am concentration are required. Use Method M.V3.3.2 to determine the corresponding Am-Pu ratio from Chart V3.3.2.

Chart V3.3.11a and V3.3.11b. Total Alpha Derived Response Levels for 2nd Year and 50 Year Long Term Objective

Figure V3-3.11a 2nd-year DRLs for WGPu (Total Alpha = marker)

Figure V3-3.11b 50-year Relocation DRLs for WGPu (Total Alpha = marker)
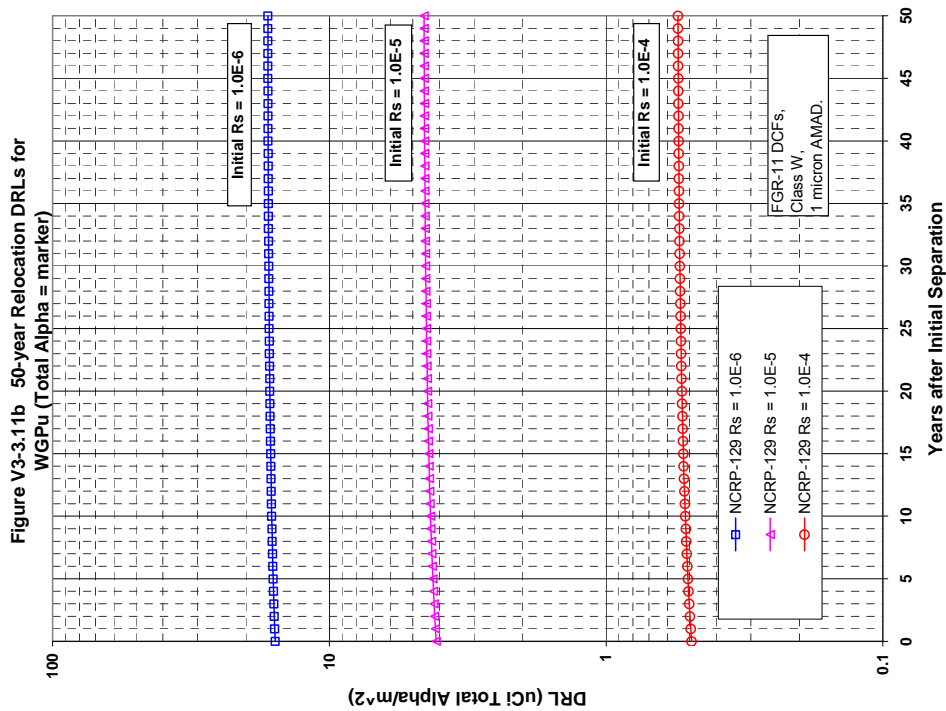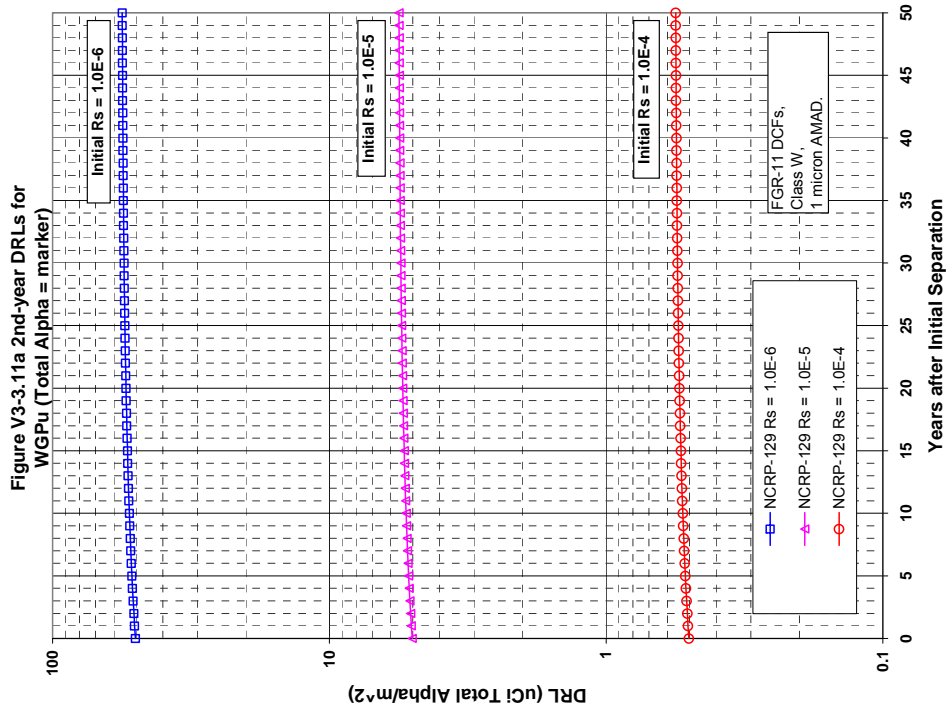
Method M.V3.3.12   Decontamination Effectiveness for Nuclear Weapons Contamination

Purpose: Assess the potential effectiveness of decontamination measures for various methods and surfaces.

Discussion: Because the Long-Term Objectives may be more restrictive than the Relocation PAG, use of mitigation measures may be needed to eliminate this complication. The following tables summarize expected performance.

Steps:

Table V3.3.12a estimates the effectiveness on particular surface types.

Tables V3.3.12b and V3.3.12c estimate the effectiveness in temperate and cold weather (snowy).

Table V3.3.12a. Decontamination Effectiveness, Material Types

Material	1[a]	2[b]	3[c]	4[d]	5[e]
Glass	98	98	100	100	97
Painted wood	99	98	99	100	91
Asphalt	72	92	98	92	22
Concrete	74	98	96	100	21
Unpainted wood	36	85	99	99	85

Source: IAEA74

[a] Vacuum method
[b] High pressure water
[c] High pressure water plus detergent
[d] Sandblasting
[e] Steam cleaning

Table V3.3.12b. Decontamination Effectiveness, Temperate Conditions

Method	Rate (100 ft²/hr)	DF
1. Roofs		
a. Firehosing		
(1) Composition shingle	12 - 60	10 - 35
(2) Tar and gravel	7 - 35	8 - 100
b. Firehosing and scrubbing		
(1) Tar and gravel	5	50
(2) Composition shingle	5	50
(3) Wood shingle	5	10
(4) Corrugated sheet metal	5	100
2. Paved areas		
a. Motorized flushers	100 - 300	25 – 50
b. Street sweepers	25 - 100	6 - 25
c. Vacuumized sweepers	25 - 100	4 - 50
d. Firehosing	5 - 25	15 - 50
3. Unpaved land areas		
a. Grading (few inches)	60	15
b. Ploughing	25	5
c. Scraping (several inches)	9	50
d. Bulldozing (several inches)	8.5	15
e. Filling		
(1) 6 inches of fill	4	7
(2) 12 inches of fill	2	50

Source: IAEA74

Table V3.3.12c. Decontamination Effectiveness, Cold Weather Conditions

Method	Rate (100 ft²/hr)	DF
1. Bare, sloped, asphalt shingles		
a. Firehosing (lobbing)		
(1) (25°F)	8	3
(2) (0°F)	8	2
2. Undisturbed snow		
a. Snow plough (blade)	330	10-35
b. Grading	125	2-20
c. Scraping	72	5-7
d. Snow plough (rotary)	53	7-50
3. Packed snow		
a. Grading (0-30°F)	70	5-6
b. Mechanical sweeping (below 20°F)	60	15
c. Vacuum sweeping (10 to 30°F)	30	6
d. Firehosing(30°F)	13	10
(1) (30°F)	13	10
(2) (15°F)	13	5
(3) (0°F)	13	4
4. Paved areas		
a. Mechanical sweeping	65	15
b. Firehosing (0°F)	20	10-15
5. Bare, frozen ground		
a. Mechanical sweeping	70	10-35
b. Vacuum sweeping	70	8
c. Firehosing	20	2

Source:IAEA74

SECTION 4. AGED FISSION PRODUCT ACCIDENT

TABLES

AGED FISSION PRODUCT ACCIDENT

4.1   Description of Generic Aged Fission Product Accident

The potential radiological consequences of aged fission product accidents range from insignificant to life threatening. Typical accident scenarios involve spent fuel storage pools, dry spent fuel storage facilities, fuel reprocessing facilities and associated liquid waste storage, transportation of spent fuel between these facilities, and transportation of work-generated solid waste to storage facilities. Mechanisms for release of these aged fission products can be either by accident or by the deliberate actions of one or more persons. The following scenarios are focused on spent reactor fuel, less than 10 percent ^{235}U enrichment, which has cooled for a minimum of 100 days. If the fuel has cooled for less than 100 days, shorter-lived radionuclides will be present and the consequences will be similar to a power reactor accident.

4.1.1   Scenario Description

As indicated above, the potential scenarios can be numerous, but there are bounding scenarios that can be tailored to fit specific events.

When nuclear fuel has reached the end of its useful life in a reactor, which could be in commercial electrical generation, research and development, or national defense programs, the spent fuel is discharged to a water-filled storage basin. The storage basin allows for the fuel to be thermally cooled while it is radiologically shielded from the employees. While the fuel is thermally cooling, there is a potential that the cooling could be affected by the loss of cooling water from the basin or the removal of the fuel from the basin. If the residual heat in the fuel is sufficient to melt the cladding around the fuel, then fission products, uranium, and some transuranic radionuclides could be released to the environment in a plume that would affect downwind areas.

Once the fuel has cooled sufficiently so that the residual heat will not melt the cladding, some fuel is stored in on-site dry fuel storage containers that provide radiological shielding and sufficient cooling for the residual heat to continue to safely dissipate. Fuel in these containers has typically aged for several years prior to being placed in the container; only the longer-lived radionuclides (half-lives greater than 1 year) are present in significant quantities. Release of radionuclides to the environment in sufficient quantity to present a hazard to the public would require a deliberate action on the part of an individual(s) to expose the fuel to the public. An explosive device that would breach the shielding container and spread radioactive particles in the environment is a likely scenario.

Transportation of nuclear fuel for reprocessing or dry storage is carried out in U.S. Department of Transportation (DOT) -approved shipping containers. Typically these shipments are made on exclusive-use trucks but may be transported by railcar. Movements are coordinated with Federal and state governments over controlled routes. An accident that could cause a breach in shielding and dispersal of radioactive material is unlikely. The deliberate actions of an individual(s) could result in the spread of radioactive materials. The radiological consequences would be comparable to the two previous scenarios.

While there is currently no reprocessing of irradiated fuel from commercial nuclear power facilities, the Federal government is in the process of stabilizing, through chemical processing, various materials including reactor fuel from research and development reactors and defense program reactors to safeguard existing inventories of fissile and fissionable materials. In general, the chemical separation process involves the dissolution of the fuel into an acidic solution. The resulting solution contains uranium, transuranic radionuclides, and associated fission products. The uranium and transuranic radionuclides are separated from the fission products through a process of solvent extraction. The extracted acidic solution, which contains the waste fission products, is chemically neutralized and transferred to remote storage tanks for final processing and disposition. The uranium is chemically separated from the transuranic materials and either recycled or stored for other uses. The separated transuranic materials receive additional treatment within the reprocessing facility prior to final stabilization and storage.

The following example illustrates the importance that transuranics can play in a scenario where inhalation during the plume phase may occur. Transuranics can represent a significant internal dose concern even at very low mass concentrations because of their higher specific activities (compared to uranium radionuclides). For a moderately soluble mixture, if ^{239}Pu contamination contributes 0.1% of the total alpha activity in uranium, then it will contribute roughly 14% of the total inhalation dose equivalent. In terms of mass, the 0.1% ^{239}Pu activity fraction corresponds to 11 parts ^{239}Pu per billion parts natural uranium.

Various accident scenarios associated with the reprocessing facilities have been assessed for impact on the public. These scenarios include loss of containment (leaks or inadvertent transfers), explosion (chemical reaction or hydrogen-generation), fire (resins or organics), inadvertent criticality, and natural phenomena (earthquake, high winds and tornado). Reprocessing facilities are designed with both active and passive features to minimize the effects of these types of events.

The final scenario involves the movement of work-related waste from the reactor or reprocessing facility to the waste disposal site. In the operation and maintenance of these facilities, waste (protective clothing, wipes, paper, tools, old equipment, etc.) is generated that is contaminated with the neutron-activated components of the plant and fission products, uranium, and transuranic materials associated with the fuel. These materials are transported in DOT-approved shipping containers to licensed nuclear waste storage facilities for disposal. The accident scenarios are comparable to the shipment of spent nuclear fuel, but the Curie content of the shipment is significantly smaller.

The DOE has begun to ship work-waste materials from its transuranic facilities to the Waste Isolation Pilot Plant in New Mexico. These shipments contain higher levels of transuranic nuclides than occur in reactor and reprocessing facility wastes.

4.1.1.1   Data Quality Objectives (Step 1 – State the Problem)

The first step in any decision making process is to define the problem that has initiated the assessment. As described in previous sections, some of the activities involved in step 1 of the DQO process, such as identifying members of the planning team, identifying the primary decision

maker of the planning team, and defining each member's role and responsibilities have been addressed. Given the range of scenarios possible for an aged fission product event, the activity of developing a concise description of the problem is of major importance. Activities that may be helpful during this phase of DQO development include describing the conditions or circumstances that are causing the event, summarizing existing information, indicating the source and reliability of the information, and breaking the problem into more manageable pieces. The Assessment Group will assign priorities to and logical relationships among the pieces of the problem.

4.1.2   Key Radiological Issues

Trying to predict the radionuclide mixture and relative abundance of aged fission products in a specific event can be difficult. Although fission product yields are well known, their presence and relative contribution to the released material that reaches the environment can be modified by many factors. In addition to the expected fission products derived from fission product yield calculations, activation products from reactor components may be present. The relative abundance of fission products can change due to the chemical environment of the release (*e.g.*, a release in an aqueous environment may deplete radionuclides such as cesium, strontium and ruthenium because of their solubility). The presence and relative abundance of uranium and transuranic radionuclides can also change significantly based on variables such as fuel type and ^{235}U enrichment, power levels, length of irradiation, time since discharge, and stage of reprocessing.

Given those considerations, until source term data are available from the facility where the event occurred or laboratory analyses are available on samples collected from the environment, the calculated fission product yield provides a valid basis for initiating the dose assessment process. Refer to Table V3.4.1 below.

Table V3.4.1.   Activity of Fission Products in Curies at Specified Times (*T*) after Removal from a Reactor That Has Operated at 1000 kW for 1 Year

Fission Product	$T = 0^{(a)}$	$T = 100$ days	$T = 1$ year	$T = 5$ years
^{85}Kr	191	187	177	132
^{86}Rb	0.26	-	-	-
^{89}Sr	38,200	10,300	321	-
^{90}Sr	1,430	1,420	1,380	1,200
^{90}Y	1,430	1,420	1,380	1,200
^{91}Y	48,900	14,500	577	-
^{95}Zr	49,200	17,000	1,000	-
^{95}Nb (90H)$^{(b)}$	687	152	15	-
^{95}Nb (35D)$^{(b)}$	48,200	28,700	2,140	-
^{103}Ru	30,900	5,920	74	-
^{103}Rh $^{(b)}$	30,900	5,920	74	-
^{106}Ru	2,180	1,800	1,090	68
^{106}Rh $^{(b)}$	2,180	1,800	1,090	68
	151	-	-	-

Fission Product	$T = 0^{(a)}$	$T = 100$ days	$T = 1$ year	$T = 5$ years
^{111}Ag	5.9	1.2	-	-
^{115}Cd	84	0.7	-	-
^{117}Sn	<64	<48	<23	<0.4
^{119}Sn	9	5	1	-
^{123}Sn	101	0.1	-	-
^{125}Sn	43	41	34	12
^{125}Sb $^{(b)}$	34	39	36	13
^{125}Te $^{(b)}$	787	-	-	-
^{127}Sb	260	123	16	-
^{127}Te (90D)$^{(b)}$	922	124	16	-
^{127}Te (9.3H)$^{(b)}$	1,590	182	0.6	-
^{129}Te (32D)	1,590	182	0.6	-
^{129}Te (70M)$^{(b)}$	25,200	4	-	-
^{131}I	252	1.5	-	-
^{131}Xe $^{(b)}$	36,900	-	-	-
^{132}Te	36,900	-	-	-
^{132}I $^{(b)}$	55,300	-	-	-
^{133}Xe	52	0.3	-	-
^{136}Cs	1,080	1,070	1,060	970
^{137}Cs	1,030	1,020	1,010	920
^{137m}Ba $^{(b)}$	51,700	230	-	-
^{140}Ba	51,700	265	-	-
^{140}La $^{(b)}$	47,800	4,740	10	-
^{141}Ce	45,300	288	-	-
^{143}Pr	26,700	20,800	10,700	268
^{144}Ce	26,700	20,800	10,700	268
^{144}Pr $^{(b)}$	21,800	40	-	-
^{147}Nd	4,900	4,800	3,950	1,360
^{147}Pm $^{(b)}$	37	37	36.6	35
^{151}Sm	74	67	52	13
^{155}Eu	109	1.2	-	-
^{156}Eu				
Total	**693,573**	**144,130**	**36,964**	**6,527**

$^{(a)}$Calculated using fission product yields (PSH70).

$^{(b)}$Daughter Product

From (Ce83)

In addition to the fission products listed above, other references include ^{134}Cs and ^{154}Eu. There is the likelihood that activation products from the fuel cladding (*e.g.*, zircalloy, reactor system components, as well as fuel activation products, such as transuranics) will be present. Activation products from fuel cladding and reactor system components may include ^{51}Cr, ^{54}Mn, ^{55}Fe, ^{59}Fe, ^{57}Co, ^{58}Co, ^{60}Co, ^{63}Ni, ^{65}Zn, ^{94}Nb, ^{95}Zr, ^{95}Nb, ^{108m}Ag, ^{110m}Ag, ^{113}Sn, ^{124}Sb, ^{125}Sb, ^{152}Eu, ^{154}Eu, and ^{181}Hf. Activation products from the fuel will include ^{233}Pa, ^{237}Np, ^{239}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu,

^{241}Pu, ^{242}Pu, ^{241}Am, ^{243}Am, ^{242}Cm, and ^{244}Cm. The unused fuel contains ^{234}U, ^{235}U, ^{236}U, and ^{238}U.

4.1.2.1   Data Quality Objectives (Step 2 – Identify the Decision)

The goal of this step is to define the questions that the Assessment Group will attempt to resolve and identify the alternative actions that may be recommended based on that assessment. In most events, several recommendations are appropriate to address the event under assessment. In these instances, the Assessment Group will organize the recommendations in order of priority and identify the most logical and efficient sequence for analyzing and resolving them.

Based on a review of the problem stated in DQO Step 1, identify the principal problem and state it as specifically as possible. This narrows the search for information needed to address the problem. Identify the possible actions that may be taken to solve the problem. Once the principal problem is identified and actions identified to solve the problem, repeat the process for the next most significant problem until all problems are addressed or resources available to address the problems are fully utilized. Priorities may change as information is developed.

4.1.3   Inputs to Radiological Assessment

The complexity of the potential source terms makes the initial tasks of the Assessment Group very difficult. However, the Assessment Group can make a number of assumptions to provide predictions that will guide initial protective actions to minimize radiation exposure to the public. As more data on the source term become available, these predictions can be refined and provided to decision makers to guide modification of the protective action recommendations to suit the circumstances.

The affected facility or shipment owner will provide the initial source term information. This may be in the form of Safety Analysis Reports, Process Hazards Reviews, shipping manifests, engineering judgment, etc.; or through monitoring data from effluent release monitors or field monitoring teams. This information, in combination with information from the FRMAC monitoring teams, will provide the basis of initial protective action recommendations. For aged fission and activation products, emergency phase (plume phase) exposure to emergency responders and members of the public is through three pathways.

•   External exposure from gamma and beta-emitting radionuclides during plume passage, predominantly fission and activation products.

•   External exposure from gamma and beta-emitting radionuclides deposited on the ground and other surfaces during plume passage, predominantly fission and activation products. The dose rates from these radionuclides will remain relatively stable because the short-lived radionuclides are not present. Weathering actions will decrease exposure rates in some areas, while increasing them in areas that receive and concentrate runoff.

•   Internal exposure from the transuranic radionuclides that may be inhaled during plume passage or resuspended from contaminated surfaces.

Typical data available to the Assessment Group during the first one or two days will be direct radiation measurements of deposited radionuclides that will allow the plotting of isodose curves. If air-monitoring data are available, initial estimates of transuranic concentrations can be made. One conservative approach is to collect air samples in both the affected and unaffected areas to allow correction for radon-thoron daughters on the air samples, then assume that any excess radioactivity is due to the transuranic material (*e.g.*, ^{239}Pu). This initial conservatism can be refined as additional information is provided by the laboratories on the actual radionuclides present and the amount of activity each radionuclide contributes to the total. The immediate priority of the Assessment Group will be to identify the actual types and amounts of fission, activation, fuel, and transuranic radionuclides present in the environment.

4.1.3.1 Protective Action Guides

EMERGENCY WORKER LIMITS

The EPA published guidance for workers performing emergency services. Federal responders (*e.g.*, FRMAC staff) are emergency workers, and their doses must be immediately and continually assessed to ensure that their doses do not exceed the EPA limits summarized in EPA's Table 2-2, "Guidance on Dose Limits for Workers Performing Emergency Services" (EPA92). These limits apply to doses incurred over the duration of the emergency. All doses (external and inhalation) received during an emergency are included in the limit. The difficulty is in estimating the inhalation dose during the Early Phase. During the Early Phase, the inhalation dose can be estimated, but later it must be confirmed by whole-body counting, chest counting, bioassay samples, or other means.

EARLY PHASE PAGs

The Early Phase begins before a major release and lasts until the risk of a major release has ended and the areas with major contamination have been identified. During this phase, early health effects and the Early Phase PAGs are the principal focus.

EPA established Early Phase PAGs which are projected doses indicating when immediate protective actions are warranted (EPA92). EPA's Table 2-1, "PAGs for the Early Phase of a Nuclear Incident", summarizes the Early Phase PAGs. The EPA PAG for evacuation ranges from 1,000 to 5,000 mrem TEDE. The upper limit (5,000 mrem) applies only for conditions that increase the risk of an evacuation. Therefore, for most conditions, evacuation is recommended at a 1,000-mrem projected dose. TEDE includes the EDE from cloud passage, four days of EDE from ground shine, and the inhalation CEDE from exposure to the plume and four days of resuspension. EPA also provides additional guidance on thyroid and skin dose, which are typically not applicable to an aged fission product scenario.

INTERMEDIATE PHASE PAGs

The Intermediate Phase begins after the source of a major release is under control. Following a nuclear incident it may be necessary to temporarily relocate the public from areas where

extensive deposition has occurred until decontamination has taken place. The period is arbitrarily defined as the period beginning after the source and releases have been brought under control and environmental measurements are available for use as a basis for decisions on protective actions and extending until these protective actions are terminated. This phase may overlap the Early and Late Phases and may last from weeks to many months. For the purpose of dose projection, it is assumed to last for one year.

LATE PHASE

The Late Phase (also referred to as the Recovery Phase) is the period beginning when recovery action designed to reduce radiation levels in the environment to acceptable levels for unrestricted use are commenced, and ending when all recovery actions have been completed. This period may extend from months to years. For the purpose of dose projection, it is assumed to last for 50 years.

4.1.3.2   Dose Projections

Each phase of the emergency brings a different set of dose projections. These projections will initially be made with limited data and will be refined as additional data are developed. There are a number of dose projection products that are relatively well defined and will be made in almost all emergencies. These will be described in more detail. Other dose projection products will be specific to the emergency and to the needs of the customers of the FRMAC Assessment organization.

EMERGENCY WORKER LIMITS

The difficulty in making dose projections for emergency workers that may have to enter the affected area is due to the unknown transuranic portion of the source term. The ratio of transuranic radionuclides to fission product radionuclides is not constant from scenario to scenario. The simplest scenario would probably consist of a commercial reactor spent fuel element that is dispersed outdoors in a dry environment. This source term would be almost identical to the calculated inventory of the fuel tube based on reactor history and time since reactor shutdown. In this case the inhalation dose would probably not exceed the external dose. In this example, if the turn-back limit were 10 rem, then the emergency worker (no respirator) should turn back at a time such that the external exposure from plume passage and ground shine does not exceed 5 rem.

Other scenarios can result in situations that have significantly higher inhalation doses. For example, during reprocessing the initial extraction of the unused uranium from the fission product waste will contain most of the transuranic materials as well. An accident could release these materials (uranium and transuranics) from the reprocessing facility or from job waste associated with the process (*e.g.*, transuranic waste shipments).

Whenever possible, emergency workers should wear respiratory protection during the plume phase. After plume passage, it may be possible to relax the respirator requirements if the

transuranic component of the source term is low or efforts are made to minimize resuspension of contamination during sample collection.

EARLY PHASE

By the time the Assessment Group is operational, the most likely scenario for the Early Phase of the event is that plume phase of the event will have terminated and people that were downwind have been either evacuated or sheltered. Initial actions by the Assessment Group will be to determine the identity of the deposited radionuclides and provide dose projections for the external exposure from the deposited radionuclides and for the internal exposure from the resuspended radionuclides. These projections will be used to modify the boundaries of the area that has been evacuated and/or sheltered. Following actions to protect the affected population, efforts will be directed to assessing the dose (internal and external) that may have been received during plume passage. If significant internal doses are suspected, then recommendations for assessment of internal dose (*e.g.*, whole body counts, chest counts, bioassay samples) may be considered. Internal dose information will be provided to medical authorities for consideration of measures (*e.g.*, chelation) to reduce significant internal doses.

INTERMEDIATE PHASE

The Intermediate Phase is a period of extensive gathering of radiation measurements and environmental samples to fully understand the extent and impact of the event. This information is used to plan decontamination activities, to plan relocation from or the reintroduction of the public to the affected areas, and to plan agricultural and business practices in the affected area.

LATE PHASE

This phase looks at long-term trends in the affected area. For example, agricultural practices that were suspended in the first year may be allowed to resume in the second year. Surface-deposited radionuclides that contaminated crops in the first year will have been incorporated into the soil column in the second year. Uptake of specific radionuclides by specific crops may restrict the introduction of those crops into the food chain, but other crops may not be affected and may be allowed back into commerce. More crops may be allowed back into commerce in later years, as the radionuclides are diluted/decayed in the soil.

4.1.3.3 <u>Data Quality Objectives (Step 3 – Identify the Inputs to the Decision)</u>

The purpose of this step is to identify the informational inputs that will be required to resolve the decision statement and determine which inputs require environmental measurements. The DQO process identifies four activities that may be necessary to meet the objective.

Identify the information that will be required to resolve the decision statement. Determine which environmental variables or other information is needed to resolve the decision statement. Consider whether monitoring or modeling approaches, or a combination or both, will be used to

acquire the information. Based on the selected data acquisition approach, identify the types of information needed to support the decision statement. Ask general questions such as, "Is information on the source term radionuclides available?" or "Is information on the chemical species or particle size distribution available?" These types of questions and their answers help identify the information needs.

Determine the sources for each item of information identified above. Identify and list the sources for the information needed to resolve the decision statement. These sources may include results of Safety Analysis Reports, process hazard reviews, professional judgment, scientific literature, or sampling data.

Identify the information that is needed to establish the dose response level. Define the basis for setting the dose response level. In this step, simply determine the criteria that will be used to set the numerical value.

Confirm that appropriate measurement methods exist to provide the necessary data. Use the list of environmental measurements identified earlier in this step to develop a list of potentially appropriate measurement methods. Note the method detection limit for each potential method. For example, the detection limit for portable contamination survey instruments may not be sufficient for food pathway limits; sample collection and laboratory analysis may be required.

4.1.4   Boundary of Consideration

Physical Extent

Typically, three physical zones are considered in an event: the ejection radius, plume footprint, and relocation pathways. If the release is triggered by an energetic event such as an explosion, then contamination will be deposited in a radial pattern around the release point. If a portion of the release is injected into the atmosphere by an explosion, fire, exhaust system, or other mechanism, then the plume will deposit a footprint of radioactive material downwind from the release point. The size and shape of the footprint may be quite complex based on the atmospheric conditions, terrain, and physical features of the area.

In some cases, relocation of contaminated material from the ejection radius and plume footprint will warrant consideration. For example, contamination deposited in the plume footprint may become resuspended and blown to areas not impacted by the initial event. Water pathways also have to be considered. The release could be to a stream, river, or canal that would transport the contamination to populations that were not downwind of the release point. Rain could erode contamination from the footprint zone and contaminate surface water supplies.

The Assessment Group has to consider all the potentially impacted areas as it evaluates the impact of the event. Priority is typically given first to emergency workers, areas to be evacuated and/or sheltered, relocation areas, and ingestion pathways.

Temporal

Decisions on emergency worker exposure recommendations and evacuation/sheltering of affected populations have to be made as soon as possible. Monitoring and assessment efforts must focus on data supporting these decisions during the early portion of the emergency phase. Data for decisions such as control of food items may wait for several days until more critical needs have been met.

EMERGENCY AND INTERMEDIATE

The initial focus of the Assessment Group will be to establish the boundaries of the offsite public sector in which doses could exceed 1 rem, warranting evacuation or sheltering. The next area of focus is the intermediate zone for which relocation of the public is recommended. Depending on the nature of the event and the radionuclides involved, this area could be very large or could be as small as the emergency zone. Establishment of these boundaries will require direct radiation measurements and sample of air, water, soil, and vegetation.

UNTIL CONDITIONS CHANGE

The boundaries may not remain static and will have to be reassessed as conditions change. Typical conditions that can cause changes in exposure to the radionuclides include snowmelt; runoff of rain into ditches, streams, ponds, and rivers; trees leafing out in spring; and erosion by water and wind. Resumption of human activities such as plowing, cutting grass, driving, etc., can change exposure pathways and affect projected doses.

UNTIL SUFFICIENT KNOWLEDGE IS ACQUIRED

Initial assumptions and conservative measures incorporated into recommendations will have to be validated. For example, it may have been assumed that all gross alpha activity was ^{239}Pu for the purpose of calculating internal exposure dose. As the source term is characterized more fully, it may be determined that a significant portion of the gross alpha activity is due to uranium radionuclides instead of ^{239}Pu or other transuranic radionuclides. This could allow the internal dose portion of the projected dose to be substantially reduced. Boundaries of controlled areas could be reduced allowing normal activities to resume in areas previously under those controls.

Data Quality Objectives (Step 4 – Define the Boundaries of the Study)

This step defines the spatial and temporal boundaries of the problem. It is difficult to interpret data that have not been drawn from a well-defined population. The term "population" refers to the total collection or universe of objects or people to be studied, from which the samples will be drawn. The purpose of this step is to define spatial and temporal components of the population that will be covered by the decision statement so that the data can be easily interpreted. These components include:

- Spatial boundaries that define the physical area to be studied and from where the samples should be taken.

- Temporal boundaries that describe the time frame the study data will represent and when the samples should be taken.

The boundaries will be used to ensure that the data collection design incorporates the time periods in which the study should be implemented, areas that should be sampled, and the time period to which the study results should apply. This will help ensure that the study data are representative of the population being studied. Practical constraints that could interfere with sampling should also be identified in this step. A practical constraint, such as snow, is a hindrance or obstacle that could potentially interfere with the full implementation of the data collection design.

