Protective Action Guidance for Radiological Incidents

Internal Review Draft
August 2007

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Introduction

This document transmits guides and recommendations that update those found in the Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA-400-R-92-001, May 1992), published by the United States Environmental Protection Agency (EPA) (EPA 1992b) (hereinafter referred to as the "1992 PAG Manual"). These guides and recommendations were developed cooperatively with the Federal Radiological Preparedness Coordination Committee (FRPCC), with representation from the Environmental Protection Agency (EPA), the Department of Energy (DOE), the Department of Defense (DOD), the Department of Homeland Security (DHS) including the Federal Emergency Management Agency (FEMA), the Nuclear Regulatory Commission (NRC), the Department of Health and Human Services (HHS), including the Centers for Disease Control (CDC) and the Food and Drug Administration (FDA), the Department of Agriculture (USDA), and the Department of Labor (DOL).

The historical and legal basis of this guidance begins with Reorganization Plan No. 3 of 1970, in which the Administrator of EPA assumed all the functions of the Federal Radiation Council (FRC), including the charge to "…advise the President with respect to radiation matters, directly or indirectly affecting health, including guidance for all federal agencies in the formulation of radiation standards and in the establishment and execution of programs of cooperation with States." Reorg. Plan No. 3 of 1970, §§ 2(a)(7), 6(a)(2) (35 FR 15623); The Atomic Energy Act of 1946, as amended, 42 USC § 2021(h). Recognizing this role, FEMA directed EPA, by regulation, to "establish Protective Action Guides (PAGs) for all aspects of radiological emergency planning in coordination with appropriate Federal agencies." 44 CFR § 351.22(a) (47 FR 10758 (March 11, 1982)). FEMA also tasked EPA with preparing "guidance for State and local governments on implementing PAGs, including recommendations on protective actions which can be taken to mitigate the potential radiation dose to the population." 44 CFR § 351.22(b). All of this information was to "be presented in the Environmental Protection Agency (EPA) 'Manual of Protective Action Guides and Protective Actions for Nuclear Incidents.'" 44 CFR § 351.22(b).

Additionally, section 2021(h) charged the Administrator with performing "such other functions as the President may assign to him by Executive order." 42 USC § 2021(h). Executive Order 12656 states that the Administrator shall "[d]evelop, for national security emergencies, guidance on acceptable emergency levels of nuclear radiation. . . ." Exec. Order No. 12656, § 1601(2), 53 FR 47491, 47507 (November 18, 1988). EPA's role in PAGs development was reaffirmed by the Federal Radiological Emergency Response Plan (FRERP) (61 FR 20944 (May 1, 1996), replacing 1985 FRERP (50 FR 46542 (November 8, 1985)), and the National Response Plan (NRP) (Dec. 2004), which superseded it.

EPA's 1992 PAG Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA 1992b), was intended for use by emergency management officials at the federal, state, tribal, and local levels, and form the basis by which emergency management officials may plan for and respond to radiological emergencies. This updated version of the 1992 PAG Manual accomplishes these additional key objectives: applying the existing 1992 protective action guides and protective actions to new radiological and nuclear scenarios of concern; lowering the recommended dose for administration of stable iodine; providing new guidance concerning consumption of water during or after a radiological emergency; and, adding guidance for dealing with long-term site restoration following a major radiological release. The reader is referred to the completely revised PAG Manual for a detailed treatment of the PAGs and their application in radiological emergencies.

This updated guidance incorporates the concepts and guidance contained in FRC Reports 5 (FRC 1964) and 7 (FRC 1965) as well as International Commission on Radiological Protection (