4.1.5   Decision Rules

The Assessment Group does not establish decision rules nor make protective action recommendations. However, published PAGs are used by the Assessment Group as decision rules for the interpretation of measurements and predictions. These PAGs are implemented as DRLs. If a measurement exceeds a specific DRL, then that location fails the test at hand. If a measurement falls short of a DRL, then it passes the test at hand.

Derived Response Levels have been defined for the following:

- Emergency Worker turn-back limits

- Evacuation based on EPA Early Phase PAG following plume passage

- Relocation (1st year), plus 2nd and 50-year long-term objectives

- Agricultural hold based on deposition concentrations

- Food condemnation (agricultural embargo) based on food concentration

- Water condemnation based on concentration

The PAGs and computational approach used by the Assessment Group may be altered upon direction of the Federal Advisory Team.

4.1.5.1 Data Quality Objectives (Step 5 – Develop a Decision Rule)

This step defines the parameter of interest, specifies the derived response level, and integrates previous DQO outputs into a single recommendation that describes a logical basis for choosing among alternative actions. The assessment process should specify the numerical value that causes a choice between alternative actions. For example, one action would be chosen if the true value of the parameter of interest is above $1 \, \mu Ci/m^2$, and a different action otherwise. Recognize that the parameter chosen in this step may be changed to an equivalent measure, as more information becomes available.

The Assessment Group should specify the parameter of interest (such as the mean, median, or percentile) for which one would like to know the true value and that the data will estimate.

4.1.6   Tolerance Limits

Assessors must establish tolerable levels of uncertainty when calculating DRLs. For example, evacuation, shelter, and agricultural product holds have a higher tolerance level than re-entry, which is higher yet than relocation, and so on. It is up to the assessor to establish these tolerable levels until a more definitive uncertainty analysis can be performed.

4.1.6.1 <u>Data Quality Objectives (Step 6 – Specify Tolerable Limits on Decision Errors)</u>

The Assessment Group is interested in knowing the true state of some feature of the environment. Since data can only estimate this state, decisions that are based on measurement data can be in error (decision error). Most of the time the correct decision will be made. However, the goal of the Assessment Group is to develop a data collection design that reduces to a tolerable level the chance of making a decision error.

There are two reasons why the Assessment Group cannot know the true value of a population parameter:

- The population of interest almost always varies over time and space. Limited sampling will miss some features of this natural variation because it is usually impossible or impractical to measure every point of a population. *Sampling design error* occurs when the sampling design is unable to capture the complete extent of the natural variability that exists in the true state of the environment.

- Analytical methods and instruments are never absolutely perfect; hence, a measurement can only estimate the true value of an environmental sample. *Measurement error* refers to a combination of random and systemic errors that inevitably arise during the various steps of the measurement process (*e.g.*, sample collection, sampling handling, sample preparation, sample analysis, data reduction, and data handling.

The combination of sampling design error and measurement error is called total study error, which may lead to a decision error. Since it is impossible to eliminate error in measurement data, basing decisions of measurement data will lead to the possibility of making a decision error.

The two types of decision errors are classified as *false positive* and *false negative* decision errors. A false positive decision error occurs when the Assessment Group presumes that the DRL is exceeded when the true value is less than the DRL. A false negative decision error occurs when the Assessment Group presumes that the DRL has not been exceeded when the true value exceeds the DRL.

While the possibility of a decision error can never be totally eliminated, it can be controlled. To control the possibility of making decision errors, the Assessment Group must control total study error. There are many ways to accomplish this, including collecting a large number of samples (to control sampling design error), analyzing individual samples several times, or using more precise

laboratory methods (to control measurement error). Better sampling designs can also be developed to collect data that more accurately and efficiently represent the population of interest. Every event will use a slightly different method of controlling decision errors, depending on where the largest components of total study error exist in the data set and the ease of reducing those error components. Reducing the probability of making decision errors generally increases costs. In many cases controlling decision error within very small limits is unnecessary for making a decision that satisfies the decision maker's needs. For instance, if the consequences of decision errors are minor, reasonable decision could be made based on relatively crude data (data with high total study error). On the other hand, if the consequences of decision errors are severe, the decision maker will want to control sampling design and measurements within very small limits.

To minimize unnecessary effort controlling decision errors, the Assessment Group must determine whether reducing sampling and measurement errors is necessary to meet the decision maker's needs. These needs are made explicit when the probabilities of decision errors are specified.

4.1.7   Optimal Design

Optimal design is initiated through preplanning of potential scenarios, training, and exercises. During an actual event, optimization of monitoring and assessment design will not be an initial consideration. As critical decisions are made and time is available to reassess objectives, then optimization of design will be initiated. By the time transition of FRMAC leadership is made for long-term recovery, the monitoring and assessment design should be close to the optimal configuration.

4.1.7.1 Data Quality Objectives (Step 7 – Optimize the Design for Obtaining Data)

The objective of this step is to identify the most resource-effective data collection design expected to generate data that satisfy the DQOs specified in the preceding steps. Review the DQO outputs generated in the preceding six steps to ensure that they are internally consistent. Generally, the goal is to find cost-effective alternatives that balance sample size and measurement performance, given the feasible choices for sample collection techniques and analytical methods. In some cases, where there is a relatively high spatial or temporal variability, it may be more cost-effective to use less expensive yet less precise analytical methods so that a relatively large number of samples can be taken, thereby controlling the sampling design error component of total study error. In other cases where the contaminant distribution is relatively homogeneous, or the action level is very near the method detection limit, it may be more cost-effective to use more expensive yet more precise and/or more sensitive analytical methods and collect fewer samples, thereby controlling the analytical measurement error component of total study error.

4.2   Default Derived Response and Intervention Levels (DRLs)

The default DRLs for aged fission products are essentially the same as for the nuclear power plant case. Therefore, the same DRLs are used here.

Table V3.4.2. Default DRLs for Releases from Irradiated Reactor Fuel

Issue	Marker	DRL	Sensitivity, Uncertainty, Spatial Density, Assumptions
Worker Protection	Exp. Rate Ext. Dose	See Table V3.2.3.	
EPA Early Phase PAG (evacuation)	Exp. Rate	10 mR/hr	Gamma exposure rate (mR/hr) indicating that evacuation or substantial shelter could be implemented in accordance with the EPA PAGs
	Predicted TEDE	1 rem	
Relocation 1st year	Exp. Rate	5 mR/hr	Gamma exposure rate (mR/hr) from deposition indicating that the population should be relocated in accordance with EPA PAGs (see Charts V3.2.2a and V3.2.2b)
	^{137}Cs concentration	3 μCi/m^2	
Ingestion PAG	Exp. Rate ^{137}Cs concentration	0.5 μR/hr	See Charts V3.2.3 and V3.2.4

4.3   Worker Protection

Doses to all workers during emergencies should, to the extent practicable, be limited to 5 rem TEDE. In some emergency situations, however, higher exposure limits may be justified. See EPA 400-R-92-001, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents* (EPA92), for specific guidance.

The difficulty in assessing the dose to emergency workers stems from the lack of source term information regarding transuranic content and inability of self-reading dosimeters to measure dose contribution by inhalation. If information on potential source term is available prior to the release, recommendations for the protection of emergency workers entering the plume can be developed. If the source term does not contain transuranic materials, the inhalation portion of the dose when compared to the external exposure will be a minor contribution to the TEDE. The use of respirators may not be required. Turn-back criteria based on gamma exposure rate and/or accumulated gamma exposure (self-reading dosimeters) during the plume phase could be established.

If transuranic material is present, then it is likely that inhalation will be the dominant dose pathway. Table V3.4.3 will aid assessment of significance and guide stay time. If the concentration of transuranic material present in the source term is known prior to the release, it may be possible for the Assessment Group to establish limits based on external exposure rates and/or accumulated external exposure that would also account for internal exposure through inhalation. If transuranics are known to be present and concentrations are not available, respiratory protection during the plume phase is generally recommended. Once air samplers can

be deployed and actual concentrations are quantified, it may be possible to reduce or eliminate the use of respiratory protection equipment.

After the plume phase, protective clothing and respiratory protection equipment will probably be based on levels of removable contamination present in the plume footprint.

Table V3.4.3.   Airborne Concentration Required to Give 5 rem CEDE
                     in Specified Time Period (μCi/mL)

Radionuclides	1 hour	100 hours	1 year[a]
^{234}U, ^{235}U, ^{236}U, ^{238}U	4×10^{-8}	4×10^{-10}	2.0×10^{-11}
^{237}Np, ^{239}Pu, ^{240}Pu, ^{242}Pu, ^{241}Am, ^{243}Am	4×10^{-9}	4×10^{-11}	2.0×10^{-12}
^{239}Np	2×10^{-3}	2×10^{-5}	1.0×10^{-6}
^{238}Pu	6×10^{-9}	6×10^{-11}	3.0×10^{-12}
^{241}Pu, ^{242}Cm	2×10^{-7}	2×10^{-9}	1.0×10^{-10}
^{244}Cm	8×10^{-9}	8×10^{-11}	4.0×10^{-12}

[a]Based upon the most restrictive annual limits on intake (ALI) for air breathed by an average worker for a working year of 2000 hours, assuming a breathing volume of 2400 m^3. (Based on 10 CFR 20.)

4.4   Emergency (Plume Phase)

The Emergency (Plume Phase), termed the Early Phase (EPA92) begins before a major release and lasts until the risk of a major release has ended and the areas with major contamination have been identified. During this phase, early health effects and the Early Phase PAGs are the principal focus.

EPA established Early Phase PAGs, which are projected doses, indicating when immediate protective actions are warranted. Table V3.4.4 summarizes the Early Phase PAGs. The EPA PAG for evacuation is 1,000 to 5,000 mrem (0.01 to 0.05 Sv) TEDE. The upper limit applies only for conditions that increase the risk of an evacuation, such as adverse weather conditions. Therefore, for most conditions, evacuation is recommended at a 1,000-mrem projected dose TEDE. TEDE includes the EDE from cloud passage and four days of ground shine, plus the inhalation CEDE from exposure to the plume and four days of resuspension. The committed dose equivalent to the skin may be 50 times higher. The emphasis for the Early Phase is in identifying those areas that have not been evacuated where the Early Phase dose may exceed the PAG.

The default recommendations of the Assessment Group will consist of DRLs based upon these PAGs. If the State or LFA and Federal Advisory Team establish different sets of PAGs, then the Assessment Group will derive a modified set of DRLs specific to the event.

Table V3.4.4. Early Phase PAGs

Protective Action	PAG (projected dose)	Comments
Evacuation (or sheltering[a])	1-5 rem[b]	Evacuation (or for some situations, sheltering[a]) should normally be initiated at 1 rem.
Administration of stable iodine	25 rem[c]	Requires approval of State medical officials.

[a] Sheltering may be the preferred protective action when it will provide protection equal to or greater than evacuation, based on consideration of factors such as source term characteristics and temporal or other site-specific conditions.

[b] The sum of the EDE resulting from exposure to external sources and the CEDE incurred from all significant inhalation pathways during the Early Phase. CDEs to the thyroid and to the skin may be 5 and 50 times larger, respectively.

[c] CDE to the thyroid from radioiodine.

4.5   Intermediate Phase

The Intermediate Phase is defined as the period beginning after the source and releases have been brought under control and environmental measurements are available for use as a basis for decisions on protective actions and extending until these protective actions are terminated. For the purpose of dose projection, it is assumed to last for one year. Although these Intermediate Phase PAGs were developed based on expected releases of radioactive materials characteristic of reactor incidents, they may be applied to any type of incident that can result in long-term exposure of the public to deposited radioactivity.

In the case of deposited radioactivity, the major relevant protective action is relocation. The principal pathways for exposure of the public occupying locations contaminated by deposited radioactivity are expected to be exposure of the whole body to external gamma radiation from deposited radioactive materials and internal exposure from the inhalation of resuspended materials. Other potentially significant exposure pathways include exposure to beta radiation from surface contamination and direct ingestion of contaminated soil.

Relocation is warranted when the projected sum of the dose equivalent from external gamma radiation and the CEDE from inhalation of resuspended radionuclides exceeds 2 rem in the first year. Relocation to avoid exposure of the skin to beta radiation is warranted at 50 times the numerical value of the relocation PAG for EDE.

It is the objective of these PAGs to assure that 1) doses in any single year after the first will not exceed 0.5 rem, and 2) the cumulative dose over 50 years (including the first and second years) will not exceed 5 rem. As indicated above, these longer-term objectives are expected to be met for reactor accidents through radioactive decay, weathering, and normal part-time occupancy in structures.

When the incident involves aged fission products, the process of radioactive decay, which plays a prominent role in meeting the long-term objectives, may have a minimal impact. It is likely that the boundary established for relocation will be based on the criterion that dose in any single year after the first will not exceed 0.5 rem. In events where the predominant contributors to dose are

long-lived radionuclides, the boundary may be based on the criterion of not exceeding a cumulative dose over 50 years of 5 rem. These boundaries may incorporate an area significantly larger than the initial evacuation zone.

4.5.1   Development of Initial DRLs

The initial set of DRLs for the Intermediate Phase will be based of the results of dose projection models using available source term information and default resuspension factors. Dose contours corresponding to 1st year, 2nd year and 50-year cumulative dose estimates will be provided to identify potentially affected populations. Based on these initial dose projections, decisions can be made on the types of additional measurements that need to be made to refine the initial DRLs.

4.5.2   Revision of Intermediate Phase DRLs

In events involving aged fission products, refinement of the DRLs and adjustment of the boundaries of the relocation zone will be an important activity of the Assessment Group. As described above, the initial boundaries and DRLs will be based on dose and deposition models using default values that are built into the computer codes.

A number of variables will have to be evaluated and incorporated into the refinement of the boundaries of the relocation zone. Several of the variables are:

- **Radioactive Decay:** Radioactive decay of the deposited material will account for reduction in dose with time. Spatial differences in the species of radioactive materials deposited (e.g., ratio of ^{137}Cs to ^{106}Ru concentrations) may vary significantly across the area. A higher relative concentration of ^{106}Ru in one area would show a quicker decrease in radiation exposure rates than in another area with a lower relative concentration.

- **Weathering:** Ranges are available of the impact of weathering on the concentration of radionuclides deposited on soil. These are affected by the chemical species of the contaminant, soil composition, precipitation, wind speed, etc.; factors that are unique to each area. The contaminant can be removed from the surface by erosion, causing exposure rates to decrease in some areas while other areas that receive the eroded materials can have increases in exposure rates. Incorporation of the contaminant into the soil column will reduce exposure rates also. The change of seasons can also affect exposure rates. In the spring, radionuclides can be incorporated into new vegetation and leaves resulting in increased exposure rates. Decreases occur in the fall when the vegetation dies and leaves fall. Snow cover in winter will shield the radioactive materials and decrease exposure rates.

- **Resuspension:** The movement of air across contaminated surfaces can cause the resuspension of radioactive material into the air. Ranges of resuspension factors are available in different climates (*e.g.*, arid, humid). Typically, resuspension factors decrease with time. Long-term measurement of site-specific resuspension factors will be required to adequately address this factor.

- **Shielding:** Some models look at the types of housing and structures in the affected area and try to account for the amount of shielding afforded by these buildings and the percentage of time that the inhabitants spend in these structures. This factor also reduces the total dose that an inhabitant would receive.

- **Decontamination:** Decontamination efforts may be applied to some portions of the contaminated area that is being considered for relocation. The value of the area or infrastructure contained in the area may warrant attempts to recover the area. Effectiveness of those decontamination efforts will need to be incorporated into the models.

These factors are typically applied to the process of making decisions on the placement of the boundary to the relocation zone. Probably portions of the affected area will be so highly contaminated that the decision to relocate is not an issue. Similarly, there will be lightly contaminated areas in the outer portions of the affected area, where relocation will not be a consideration. But there will be an area between these two regions where both political and practical considerations will significantly impact placement of the relocation boundary. Additional field monitoring and application of modifying conditions will play an important role in establishing this boundary.

4.6   Ingestion

In August 1998, the FDA issued revised guidance, "*Accidental Radioactive Contamination of Human Food and Animal Feeds*" (FDA98), for the protection of the ingestion pathway from accidental contamination of food. The new guidance established revised PAGs that restrict the CEDE to 500 mrem or the CDE to any specific organ to 5,000 mrem, whichever is most limiting. The PAGs are then expressed in terms of measurable quantities called Derived Intervention Levels (DILs). A DIL corresponds to the concentration of radioactivity in food, which could lead to an individual in the most sensitive population receiving a dose equal to the PAG if no intervention were taken for the year.

The DILs were computed with respect to the most sensitive population for five types of nuclear accidents (reactor, fuel reprocessing, waste, weapons, and spacecraft). Analysis of these scenarios leads to the identification of nine radionuclides that essentially control dose. The DILs are applicable to foods as prepared for consumption. Assessment of dehydrated food should consider reconstitution. Both raw foods and prepared foods are assessed as-is. The procedures are applicable to water to the extent it is part of human food. However, water, in general, is still governed by the Clean Water Act.

The DILs for the five principal nuclide groups are listed in Table V3.4.5.

Table V3.4.5. DILs for the Five Principal Nuclide Groups

Radionuclide Group	Bq/kg	pCi/kg
^{90}Sr	160	4,300
^{131}I	170	4,600
^{134}Cs + ^{137}Cs	1,200	32,000
^{238}Pu + ^{239}Pu + ^{241}Am	2	54
^{103}Ru + ^{106}Ru	C ^{103}Ru/6,800 + C ^{106}Ru/450 <1	C ^{103}Ru /180,000 + C ^{106}Ru /12,000 <1

Circumstances may exist where contributions by radionuclides other than those in the five groups may be controlling. The FDA has provided 15 additional DILs in Appendix E of their guidance document for those radionuclides next most likely to be of significance (FDA98). These nuclides consist of ^{89}Sr, ^{91}Y, ^{95}Zr, ^{95}Nb, ^{132}Te, ^{129}I, ^{133}I, ^{140}Ba, ^{141}Ce, ^{144}Ce, ^{237}Np, ^{239}Np, ^{241}Pu, ^{242}Cm, and ^{244}Cm. All of the FDA DILs are presented in Table 5.2 in Volume 2.

NOTE: The five principal DILs are applied separately, not summed as in the previous guidance. Similarly, the 15 additional DILs are applied separately. If any one DIL is exceeded, then the food is determined to exceed the PAG.

SECTION 5. URANIUM FUEL ACCIDENT

TABLES

URANIUM FUEL ACCIDENT

5.1   Description of Generic Uranium Fuel Facility Accident

For uranium fuel facilities that chemically convert uranium, separate uranium radionuclides, or manufacture fuel for nuclear power plants and other types of nuclear facilities, the overall degree of risk to the public is small. A historical review of past accidents published in NUREG-1140 (NRC85) concludes that the maximally exposed individual effective dose (using adverse meteorology and exposure times) from such an accident would be less than 5 to 10 rem. However, the actual effective dose that a person could realistically receive is lower. The probability of accidents at these types of facilities is on the order 10^{-4} per year or smaller.

A PAG could be exceeded in three postulated accident scenarios:

- Uranium hexafluoride (UF_6) cylinder rupture
- Large fire at a facility handling very large quantities of dispersible uranium
- Long-term, pulsating criticality accident at a facility that handles highly enriched uranium (covered in Section 7)

This section covers the first two types of accidents; the long-term criticality accident is discussed in Section 7. The most hazardous type of fuel facility accident is the sudden rupture of a heated cylinder of uranium hexafluoride (UF_6). Past accident reports and previous postulated accidents indicate that acute fatalities and permanent injuries can be caused by the chemical toxicity of UF_6 and its byproducts. Radiation doses are not as significant as chemical toxicity hazards.

The chemical toxicity of uranium is also a factor in uranium fuel facility accidents. The heavy metal, uranium, is chemically toxic to the kidneys. Exposure to soluble forms of uranium can cause renal injury. Factors that contribute to chemical toxicity risk include enrichment, mode of entry, and solubility. See Method M.V3.5.0.

5.1.1   Scenario Description

5.1.1.1   Types of Uranium Fuel Facilities

Four types of uranium fuel facilities have been considered. These are briefly described below. Only one the uranium enrichment facility has the sufficient potential for a radiological accident to warrant for further consideration.

5.1.1.1.1 URANIUM ENRICHMENT FACILITY

Two uranium enrichment facilities are operational in the United States and about 20 plants worldwide, according to IAEA information. The plants in the United States use the gaseous diffusion process to enrich the uranium in the radionuclide ^{235}U.   The process separates a stream of heated UF_6 gas into the two fractions, one enriched in ^{235}U and the other depleted in ^{235}U. The

process is repeated a number of times until the uranium is enriched to the desired enrichment. The enriched UF_6 is generally used to manufacture fuel for nuclear reactors. The depleted UF_6 is stored as a solid in large metal cylinders at the enrichment facility. Since UF_6 is the predominant form of uranium at uranium enrichment facilities, the UF_6 dispersion scenario noted above is applicable to these facilities.

5.1.1.1.2 CHEMICAL CONVERSION PLANT

Chemical conversion plants convert various chemical species of uranium to other forms for production of nuclear fuels. In the United States, uranium ore concentrate (uranium oxide, commonly called yellowcake) is shipped from uranium mills and converted into impure UF_6 and then purified. According to information from the International Atomic Energy Agency (IAEA), one plant is operational in the United States, with about 30 facilities worldwide. At the end of the production process, liquid UF_6 is placed into 10-ton or 14-ton cylinders for storage and transportation to uranium enrichment facilities.

The uranium is handled in many different forms in conversion plants, but the UF_6 is the only chemical form of uranium that is readily dispersible (NUREG-1140).

5.1.1.1.3 URANIUM FUEL FABRICATION PLANT

Fuel fabrication plants convert UF_6 to uranium dioxide (UO_2) using one of two processes. One process uses deionized water to hydrolyze the UF_6 to form UO_2F_2. Ammonium hydroxide is added to the solution to precipitate ammonium diuranate. This precipitate is then reduced to UO_2 in a calciner operating at high temperatures in a hydrogen-reducing atmosphere. The other process converts UF_6 to UO_2 powder using fluidized bed reactors operation under a reducing atmosphere. There are five operational fuel fabrication facilities in the United States and about 46 plants worldwide.

In most cases the ^{235}U is enriched to less than 5%. One NRC-licensed plant currently handles highly enriched UF_6, the only chemical form readily dispersible to the environment.

5.1.1.1.4 DEFENSE FACILITY

A number of facilities within the DOE complex conduct similar or identical functions for defense purposes. Some of the facilities are still active, while others are in various stages of decontamination and decommissioning. Many of the DOE facilities also used recycled uranium, so trace contamination of uranium by ^{99}Tc, ^{237}Np and ^{239}Pu is possible. Table V3.5.1 shows contaminants produced at the three facilities.

Table V3.5.1. Contaminants in Recycled Uranium and Depleted Tails

Site	Enriched Product	Depleted Tails
Oak Ridge K-25	Pu < 0.05 ppb Np < 5 ppb Tc < 1 ppm	Pu < 0.01 ppb Np < 5 ppb Tc < 10 ppb
Portsmouth	Pu < 0.037 ppb Np < 3.19 ppb Tc < 0.69 ppm	Pu < 0.007 ppb Np < 0.6 ppb Tc < 0.4 ppb
Paducah	Pu < 0.01 ppb Np < 5 ppb Tc < 20 ppm	Pu < 0.01 ppb Np < 5 ppb Tc < 10 ppb

5.1.1.2   Uranium Hexafluoride Release

The rupture outdoors of a large, heated cylinder of UF_6 is considered by the NRC to be a standard accident for regulatory analysis. The compound UF_6 is a volatile, white crystalline solid at ambient temperature. A large decrease in UF_6 density occurs when UF_6 changes from the solid to the liquid state, resulting in a large increase in volume. The thermal coefficient of expansion of liquid UF_6 is also high when UF_6 is heated (about 0.1% per degree F). Based on these physical phenomena, it is vital to maintain operational control of the mass and physical state of UF_6 to avoid container ruptures.

Of the several different cylinder types used to store UF_6, the vast majority has a 14-ton capacity, while some have a 10-ton capacity. The 14-ton cylinder is 3.7 m long by 1.2 m in diameter, with a wall thickness of 0.79-cm steel.

Four UF_6 accident scenarios are considered: depleted uranium, natural uranium, low enrichment uranium (5%), and highly enriched uranium (93%). Uranium isotopic abundances are defined for each enrichment category in Table V3.5.2 (DOE01a).

Table V3.5.2. Nuclide Activities for Various Enrichments of ^{235}U

Enrichment	Nuclide Contribution to Total Activity (%)			Specific Activity (Ci g^{-1})
	^{234}U	^{235}U	^{238}U	
Depleted	8.4	1.5	90	4×10^{-7}
Natural	49	2	49	7.06×10^{-7}
Low Enriched (5%)	88.8	2.8	8.4	2.7×10^{-6}
Highly Enriched (93%)[1]	96.9	2.9	0.03	7×10^{-5}

[1]The remaining activity is attributed to ^{236}U.

For the gaseous diffusion plant and chemical conversion plant scenario, a fire affects three 14-ton cylinders. This scenario has been thoroughly analyzed by Brown, *et al.* (Br97). The fire heats a cylinder; the UF$_6$ sublimes and builds pressure within the cylinder. The cylinder ruptures due to the high pressure, and hot UF$_6$ gas is jetted to the atmosphere. The UF$_6$ gas and atmospheric water vapor form uranyl fluoride (UO$_2$F$_2$) and hydrogen fluoride (HF), which is an exothermic reaction (+52 kcal/mole, NUREG-1140) that helps to continue heating the cylinders. Brown, *et al.* assumed in one of their analyses that the release time is 121.4 minutes, with a resulting ground level release of UF$_6$, initially.

For regulatory analysis and for comparison to Brown, *et al.*, NUREG-1140 assumed a rupture of a hot 14-ton UF$_6$ cylinder. The ruptured cylinder was assumed to release 9500 kg of UF$_6$. It was further assumed that 4800 kg of natural uranium became airborne, while 1600 kg settled on the ground quickly due to agglomeration and impaction. The release was estimated to last 15 minutes.

NUREG-1189 (NRC86) provides an analysis of an actual UF$_6$ accident at a fuel facility, giving an interesting comparison to the model scenarios above. During this accident, an 8-inch by 52-inch split in a 14-ton UF$_6$ container occurred during a container heating, resulting in a rapid excursion of UF$_6$. Approximately 75% of the UF$_6$ was released in 5 minutes, and the remainder in 40 minutes. Approximately 14,750 lbs. of UF$_6$ was released to the environment, along with 12,900 lbs of UO$_2$F$_2$ and 3350 lbs of HF.

NUREG-1140 considers two UF$_6$ release scenarios for fuel fabrication plants, one for low-enrichment UF$_6$ and the other for high-enrichment UF$_6$. For low-enriched UF$_6$, the worst-case accident assumed a full 2500-kg cylinder heated to high temperatures. The cylinder failed and released 22% of its contents in 15 minutes. For high-enriched UF$_6$, a leak from a 15-kg cylinder

was assumed, with 9 kg released and 6 kg remaining in the cylinder. The quantities of UF_6 and release times assumed in this section for the types of facilities are listed by facility type in Table V3.5.3.

Table V3.5.3 Summary of UF_6 Source Terms

Type of Facility	Amount of UF_6 Released (kg)	Release Time (minutes)	Total Quantity of UF_6 Released (kg)	Release Level	Particle Size of UO_2F_2 (μ, AMAD)	Source of Data
Chemical Conversion (Three Cylinders)	0 5,221 4,048 1,622	0-12.2 12.2 12.2-30 30-121.4	10,892	Ground	1	Brown, *et al*
Gaseous Diffusion (Three Cylinders)	0 5,221 4,048 1,622	0-12.2 12.2 12.2-30 30-121.4	10,892	Ground	1	Brown, *et al*
Fuel Fabrication, Low Enrichment (Single Cylinder)	540	15	540	Ground	1	NUREG-1140
Fuel Fabrication, High Enrichment (Single Cylinder)	9	5	9	Ground	1	NUREG - 1140

5.1.1.3 Uranium Facility with Large Inventory of Dispersible Uranium

NUREG-1140 analyzed the accident history and probability of non-UF_6 accidents occurring at various facilities. The analysis showed that fires and explosions caused little exposure or contamination. None of the accidents caused offsite doses approaching the 1-rem lower limit of the PAG. Thus, the conservative source term for all facilities is the UF_6 source term listed in Table V3.5.3 above.

5.1.2 Data Quality Objective Process

The seven steps of EPA's DQO process are applied to the uranium nuclear fuel accident scenario in the discussion below. Application detail is minimal in the manual but will develop as incident-specific work proceeds. A complete and formal DQO treatment is not expected until the Recovery Phase, specifically, at the beginning of long-term monitoring.

5.1.2.1 DQO Step 1—State the Problem

The key element of step 1 of EPA's DQO process, State the Problem, is addressed by the scenario description above. The remaining elements, which are not scenario specific, are covered in the organizational overview of the Assessment Group, Volume 1 of the *FRMAC Assessment Manual*. These elements include team composition, customer/decision maker interface, and resources. This overview is primarily presented in the Introduction. Team members and roles are described in Volume 1, Section 1, Introduction, and in Section 6: Administration, Internal Procedures, and Tools. The specific identity of the decision maker(s) is dynamically defined by consultations between the FRMAC Manager and representatives of the Lead Federal Agency (LFA) and state(s). As part of the FRMAC team, the Assessment Group is responsible for providing the assessments of measurements and predictions necessary for decision-makers to protect the public and emergency workers from excessive exposure to radioactive materials. The *FRMAC Assessment Manual* is the technical basis for these assessments. Assessment results are interpretations of measurements in terms of published PAGs, or as otherwise directed by the LFA and Advisory Team. Should the *FRMAC Assessment Manual* not provide the information needed to address a specific issue, then technical experts outside the FRMAC will be enlisted.

5.1.2.2 DQO Step 2—Identify the Decisions

The key element of step 2, Identify the Decisions, is the enumeration of the major protective actions and their respective "triggers." The FRMAC does not make PARs, but it does identify those areas where specific actions may be technically warranted. It may also identify potential mitigating measures. The decision-maker is expected to consult with the staff to develop the alternative actions for each decision.

5.1.2.2.1 POTENTIAL CONSEQUENCES

The potential consequences from a UF_6 release are broken down into three main effects: radiation doses from uranium intakes and direct exposures, chemical toxicity doses from uranium, and chemical toxicity doses from HF.

5.1.2.2.1.1 Uranium Issues

Uranium released to the environment by an accident may be inhaled, ingested or injected directly into the bloodstream. The most likely routes of intake are through inhalation and ingestion. Inhalation hazards from uranium result primarily from alpha particle emissions. As noted above, radiation or chemical toxicity effects may be the predominant effects from uranium, depending on the chemical form of uranium, enrichment and route of entry. Since the expected chemical compound is UO_2F_2, the inhalation classification is D, and chemical effects are expected to be dominant.

5.1.2.2.1.2 HF Issues

The primary hazard of a UF_6 accident is from the release of hydrogen fluoride (HF), a chemical, not a radiological hazard. The National Institute of Occupational Safety and Health (NIOSH) exposure limit (Time Weighted Average) for HF is 3 ppm; the concentration that causes imminent danger to life or health (IDLH) is 30 ppm. Table V3.5.4 summarizes HF health effects.

Table V3.5.4. HF Health Effects

Health Effect	HF Concentration (mg m^{-3})
Lethal (15 minutes)	3500
Unbearable (for one minute)	100
Irritation (15 minutes)	13
Detectable by smell but no health effects	2.5

(Ju84)

The compound HF is produced when gaseous UF_6 reacts with atmospheric moisture, undergoes hydrolysis, and produces UO_2F_2 and the highly corrosive and toxic gas HF. This type of scenario may occur when a UF_6 cylinder is heated above the triple point of UF_6 (147°F/64°C), and hydraulically ruptures. When the rupture occurs, there is a rapid outflow of UF_6 until the pressure drops sufficiently to start the solidification process. The rate of outflow then decreases but continues until the contents cool to about 133°F/56°C, which is the atmospheric sublimation temperature (DOE99).

Finally, as noted above, HF is a chemical poison that must be considered during the overall accident assessment. Plumes of HF, when highly concentrated, are visible and immediately irritating to the lungs. Several authors have reported injuries and fatalities from past accidents (NUREG-1140, NUREG-1198 [NRC86a]). The most serious injuries have been HF irritation of the skin, eyes, mucous membrane of the upper respiratory tract, esophagus, larynx and bronchi.

The release of UO_2F_2 forms a particulate in air, which behaves as any particulate in the environment. It is readily dispersible by the wind, and it deposits on the ground and resuspends back to the air according to its particle size and other physical factors. HF behaves as a gas and will disperse in air as any gas (NUREG-1140).

5.1.2.2.2 POTENTIAL ACTIONS

The Assessment Group provides decision-makers with the technical basis for protective actions. The major potential actions are those necessary to reduce risk due to exposure to acceptable levels. The actions are time-sensitive and prioritized to address the most serious and time-sensitive potential effects first.

The major protection actions include (generally prioritized):

1. Expedited evacuation where potential for early health effects exists due to HF and/or UO_2F_2 exposure
2. Evacuation of immobile populations (*e.g.*, hospitals and prisons)
3. Evacuation of the general public. On the basis of the study completed by Brown, *et al.*, (Br97) evacuation distances to prevent any effects (for adverse meteorological conditions) could extend as far 6.2 miles. In order to prevent irreversible effects, Brown, *et al.* recommends an evacuation distance of about 100 meters. NUREG-1140 recommends evacuation distances of one mile.
4. Control of access to the area
5. Sheltering of public and immobile populations
6. Triage of potential health effect victims (*i.e.*, identification of unevacuated areas where the population may have the potential for early health effects so that these individuals may be identified and treated as required)
7. Relocation of unevacuated populace to avoid future risk
8. Suspension of agricultural production
9. Condemnation of foods

Other decisions may include:

1. Exposure planning for emergency workers
2. Selection of measurements and monitoring locations
3. Guidelines for reentry
4. Identification and selection of mitigation options

5.1.2.3 DQO Step 2--Inputs to the Decisions

5.1.2.3.1 INFORMATION INPUTS

Because decisions are very time sensitive, particularly early in an accident, the radiological and toxic chemical assessments must proceed with whatever quantity and quality of data are available. Initial radioactivity concentration measurements and visible sightings of an HF plume may be used to validate or renormalize a model. As quickly as possible, sufficient measurements must be acquired to replace dependence on the model. As time progresses and decisions become less critical, the quantity and quality of data improves. Eventually, guidelines will be implemented on the collection and analysis of measurements and models will become interpolation tools.

During the Early and much of the Intermediate Phase of an accident, the assessment methods and reference data in the *FRMAC Assessment Manual* should be sufficient for radiological assessment. Default decision levels (DRLs) in Table V3.5.6 are to be used until sufficient data have been collected to eliminate assumptions. Revision of a DRL is acceptable only if an assumption can be eliminated. Several revisions may occur over time as assumptions are eliminated.

From a radiological standpoint, uranium air concentrations serve as the measurement to identify where protective actions and relocation are warranted. If the degree of uranium enrichment is not known, the most conservative DCFs should be used.

5.1.2.3.2 MEASUREMENT AND PREDICTION INPUTS

Table V3.5.5 outlines the measurements and model results that will be needed in relative order of necessity.

Table V3.5.5. Measurement and Predictive Inputs

Predictions	Plume Passage-TEDE Post Plume Passage-Exposure Rate and Fallout Patterns
Field Measurements*	Handheld exposure rate meters Handheld GM meters Alpha contamination detectors Field gamma detection and spectroscopy Air sample gross counting analysis
Sample Analysis	Air samples Soil samples Water samples Crop samples

*Typically, detection of uranium contamination has been performed with alpha particle detectors. For some conditions and situations, detection of beta particles or gamma rays may be more appropriate. For natural, depleted and low enrichment uranium that are in equilibrium with their progeny, the detection sensitivity for beta/gamma radiation is about 5 times more sensitive than by detection of the alpha particles alone. If the uranium is highly enriched or has been very recently processed, detection using the alpha particles alone is required (DOE01b).

5.1.2.3.3 COMPLICATING FACTORS

Environmental data must be representative in order to be a valid basis for revising the DRLs. Environmental data may not representative because:

- The fraction of equilibrium between uranium radionuclides and their progeny could confuse measurements and assessments.

- The expected chemical form of uranium released during a UF_6 accident is UO_2F_2. If the actual chemical form is different, the International Commission on Radiological Protection

(ICRP) transportability class could change, changing the chemical or radiological assessment.

- A mixture of UF_6 enrichments could be released, confusing the need for radiological and/or toxicity assessments.

5.1.2.4 DQO Step 4—Boundary of Consideration

5.1.2.4.1 PHYSICAL BOUNDARY

The area for which assessments are needed is the entire area impacted by the plume, as well as a sufficient area outside of that to ensure that the extent of the area affected by the plume could be defined. Initially, the extent will be the area potentially subject to evacuation or relocation. After these concerns are addressed, the limits will be extended to surrounding agricultural production areas where the FDA PAGs may be exceeded. It may also be necessary to include food-processing facilities well outside the affected region, where contaminated foods may have been transported.

5.1.2.4.2 TEMPORAL BOUNDARY

The FRMAC Assessment tools provided here are sufficient for the appraisal of doses during the Early and a significant portion of the Intermediate Phase.

Detection of conditions that threaten additional UF_6 cylinders could result in declaration of higher-level emergency classifications. These conditions include a fire that heats other cylinders, or other similar cylinder-heating scenarios, such as a facility fire. A FRMAC may be requested if the emergency progresses and involves a number of containers resulting in a major release. Releases could occur as long as there is a heat source to continue heating the UF_6 cylinders. Assessment of initial measurements is likely to occur after the release has begun, as local emergency response organizations would likely be capable of addressing releases from single or multiple UF_6 containers.

5.1.2.4.3 CONSTRAINTS

Some of the potential constraints on measurements include:

1. Remonitoring if new releases occur
2. Deposition on snow cover
3. Deposition on a leaf canopy
4. Delays in monitoring due to adverse weather
5. Access denied by property owners
6. Inaccessible terrain

5.1.2.5 DQO Step 5--Decision Rules

The Assessment Group does not establish decision rules nor make PARs. However, published PAGs are used by the Assessment Group as decision rules for the interpretation of measurements and predictions. These PAGs are implemented as DRLs. If a measurement exceeds a specific DRL by any margin, then that location fails the test at hand. If a measurement falls short of a DRL by any margin, then it passes the test at hand.

DRLs have been defined for the following:

- Emergency Worker Turn-back limits

- Evacuation based on EPA Early Phase PAG following plume passage

- Relocation (1st year), plus 2nd and 50-year long-term objectives

- Agricultural hold based on deposition

- Food condemnation (agricultural embargo) based on food concentration

- Water condemnation based on concentration

The PAGs and computational approach may be altered by the Federal Advisory Team.

5.1.2.6   DQO Step 6 Tolerance Limits

Assessors must establish tolerable levels of uncertainty when calculating DRLs. For example, evacuation, shelter, and agricultural product holds have a higher tolerance level than re-entry, which is higher yet than relocation, and so on. It is up to the assessor to establish these tolerable levels until a more definitive uncertainty analysis can be performed.

Sensitivities of measurements must always be adequate to detect the DRL level for the question at hand. Acceptable uncertainties are listed below.

If assessments are being used for:

- Evacuation, sheltering, or agricultural hold - the tolerance limit is a factor of 10.

- Re-entry considerations - the tolerance limit is a factor of 2.

- Relocation - the tolerance limit will be negotiated, but is expected to be approximately 30%.

- Return - the tolerance limit will be negotiated and will likely be much smaller.

- Condemning foods or water - the same criteria used by USDA for evaluation of non-radiological contamination will be applied (10%).

5.1.2.7   DQO Step 7 Optimal Design

FRMAC can do little initially to optimize design, which is primarily the responsibility of the EPA under its management of FRMAC during the Recovery Phase.

5.2   Default Derived Response and Intervention Levels (DRLs)

Default decision levels are presented in Table V3.5.6.

Table V3.5.6. Default Derived Response and Intervention Levels

Issue	Marker	DRL	Sensitivity, Uncertainty, Spatial Density, Assumptions
Worker Protection	Predicted Doses	See Method V1.M.2.1. Factors needed for this method are listed in Table V3.5.13. Be sure to include contaminants in Table V3.5.13, as appropriate.	Inhalation intakes are, by many orders of magnitude, the largest contributor to dose during the Early Phase. See Table V3.5.7 for Post Plume Phase
EPA Early Phase PAG	Predicted Doses	See Methods V1.M.3.1- 3.9. Factors needed for this method are listed in Table V3.5.13. Be sure to include contaminants in Table V3.5.13, as appropriate. **DRL for combined pathways is 3.4 E-7 μCi cm^{-3} h.**	Inhalation intakes are, by many orders of magnitude, the largest contributor to combined pathways dose during the plume phase. The DRL varies for different enrichments and ICRP-30 lung clearance class. The values provided are for Class D uranium. [Note: Skin PAGs cannot be reached with any uranium chemical compound.]
Relocation 1st year	Predicted Doses	See Methods V1.M.4.2- 4.8	Factors needed are listed in Table V3.5.14. Be sure to include contaminants in Table V3.5.14, as appropriate.
Ingestion PAG	Exp. Rate Predicted Doses	0.5 μR/hr See Methods M.5.2-M.5.14 in Volume 1	Uranium values listed in Volume 2, Tables 5.2-5.9 are acceptable for use in Methods V1.5.2-5.14.

5.3 Worker Protection

As indicated in Volume 1, the major responsibility of the Assessment Group in the area of worker protection is to provide information for determining turn-back guidance for emergency workers. The guidance is based on dose limits (TEDE) but must be presented in a manner useful to field personnel. This typically means that it must be presented in terms of exposure rate (as measured on a hand-held instrument) or integrated dose (as measured on a self-reading or electronic dosimeter).

Table V3.5.7 contains default turn-back guidance for accidents involving UF_6. The guidance is expressed in terms of integrated exposure as registered on a self-reading dosimeter. The guidance values for the various emergency activities reflect the total integrated dose from external exposure resulting from the emergency. Careful coordination must occur among emergency workers, and Field Monitoring and Health and Safety Management to ensure that workers are not allowed to exceed applicable limits. The administrative limits are designed to assist in this effort.

Table V3.5.7.   Default Federal Emergency Worker Dose Limits and Turn-back Guidance for Events Involving UF_6 for Post Plume Phase

Dose Limit Category or Emergency Activity	Inhalation Intake Possible (mrem)	No Inhalation Intake Possible (Post Plume) (mrem)[1]
Administrative limits	Estimate using Volume 1, Method M2.1	
Investigation level		1,500
Administrative level		2,500
Emergency activity		
All		5,000
Protecting major property		10,000
Life saving or protecting large populations		25,000
Life saving or protecting large populations[2]		>25,000

[1]Significant inhalation hazards are indicated by 1) breaches of UF_6 containers, 2) possible airborne plume or 3) ineffective respiratory protection provided to minimize resuspension intakes.
[2]Only on a voluntary basis to personnel fully aware of the risks involved.

Table V3.5.7 can be used as a guideline for determining turn-back levels. For example, a responder will be available for 7 days and receives about 200 mrem per day. The Investigation Level will nearly be achieved in that time interval. If no inhalation intake is possible, only dose rates are used for this evaluation. Table V3.5.7 can be used to alter these values if mitigation methods are employed. The *Radiological Emergency Response Health and Safety Manual* (DOE01) specifies a turn-back exposure rate of 600 mR/hr, but this exposure rate will never be achieved during a UF_6 accident.

5.3.1 Computation of Turn-back Guidance

Revised turn-back guidance may be calculated using Method M.2.1 in Volume 1 of the *FRMAC Assessment Manual* or RASCAL 3.0 Field Monitoring to Dose Option.

5.4 Early (Plume) Phase

As indicated in Volume 1, the Early Phase is considered to last for about 96 hours for the purpose of dose assessment. The EPA evacuation PAG is 1 to 5 rem, where the dose considered is the sum of the EDE from external sources and the CEDE from significant inhalation pathways. The CDE to the thyroid and skin may be 5 and 50 times higher, respectively. It is expected that the public may be evacuated upon a release of UF_6 if they live or work close enough to be affected by the HF and/or UO_2F_2 plume. The emphasis is, therefore, to identify areas that have not been evacuated where the Early Phase dose may exceed the PAG.

5.4.1 Default Derived Response Level

The DRL for the evacuation PAG is 3.4×10^{-7} µCi cm^{-3} h for the combined pathways, inhalation, immersion, and exposure to contaminated ground.

5.4.2 Revision of the Evacuation DRL

Instead of revising the DRL, the preferred method of identifying areas where the Early Phase PAG may be exceeded is by performing a dose projection using a computer model (such as SHARC) or by completing Method M.3.1 in Volume 1.

5.5 Intermediate Phase -- Relocation

As indicated in Volume 1, the Intermediate Phase is the period of time which begins after releases have been brought under control and reliable measurements are available to use for determining additional PARs. It continues until the additional protective actions are terminated. The major protective actions involve relocation and restrictions on the use of contaminated food and water.

EPA guidance (EPA 92) states that relocation is warranted if the dose from deposited materials and inhalation of resuspended radioactive material is projected to be greater than 2000 mrem the first year (or beta skin dose 50 times higher). Dose reduction due to part-time occupancy and decontamination is not to be considered. Dose reduction due to decay and weathering is to be included and has been considered in the dose factors used in the manual. The levels of uranium contamination should be used to develop relocation DRLs. These DRLs may change spatially and temporally (due to in-growth of progeny or the identification of significant recycling

contaminants or concentrated contaminant in depleted tails), and should be reevaluated periodically.

EPA guidance also established objectives to ensure the dose in the second year does not exceed 500 mrem and the cumulative dose over 50 years does not exceed 5000 mrem. For contamination resulting from a UF_6 accident, meeting the 2000 mrem PAG for the first year is expected to result in the longer-term objectives being met though decay, weathering and part-time occupancy. The process for determining if these objectives are being met will be developed as part of the long-term assessment plan and is beyond the scope of this document.

5.5.1 Default Relocation Derived Response Levels

To help determine areas where the PAG may be exceeded, we can define DRLs that correspond to the 2000 mrem first-year effective dose. See Volume 1, Methods M4.1- 4.8. As mentioned above, for a UF_6 accident the DRL for relocation can be expressed in terms of exposure rate or concentration of deposited uranium on the ground. The DRL levels may need to be adjusted if the state in which the FRMAC is operating has a relocation PAG different from the EPA's. In that case, follow method Volume 1, Method M.4.1 to make the adjustment.

5.5.2 Revision of Relocation Derived Response Levels

The DRLs are a function of a mixture of uranium radionuclides, their progeny and the level of recycling and tails contaminants in the deposited material. Samples should be obtained and analyzed to assure that the values used in the calculations are representative of the entire affected area.

5.5.2.1 Criteria

The DRLs should be reevaluated:

- Initially when the actual radionuclide mix of the release is known.

- Weekly for the first month to account for further changes in weathering.

- Monthly thereafter until weathering is no longer having significant impact.

The characteristics of the release and changing meteorological conditions may dictate that a single value for each type of DRL may not be appropriate for the entire affected area.

5.6 Intermediate Phase -- Ingestion

As indicated in Volume 1, the Intermediate Phase is the period of time which begins after any releases have been brought under control and reliable measurements are available to use for determining additional protection action recommendations. It continues until the additional

protection actions are terminated. The major protective actions taken during the Intermediate Phase involve relocation and restrictions on the use of contaminated food and water. The FDA issued recommendations regarding contaminated food in 1998 (FDA98). Key points in these recommendations are the DILs, concentrations in food at which some action should be taken to limit or preclude the use of the food product.

5.6.1 Default Ingestion Derived Response Levels

As with other PAGs, DRLs are used that indicate DILs may be approached or exceeded. The DRLs may be expressed in terms of many measurable quantities. One of these is the concentration of the deposited materials (*i.e.*, the radionuclide mix). The default deposition concentration DRLs for ingestion are determined using Volume 1, Method M.5.8, and Tables 5.3-5.9 in Volume 2, which are applicable to UF_6 releases.

5.6.2 Revision of Ingestion Derived Response Levels

The DRLs are a function of the mixture of uranium radionuclides, progeny, and potential contaminants in the deposition. Samples should be taken and analyzed to determine the values used in the calculations are representative of the entire affected area.

5.6.2.1 Criteria

The DRLs should be reevaluated:

- When the actual radionuclide mix of the release is known.
- When site-specific transfer factors have been determined.
- Every month thereafter until weathering no longer has a major impact.

5.7 Decay Corrections

5.7.1 Discussion of Complications Due to Decay and In-growth of Progeny

Due to changes in activity caused by radioactive decay and other processes, data collected at one point in time may be difficult to correlate with data collected at another point in time. The decay of uranium atoms produces other radioactive species. If the released uranium was newly enriched uranium, the radioactive progeny associated with the uranium would have been removed during the separation process. As time proceeds, the radioactive progeny build in activity and reach equilibrium. The fraction of secular equilibrium can be estimated by:

$$1 - e^{-(\lambda_2 - \lambda_1)t}$$

Where:

λ_2 = is the decay constant of the progeny,

λ_1 = is the decay constant of the uranium parent, and

t = the time post enrichment.

Correcting measured or predicted data to create a set of data related to a common point in time should allow analysts to better identify trends and association in various data sets.

5.7.2 Revision of DRLs for Decay of Uranium and Equilibrium of Progeny

Revising DRLs for decay may be as simple as re-running a computer code or revising a spreadsheet so that the DRL for another point in time is calculated. Revising the DRL may simply consist of reading a new value from the appropriate chart (or a similar chart constructed for the actual mix involved in the release). If the release is aged uranium, then it is likely that no correction for decay and in-growth will be needed.

5.7.3 Correction of Data for Weathering and In-Growth

Data most likely to require correction are exposure rate, in situ spectroscopy and soil sampling data. These data points may be corrected to a common point in time be applying a factor determined from a curve. The first method for creating the curve is to make repeated exposure rate readings over time at one location, plot the resultant curve and determine the decay factor. One caution in using this method is that if it has rained during the evaluation period, weathering may have occurred, so comparison may be very difficult. Since UO_2F_2 is very soluble in water, it is likely that rain will affect comparisons over time. The second method is to determine the nuclide mix (from sample analysis or in situ spectroscopy and knowledge of the source), compute the decay and in-growth curve and determine the factor. The factor can then be used within the FRMAC database to correct the exposure rate readings to a common time.

Results of analysis may also be corrected, but this will probably not occur very frequently for uranium radionuclides and their progeny.

5.8   DRL Revision

Revision to DRLs may be made when assumptions can be eliminated or when necessitated by decay and in-growth or changing conditions. The frequency required to compensate for decay and in-growth depends on the makeup of the release and the particular DRL under consideration. The methods in this section are for those DRLs specific to UF_6 accident scenarios; those in Volume 1 are for DRLs common to other accident scenarios.

5.9   Ancillary Information and Methods

Method M.V3.5.0   Chemical and Radiological Dividing Line for Uranium

Table V3.5.8 lists the various ICRP-30 clearance classes and the various dividing line enrichments for each class (adopted from DOE01b). Radiological concerns are predominant above the dividing line, while chemical toxicity hazards are predominant below the dividing line.

Table V3.5.8. Dividing Line Enrichment for Chemical and Radiological Hazards

ICRP-30 Class	Inhalation Annual Limit on Intake (μCi)	Specific Activity of Dividing Line Enrichment (μCi g^{-1})	Dividing Line Enrichment for Radiological Dose Limit (%)
D	1	169.5	(a)
W	0.7(b)	29.9	52.8
Y	0.05	0.71	0.82

[a] The resulting enrichment is greater than 100%. Consequently, chemical toxicity is limiting for acute exposures to Class D uranium.

[b] ICRP-30 lists Class W annual limits of intake (ALIs) of 0.7 μCi for ^{234}U and 0.8 μCi for ^{235}U and ^{238}U. The differences are the result of rounding to one significant digit. Non-rounded values are all approximately 0.75 μCi.

Uranium Heavy Metal Poisoning

NUREG-1140 provides a concise summary of the health effects of uranium heavy metal poisoning. The effects are summarized in Table V3.5.9.

Table V3.5.9. Levels of Acute Intakes of Uranium Leading to Health Effects

Health Effects	Acute Intake (mg)
50% Lethality	243
Permanent Damage	45
Renal Injury (transient)	8.6
No Effect	4.5

Method M.V3.5.1   Absorbed Dose Rates in Air from Uranium Compounds

Uranium radionuclides and their progeny emit alpha and beta particles, and gamma rays. The surface-absorbed dose rates for the various chemical forms of uranium (and progeny) are listed in Table V3.5.10.

Table V3.5.10. Absorbed Dose Rates in Air for Various Chemical Forms of Uranium

Chemical Form	Surface Absorbed Dose Rate in Air (mrad/hour)
U-Nat metallic	233
UO_2	207
UF_4	179
$UO_2(NO_3)_2 6H_2O$	111
UO_3	204
U_3O_8	203
UO_2F_2	176
$Na_2U_2O_7$	167

Absorbed dose rate measured through a 7 mg cm^{-2} filter from
infinitely thick slabs of materials.

Method M.V3.5.2   Radiological Properties of Uranium and Progeny

Table V3.5.11 gives the radiological properties of uranium.

Table V3.5.11. Radiological Properties of ^{234}U, ^{235}U, and ^{238}U and Their Radioactive Progeny

Radionuclide	Half-life (y)	Energies (MeV) and Abundance of Major Radiations		
		Alpha	Beta (Max)	Gamma
Primary U Nuclides				
^{238}U	4.51×10^9	4.15, 21%		
		4.2, 79%		
^{235}U	7.1×10^8	4.21, 6%		0.144, 11%
		4.37, 17%		0.163, 5%
		4.40, 55%		0.186, 57%
		4.6, 5%		0.205, 5%
^{234}U	2.47×10^5	4.72, 28%		0.053, 0.12%
		4.77, 72%		
U Progeny				
^{234}Th	6.60×10^{-2}		0.13, 21%	0.013, 9.8%
			0.193, 79%	0.063, 3.5%
				0.092, 3%
				0.093, 4%
^{234m}Pa	2.23×10^{-6}		2.29, 98%	0.765, 0.30%
				1.001, 0.6%
^{231}Th	2.91×10^{-3}		0.206, 13%	0.026, 2%
			0.287, 12%	0.085, 10%
			0.288, 37%	
			0.305, 35%	

Method M.V3.5.3   Neutron Fluence Rates from UF_6

The interaction of alpha particles from uranium with the nuclei of fluorine and other low-A atoms generates 2 MeV neutrons. The neutron fluence rate varies with the enrichment and the chemical form of uranium. In the case of UF_6 the neutron dose equivalent rate (measured with a 9-inch-diameter, spherical, BF_3 survey meter) for cooled storage cylinders is (DOE01b):

Natural to 5% enrichment   0.01-0.2 mrem per hour

Very high enrichment (97%+)   2-4 mrem per hour (contact)

                                                  1-2 mrem per hour (3 feet)

Method M.V3.5.4   Physical and Chemical Properties of Uranium Compounds

A number of chemical forms of uranium are used in uranium fuel facilities. Table V3.5.12 summarizes some physical properties and chemical properties of various uranium compounds.

Table V3.5.12. Physical and Chemical Properties of Uranium Compounds

Compound	Melting Point (Degrees C)	Density (g cm^{-3}) Crystal/ Particle	Density (g cm^{-3}) Bulk	Solubility in Water at Ambient Temperature	ICRP-30 Inhalation Class	Comments
UF$_6$	64.1	5.1	5.1	Decomposes to UO$_2$F$_2$	D	Volatile, white crystalline solid
UF$_4$	960	6.7	2.0-4.5	Very slightly soluble	W	A green solid
UO$_2$F$_2$	Decomposes to U$_3$O$_8$ at 300			Very soluble	D	Brilliant orange hygroscopic solid
U$_3$O$_8$	Decomposes to UO$_2$ at 1300	8.3	1.5-4.0	Insoluble	W	Stable, olive green solid
UO$_2$	2878	10.96	2.0-5.0	Insoluble	W	Most common form used for fuels
U Metal	1132	19.05	19	Insoluble	Y	Heavy, silvery white metallic element. Powder ignites spontaneously in air at ambient temperatures.

Table V3.5.13. Early Phase Deposition, Air Submersion Dose, and Intake to DCFs

Uranium Enrichment	Deposition External Exposure Rate ECF_{gi}	Deposition External EDE Rate DCF_{gi}	Deposition Four Day Dose ($R_S = 5.2 \times 10^{-7}$ m^{-1}) Resuspension DCF_{EPgi}	Air Submersion External EDE Rate DCF_{ai}	Intake Dose Based on Air Concentration $DCF_{e,50}$
	mr/hr per μCi/m^2	mrem/hr per μCi/m^2	mrem per μCi/m^2	mrem/hr per μCi/m^3	mrem/hr per μCi/m^3
Depleted U [a]	3.17×10^{-4}	2.26×10^{-4}	2.82×10^{-1}	2.25×10^{-2}	2.94×10^{3}
Natural U [a]	2.03×10^{-4}	1.45×10^{-4}	2.95×10^{-1}	2.31×10^{-2}	3.08×10^{3}
Enriched 5% [a]	9.60×10^{-5}	6.86×10^{-5}	3.08×10^{-1}	2.38×10^{-2}	3.21×10^{3}
Enriched 93% [a]	7.26×10^{-5}	5.19×10^{-5}	3.10×10^{-1}	2.39×10^{-2}	3.23×10^{3}
Depleted U [b]	3.63×10^{-5}	2.59×10^{-5}	N/A	1.49×10^{-3}	N/A
Natural U [b]	4.69×10^{-5}	3.35×10^{-5}	N/A	1.99×10^{-3}	N/A
Enriched 5% [b]	6.33×10^{-5}	4.52×10^{-5}	N/A	2.78×10^{-3}	N/A
Enriched 93% [b]	6.55×10^{-5}	4.68×10^{-5}	N/A	2.88×10^{-3}	N/A

[a] Assumes secular equilibrium with progeny
[b] Assumes no equilibrium

Table V3.5.14. Intermediate Phase

Uranium Enrichment	First Year Relocation DCF_{Rgi} $(R_S=2.74 \times 10^{-9}m^{-1})^c$ mrem/µCi m^{-2}	Second Year Relocation DCF_{Rgi} $(R_S=1.37 \times 10^{-9}m^{-1})^d$ Mrem/µCi m^{-2}	Relocation Dose Conversion Factor for Inhalation of Resuspended Material DCF_{Rai} mrem per µCi m^{-3}
Depleted U [a]	7.0×10^{-1}	3.5×10^{-2}	2.6×10^{7}
Natural U [a]	7.4×10^{-1}	3.7×10^{-2}	2.7×10^{7}
Enriched 5% [a]	7.8×10^{-1}	3.9×10^{-2}	2.8×10^{8}
Enriched 93% [a]	7.8×10^{-1}	3.9×10^{-2}	2.8×10^{8}
Depleted U [b]	N/A	N/A	N/A
Natural U [b]	N/A	N/A	N/A
Enriched 5% [b]	N/A	N/A	N/A
Enriched 93% [b]	N/A	N/A	N/A

[a] Assumes secular equilibrium with progeny
[b] Assumes no equilibrium
[c] Resuspension factor at day 365, $1 \times 10^{-6}/365$ (reference NCRP99)
[d] Resuspension factor at day 730, $1 \times 10^{-6}/730$ (reference NCRP99)

SECTION 6. RADIOISOTOPIC THERMOELECTRIC GENERATOR ACCIDENT

TABLES

RADIOISOTOPIC THERMOELECTRIC GENERATOR ACCIDENT

6.1   Description of a Generic RTG Accident

This introduction defines the scenario and outlines the DQOs for Radionuclide Thermoelectric Generator (RTG) accidents.

6.1.1   Scenario Description

Nuclear power sources have been used for several decades to meet the thermal and electrical energy requirements for some spacecraft. Because of mission power and longevity requirements, U.S. mission planners have relied exclusively on the use of static radionuclide converters, usually referred to as RTGs. RTGs use the heat from the radioactive decay of a radionuclide to produce electricity through a thermoelectric converter. The most commonly used radionuclide fuel for RTGs is ^{238}Pu, due to its relatively high heat-to-mass ratio and long half-life of 87.7 years. ^{238}Pu is primarily an alpha emitter with some low-level gamma radiation. Neutrons are also emitted at low levels due to spontaneous decay and alpha-neutron reactions.

Spacecraft carrying RTGs can incur several types of accidents, including a first-stage accident at launch, orbital decay resulting in re-entry to the earth's atmosphere, and re-entry at higher-than-orbital velocities during a fly-by maneuver for deep space missions. The fuel used in RTGs consists of ^{238}Pu dioxide in ceramic pellets. The duel pellets are packed in side graphite impact shells (2 pellets per shell). Two shells are then placed inside a carbon reentry shell comprising a general-purpose heat source. RTGs are designed to contain their fuel under most accident conditions; however, releases can occur as a result of impact with concrete or steel during a launch accident or as a result of impact on rock following re-entry. An earth gravitational assist inadvertent re-entry represents the most severe accident scenario to which an RTG would be subjected. It would lead to a range of fuel-end states that include intact or damaged modules, intact graphite shells, and fuel released at high altitude in both particulate and vapor form.

An RTG typically contains 18 general-purpose heat sources. Total ^{238}Pu content of an RTG is about 4.8×10^{15} Bq (0.13 MCi). The principal dose mechanism following an RTG accident would be internal dose from inhalation of released or resuspended ^{238}Pu.

6.1.2   Data Quality Objective Process

The seven steps of EPA's DQO process are applied to the RTG accident scenario in the discussion below. Application detail is minimal in the manual but will develop as incident-specific work proceeds. A complete and formal DQO treatment is not expected until the Recovery Phase, specifically at the beginning of long-term monitoring.

6.1.2.1   DQO Step 1 – State the Problem

The key element of step 1 of EPA's DQO process, State the Problem, is addressed by the scenario description above. The remaining elements, which are not scenario specific, are covered in the organizational overview of the Assessment Group, Volume 1 of the *FRMAC Assessment Manual*. These elements include team composition, customer/decision maker interface, and resources. This overview is primarily presented in the Introduction. Team members and roles are described in the Introduction, and Administration, Internal Procedures and Tools. The specific identity of the decision maker(s) is dynamically defined by consultations between the FRMAC Manager and representatives of the Lead Federal Agency (LFA) and state(s). As part of the FRMAC team, the Assessment Group is responsible for providing the assessments of measurements and predictions necessary for decision-makers to protect the public and emergency workers from excessive exposure to radioactive materials. The *FRMAC Assessment Manual* is the technical basis for these assessments. Assessment results are interpretations of measurements in terms of published PAGs, or as otherwise directed by the LFA and Advisory Team. Should the *FRMAC Assessment Manual* not provide the information needed to address a specific issue, then technical experts outside the FRMAC will be enlisted.

6.1.2.2   DQO Step 2 – Identify the Decisions

The key element of step 2, Identify the Decisions, is the enumeration of the major protective actions and their respective "triggers." The FRMAC does not make PARs, but it does identify those areas where specific actions may be technically warranted. It may also identify potential mitigating measures. The decision-maker is expected to consult with the staff to develop alternative actions for each decision.

6.1.2.2.1 POTENTIAL CONSEQUENCES

The potential consequences of an RTG accident are the result of releases from an explosion at launch or the breakup of an RTG upon contact with hard objects following orbital decay and re-entry. These releases consist of ^{238}Pu in a ceramic, oxide form.

RTG fuel sources are expected to survive re-entry as intact modules that impact independently. Small, localized releases are expected only if impact with hard objects occurs. In addition, the modular nature of RTGs may result in limited releases in multiple areas. ^{238}Pu has a high specific activity (alpha) with some low-level gamma radiation and low-level neutrons due to spontaneous fission and alpha-neutron reactions in the fuel. The primary radiation hazard associated with RTG releases involves the inhalation of ^{238}Pu particles during plume passage or as a result of resuspension.

6.1.2.2.2 POTENTIAL ACTIONS

The Assessment Group provides decision-makers with the technical basis for protective actions
(radiological assessment). The major potential actions are those necessary to reduce risk due to
exposure to acceptable levels. The actions are prioritized to address the most serious and time-
sensitive potential effects first.

The major protective actions include (generally prioritized):
- Expedited evacuation where potential for early health effects exists
- Evacuation of immobile populations (*e.g.*, hospitals, prisons)
- Evacuation of general public
- Sheltering of public and immobile populations
- Identification/treatment of potential early health effect victims
  (*i.e.*, identification of unevacuated areas where the population may have the potential for
  early health effects so that these individuals may be identified and treated, as needed)
- Relocation of unevacuated populace to avoid future risk
- Suspension of agricultural production
- Condemnation of foods

Other decisions might include:

- Exposure planning for emergency workers
- Selection of measurements and monitoring locations
- Guidelines for re-entry
- Identification and selection of mitigation options

6.1.2.3   DQO Step 3 – Inputs to the Decisions

6.1.2.3.1 INFORMATIONAL INPUTS

Because the decisions are very time sensitive, particularly early in an accident, the radiological
assessment must proceed with whatever quantity and quality of data are available. Most initial
decisions will be based on plume models. Initial measurements will be used to validate or
renormalize the model. As quickly as possible, sufficient measurements must be acquired to
replace dependence on the model. As time progresses and decisions become less time-critical,
the quantity and quality of data will improve. Eventually guidelines will be implemented on the
collection and analysis of measurements and models will become interpolation tools.

During the Early (and much of the Intermediate) Phase of the accident, the assessment methods
and reference data provided in the *FRMAC Assessment Manual* should be sufficient for the
radiological assessment. Default decision levels (DRLs) presented in Table V3.6.2. are to be used
until sufficient data have been collected to eliminate assumptions. Revision of a DRL is

acceptable only if an assumption can be eliminated. Several revisions may occur over the course of time as assumptions are eliminated.

Normally, the gamma exposure rates from radioactive material dispersed as a result of an RTG accident will be too low for use in identifying where urgent protective actions and relocation are warranted. As a result, DRLs for plutonium contamination levels must be used to determine where evacuation and relocation from contaminated areas are warranted in accordance with the EPA PAGs.

6.1.2.3.2 MEASUREMENT AND PREDICTION INPUTS

Table V3.6.1 outlines the measurements and model results in relative order of necessity.

Table V3.6.1. Measurements and Predictive Inputs

Predictions	Plume Passage – CEDE
	Post Plume Passage – CEDE, resuspension
Field Measurements	AMS quick-look survey
	Handheld alpha survey meters
	FIDLER measurements
	Infrared measurements
Sample Analyses	Soil samples
	Air samples
	Crop samples
	Water samples

6.1.2.3.3 COMPLICATING FACTORS

The environmental data must be representative in order to be a valid basis for revising the DRLs. The environmental data may not be representative because:

- The footprint impacted by an RTG may cover an area several hundred kilometers long and up to 50 kilometers wide in the case of an orbital reentry.

- Widespread contamination is not expected, however, the modular nature of RTG fuel may result in limited releases in multiple areas.

- Attenuation of alpha particles will make the detection and measurement of dispersed material using handheld alpha survey instruments difficult.

- Resuspension varies.

- The deposition density can be very complex, varying a by factor of 10 or more over short distances.

6.1.2.4   DQO Step 4 - Boundary of Consideration

6.1.2.4.1 PHYSICAL BOUNDARY

The area for which assessments are needed is the entire area impacted by the plume as well as sufficient area outside of that to ensure that the extent of the area affected by the plume can be defined. Initially the extent will be the areas potentially subject to evacuation or relocation. After these concerns are addressed, the limits will be extended to the surrounding agricultural production area where the FDA PAGs may be exceeded. It may also be necessary to include food processing facilities well outside the affected region, to which contaminated foods may have been transported. For an RTG accident involving orbital reentry, the most difficult task will be the determination of areas which are impacted. Widespread contamination is not expected. Any contamination released would be expected to be localized; however, the potential affected area may cover several thousand km^2.

6.1.2.4.2 TEMPORAL BOUNDARY

The FRMAC Assessment tools provided here are sufficient for the appraisal of doses during the Early and a significant portion of the Intermediate Phase.

Detection of conditions that could lead to the possible re-entry of a spacecraft carrying RTGs will usually occur days to months in advance, and tracking of the system with projection of possible impact locations and debris footprints will be initiated. Response assets (*e.g.*, RAP, FRMAC) will be activated and possibly predeployed to locations near the projected impact area; however, deployment to the actual impact site will not occur until after re-entry and impact.

Although the life cycle of the accident response continues through the Recovery Phase, the scope of the treatment in the *FRMAC Assessment Manual* is intended to be valid only until the end of the Intermediate Phase. At that time, a Recovery Plan for long-term monitoring will be created. During that period, only portions of this manual may remain applicable.

6.1.2.4.3 CONSTRAINTS

Some of the potential constraints on measurements include:

- Deposition on snow cover
- Deposition on a leaf canopy
- Delays in monitoring due to adverse weather
- Access denied by property owners

- Inaccessible terrain

- Potential releases over a very large area

6.1.2.5   DQO Step 5 - Decision Rules

The Assessment Group does not establish decision rules nor make PARs. However, published PAGs are used by the Assessment Group as decision rules for the interpretation of measurements and predictions. These PAGs are implemented as DRLs. If a measurement exceeds a specific DRL by any margin, then that location fails the test at hand. If a measurement falls short of a DRL by any margin, then it passes the test at hand.

Derived Response Levels have been defined for:

- Emergency Worker Turn-back limits

- Evacuation based on EPA Early Phase PAG following plume passage

- Relocation (1st year), plus 2nd and 50-year long-term objectives

- Agricultural hold based on deposition

- Food condemnation (agricultural embargo) based on food concentration

- Water condemnation based on concentration

The PAGs and computational approach may be altered by the Federal Advisory Team.

6.1.2.6   DQO Step 6 Tolerance Limits

Assessors must establish tolerable levels of uncertainty when calculating DRLs. For example, evacuation, shelter, and agricultural product holds have a higher tolerance level than re-entry, which is higher yet than relocation, and so on. It is up to the assessor to establish these tolerable levels until a more definitive uncertainty analysis can be performed.

Sensitivities of measurements must always be adequate to detect the DRL level for the question at hand. The acceptable uncertainties are listed below.

If assessments are being used for:

- Evacuation, sheltering or agricultural hold - the tolerance limit is a factor of 10.

- Re-entry considerations - the tolerance limit is a factor of 2.

- Relocation - the tolerance limit will be negotiated, but is expected to be approximately 30%.

- Return - the tolerance limit will be negotiated and will likely be much smaller.

- Condemning foods or water - the same criteria used by USDA for evaluation of non-radiological contamination will be applied (10%).

6.1.2.7   DQO Step 7 Optimal Design

FRMAC can do little initially to optimize design, which is primarily the responsibility of the
EPA under its management of FRMAC during the Recovery Phase.

6.2   Default Derived Response Levels (DRLs)

Table V3.6.2 presents default DRLs for assessment of releases from accidents involving RTGs
containing ^{238}Pu fuel. The default DRLs are calculated using initial assumptions (*e.g.*,
resuspension factors, fraction retained on food). These defaults should be recalculated as soon as
the appropriate inputs (*e.g.*, ^{238}Pu ground density, resuspension) have been determined based on
environmental measurements. The goal is to eliminate as many assumptions as possible from the
calculation of the DRLs.

Table V3.6.2. Default DRLs for Releases from RTGs

Issue	Marker	DRL	Sensitivity, Uncertainty, Spatial Density, Assumptions
Worker Protection[a]	Predicted TEDE	Stay time	
EPA Early Phase PAG (evacuation)	^{238}Pu ground density	37 μCi/m^2	Ground density (μCi/m^2) indicating that evacuation or substantial shelter could be implemented in accordance with the EPA PAGs
Relocation 1st year	^{238}Pu ground concentration	29 μCi/m^2	Ground density (μCi/m^2) from deposition indicating that the population should be relocated in accordance with EPA PAGs
Relocation 2nd year	^{238}Pu ground concentration	62 μCi/m^2	Ground density (μCi/m^2) from deposition indicating that the population should be relocated in accordance with EPA PAGs
Relocation 50th year	^{238}Pu ground concentration	21 μCi/m^2	Ground density (μCi/m^2) from deposition indicating that the population should be relocated in accordance with EPA PAGs
Ingestion PAG	^{238}Pu ground density	20 Bq/m^2, 540 pCi/m^2	^{238}Pu ground density indicating that consumption of locally produced food or milk should be restricted in accordance with HHS DILs
	^{238}Pu concentration in food	2 Bq/kg, 54 pCi/kg	^{238}Pu concentration in food indicating that consumption should be restricted in accordance with HHS guidance
Ingestion PAG	Grazing Cow Deposition	1.4 μCi/m^2	^{238}Pu ground density capable of producing milk ingestion dose equal to PAG in limiting age group
Ingestion PAG	Cow Forage Concentration	9.8×10^{-1} μCi/kg	^{238}Pu in cow forage capable of producing milk ingestion dose equal to PAG in limiting age group

Ingestion PAG	Cow Water Concentration	8.2×10^{-1} μCi/L	^{238}Pu concentration in water capable of producing milk ingestion dose equal to PAG in limiting age group

(a) Administrative stay-time limits based on estimates for CEDE ($H_{e,50}$) from inhalation.

6.3   Worker Protection

6.3.1   Discussion of Assessment in Worker Protection

As indicated in Volume 1, the major responsibility of the Assessment Group in worker protection is to provide information for determining the turn-back guidance for emergency workers. For incidents involving RTGs, this guidance is based on dose limits (CEDE) but must be presented in a manner that is useful to field personnel. For most incidents, turn-back guidance is presented in terms of exposure rate (as measured on a hand-held instrument) or integrated dose (as measured on a self-reading or electronic dosimeter). For RTG incidents, the gamma exposure rates from dispersed radioactive material (^{238}Pu) are too low to be used for assessing adherence to emergency worker turn-back limits. As a result, administrative "stay-time" limits based on estimates for CEDE ($H_{e,50}$) from inhalation should be developed. The stay-time limits should be based on conservative model projections during the Early Phase of the incident and should be refined based on environmental samples and measurements. Once the Early Phase is over, the total dose incurred (during the Early Phase) must be confirmed before an emergency worker is allowed to perform activities that may result in additional dose.

EPA has recommended dose limits (L_{ew}) for individuals performing various types of emergency services (Table V3.6.3). These dose limits are in terms of TEDE and include CEDE from inhalation and external dose from penetrating radiation. Since ^{238}Pu released during an RTG accident presents primarily an internal radiation hazard, the CEDE from inhalation will be compared to the Emergency Worker Turn-Back Limits.

Table V3.6.3.   Default Federal Emergency Worker Dose Limits and Turn-back Guidance for Events Involving RTGs

	Turn-Back Guidance Expressed in Terms of Total Effective Dose Equivalent (TEDE), mrem
Emergency Activity	
All	5,000
Protecting Major Property	10,000
Life Saving or Protecting Large Populations	25,000
Life Saving or Protecting Large Populations [1]	>25,000

[1] Only on a voluntary basis to personnel fully aware of the risks involved.

6.4   Early (Plume) Phase

As indicated in Volume 1, the Early Phase is considered to last until the release has ended and areas of major contamination have been identified. For dose assessment purposes, the Early Phase is usually considered to last for about 96 hours. The EPA evacuation PAG is 1 to 5 rem, where the dose is the sum of the EDE from external sources and the CEDE incurred from significant inhalation pathways. In an accident involving RTGs containing ^{238}Pu, the inhalation dose from plume passage is the most important consideration for dose assessment. Since the time and area of impact of an RTG cannot be accurately predicted, evacuation of populations will usually not be conducted prior to re-entry. The low probability of dispersal of ^{238}Pu and the likelihood that affected areas will be of limited size make sheltering the most likely initial protective action. The emphasis for the Early Phase is in identifying those areas that have not been evacuated where the Early Phase dose may exceed the PAG.

6.4.1   Default DRL for Evacuation

The DRL for the evacuation PAG is given in terms of ^{238}Pu ground concentration. Given that the Early Phase lasts for about 96 hours and the EPA evacuation PAG of 1 to 5 rem, the default value for the evacuation DRL for an RTG accident is 37 μCi/m^2 as presented in Table V3.6.2. This is calculated by dividing the evacuation PAG of 1 rem by the Early Phase DCF for ^{238}Pu in Table 3.5 of the *FRMAC Assessment Manual*, Volume 2 (2.7×10^1 mrem/μCi/m^2).

6.4.2   Revision of Evacuation DRL

The evacuation DRLmay be revised according to Method M.3.4 in Volume 1 based on an assessment of resuspension.

6.5   Intermediate Phase -- Relocation

As indicated in Volume 1, the Intermediate Phase is the period of time which begins after any releases have been brought under control and reliable measurements are available to use for determining additional PARs. It continues until the additional protective actions are terminated. The major protective actions taken during the Intermediate Phase involve relocation and restrictions on the use of contaminated food and water.

The EPA guidance (EPA92) states that relocation is warranted if the dose from gamma exposure from deposition and inhalation of resuspension is projected to be greater than 2000 mrem the first year (or beta skin dose 50 times higher). Dose reduction due to part-time occupancy and decontamination are not to be considered. Dose reduction due to decay and weathering is to be included and has been considered in the dose factors used in the manual. Due to the nature of the

radioactive decay of ^{238}Pu, the exposure rate cannot be used as the DRL for locating areas where relocation is warranted following an RTG accident. As a result, the areal density of ^{238}Pu must be used to develop a relocation DRL that will change temporally (due to decay and weathering) and spatially and must be re-evaluated periodically.

EPA guidance also established objectives to ensure the dose in the second year does not exceed 500 mrem and the cumulative dose over 50 years does not exceed 5000 mrem. Due to the 87.4 y half-life of ^{238}Pu, meeting the long-term objectives will be difficult. Decontamination of the areas or continued exclusion of the public from contaminated areas may be necessary to meet the long term objectives. The process for determining whether these objectives are being met will be developed as part of the long term assessment and is beyond the scope of this Manual.

6.5.1   Default Relocation DRLs

To help determine areas where the PAG may be exceeded, DRLs can be defined that correspond to the first year, second year and 50-year doses. For an RTG accident, the DRL for relocation must be expressed in terms of areal concentration of ^{238}Pu on the ground. The default deposition concentration DRLs for relocation are 29.0 μCi/m^2 for the first year, 62 μCi/m^2 for the second year, and 21 μCi/m^2 for the 50-years. The default DRL values may need to be adjusted if the state in which the FRMAC is operating has a relocation PAG different from the EPA PAG. In that case, follow Method M.4.1 of Volume 1 to make the adjustment.

6.5.2   Revision of Relocation DRLs

The DRLs are a function of the mixture of radionuclides in the deposition. Samples should be taken and analyzed to assure that the values used in the calculations are representative of the entire affected area.

6.5.2.1   Criteria

The DRLs should be re-evaluated:

- When resuspension is evaluated.

- When significant changes in resuspension occur.

The characteristics of the release and changing meteorological conditions may dictate that a single value for each type of DRL will not be appropriate for the entire affected area.

6.5.2.2   Procedure

The default deposition DRL for an RTG accident is calculated using Method M.4.1 and is based on the default resuspension function described in NCRP99, which is 10^{-6} for the first day, 10^{-6}/t (where t is time in days) out to 1000 days, and 10^{-9} after 1000 days. Actual measurements of

resuspension should be conducted in accordance with Method M.4.7 (Volume 1), and the DCFs and associated DRLs should be recalculated in accordance with Method M.4.2 (Volume 1).

6.6   Intermediate Phase – Ingestion

As indicated in Volume 1, the Intermediate Phase is the period of time that begins after any releases have been brought under control and reliable measurements become available for use in determining additional PARs. It continues until the additional protective actions are terminated. The major protective actions taken during the Intermediate Phase involve relocation and restrictions on the use of contaminated food and water. The FDA issued recommendations regarding contaminated food in 1998 (FDA98). Key points in these recommendations were the DILs, concentrations in food at which some action should be taken to limit or preclude the use of the food product.

Food may become contaminated through several pathways. This section considers the simplest case, direct deposition onto produce (*e.g*., leafy vegetables), as well as the grass-cow-milk-man pathway.

6.6.1   Default Ingestion DRLs

The FDA recommendations contain limiting concentrations (*i.e*., DILs) for five groups of radionuclides in food. The five DILs are applied separately, and for the RTG case only the DIL for the nuclide group of ^{238}Pu, ^{239}Pu and ^{241}Am would be applicable. This DIL is 54 pCi/kg (2 Bq/kg). Since the food concentrations of ^{238}Pu cannot be measured directly in the field, we use DRLs that indicate the DILs may be approached or exceeded. The default DRLs are calculated using Method M.5.8. The default DRLs are presented in Table V3.6.4.

Table V3.6.4. Default DRLs for Ingestion of ^{238}Pu

Pathway	Bq/m^2	pCi/m^2	μCi/m^2
Fresh Produce	20	5.4×10^2	5.4×10^{-4}
Grass-Cow-Infant	5×10^4	1.4×10^6	1.4

6.6.2   Revision of Ingestion Derived Response Levels

The ingestion DRLs for an RTG event are a function of the relationship between ^{238}Pu deposition density and food concentration. The default DRLs are based on assumed values for productivity and retention. Temporal and spatial variations in deposition rates, retention and the possibility of multiple releases with different release characteristics can result in different relationships

between the ^{238}Pu deposition density and food concentration. Samples should be taken and analyzed to assure that the values used in the calculations are representative of the entire affected area.

6.6.2.1   Criteria

The DRLs should be re-evaluated:

- When the actual area affected by the release is known.

- When site-specific transfer factors have been determined.

- Weekly for the first month to account for major changes in the composition of the deposition due to weathering.

- Monthly thereafter, until weathering no longer has a major impact.

6.6.2.2   Procedure

The areal concentration ingestion DRLs for an RTG accident are calculated using Method M.5.8.

6.7   DRL Revision

Revisions to DRLs may be made when assumptions can be eliminated or when necessitated by decay or changing conditions. The frequency required to compensate for decay depends on the makeup of the release and the particular DRL under consideration. For the case of an RTG accident, the half-life of ^{238}Pu (87.7 years) makes the correction of DRLs or data due to radioactive decay unnecessary during the time frame associated with a FRMAC response. Significant changes can occur due to weathering and changes in resuspension.

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SECTION 7. DOMESTIC NUCLEAR EXPLOSION

This section has not yet been developed. The Department of Energy's Consequence Management organization has been asked to develop this material during FY02. After development it will be posted as an addendum until another revision of the Assessment Manual is published.

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FRMAC Assessment Manual

Volume 3

Section 8. Radiological Dispersal Device Accident

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FRMAC is an acronym for Federal Radiological Monitoring and Assessment Center.

SECTION 8. RADIOLOGICAL DISPERSAL DEVICE ACCIDENT

CHARTS

TABLES

ABBREVIATIONS AND ACRONYMS

ALARA   as low as reasonably achievable

ALI   annual limit of intake

CEDE   committed effective dose equivalent

DCF   dose conversion factor

DRL   derived response level

DTPA   SEE PAGE 21.

ECF_{gi} exposure rate conversion factor from ground deposition

ECF_{TEDE} exposure to TEDE conversion factor

ERAD   explosive release atmospheric dispersion

ETI   early transient incapacitation

GM   Geiger-Mueller

HPGe   high-purity germanium

PAG   protective action guide

RDD   radiological dispersal device

RPF   respiratory protection factor

RTGs   radioisotopic thermoelectric generators

TEDE   total effective dose equivalent

WGPu   weapon-grade plutonium

RADIOLOGICAL DISPERSAL DEVICE ACCIDENT

8.1   Introduction

This report evaluates the radiation doses that may result from radiological dispersal devices (RDDs) that are likely to be constructed from the more common commercially available sealed-source radionuclides. It is intended to be of use during the first one to two days after an RDD detonation. During this time exposure rate, gross alpha (α), and gross beta (β) radiation data may be available; however the radionuclide identities, dispersal estimates, and deposition estimates are not available. Once the radionuclides are identified, better guidance can be accessed using approved software-based assessment tools. The contents of this section define how the FRMAC resources anticipate responding to the DHS National Planning Scenario #11 (Radiological Attack – Radiological Dispersal Devices).

It is assumed that an RDD will incorporate one or more substantial industrial or medical sources. This assumption eliminates most radionuclides from consideration, leaving γ-emitters (^{137}Cs, ^{60}Co, and some short-lived nuclides), ^{90}Sr, and ^{241}Am as the most likely radionuclides to be used in an RDD.

Other than ^{137}Cs, γ-emitters are not considered in detail because assuming that an unknown γ-emitter is ^{137}Cs will generally provide a reasonable dose estimate during the Early Phase of an emergency. Although there are more radiotoxic γ-emitters, such as ^{131}I, they are not generally available in sufficient quantities for a large-scale RDD.

^{90}Sr (a β-radiation emitter) is considered in detail. It has essentially no γ emissions, presents detection difficulties, and causes a high radiation dose per unit intake. It is a common constituent of currently manufactured industrial and medical sources, as well as high activity radioisotopic thermoelectric generators (RTGs).

Neither ^{239}Pu nor ^{238}Pu is likely to be accessible in quantities that are sufficient to produce a large-scale RDD. The consequences of ^{238}Pu and ^{239}Pu dispersal are addressed in NNSA03b (2003) in the sections titled "Radioisotopic Thermoelectric Generator Accident" and "Nuclear Weapon Accident."

^{241}Am was evaluated in more detail than the other α-emitters because it is considered to be the most available in relatively large quantities and it causes a high radiation dose per unit intake. ^{241}Am has been used extensively in the manufacture of neutron sources and well-logging sources.

The operational guidelines task group (OGT) has developed complimentary guidance to this pre-assessed scenario. The OGT guidance chooses several of the anticipated dispersal radionuclides and evaluates the potential doses from reoccupation and post-event use of varying facilities or community areas.

8.2   Modes of Dispersal

RDDs are generally assumed to disperse radionuclides by explosion. This pre-assessed scenario concentrates on explosive dispersal. It is recognized, however, that fires, aerosolization devices, and direct contamination might also distribute radionuclides.

8.2.1   Fire

Fire is capable of distributing radioactive contamination within structures to some degree. Airborne release fractions tend to be low, however, unless the radionuclide is volatile (such as tritium, for instance). The recommended release fractions are shown in Table 8-1 (simplified from DOE94a, page A-16, 1994).

Table 8-1. Simplified Airborne Release Fractions and Respirable Fractions Due to Fire

Material	Percent Airborne
Tritium and other volatile radioactive materials	100
Non-volatile radionuclides	1

8.2.2   Insecticide Fogging and Crop-Dusting Equipment

Insecticide-fogging vehicles and crop-dusting aircraft are means of dispersing radioactive materials.

8.2.3   Dispersal Into Buildings

Radioactive materials could be dispersed into buildings using aerosol generators or by directly contaminating the floor. Table 8-2 illustrates that 1 mCi of selected radioactive material can contaminate significant areas to the levels given in NRC Regulatory Guide 1.86 (NRC74, 1974). These surface contamination levels do not create any risk of acute health effects or a significant risk of latent health effects (cancers); however, these levels might be judged to require mitigation eventually.

Table 8-2. The Area That Can Be Contaminated by a 1 mCi Source

Radionuclide	Maximum Area Contaminated to NRC Regulatory Guide 1.86 Values, m^2	NRC Regulatory Guide 1.86 Limit for Removable Contamination, dpm/ 100 cm^2
^{241}Am, ^{244}Cm, ^{252}Cf, ^{238}Pu, weapon-grade Pu (WGPu), ^{226}Ra, ^{228}Ra	1,100,000	20
^{210}Po	22,000	1,000
^{137}Cs, ^{60}Co, ^{192}Ir	22,000	1,000

Radionuclide	Maximum Area Contaminated to NRC Regulatory Guide 1.86 Values, m^2	NRC Regulatory Guide 1.86 Limit for Removable Contamination, dpm/ 100 cm^2
^{90}Sr	110,000	200

8.2.4   Malicious Irradiation

Malicious irradiations that lead to serious acute radiation injuries or deaths are a possible concern, especially with γ-emitting sources. An example would be placing a radioactive source under a bench in a park. Malicious irradiations are not addressed in this document.

8.3   Radionuclides of Concern

A recent RDD evaluation document, LANL03 (2003), published two figures that summarize the radionuclides used in sealed sources, their applications, and relative sizes. These figures are reproduced as Figures 8-1 and 8-2.
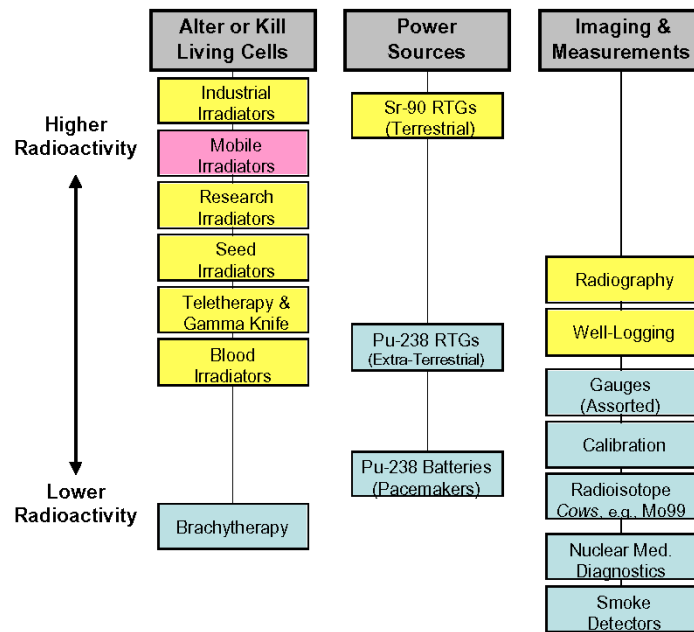
Figure 8-1. Types and Sizes of Sealed Sources (LANL03, 2003)
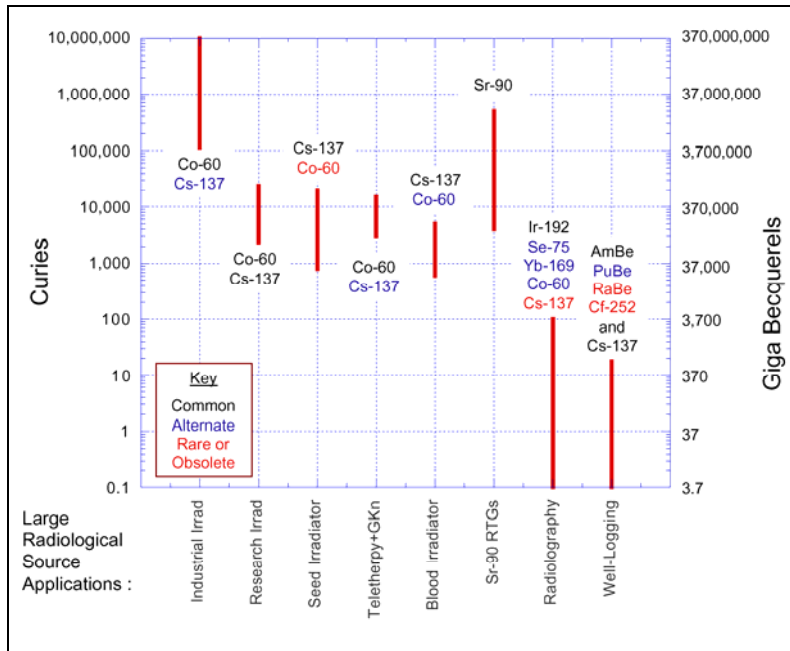
Figure 8-2. Radionuclides Used in Sealed Sources, by Application (LANL03, 2003)

Table 8-3 summarizes the radioactive sources that might be used in RDDs, based on ANS03 (2003). Additional information on each of these radionuclides is provided in subsequent sections.

Table 8-3. Relative Importance of Various Exposure Pathways

Radionuclide	Inhalation		Direct Exposure		Ingestion
	Plume	Deposited	Plume	Deposited	
^{137}Cs	High	Low	Low	High	Low
^{60}Co					
^{192}Ir					
^{252}Cf		High	Low	Low	High
$^{226, 228}$Ra					
^{241}Am					
^{238}Pu					
WGPu					
^{210}Po					
^{244}Cm					
^{90}Sr					Medium

8.3.1   γ- and βγ-emitting Radionuclides

This group of radionuclides includes ^{137}Cs, ^{60}Co, and ^{192}Ir (ANS03, 2003). These radionuclides are primarily an external γ radiation hazard.

8.3.1.1   Self-Protection of γ and βγ Sources

Large doses of γ radiation that are received in a short period of time produce early transient incapacitation (ETI) (DOA89, 1989). ETI typically occurs within 5 to 10 minutes after test animals receive doses of 1000 rad or more of γ radiation. This makes large γ radiation sources self-protecting to some degree, because persons receiving high doses would quickly become ineffective at performing complicated tasks or fine work.

Table 8-4 lists approximate source activities that are self-protective due to the high γ radiation hazard. At the exposure rates expected from these sources, onset of ETI is likely within minutes (and death within days). These levels preclude extensive work with the source unless it is thoroughly and carefully planned and protective measures are in place. Table 8-4 values should not be considered exact thresholds that clearly distinguish easily handled sources from sources that are difficult to handle.

Persons who are working with large sources of γ-emitters may attempt to reduce external dose by using shielding (such as lead or concrete) or remote handling devices or by dividing the work among several individuals. Respirators, glove bags, or fume hoods might be employed to reduce inhalation dose to persons who plan to breach any radiation source.

Table 8-4. Self-Protective γ Sources

Radionuclide	No Shielding	Limited Improvised Shielding for Head and Body
^{60}Co	2000 Ci	4000 Ci
^{192}Ir	6000 Ci	12,000 Ci
^{137}Cs	9000 Ci	17,000 Ci

Substantial γ sources are readily transportable if they are shielded. A lead shield that is sufficient to protect passengers from a 40,000 Ci ^{60}Co source for hours or days is light enough to be transported in a conventional automobile. It may not be feasible, however, to transport such sources with enough shielding to pass through a high-sensitivity vehicle radiation monitor, even if a light truck were used. Less shielding would be required to provide a given degree of attenuation for ^{137}Cs or ^{192}Ir sources, owing to their lower energy γ spectra.

8.3.1.2   Assumptions for Unknown γ-Emitter

In the first hours after an RDD detonation, the identity of any high-energy γ-emitters may not be known. Under these circumstances, assume that the high-energy γ-emitter is ^{137}Cs. The reasons for this recommendation follow:

- The most likely large high-energy, γ-emitter sources to be incorporated into an RDD are assumed to be ^{60}Co, ^{137}Cs, or ^{192}Ir because of their wide use in medical, radiography, well-logging, and sterilizer sources. These radionuclides are similar in that they are primarily an inhalation hazard in the direct plume. After plume passage, the deposited radioactivity is primarily an external radiation hazard. Initially mistaking one of these radionuclides for another is of little consequence.

- ^{131}I and ^{99}Mo are possible γ-emitting sources, but most facilities (such as hospitals) that use them do not keep an inventory of more than a few curies and they are short-lived. Thus there would be logistical difficulties in amassing of the materials at one place and time to create a major RDD release.

- If ^{241}Am is the γ-emitter, then air samples and deposited material would exhibit very high α activities.

Tables 8-5, 8-6, and 8-7 provide physical properties and typical applications of ^{137}Cs, ^{60}Co, and ^{192}Ir, respectively (HACKER01, 2001; LANL03, 2003; and MIIS04, 2004).

8.3.1.2.1 CESIUM-137

^{137}Cs is typical of the βγ group of radionuclides; it is evaluated in greater detail in subsequent sections. ^{137}Cs is typically in equilibrium with a short-lived βγ-emitter, ^{137m}Ba, and the symbol '^{137}Cs' is understood to refer to the equilibrium mixture in this pre-assessed scenario except where noted otherwise.

Table 8-5. Properties of ^{137}Cs and the More Common ^{137}Cs Sources

Half-life	30.1 years
Decay Mode	β
Principal a Emissions (KeV)	661 (85%) from ^{137m}Ba
Typical Uses	**Size Range (Ci)**
Industrial Sterilization	170,000– 3,000,000
Blood Irradiators	7,000 – 15,000
Medical Diagnosis and Therapy	3 – 1,400
Calibration	<1– 60
Level and Conveyor Gauges	3– 5
Well Logging	2

8.3.1.2.2 COBALT-60

^{60}Co is a commonly used radionuclide in industrial and medical irradiators. ^{60}Co emits more photons and higher energy photons than ^{137}Cs. Table 8-6 provides properties of ^{60}Co sources.

Table 8-6. Properties of ^{60}Co and the More Common ^{60}Co Sources

Half-life	5.27 years
Principal γ Emissions (KeV)	1173 (100%), 1332 (100%)
Typical Uses	Size Range (Ci)
Industrial Sterilization	30,000– 4,000,000
Blood Irradiators	2,400 – 25,000
Medical Diagnosis and Therapy	10 – 7,000
Calibration	<1 – 20
Level and Conveyor Gauges	5

8.3.1.2.3 IRIDIUM-192

^{192}Ir is widely used in radiography and medical sources. Table 8-7 provides properties of ^{192}Ir sources.

Table 8-7. Properties of ^{192}Ir and the More Common ^{192}Ir Sources

Half-life	74 days
Decay Mode	B
Principal γ Emissions (KeV)	317 (82%), 468 (48%), 308 (30%), 295 (29%), Others
Typical Uses	Typical Size Range (Ci)
Industrial Radiography	30 – 100
Medical Diagnosis and Therapy	6

8.3.1.3   Detection of γ-emitters

Appreciable deposition levels can be detected and measured in the field by relatively insensitive equipment such as conventional ion chambers and steel wall or energy-compensated Geiger-Mueller (GM) detectors.

Near-background levels are commonly detected in the field by plastic scintillators, pancake GM detectors, inorganic γ scintillators, thin window proportional detectors, or high-purity germanium (HPGe) detectors.

8.3.2   α-emitting Radionuclides

This group of radionuclides includes ^{241}Am, ^{252}Cf, ^{244}Cm, ^{238}Pu, WGPu, ^{210}Po, ^{228}Ra, and ^{226}Ra. Of these radionuclides, ^{210}Po and radium are the least toxic via the inhalation pathway. ^{241}Am is used as the representative nuclide for an unknown α-emitter in subsequent scenario evaluations and data tabulations because it is the most available transuranic α-emitter.

8.3.2.1   Self-Protection of α-emitting Radionuclide Sources

Most α sources are not significantly self-protective. Individuals can handle large α-emitting radionuclide sources (such as ^{241}Am, ^{238}Pu, or WGPu) for extended periods of time without developing radiation sickness. However inhalation of large amounts of these materials, as may occur in a casual work environment, can result in severe lung disease and death after a latent period of weeks to years.

Persons who plan to breach an α-emitting radionuclide source may use respirators, a glove bag, or an improvised fume hood to reduce inhalation dose and extend working times with these materials.

8.3.2.2   Detection of α-emitting Radionuclides

Detection of α particles that are emitted from contaminated surfaces can be accomplished using thin window proportional detectors, α scintillation detectors, and pancake GM detectors. Intact sealed sources may not exhibit any α radiation, however.

^{241}Am, WGPu, ^{238}Pu, and ^{244}Cm may be detected in the field with detectors that are reasonably efficient at detecting x-rays and low-energy γ-rays. These radionuclides are most available as α-n (AmBe, PuBe, etc.) neutron sources. As such they should have a high-enough neutron emission rate to be easily detected by neutron detectors. In addition any large WGPu source will also emit detectable neutrons owing to spontaneous fission of ^{240}Pu.

^{252}Cf and ^{244}Cm sources have high spontaneous fission rates, so the sources that contain them should exhibit appreciable γ radiation and neutron radiation.

In general airborne concentrations of α-emitting radionuclides must be characterized by collecting air samples on filters and then measuring the activity in a counter. Exposure rate measurements for airborne plumes of α-emitting radionuclides by themselves are not sensitive enough to provide prudent radiation protection guidance.

^{241}Am is likely the only common transuranic α-emitting radionuclide that provides enough γ radiation to make exposure rates a useful indicator of deposited radioactivity in the Early Phase.

8.3.2.3   Availability, Typical Uses, and Characteristics

Table 8-8 characterizes α-emitting radionuclides and their uses. Tables 8-9 through 8-17 detail each major radionuclide (HACKER01, 2001; LANL03, 2003; and MIIS04, 2004).

Table 8-8. Objects That Contain Significant Quantities of α-emitting Radionuclides

Product	Radionuclide	Subject to a Specific License or DOE Control in the United States?
RTG	^{238}Pu	Yes
Well Logging or Neutron Source	^{241}Am, ^{244}Cm, ^{252}Cf, ^{238}Pu, ^{239}Pu; formerly ^{226}Ra was used	Yes
Smoke Detector	^{241}Am (≤ 1 μCi)	No
Static Eliminator	^{210}Po (≤ 500 μCi)	No
Scrap Radio-Luminescent Items	^{226}Ra (μCi range)	No

8.3.2.3.1 CALIFORNIUM-252

Use of ^{252}Cf in well logging sources is an uncommon or obsolete practice (LANL03, 2003).

Table 8-9. Properties of ^{252}Cf and the More Common ^{252}Cf Sources

Half-life	2.6 years
Decay Mode	α, spontaneous fission (0.12 neutron/decay)
Typical Uses	Typical Size (Ci)
Well Logging	0.03

8.3.2.3.2 RADIUM

Radium was formerly used in a variety of applications; however it was always an expensive radionuclide and only limited amounts were ever extracted from ore materials. There are probably not a large number of sizeable sources of radium available for incorporation into an RDD. It is no longer used to any degree in the manufacture of luminescent items and other α-emitting radionuclides are usually preferred in the manufacture of neutron sources. Enough radium can readily be accumulated from secondary markets to contaminate a significant area within a structure, however.

Table 8-10. Properties of ^{226}Ra + Daughters and ^{228}Ra + Daughters

Half-life	^{226}Ra: 1,600 years ^{228}Ra: 5.75 years	Comments
Decay Mode	α, β	These radionuclides were once used in neutron sources, diagnosis and therapy, and luminescent paint. However, other radionuclides have largely replaced radium.
Principal γ Emissions (KeV)	Numerous	

8.3.2.3.3 AMERICIUM-241

Smoke detectors that contain small amounts of ^{241}Am are readily available to the general public in the United States. ^{241}Am is also a common constituent of neutron sources.

Table 8-11. Properties of ^{241}Am and the More Common ^{241}Am Sources

Half-life	432 years	Comments
Decay Mode	A	^{241}Am neutron sources are used for well logging and other purposes. Smoke detectors that contain up to 1 μCi of ^{241}Am are exempt from licensing in the United States.
Principal γ Emissions (KeV)	59.5 (36%), 13.9 (43%)	
Typical Uses	Size Range (Ci)	
Well Logging (AmBe)	Up to 20	
Calibration	Up to 10	
Smoke Detectors	10^{-6}	

8.3.2.3.4 PLUTONIUM-238

^{238}Pu is much more difficult to detect in the field than either ^{241}Am or WGPu. Section 6 of NNSA03b (2003), "Radioisotopic Thermoelectric Generator Accident," is largely concerned with ^{238}Pu. Plutonium-238 RTGs are highly valued devices that are generally closely controlled and they have seen little use in industry. A Plutonium-238 RTG detonation is assumed to be unlikely, so it is not addressed in this pre-assessed scenario.

Cardiac pacemakers containing ^{238}Pu have been manufactured and used, but they were never common.

Table 8-12. Properties of ^{238}Pu and the More Common ^{238}Pu Sources

Half-life	88 years	Comments
Decay Mode	α	RTG containing ^{238}Pu are used in space programs and little else. These sources should be closely controlled.
Principal γ Emissions	13.6 KeV (11.6%)	
Typical Uses	Size Range (Ci)	

Radioisotope Thermoelectric Generator (NASA95)	1,500 – 260,000	[238]Pu neutron sources are used for well logging and other purposes. [238]Pu pacemakers were produced in limited numbers.
PuBe Neutron Sources	10	
Cardiac Pacemakers	1	

8.3.2.3.5 PLUTONIUM-239

[239]Pu is usually a synonym for WGPu, which is about 93% [239]Pu by mass. The composition of WGPu varies with its age. Mass and activity percentages for 30-year-old WGPu are provided in Table 8-13. Gamma emission rates for 30-year-old WGPu are provided in Table 8-14. Thirty-year-old WGPu emits about 1/7th the amount of γ radiation per α disintegration as [241]Am. Applications and decay properties are provided in Table 8-15. Composition and gross α dose conversion factors (DCFs) exposure rate conversion factors for ground deposition (ECF$_{gi}$) for WGPu based on ICRP77 (1977) are provided in Table 8-16. These factors change slowly with time, depending on the age of the WGPu.

Further information on WGPu is provided in Section 3 of NNSA03b (2003), "Nuclear Weapons Accident."

Table 8-13. Physical Data for 30-Year-Old WGPu

Radionuclide	Mass (%)	Activity (%)	Half-life (Yr)
[238]Pu	0.03	2	88
[239]Pu	93	25	24,100
[240]Pu	6	6	6,540
[241]Pu	0.1	61	14.3
[242]Pu	0.04	0.001	376,000
[241]Am	0.4	6	432

Table 8-14. γ Emission Rates for 30-Year-Old WGPu

γ Energy (MeV)	Photons/sec/gram WGPu
0.0136	1.73E+08
0.0139	2.32E+08
0.0263	1.31E+07
0.0332	5.77E+05
0.0543	2.64E+05
0.0553	9.45E+04

γ Energy (MeV)	Photons/sec/gram WGPu
0.0595	1.95E+08
0.0692	9.75E+05
0.1129	1.02E+06
Totals	6.16E+08

Table 8-15. Additional Properties of WGPu and the More Common WGPu Sources

Half-life	24,100 years	Comments
Decay Mode	α and β	^{239}Pu neutron sources have been used for well logging and other purposes.
Typical Uses	Size Range (Ci)	
PuBe Neutron Sources	10	WGPu emits low-energy β-particles that are undetectable with the instruments that are commonly available to first responders.

Table 8-16. Activity Fraction ^{241}Am and Gross-α Conversion Factors for WGPu

Age, Yr	Activity Ratio: $\frac{^{241}Am}{Gross\ \alpha}$	$\frac{DCF_{e,50i}}{mrem\ m^3}$ $\frac{}{\mu Ci\ \alpha\ hr}$	$\frac{ECF_{gi}}{mR\quad m^2}$ $\frac{}{\mu Ci\ \alpha\ hr}$	$\frac{DCF_{ai}}{mrem\ m^3}$ $\frac{}{\mu Ci\ \alpha\ hr}$
0	0.00	5.87E+05	6.69E-06	2.82E-05
10	0.09	5.56E+05	3.88E-05	9.98E-04
20	0.14	5.40E+05	5.57E-05	1.51E-03
30	0.16	5.30E+05	6.52E-05	1.79E-03
40	0.18	5.25E+05	7.05E-05	1.96E-03
50	0.18	5.22E+05	7.34E-05	2.04E-03
60	0.19	5.20E+05	7.50E-05	2.09E-03

8.3.2.3.6 POLONIUM-210

Table 8-17 provides information on ^{210}Po. Polonium sources have a short shelf life and large ones are no longer in wide use.

Table 8-17. Properties of ^{210}Po and the More Common ^{210}Po Sources

Half-life	138 days
Decay Mode	A

Principal γ Emissions	None
Typical Uses	
RTG and PoBe neutron sources.	Formerly used in large amounts in neutron sources and RTG. Due to the short half-life of ^{210}Po, the current activities of most of these sources are insignificant.
Po static eliminators.	^{210}Po static elimination devices having an activity of less than 500 µCi are exempted from licensing and are readily available in the United States.

8.3.3   β-emitting Radionuclides

^{90}Sr is the most common of the essentially pure β-emitting radionuclides. It is used in industrial, medical, and RTG applications. It is typically in equilibrium with ^{90}Y; the equilibrium mixture of the two radionuclides is often referred to as ^{90}SrY. As the term is used in this document, two curies of ^{90}SrY consist of 1 curie of ^{90}Sr and 1 curie of ^{90}Y. Figures in this pre-assessed scenario that refer to an unknown β-emitting radionuclide are based on this ^{90}SrY activity convention.

8.3.3.1   Self-Protection of ^{90}SrY Sources

^{90}SrY sources are generally weak γ-emitting radionuclides and sealed multicurie-sized sources can be handled without appreciable penetrating radiation exposure.

Strontium-Yttrium sources in RTGs are often in the 100,000-curie range and these sources may exhibit an exposure rate of 400 to 800 R/hour at a distance of 0.5 m (BELLONA03, 2003). The exposure rate is primarily due to bremsstrahlung x-rays, which result from the interaction of β-radiation with the materials of construction. RTG capsules have high-enough exposure rates that protective facilities and planning would be required for extensive handling. Strontium-Yttrium RTG sources can certainly be handled for short periods of time, however.

Much less shielding is required to transport β-emitting radionuclide sources than γ-emitting radionuclide sources of a given activity. Consequently, very large ^{90}SrY sources can be transported in a conventional automobile without endangering the occupants.

8.3.3.2   Detection of β-emitting Radionuclides

β-emitting radionuclides are difficult or impossible to distinguish from low-energy γ-emitting radionuclides with common field equipment. Large β-emitting radionuclide sources and high deposition levels of β-emitting radionuclides are easily detected in the field with relatively insensitive equipment such as conventional ion chambers and steel wall or energy-compensated GM detectors. Low levels of β-emitting radionuclides can be detected in the field by plastic scintillators and pancake GM detectors. Inorganic γ-scintillators that have thin windows may also respond to β-emitting radionuclides.

8.3.3.3 ^{90}SrY

Properties and uses of ^{90}SrY are provided in Table 8-18 (HACKER01, 2001; LANL03, 2003; and MIIS04, 2004).

Table 8-18. Properties of ^{90}SrY and ^{90}SrY Sources

Half-life	28.6 years	Comments
Decay Mode	B	^{90}SrY has been used in large quantities in RTG in the United States and in the former Soviet Union. Many of these RTG were used to supply electric power in remote locations and for military applications.
Principal γ Emissions	Bremsstrahlung X-rays.	
Typical Uses	**Typical Size (Ci)**	
RTG	20,000 – 400,000	
Calibration Sources	Up to 1	
Intravascular Brachytherapy Source	0.8	
Thickness Gauge	0.05	

8.4   Charts for Early-Event First Responders

This section provides charts useful to local first responders, i.e., law enforcement and fire crews. These tables are the result of a realistic scenario analysis performed by Sandia National Laboratories.

Table 8-19 provides a summary of nuclides of concern, the form of the nuclide, the size of the source used in the analysis, and the justification for the source size (HARPER05). If the responder can identify the type of radiation present, source size can be estimated. Likewise, if intelligence estimates the application used in the RDD, nuclide and source size can be estimated.

The primary exposure pathways for RDDs are groundshine, inhalation, and deposition on the skin, hair, and clothes. The relative importance of the pathways depends on the material and the device geometry. Cobalt is primarily a localized groundshine problem, Strontium is primarily an inhalation problem, and Cesium could be either a groundshine or inhalation problem depending on the device design. A summary of the results from the effects analysis discussed in the previous section is presented in Table 8-20. The hazard boundaries from the devices and scenarios that resulted in the most conservative boundary were selected for this table (HARPER05).

Observations from Table 8-20:

- The area of highest concern is limited to the area within 500 m of the release in the more probable scenarios. The area of highest concern is defined as the area in which acute effects, a lifetime inhalation dose of 1 Sv (100 rem) or a 50 mSv (5 rem) groundshine dose (5-hour exposure), might occur. This implies that the initial response should be to set up a Hot Zone, or boundary, within 500 m (if nothing is known about the release) or at 0.01 Gy/h (1 rad/h) if exposure rate measurements are available (HARPER05).

- The inhalation hazard boundaries for the selected dose levels were minimal for the basic engineering scenarios (including the large source scenarios). Precautions should still be taken to reduce the inhalation dose to as low as reasonably achievable (ALARA). There could be localized areas with high concentrations of respirable aerosol during plume passage. (Models tend to underpredict the maximums and overpredict the minimums.) However, for the basic engineering cases, any significant inhalation dose would come from plume passage within a few hundred meters of the release. In this area, the plume would probably arrive within 10 minutes (and would be gone by the time most of the early first responders arrive). This implies that for the most probable scenarios, early emergency response should focus on the groundshine exposure and contamination and treat inhalation as a lower priority ALARA issue (HARPER05).

- Areas further downwind may meet the EPA guides for Intermediate and Late Phases and require additional intervention after the Emergency Phase is assessed (depending on the device) (HARPER05).

Table 8-20 was developed using the explosive release atmospheric dispersion (ERAD) model, which does not include the effect of buildings on urban aerosol transport. Initial studies suggest that although the localized dispersion patterns are modified significantly by the presence of buildings, the hazard boundaries in Table 8-20 should not be altered to account for urban aerosol transport (HARPER05).

Tables 8-19 and 8-20 incorporate an experimentally based bi-modal particle-size distribution.

Table 8-19. Summary of Sensitivity Studies Performed (HARPER05)

Nuclide	Primary Radiation Type (half-life)	Primary Form	Size of Source for Calculation, in GBq (Ci)	Application That Forms the Basis for Size of Source
^{90}Sr	Beta (28.6 years)	Ceramic (SrTiO$_3$)	1.11×10^7 GBq (300,000 Ci)	Large radioisotopic thermal generator (RTG) (Russian IEhU-1)
^{137}Cs	Beta + Ba-137m Gamma (30.17 years)	Salt (CsCl)	7.4×10^6 GBq (200,000 Ci)	Irradiator
^{60}Co	Beta, Gamma (5.27 years)	Metal	1.11×10^7 GBq (300,000 Ci)	Irradiator
^{238}Pu	Alpha (87.75 years)	Ceramic (PuO$_2$)	4.92×10^6 GBq (133,000 Ci)	RTG used for the Cassini Saturn space probe
^{241}Am	Alpha (432.2 years)	Pressed ceramic powder (AmO$_2$)	7.4×10^2 GBq (20 Ci)	Single well-logging source

Nuclide	Primary Radiation Type (half-life)	Primary Form	Size of Source for Calculation, in GBq (Ci)	Application That Forms the Basis for Size of Source
^{252}Cf	Alpha (2.64 years)	Ceramic (Cf_2O_3)	7.4×10^2 GBq (20 Ci)	Several neutron radiography or well-logging sources
^{192}Ir	Beta, Gamma (74.02 days)	Metal	3.7×10^4 GBq (1000 Ci)	Multiple industrial radiography units
^{226}Ra	Alpha (1600 years)	Salt ($RaSO_4$)	3.7×10^3 GBq (100 Ci)	Old medical therapy sources

Table 8-20. Range of Specified Hazard Boundaries from Point of Release
(from Realistic Scenario Analysis) (HARPER05)

Selected Dose Limit	Significance of Selected Dose Limit	Realistic RDD Hazard Boundaries for Varying Device Designs		
		Intermediate Size Source[a]	Very Large Source[b]	Very Large Source[b], Sophisticated Engineering
Groundshine dose of 1 Gy (100 rad), 24-hour exposure assumed	Acute groundshine threshold – Lower level where one might see deterministic effects	0	~ 300 m	~ 300 m
Inhalation dose of 1 Gy (100 rad) to the bone marrow (30-day committed dose)	Acute hematopoietic syndrome threshold dose from inhalation	0	0	~ 200 m
Inhalation dose of 2.7 Gy (270 rad) to the lung (30-day committed dose)	Acute pneumonitis threshold dose from inhalation	0	0	~ 2 km
Lifetime inhalation dose of 1 Sv (100 rem) (50-year committed dose)	Chronic radiation sickness threshold	0	0	~ 7 km
50 mSv (5 rem) groundshine dose (5-hour exposure assumed)	Level at which all emergency workers can work unrestricted for 5 hours	~ 100 m	~ 600 m	~ 600 m
10 x ALI for inhalation (Annual Limit of Intake)	Use of Prussian Blue for Cs internal or DTPA for transuranic internal dose is highly recommended	0	0	< 10 km
500 mSv (50 rem) (50-year committed dose)	Evacuation is suggested	< 150 m	< 1 km	~15 km
50 mSv (5 rem) (50-year committed dose)	Sheltering is suggested	< 600 m	< 3.3 km	< 100 km
10 mSv (1 rem) (50-year committed dose)	EPA suggests protective actions initiated	2 km	~10 km	> 100 km
20 mSv (2 rem) in one year	EPA prescribes relocation	8 km	~ 100 km	> 100 km

Selected Dose Limit	Significance of Selected Dose Limit	Realistic RDD Hazard Boundaries for Varying Device Designs		
		Intermediate Size Source[a]	Very Large Source[b]	Very Large Source[b], Sophisticated Engineering
– derived deposition limit				

[a]Intermediate Size Source: 3.7×10^5 GBq (10000 Ci), Basic Engineering

[b]Very Large Size Source: 7.4×10^6 GBq (200000 Ci), Basic Engineering

Notes: 1) Deterministic effects are defined as health effects for which thresholds exist and for which severity varies with dose.

2) Hazard boundaries of 0 mean that the selected dose limit was not observed for any of the scenarios.

8.5   Charts for Dose Limitation and Estimation

This section provides a series of charts intended for use when an RDD detonation has occurred, but the identities of specific radionuclides are unknown. If the responder can provide an estimate of the relative amounts of α- and β-activity, these charts can be used to estimate a number of useful factors. Once the radionuclides are identified, better guidance will be readily available from TurboFRMAC and other assessment tools; use of these charts should then be discontinued.

Figures intended primarily for the protection of workers include:

- Figures 8-3 and 8-5: Maximum worker exposure rates *after* the direct plume has passed that correspond to a TEDE of approximately 5000 mrem in an hour under low-dust conditions.

- Figures 8-4 and 8-6: Maximum worker exposure rates *after* the direct plume has passed that correspond to a TEDE of approximately 5000 mrem in an hour under high-dust conditions.

Figures intended primarily for the protection of the public include:

- Figures 8-7 and 8-8: Exposure rate derived response levels (DRLs) for evacuation and sheltering, for use *after* the plume has passed to estimate exposure rates that correspond to a TEDE of 1 rem in 96 hours from inhalation plus external dose. Values are provided for both high-dust and low-dust conditions.

- Figures 8-9 through 8-12: Deposition DRLs for evacuation and sheltering, for use *after* the plume has passed to estimate deposition levels that correspond to a TEDE of 1 rem in 96 hours from inhalation plus external dose. Values are provided for both high-dust and low-dust conditions.

Studies with mock RDDs indicate that usually *most* of the particulate that is produced will be non-respirable and may be expected to be deposited locally. Consequently, figures are provided for both the near-field and far-field. First responders should consider the term 'near-field' as the

area within a 500-m radius of the detonation. Studies indicate that the area impacted during the Emergency Phase by an explosive RDD where acute health effects are possible, as well as lesser affected areas that have levels of contamination that meet or exceed the EPA PAGs for evacuation, can be assumed to be bounded within a 500-m radius (MUSOLINO06). It is possible that this area may be considerably smaller than 500 m; however this value has been chosen for conservatism. Conversely, areas outside a 500m radius should be considered 'far-field' by first responders while interpreting these figures.

Values provided by the near-field and far-field figures differ due to particle size of the radionuclide. As stated earlier, larger particulate which may be non-respirable is expected to be deposited locally; thus exposure rate in the near-field is due primarily to external exposure. Respirable particles are expected to be deposited further away from the detonation site; thus far-field exposure rate accounts for exposure due to external exposure and inhalation.

Figures 8-3 through 8-12 are based on the respirable fraction of aerosols released in an RDD event, considering a reasonable bi-modal particle size distribution. Table 8-21 describes the dispersion modeling for these figures.

Table 8-21. Scenarios Used For Dispersion Modeling, Figures 8-3 through 8-12

Scenario	Source	Size	Respirable			Non-respirable			10 lb C4 Explosive
			Median (µm)	GSD	Fraction	Median (µm)	GSD	Fraction	
1	^{137}Cs	10,000 Ci	2	3	0.5	200	3	0.5	
2	^{90}Sr RTG	100,000 Ci	2	3	0.1	250	3	0.9	
3	^{241}Am (AmBe)	200 Ci	2	3	0.35	100	3	0.65	

Figures 8-3 through 8-12 incorporate a ground roughness factor of 0.82 for external exposure and dose.

Figures 8-3 through 8-6 are based on the respiratory protection factors described in Table 8-22.

Table 8-22. Respiratory Protection Factors Against Radioactive Particulate

Respirator Type	Filter Type	Protection Factor (RPF, unitless)
No Respirator	NA	1
Half-face Air-Purifying Respirator	Any cartridge designated as P100, N100, or R100.	10
Full-face Air-Purifying Respirator		100
Positive Pressure Self-Contained Breathing Apparatus	NA	1000

All of the figures are scaleable. The results in Figures 8-3 through 8-6 are scaleable for other Emergency Worker Dose Limits and exposure times. The Emergency Worker Dose Limit is

usually 5 rem (5000 mrem), although higher values are justifiable in special circumstances provided that a lower dose limit is not practical (EPA91, 1991). Examples of such exceptions include:

- Protecting valuable property, 10 rem, and

- Life-saving or protection of large populations, 25 rem.

On a voluntary basis emergency workers may receive higher doses for activities involving life-saving or protecting large populations (EPA91, 1991).

The results in Figures 8-7 through 8-12 are scaleable for protective action guides (PAGs) other than 1000 mrem. The EPA Evacuation/Sheltering PAG is in the range of 1 to 5 rem (1000 to 5000 mrem) TEDE for the Early Phase. A 1000 mrem PAG for the Early Phase was used in the construction of DRL charts, Figures 8-7 through 8-12.

When interpreting the charts, note that:

- The charts assume that a first responder will be able to estimate approximate percentages of alpha, beta, and gamma activity present.

- All charts involving an α-emitting radionuclide are based on ^{241}Am.

- All charts involving a β-emitting radionuclide are based on ^{90}SrY.

- The horizontal axis (% β-activity) of all figures involving a γ-emitting radionuclide are based on ^{137}Cs in equilibrium with ^{137m}Ba.

- The vertical axis (μCi/m^2) of Figure 11 is based on ^{137}Cs without ^{137m}Ba .

- Exposure rate values are estimates.

- The x-axis of the chart is not necessarily 0 (zero). If a curve falls below the x-axis, it is below a detectible level.

Dose estimation for plume submersion conditions has not been considered in this evaluation. EPA PAGs also do not consider plume submersion, as the goal of these studies is to reduce avoidable dose following the plume passage. These studies have been developed to indicate what actions may be taken following passage of the plume.

8.5.1   Worker Protection

The charts in this section are primarily intended for protection of workers.

After plume passage, the principal exposure pathways of concern to emergency response personnel will be direct irradiation (ground shine) and inhalation of resuspended material. The ECF$_{TEDE,}$ as defined in Equation 1, relates TEDE to measured exposure rates in this situation.

$$\text{ECF}_{\text{TEDE}} = DEXP + \frac{R_s \times \sum_i C_i \times DCF_{e,50i}}{RPF \times \left(\frac{0.82}{0.7}\right) \sum_i C_i \times ECF_{gi}}$$

$$\left(mrem/mR\right) = \left(mrem/mR\right) + \frac{\left(m^{-1}\right) \times \sum \left(\mu Ci/m^2\right) \left(\frac{mrem/hr}{\mu Ci/m^3}\right)}{\left(unitless\right) \times \left(\frac{unitless}{unitless}\right) \sum \left(\mu Ci/m^2\right) \left(\frac{mR/hr}{\mu Ci/m^2}\right)}$$

(1)

In Equation 1:

- R_s is the resuspension factor (m^{-1}), 10^{-4} is used for high-dust conditions, and 10^{-6} is assumed to represent low-dust conditions;

- $DCF_{e,50i}$ is the committed effective dose conversion factor from inhalation $\left(\frac{mrem/hr}{\mu Ci/m^3}\right)$ from Table 3.3 of the FRMAC Manual, Volume 2 (NNSA03a, 2003).

- RPF is the respiratory protection factor from Table 8-22 (unitless);

- ECF_{gi} is the exposure rate conversion factor for ground deposition ($\frac{mR/hr}{\mu Ci/m^2}$) from Table 3.4 of the FRMAC Assessment Manual, Volume 2 (NNSA03a, 2003);

- C_i is the deposited concentration of radionuclide "i" ($\mu Ci/m^2$);

- DEXP is the dose to exposure factor. The value 0.7 mrem/mR is the nominal conversion from exposure to effective dose equivalent to deep tissue (bone marrow); and

- The factor $\frac{0.82}{0.7}$ corrects ECF_{gi} values from NNSA03a (2003) to reflect a ground roughness factor of 0.82.

The suggested maximum exposure rates for a 1-hour entry are calculated using Equation 2.

$$Max \quad Exposure \quad Rate = \frac{Emergency \ \text{Worker} \ Dose \ Limit / Duration}{\text{ECF}_{\text{TEDE}}}$$

$$\left(mR/hr\right) = \frac{\left(mrem\right)/\left(hr\right)}{\left(mrem/mR\right)}$$

(2)

Figures 8-3, 8-4, 8-5, and 8-6 are based on Equation 2. These figures provide suggested maximum worker exposure rates for emergency workers needing to limit dose to 5000 mrem (5 rem) during a 1-hour entry. These charts are scaleable for other emergency worker dose limits and exposure times. The emergency worker dose limit is usually 5 rem (5000 mrem), although

higher values are justifiable in special circumstances, provided that a lower dose limit is not practical (EPA91, 1991). Examples of such exceptions include:

- Protecting valuable property, 10 rem; and

- Life-saving or protection of large populations, 25 rem.

On a voluntary basis emergency workers may receive higher doses for activities involving life-saving or protecting large populations (EPA91, 1991).

Figures 8-3 and 8-4 may be rather conservative at locations within a few hundred meters of the point of detonation because much of the locally deposited material will be of a large particle size, which is not easily resuspended.

Use Figure 8-3a for:

- FAR-FIELD
- The incident involves βγ- and α-emitting radionuclides,
- Low-dust conditions,
- Exposure rates can be measured, and
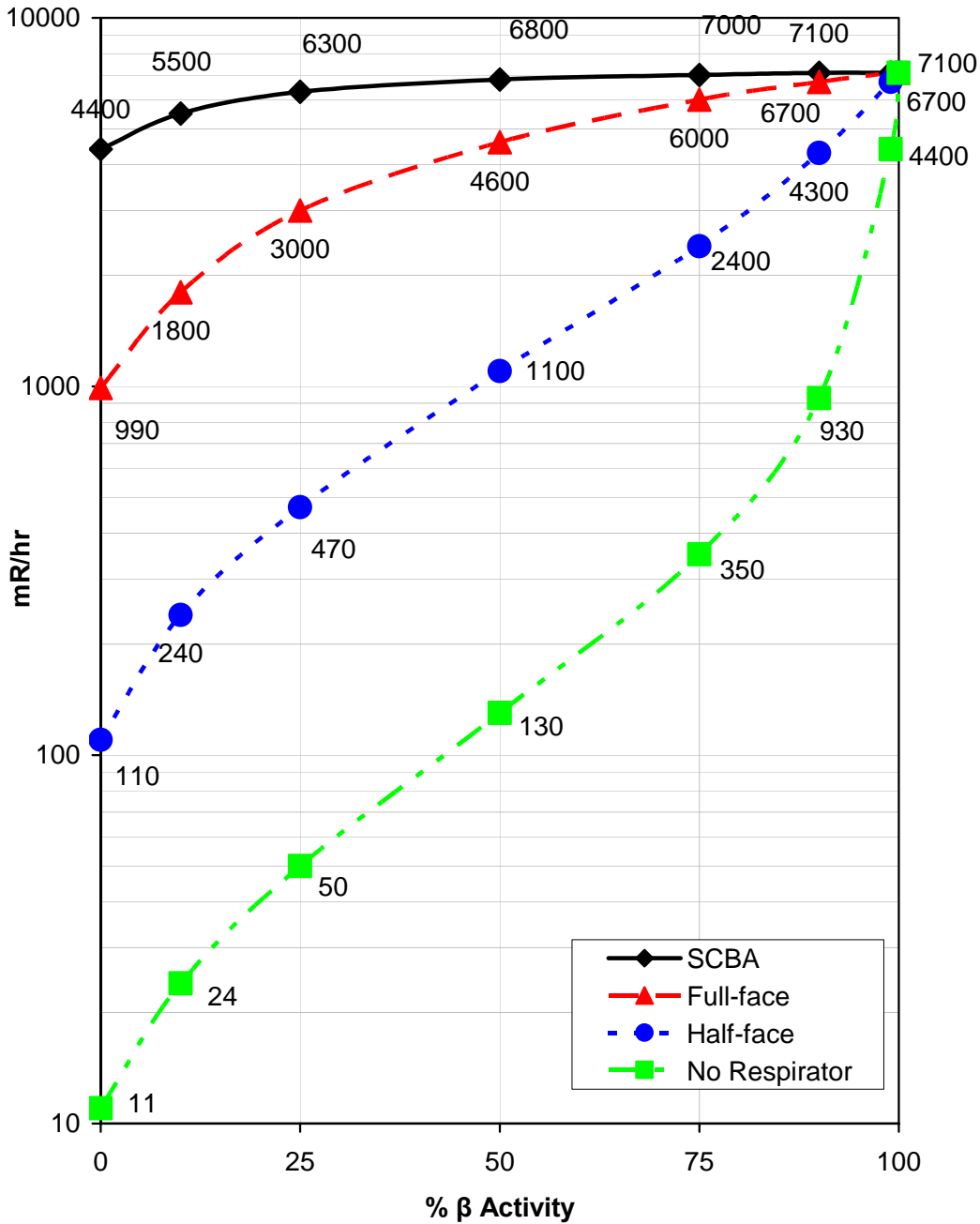- The fractions of α- and β-activity can be estimated.
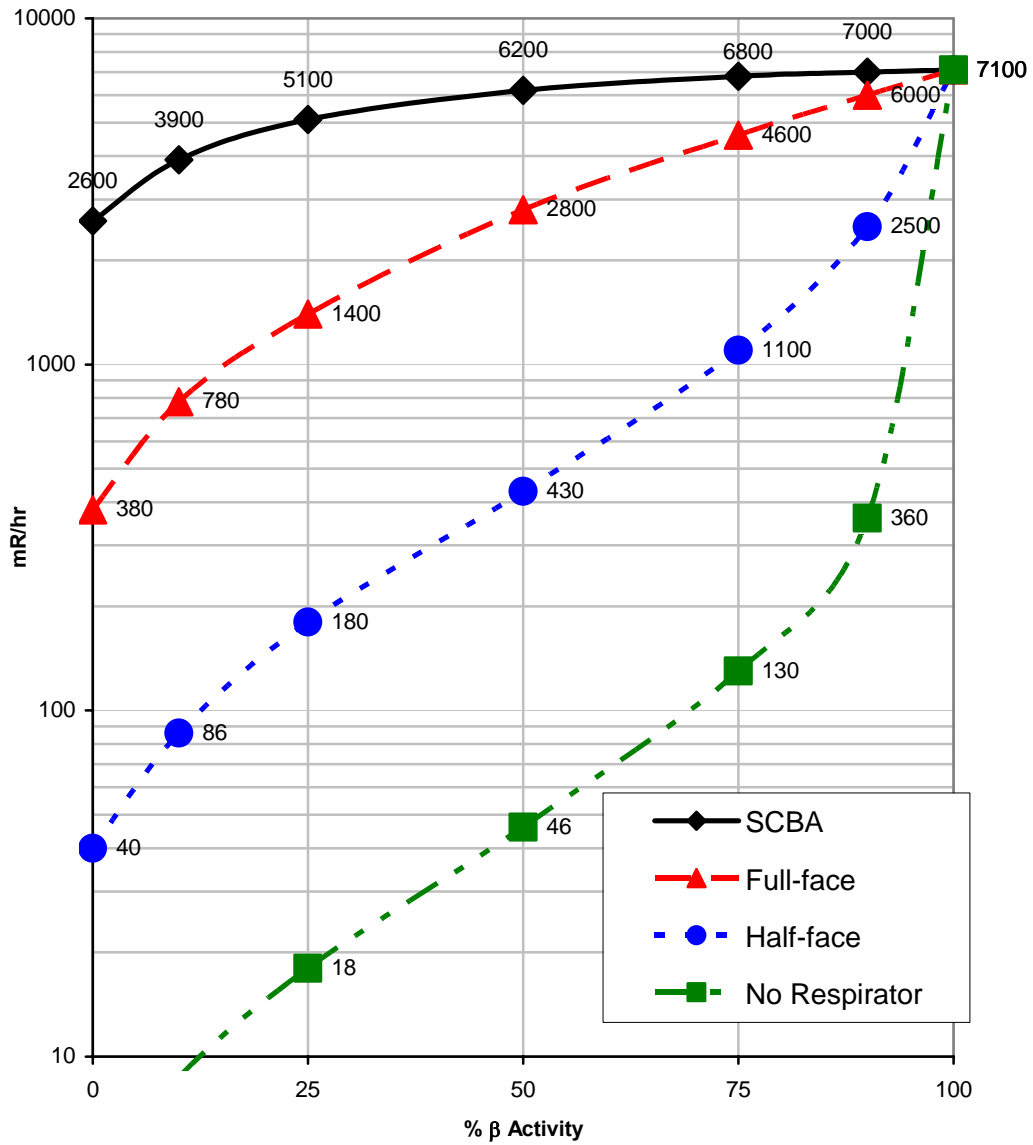
Figure 8-3a. Far-Field Maximum Worker Exposure Rates, 5 rem in 1 Hour,
Deposited βγ- and α-emitting Radionuclides, Low Dust

Use Figure 8-3b for:

- NEAR-FIELD
- The incident involves βγ- and α-emitting radionuclides,
- Low-dust conditions,
- Exposure rates can be measured, and
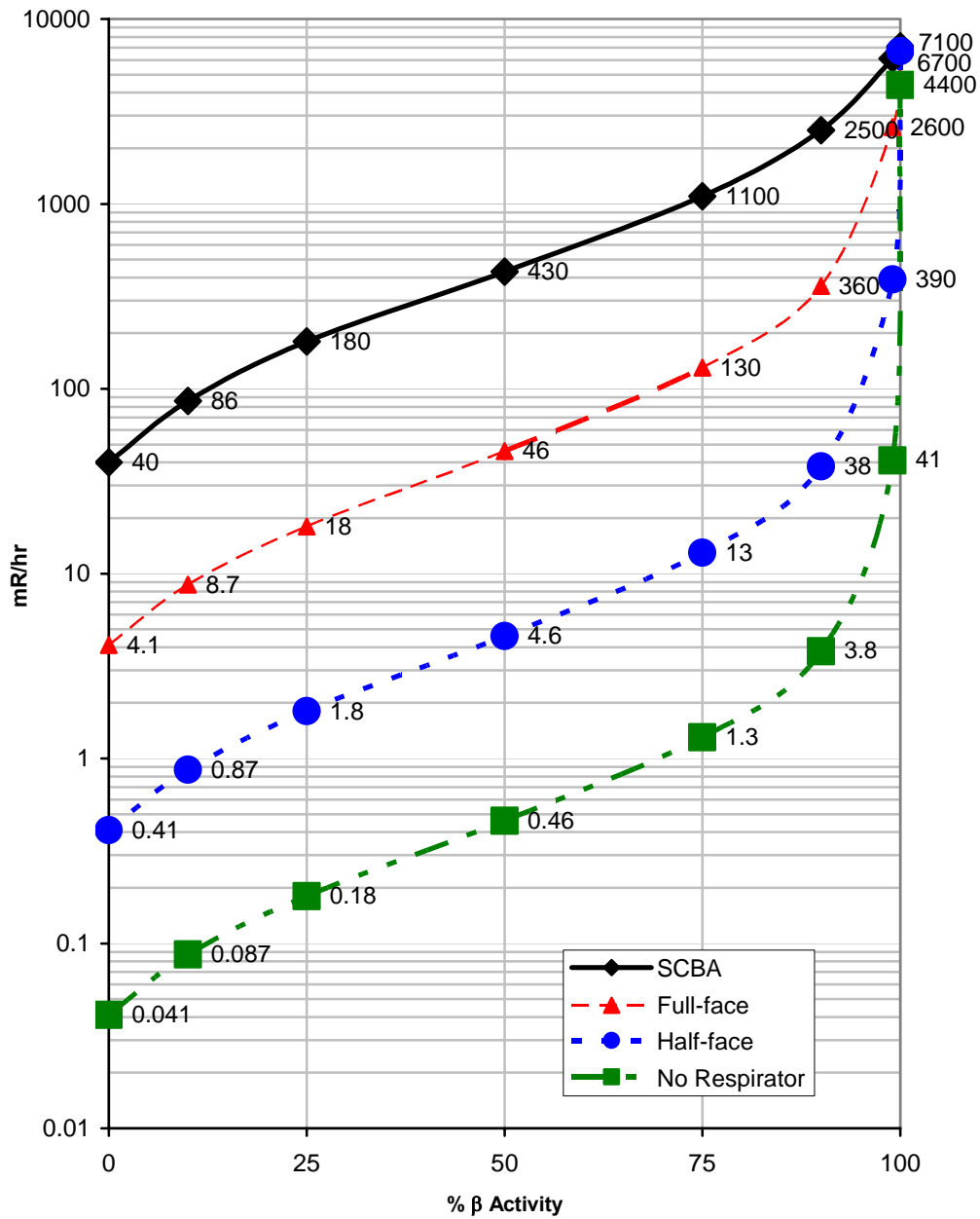- The fractions of α- and β-activity can be estimated.

Figure 8-3b. Near-Field Maximum Worker Exposure Rates,
5 rem in 1 Hour, Deposited βγ- and α-emitting Radionuclides, Low Dust

Use Figure 8-4a for:

- FAR-FIELD
- The incident involves βγ- and α-emitting radionuclides,
- Dusty conditions,
- Exposure rates can be measured, and
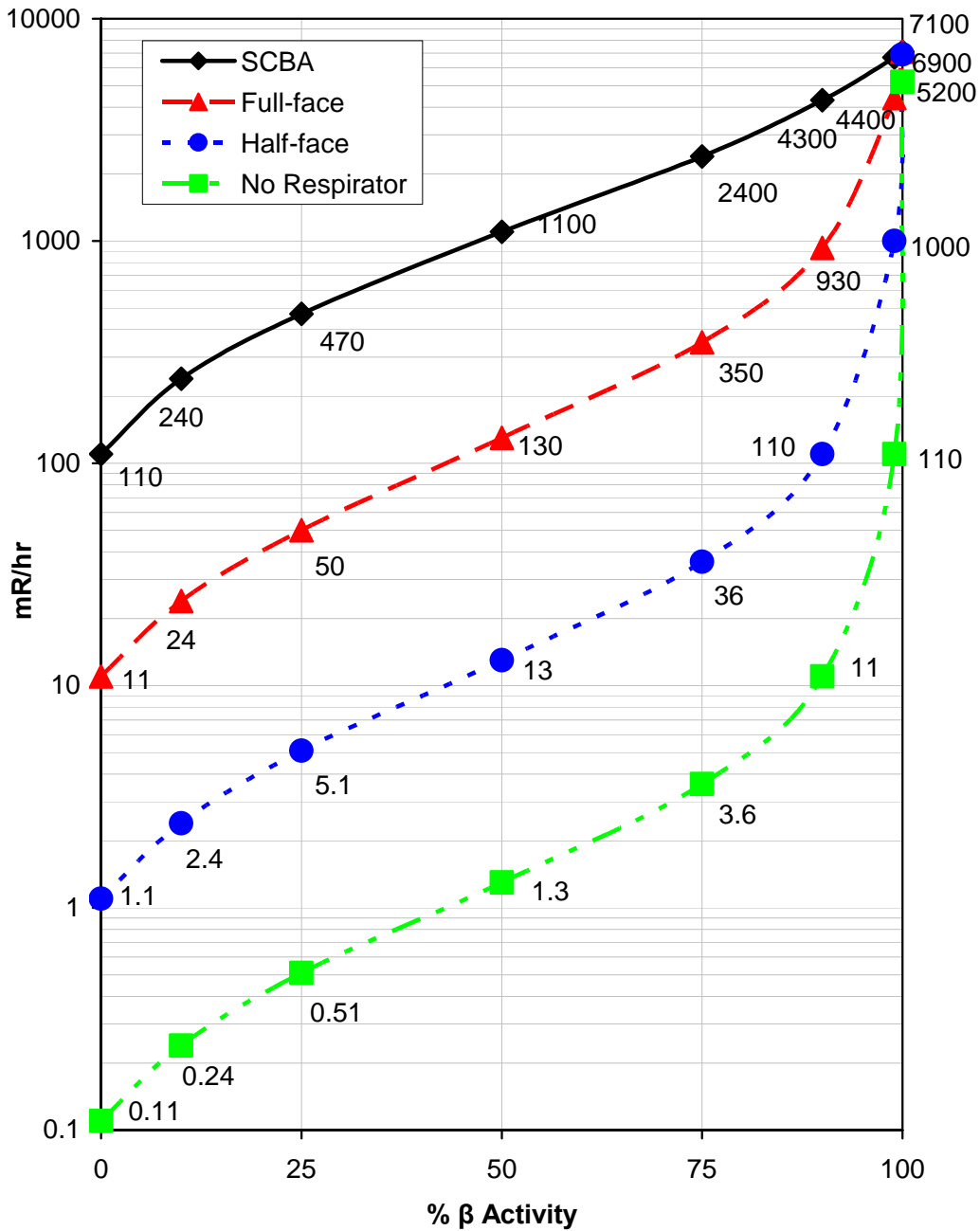- The fractions of α- and β-activity can be estimated.

Figure 8-4a. Far-Field Maximum Worker Exposure Rates, 5 rem in 1 Hour,
Deposited βγ- and α-emitting Radionuclides, High Dust

Use Figure 8-4b for:

- NEAR-FIELD
- The incident involves βγ- and α-emitting radionuclides.
- Dusty conditions,
- Exposure rates can be measured, and
- The fractions of α- and β-activity can be estimated.

Figure 8-4b. Near-Field Maximum Worker Exposure Rates, 5 rem in 1 Hour, Deposited βγ- and α-emitting Radionuclides, High Dust

Use Figure 8-5a for:

- FAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- Low-dust conditions,
- Exposure rates can be measured, and
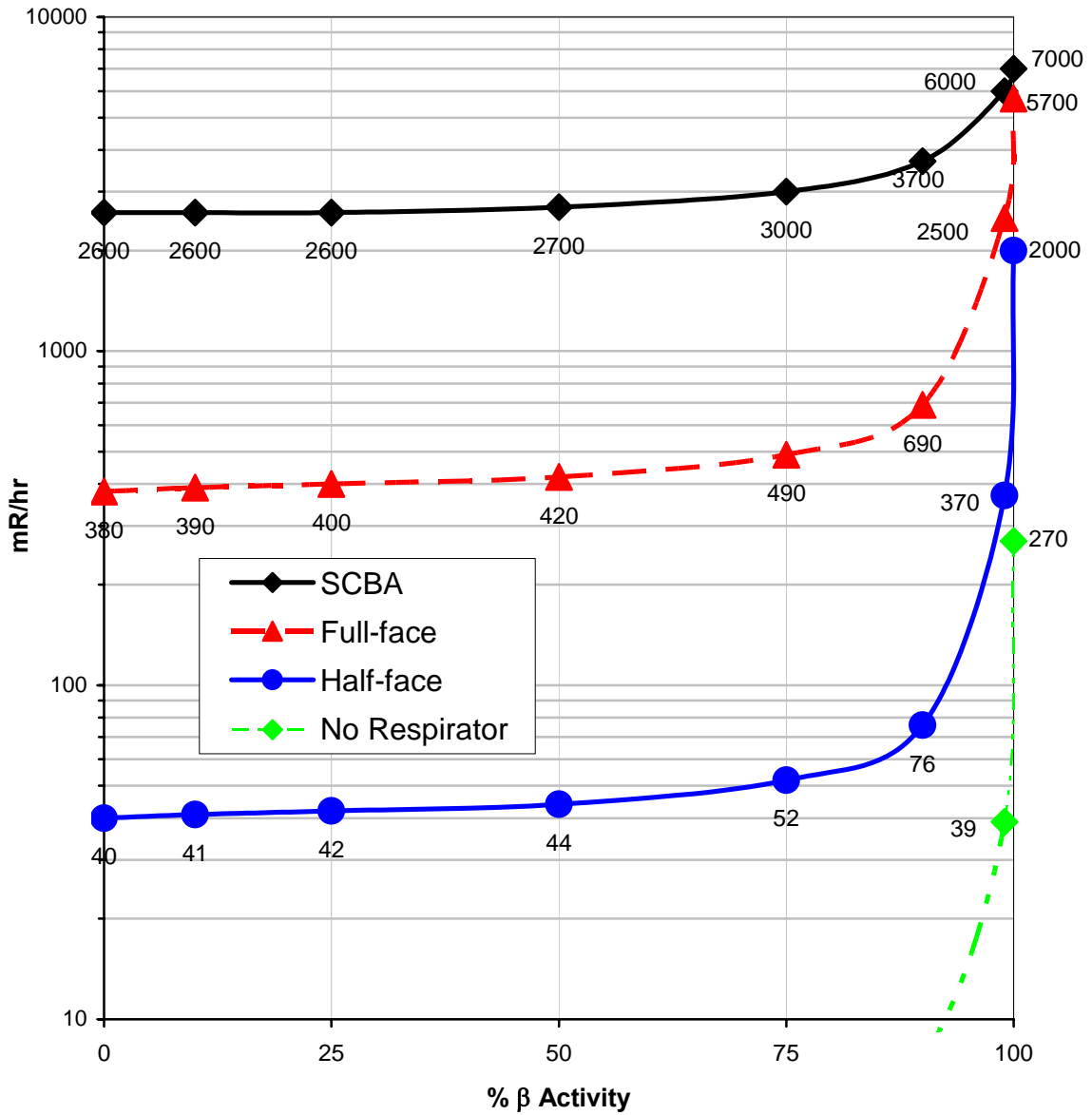- The fractions of α- and β-activity can be estimated.

Figure 8-5a. Far-Field Maximum Worker Exposure Rates, 5 rem in 1 Hour,
Deposited β- and α-emitting Radionuclides, Low Dust

Use Figure 8-5b for:

- NEAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- Low-dust conditions,
- Exposure rates can be measured, and
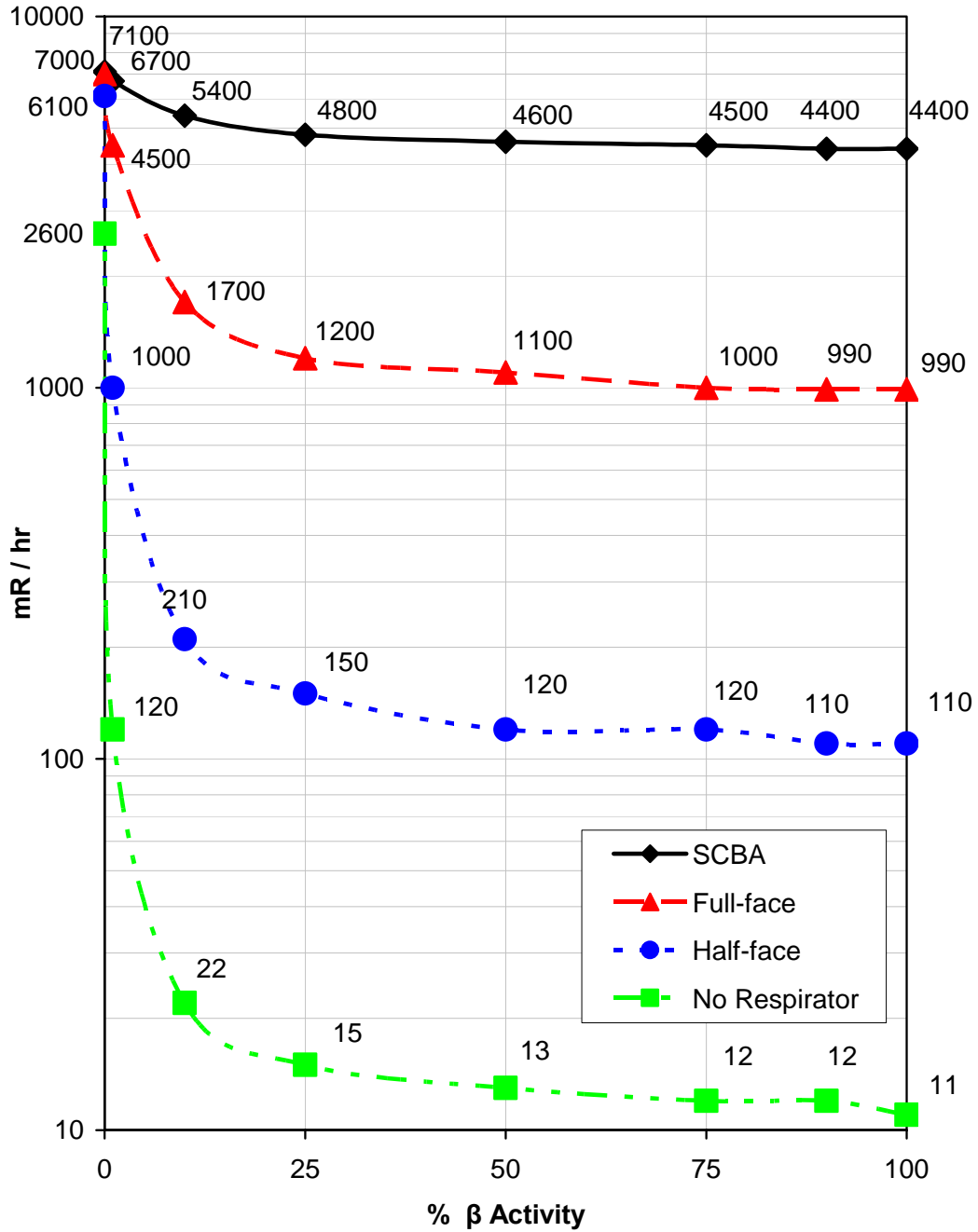- The fractions of α- and β-activity can be estimated.

Figure 8-5b. Near-Field Maximum Worker Exposure Rates, 5 rem in 1 Hour, Deposited β- and α-emitting Radionuclides, Low Dust

Use Figure 8-6a for:

- FAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- Dusty conditions,
- Exposure rates can be measured, and
- The fractions of α- and β-activity can be estimated.

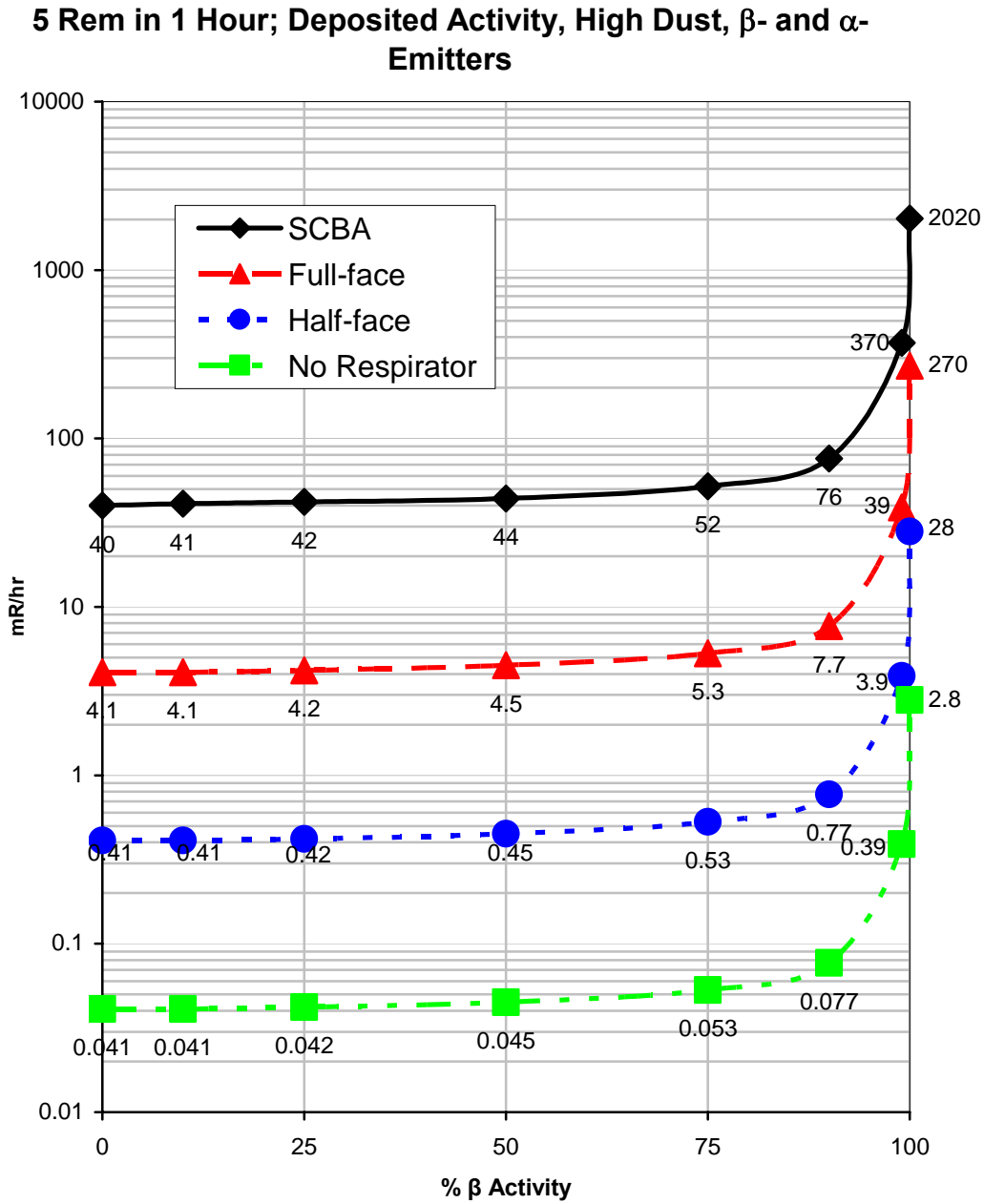**5 Rem in 1 Hour; Deposited Activity, High Dust, β- and α-Emitters**

Figure 8-6a. Far-Field Maximum Worker Exposure Rates, 5 rem in 1 Hour, Deposited β- and α-emitting Radionuclides, High Dust

Use Figure 8-6b for:

- NEAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- Dusty conditions,
- Exposure rates can be measured, and
- The fractions of α- and β-activity can be estimated.

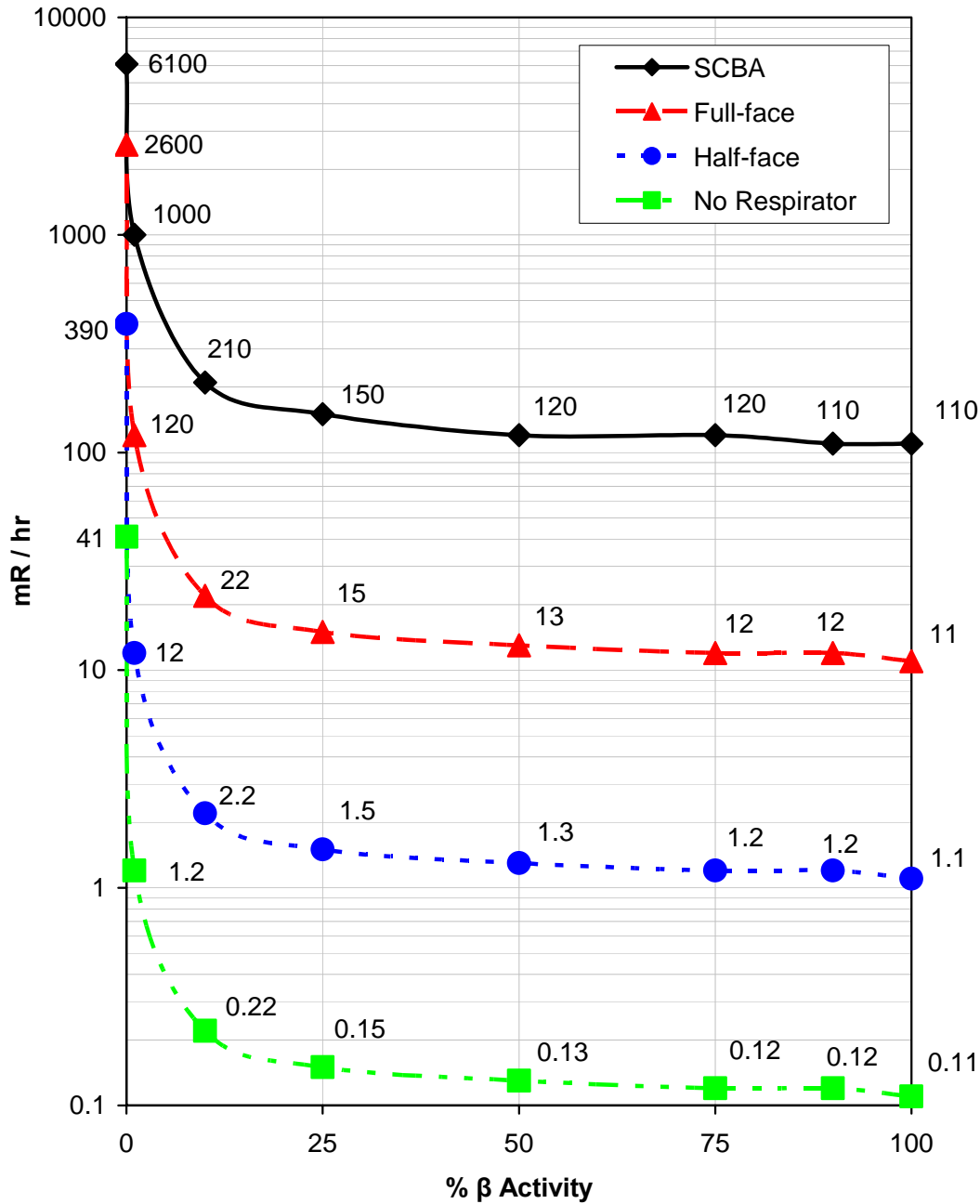

Figure 8-6b. Near-Field Maximum Worker Exposure Rates, 5 rem in 1 Hour,
Deposited β- and α-emitting Radionuclides, High Dust

8.5.2   Public Protection

Figures in this section are intended to relate deposited activity to protecting members of the public. These figures are valid only after the plume has passed.

Figures 8-7 through 8-12 provide estimates of evacuation/sheltering DRLs for use with mixtures of unknown radionuclides during the first 96 hours after a release. The results in Figures 8-7 through 8-12 are based on a 1000-mrem TEDE PAG, but they are scaleable for other PAGs. The EPA evacuation/sheltering PAG is in the range of 1 to 5 rem (1000 to 5000 mrem) TEDE for the Early Phase (first 96 hours after a release). Cognizant authorities may establish higher or lower PAGs.

Doses to the public from resuspension and from external exposure to deposited activity should be less than the PAG when deposition levels do not exceed the DRL. The DRLs provided in Figures 8-7 through 8-12 are in units of exposure rate (mR/hr) or deposited activity level (e.g., $\mu Ci/m^2$ of gross-α, gross-β, or gross-γ radioactivity).

8.5.2.1   Exposure Rate DRLs for Evacuation/Sheltering

Exposure rate DRLs for mixtures are provided in a series of figures in this section. Exposure rate DRLs account for dose from radiation deposited on the ground surface and inhalation of resuspended material only. FRMAC Method M.3.3 Step B (NNSA03, 2003) was used to calculate the results presented in Figures 8-7 and Figure 8-8.

To use the graphs, the user must have an estimate (however rough) of the relative amounts of α and β activity that have been deposited.

Caution: Under high-resuspension conditions, exposure rate DRLs that are based entirely on α-emitting radionuclides or essentially pure β-emitting radionuclides may not be sensitive enough to support evacuation/relocation decisions. It is difficult to use exposure rate DRLs that are less than levels about two times the smallest exposure rate that can be measured with the instruments that responders have available to them.

Use Figure 8-7a for:

- FAR-FIELD
- The incident involves βγ- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rates can be measured, and
- The fractions of α- and β-activity can be estimated.

Figure 8-7a. Far-Field Early Phase Exposure Rate DRL;
Deposited βγ- and α-emitting Radionuclides, 1 Rem

Use Figure 8-7b for:

- NEAR-FIELD
- The incident involves βγ- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rates can be measured, and
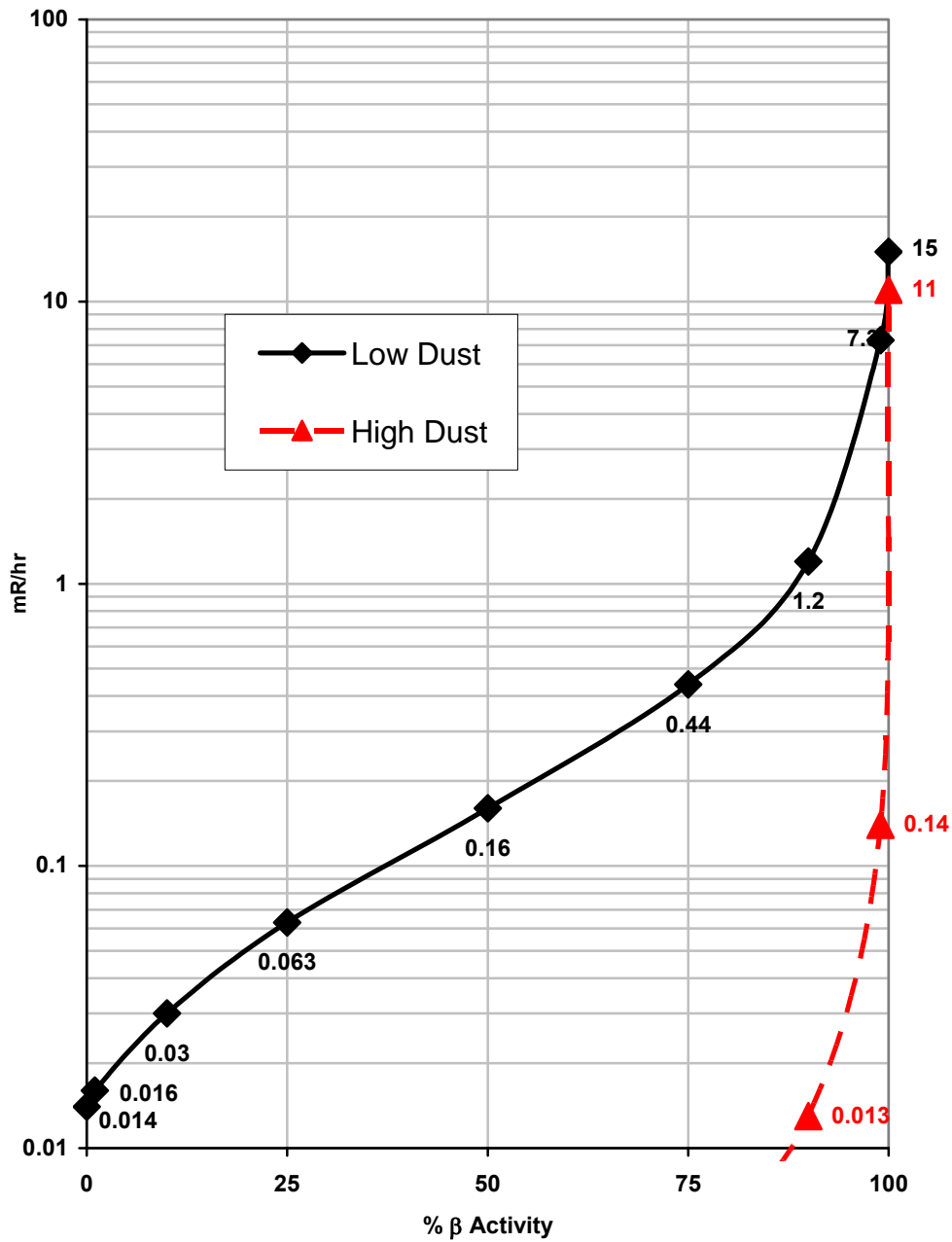- The fractions of α- and β-activity can be estimated.

Figure 8-7b. Near-Field Early Phase Exposure Rate DRL;
Deposited βγ- and α-emitting Radionuclides, 1 Rem

Use Figure 8-8a for:

- FAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rates can be measured,
- The fractions of α- and β-activity can be estimated.

Figure 8-8a. Far-Field Early Phase Exposure Rate DRL,
Deposited β- and α-emitting Radionuclides, 1 Rem

Use Figure 8-8b for:

- NEAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rates can be measured,
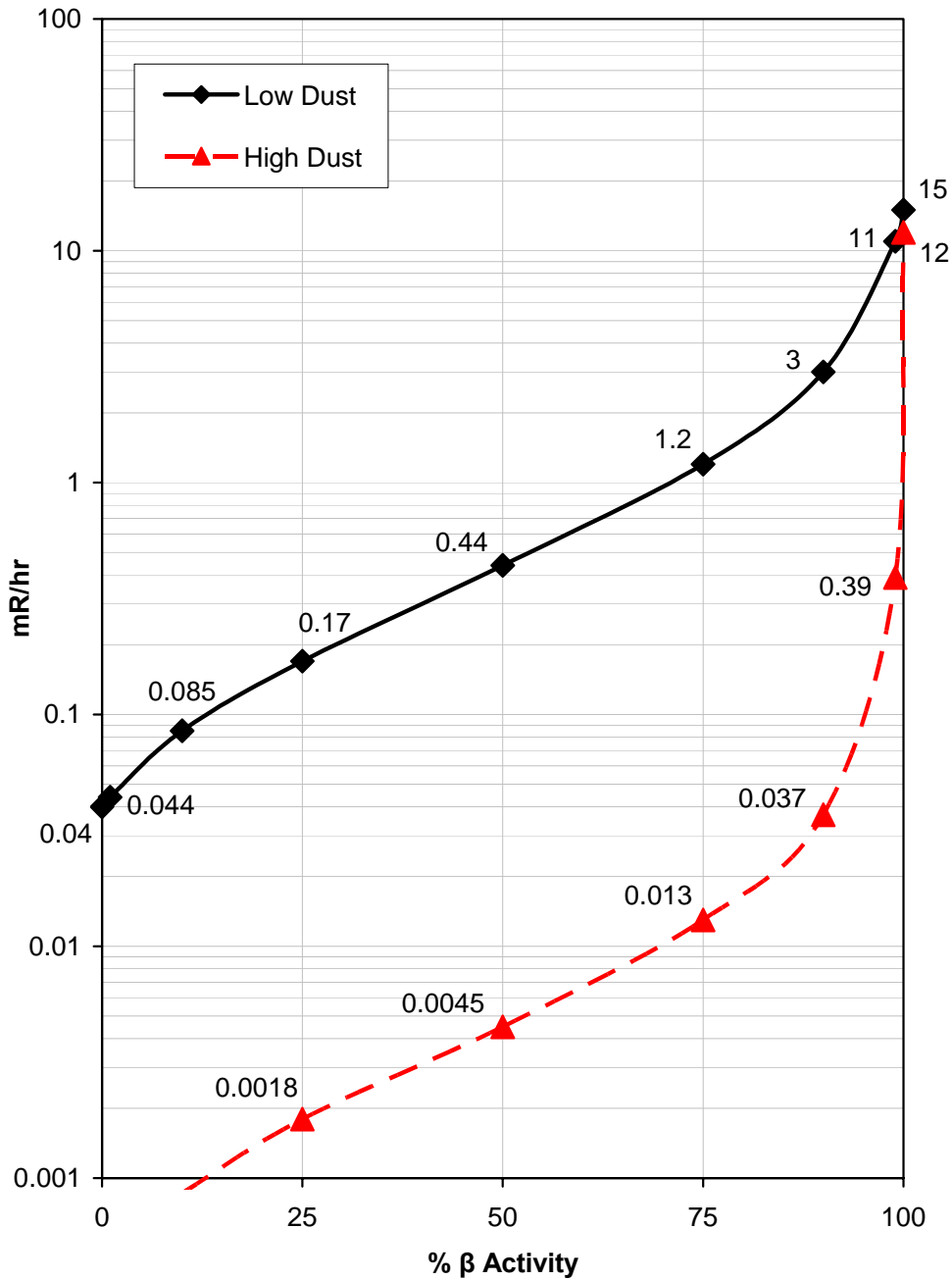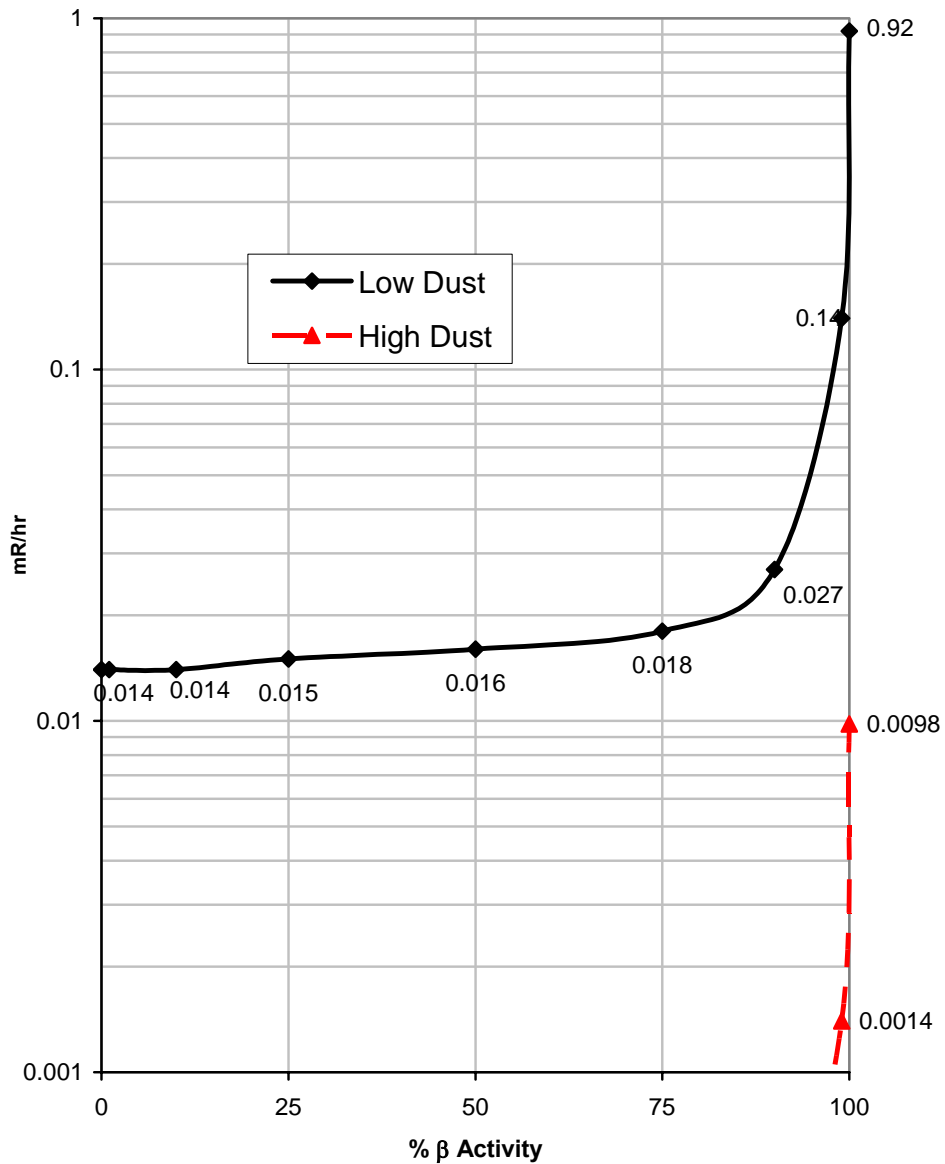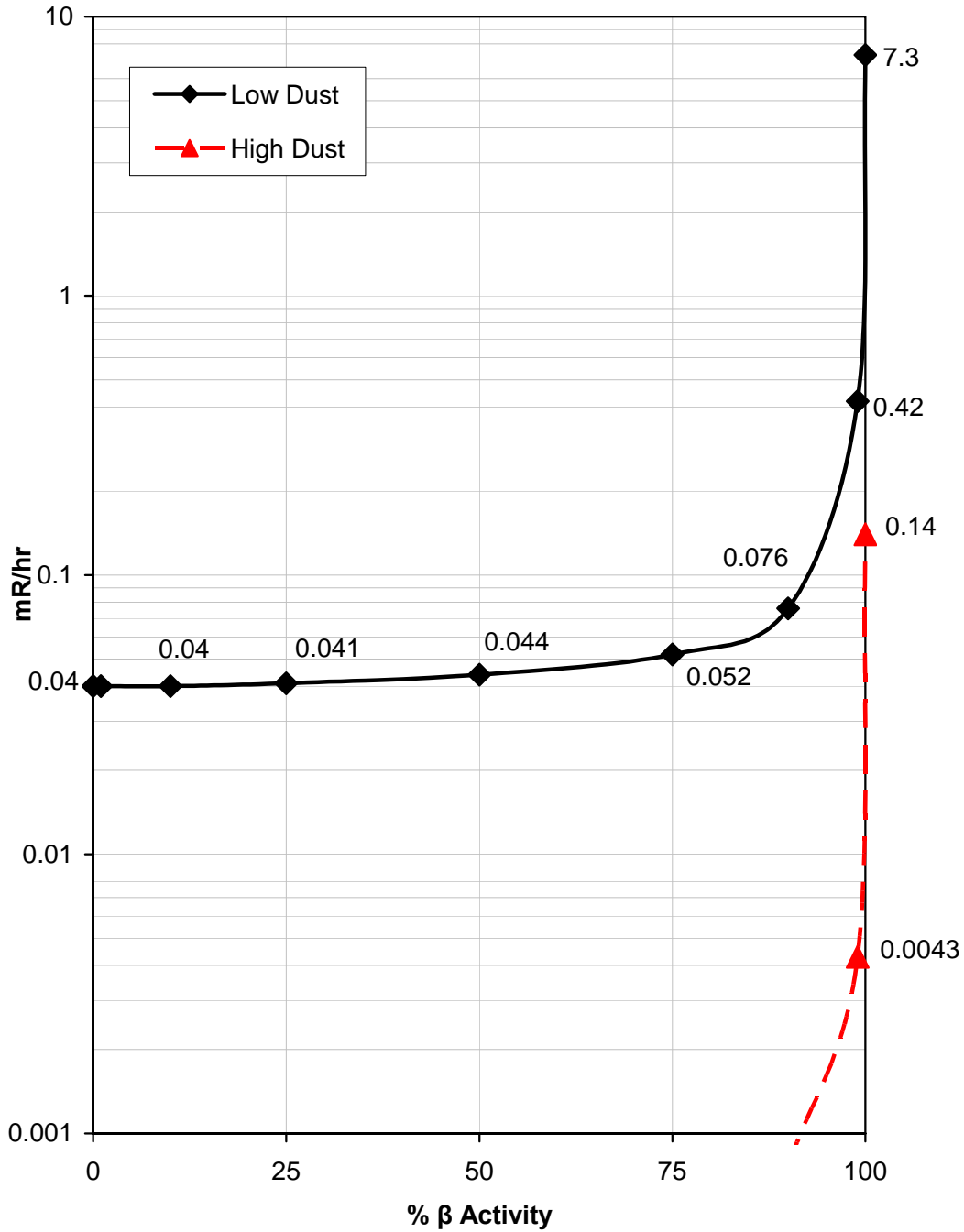- The fractions of α- and β-activity can be estimated.

Figure 8-8b. Near-Field Early Phase Exposure Rate DRL,
Deposited β- and α-emitting Radionuclides, 1 Rem

8.5.2.2 <u>Early Phase Marker Nuclide Deposition DRLs</u>

Deposition DRLs for marker nuclides are presented in this section. DRLs in this section estimate TEDE from radioactive material deposited on the ground surface. These DRLs only account for dose from external exposure and inhalation of resuspended material. The figures were calculated for the Early Phase using FRMAC Manual Method M.3.3 (NNSA03, 2003).

Figures 8-9 through 8-12 are primarily used to estimate gross-α, gross-β, and gross-γ activity deposition DRLs for the Early Phase for *unknown* radionuclides when exposure rate measurements are undesirable or not feasible. If the identities of the radionuclides are known, then TurboFRMAC or other software tools should be used to obtain radionuclide-specific guidance.

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Use Figure 8-9a for:

- FAR-FIELD
- The incident involves βγ- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rate measurements are undesirable or not feasible,
- Deposition levels for strong γ-emitting radionuclides can be estimated, and
- The fractions of α- and β-activity can be estimated.

Figure 8-9a. Far-Field Early Phase Deposition DRL,
α-emitter with βγ-emitter Marker, 1 Rem

Use Figure 8-9b for:

- NEAR-FIELD
- The incident involves βγ- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rate measurements are undesirable or not feasible,
- Deposition levels for strong γ-emitting radionuclides can be estimated, and
- The fractions of α- and β-activity can be estimated.

Figure 8-9b. Near-Field Early Phase Deposition DRL,
α-emitter with βγ-emitter Marker, 1 Rem

Use Figure 8-10a for:

- FAR-FIELD
- The incident involves βγ- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rate measurements are undesirable or not feasible,
- Deposition levels for α-emitting radionuclides can be estimated, and
- The fractions of α- and β-activity can be estimated.

Figure 8-10a. Far-Field Early Phase Deposition DRL,
βγ-emitter With α-emitter Marker, 1 Rem

Use Figure 8-10b for:

- NEAR-FIELD
- The incident involves βγ- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rate measurements are undesirable or not feasible,
- Deposition levels for α-emitting radionuclides can be estimated, and
- The fractions of α- and β-activity can be estimated.
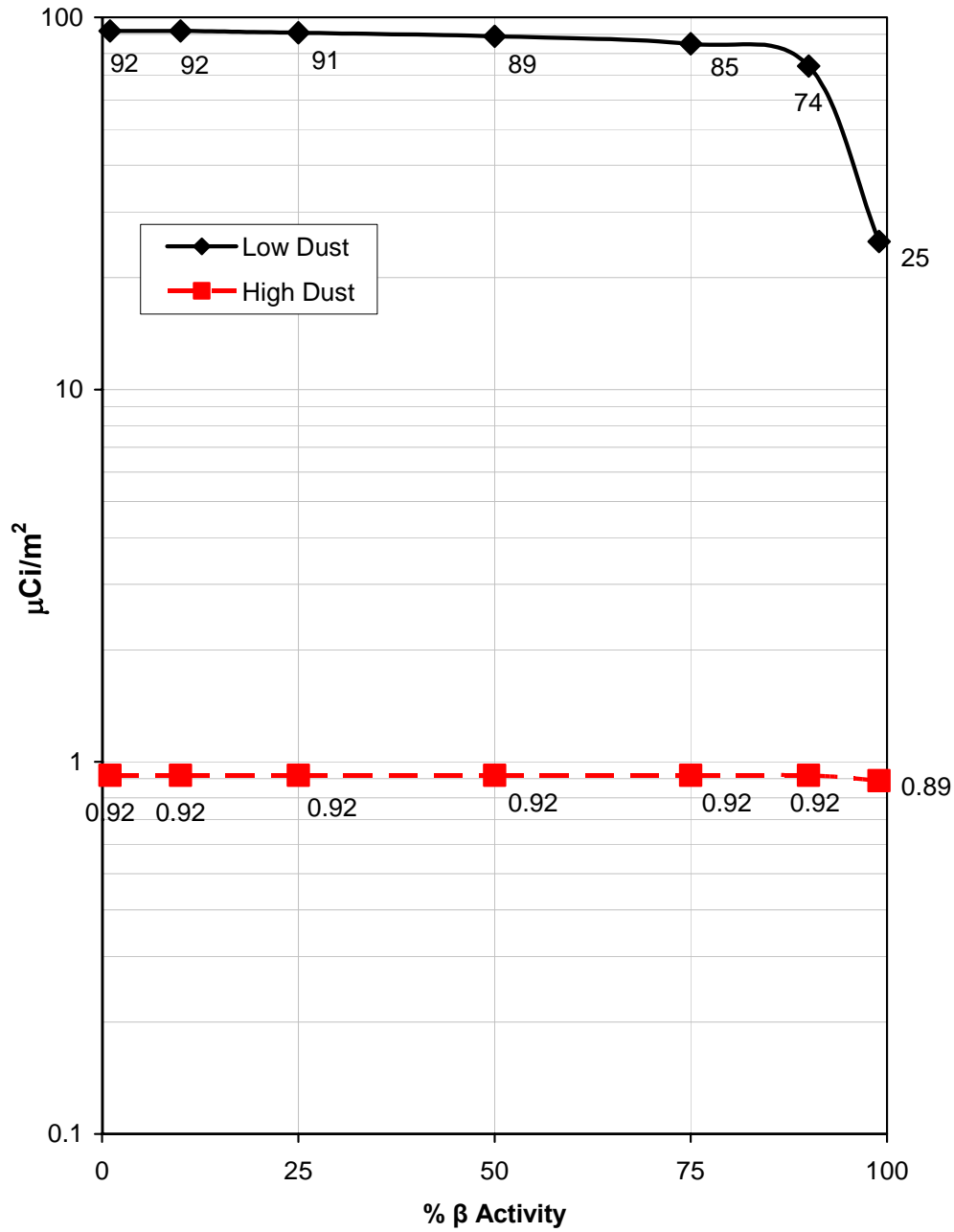
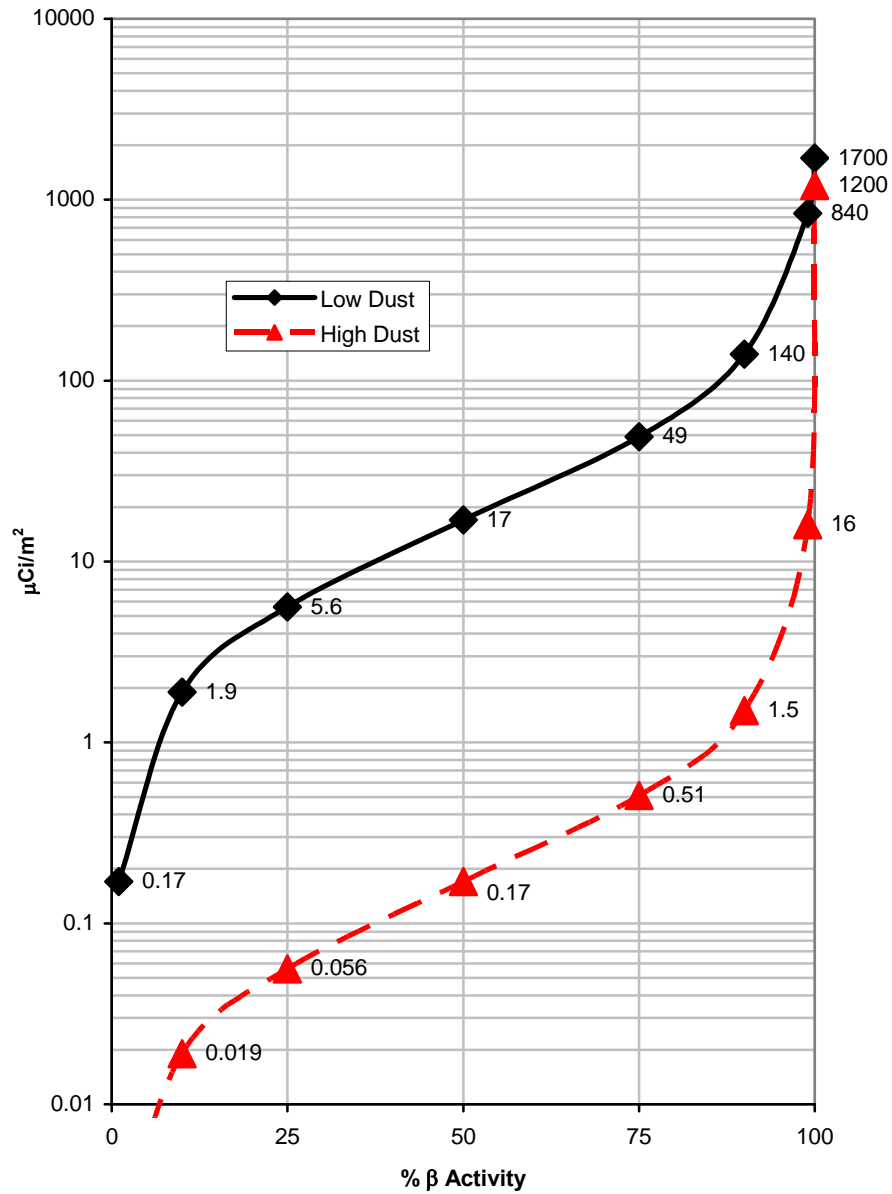Figure 8-10b. Near-Field Early Phase Deposition DRL,
βγ-emitter with α-emitter Marker, 1 Rem

Use Figure 8-11a for:

- FAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rate measurements are undesirable or not feasible,
- Deposition levels for β-emitting radionuclides can be estimated, and
- The fractions of α- and β-activity can be estimated.
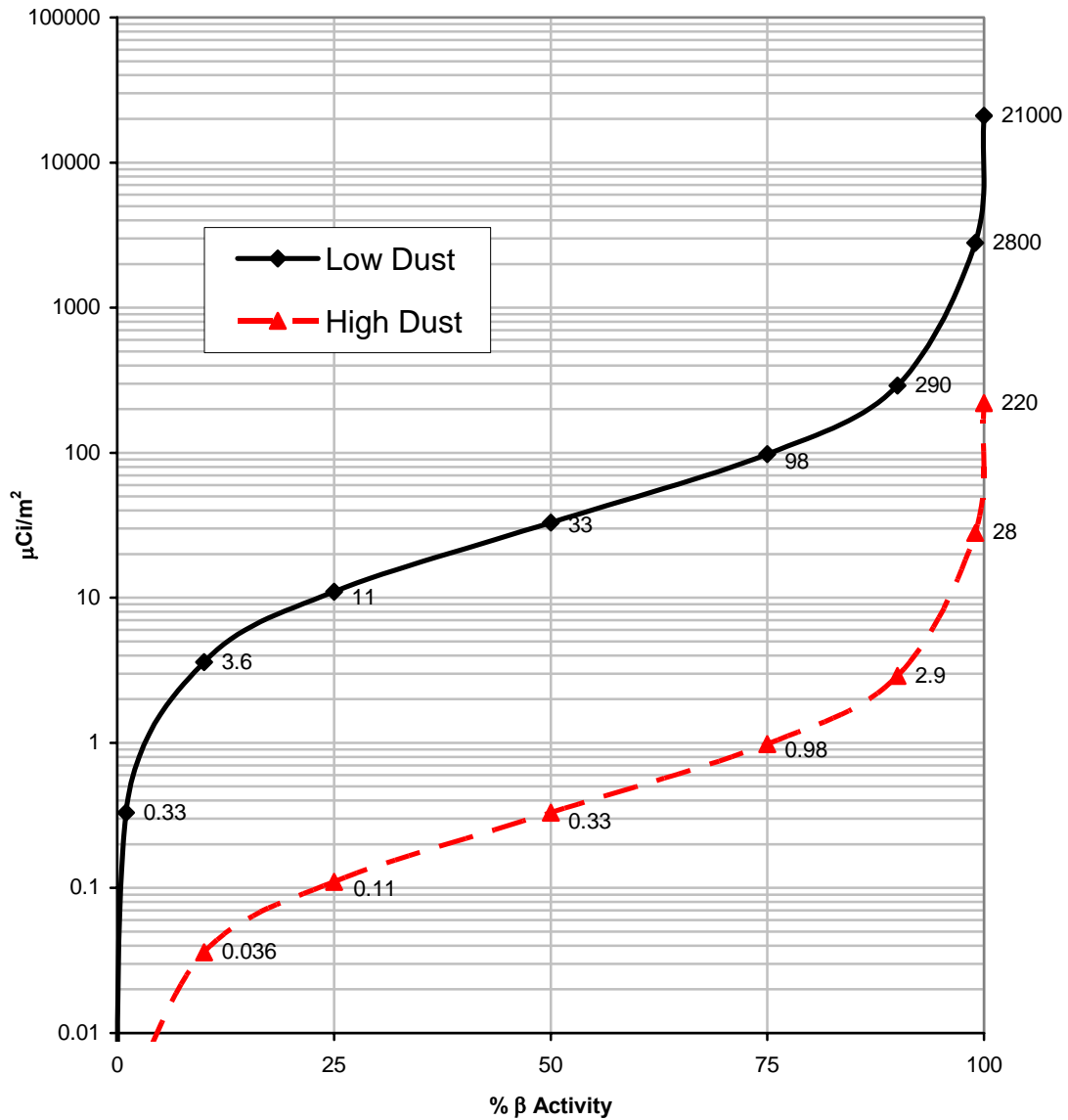
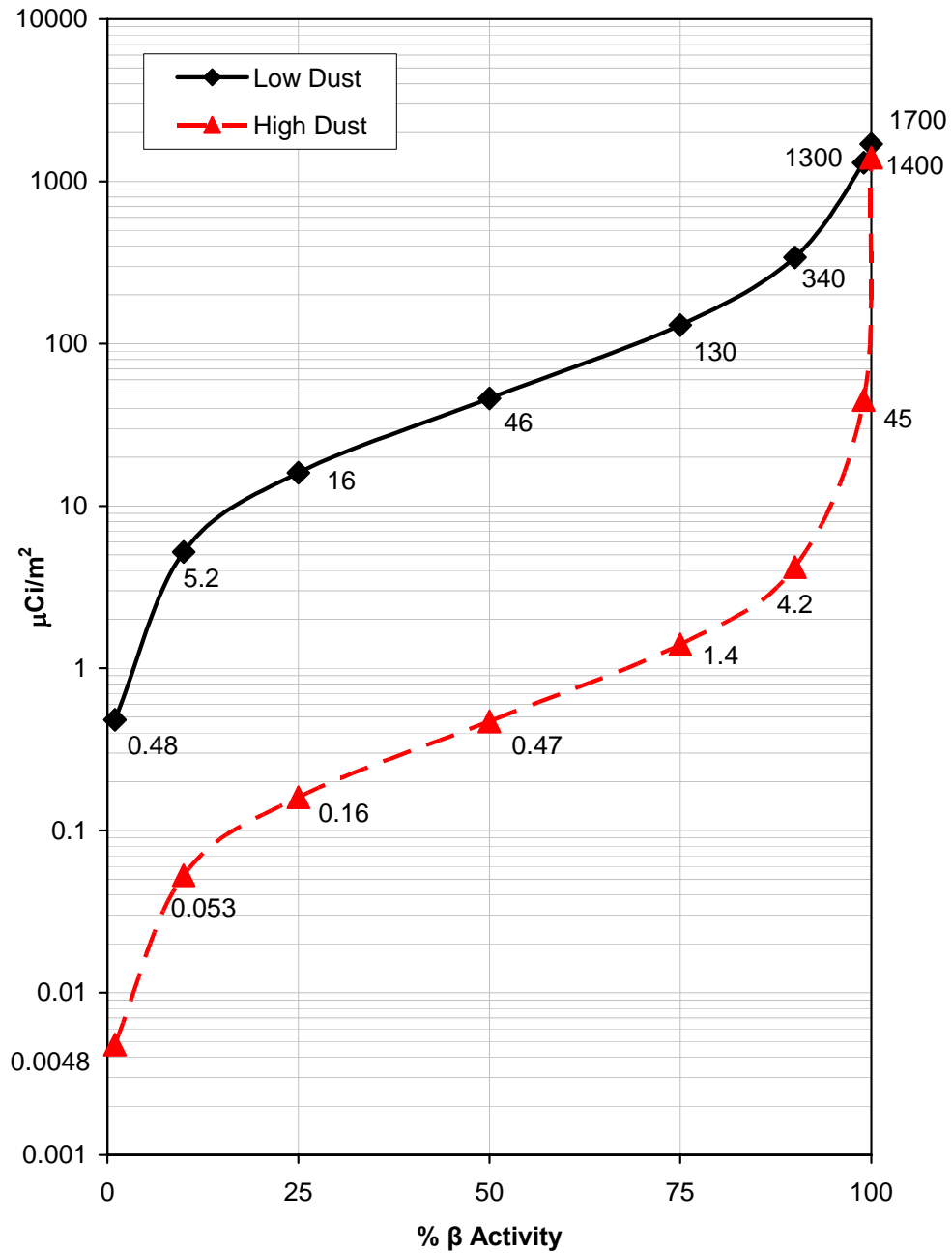Figure 8-11a. Far-Field Early Phase Deposition DRL,
α-emitter With β-emitter Marker, 1 Rem

Use Figure 8-11b for:

- NEAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rate measurements are undesirable or not feasible,
- Deposition levels for β-emitting radionuclides can be estimated, and
- The fractions of α- and β-activity can be estimated.

Figure 8-11b. Near-Field Early Phase Deposition DRL,
α-emitter With β-emitter Marker, 1 Rem

Use Figure 8-12a for

- FAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rate measurements are undesirable or not feasible,
- Deposition levels for α-emitting radionuclides can be estimated, and
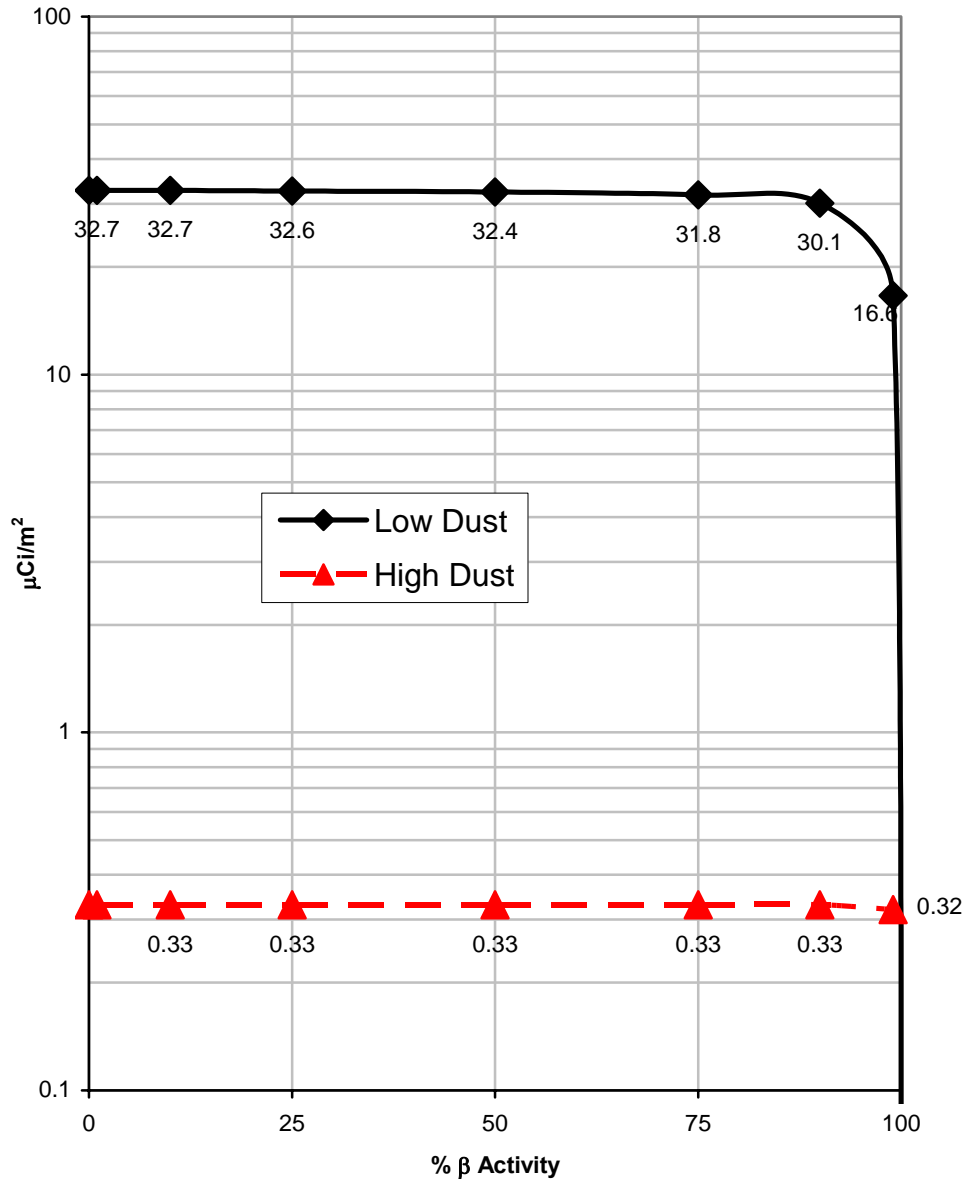- The fractions of α- and β-activity can be estimated.

Figure 8-12a. Far-Field Early Phase Deposition DRL,
β-emitter With α-emitter Marker, 1 Rem

Use Figure 8-12b for

- NEAR-FIELD
- The incident involves β- and α-emitting radionuclides,
- TEDE needs to be estimated over the next 96 hours,
- Exposure rate measurements are undesirable or not feasible,
- Deposition levels for α-emitting radionuclides can be estimated, and
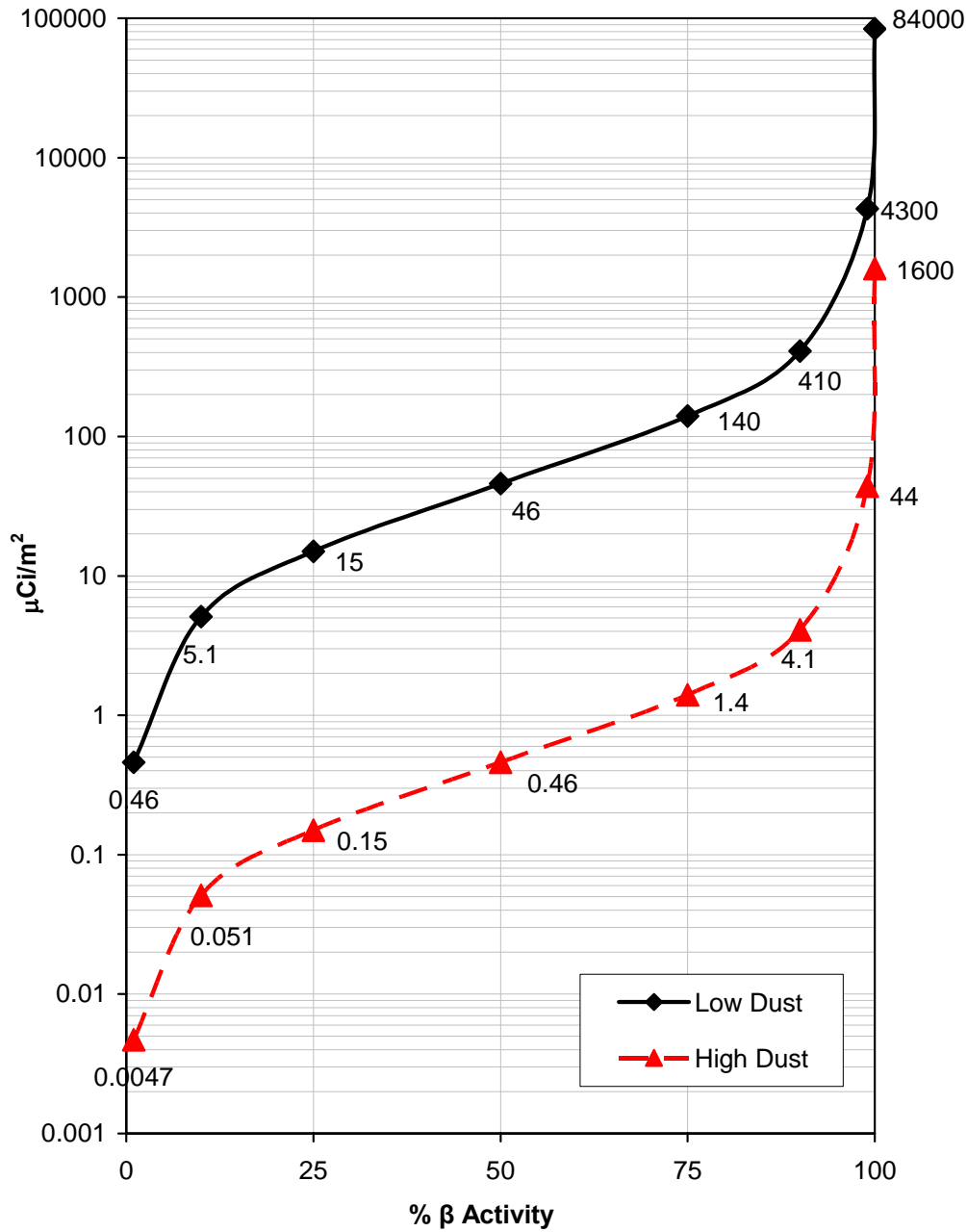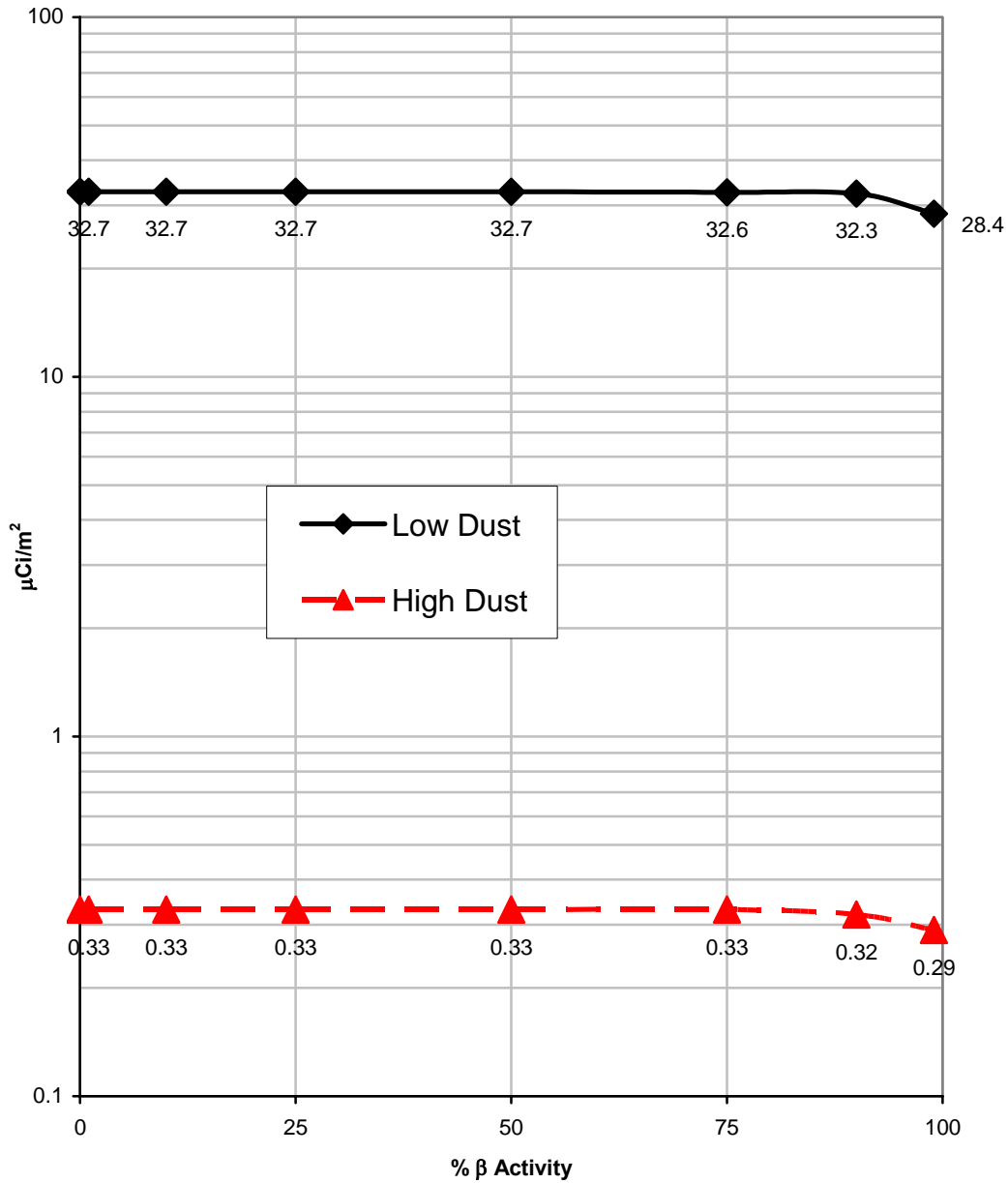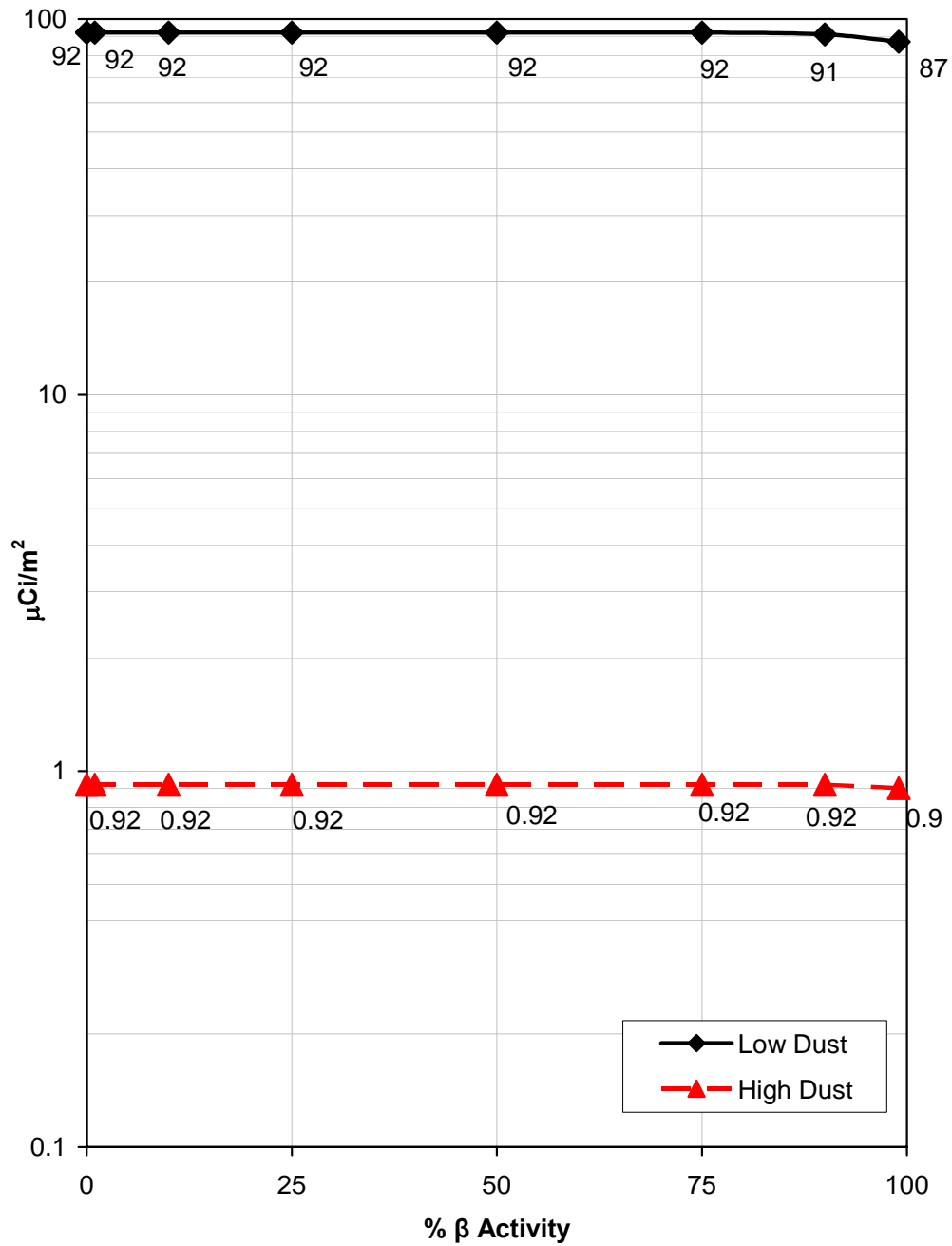- The fractions of α- and β-activity can be estimated.

Figure 8-12b. Near-Field Early Phase Deposition DRL,
β-emitter With α-emitter Marker, 1 Rem

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APPENDIX A. GLOSSARY

The glossary contains definitions for terms and acronyms used in the manual.

Acute Health Effects	See Early Health Effects
AGL	Above Ground Level
ALI	Annual Limit of Intake
AMAD	Activity Median Aerodynamic Diameter
ARAC	Atmospheric Release Advisory Capability
AMS	Aerial Measuring System
ARF	Airborne release fraction
ARG	Accident Response Group
ASHG	Accident Site Health Group
BN	Bechtel Nevada
BWR	Boiling-Water Reactor
CDC	Centers for Disease Control and Prevention
Child	A 10-year-old person.
Cloud Shine	Gamma radiation from radioactive materials in the air (plume).
CMRT	Consequence Management Response Team
Committed Dose Equivalent (CDE)	The dose equivalent to a specific organ for 50 years following intake (inhalation or ingestion). It does not include contributions from external dose.
Committed Effective Dose Equivalent (CEDE)	The sum of the dose equivalent for 50 years following intake (inhalation or ingestion) of a radionuclide to each organ multiplied by a weighting factor. CEDE is used to estimate the risk from delayed health effects.
Concentration	The amount of activity per unit of measure (volume, mass, area, sample, etc.) considered for a sample.

Critical	Most important source of dose, the most important organ, or the most important group of (potentially) exposed population. That is, effects will be dominated by this source of dose or effects (*e.g.*, deaths) will occur first as a result of exposure to this organ or population (*e.g.*, infants) when exposed to radiation through a certain pathway.
DAC	Derived Air Concentration
DCENT	Computer database maintained by the FRMAC Data Center
DCF	Dose Conversion Factor
Delayed Health Effects	A wide range of cancers and hereditary effects that usually occur many years after exposure. In contrast to early health effects, it is assumed there are no dose thresholds below which these effects do not occur.
Deposition	The contamination found on the surface of the ground.
Derived Intervention Level (DIL)	The concentration of a radionuclide in food derived from the protective action guide and at which introduction of protective measures should be considered.
Derived Response Level (DRL)	A calculated value (*e.g.*, exposure rate or radionuclide concentration) that corresponds to an early health effect threshold, a PAG, or a DIL. DRLs can be used to relate environmental measurements or laboratory analysis to the potential for early health effects or need for protective actions. Used to facilitate prompt assessments.
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
Dose Conversion Factor (DCF)	The dose equivalent per unit intake of a radionuclide (mrem/µCi).
DOT	U.S. Department of Transportation
DQO	Data Quality Objective
DRL	Derived Response Level

Early Health Effects

Health effects that will be occur shortly after exposure (hours, weeks) resulting from high doses over a short period (acute doses) to specific organs and involve thresholds below which these health effects are not expected to occur.

Early Phase

The period of time that extends from the time the threat of a major release is identified (before the release) until the release or threat of major release has ended, and areas of major deposition have been identified.

Effective Dose Equivalent (EDE)

The sum of the dose equivalent from external exposure to each organ multiplied by a weighting factor. EDE is used to estimate the risk from delayed health effects. EDE rate from air submersion and ground shine is assumed to equal the exposure rate.

Emergency Worker Guidance

Guidance on the external dose and CEDE incurred by workers (other than a pregnant woman) while performing emergency services.

EPA

U.S. Environmental Protection Agency

ESRI

Environmental Systems Research Institute, Inc.

External Dose

The dose of radiation received by an individual from a source of ionizing radiation outside the body.

Facility Operator

The organization that operates the facility.

FDA

U.S. Food and Drug Administration

FRMAC

Federal Radiological Monitoring and Assessment Center.

GIS

Geographic Information System

GM

Geiger-Mueller

Groundshine

Gamma radiation from radioactive materials deposited on the ground.

H&S

Health and Safety

HHS

U.S. Department of Health and Human Services.

Immersion

To be surrounded or engulfed by the radioactive cloud.

Infant	A child one year of age or younger.
Intermediate Phase	The period beginning after the release and potential for further major release is over and reliable environmental data are available for use as a basis for relocation and ingestion protective actions. Usually one year of deposition and 30 days of ingestion calculation.
IR	Radionuclide Ratio
LANL	Los Alamos National Laboratory
LET	Linear Energy Transfer
LFA	Lead Federal Agency
LLNL	Lawrence Livermore National Laboratory
Light-Water Reactor (LWR)	A nuclear reactor that uses natural water as a coolant and moderator. All U.S. commercial power reactors in the United States are LWRs, as are the Russian-constructed VVERs.
Marker Nuclide	A nuclide contained in deposition or samples that is easily identified in the field or laboratory. It is used to determine areas of concern before performing a comprehensive nuclide analysis.
MDA	Minimum Detectable Activity
Mix	The nuclide ratio (relative abundance) of the radionuclides in a sample or deposition.
NARAC	National Atmospheric Release Advisory Center
NDA	National Defense Area
NNSA	National Nuclear Security Administration
NRC	U.S. Nuclear Regulatory Commission
NSA	National Security Area
Pathways	The paths radionuclides follow from the source through the environment, including vegetation and animals, to reach an individual or a population.

Protective Action Guide (PAG) — The projected dose, from an accidental release of radioactive material, where specific actions to reduce or avoid dose are warranted.

PPE — Personal Protective Equipment

PWR — Pressurized-Water Reactor

Quality Factor (QF) — The principal modifying factor that represents the biological effectiveness of different radiation types with respect to induction of stochastic effects. It is used to calculate the dose equivalent from the absorbed dose. The absorbed dose, expressed in rad or Gy, is multiplied by the appropriate quality factor to obtain the dose equivalent

RAP — Radiological Assistance Program

Relative Biological Effectiveness (RBE) — The RBE of a given type of ionizing radiation is a factor used to compare the biological effectiveness of absorbed radiation doses (i.e., rads) due to one type of ionizing radiation with that of other types of ionizing radiation; more specifically, it is the experimentally determined ratio of an absorbed dose of a radiation in question to the absorbed dose of a reference radiation required to produce an identical biological effect in a particular experimental organism or tissue.

Resuspension — Reintroduction to the atmosphere of material originally deposited onto surfaces.

RF — Respirable Fraction or Resuspension Factor (depending on context)

RHU — Radioisotope Heater Unit

RTG — Radioisotopic thermoelectric generator

RTM — Response Technical Manual

SCA — Single Channel Analyzer

SCBA — Self-Contained Breathing Apparatus

SI — International System of Units

SNL — Sandia National Laboratories

SNM	Special Nuclear Material
Stability Class	A class that describes conditions of atmospheric turbulence. Classes are generally grouped into six classes ranging from class A, very unstable, through class F, very stable.
TNT	Trinitrotoluene
Total Early Organ Dose (TEOD$_{Organ}$)	Dose estimates used to determine if early health effects are possible from dose to the indicated organ. The organs considered in this manual are the small intestine, red bone marrow, thyroid, and lung. Bone marrow is a critical organ when considering deaths from LWR accidents. TEOD projections in the manual include (1) EDE from air submersion, (2) four days of EDE from ground deposition, (3) early inhalation dose from the plume (dose to the organ for 30 days after inhalation of the radioactive material), and (4) early inhalation from four days of resuspension.
Total Effective Dose Equivalent (TEDE)	The sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures). Dose projections used for comparison with the EPA Early Phase PAG (EPA92), which is expressed in terms of TEDE, include: (1) the EDE from air submersion, (2) four days of EDE from ground deposition, (3) the inhalation CEDE dose from the plume, and (4) CEDE from inhalation of four days of resuspension.
Total Effective Exposure Period	The time span, considering decay, that will approximate the integrated dose over a period of time when multiplied by the dose rate at the beginning.
Turn-Back Guidance	Guidance given to emergency workers indicating when they should seek areas of lower exposure rate or potential. This guidance is usually implemented via a DRL expressed as an integrated dose reading on a self-reading dosimeter, an exposure rate, or a deposition concentration indicating that the emergency worker should leave the area where further exposure is possible.
USDA	U.S. Department of Agriculture
Weathering	Reduction of dose from deposited radionuclides (external and resuspension) over time due to movement of contamination below the surface or binding on surface materials.

Weighting Factors   An estimate of the mortality risk from delayed health effects arising from irradiation of a particular organ and used to calculate CEDE and EDE.

Organ	Weighting Factor[a]
Gonads	0.25
Breasts	0.15
Red Bone Marrow	0.12
Lungs	0.12
Thyroid	0.03
Bone Surface	0.03
Remainder	0.30

[a]Source: EPA88 (page 6)

WGPu   Weapons-Grade Plutonium

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APPENDIX B. SUPPORTING REFERENCES

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APPENDIX C
REQUIRED DEPLOYMENT REFERENCES AND CODES

The following list contains the references and codes used that should be readily at hand when using the manual.

Reference/Code Title

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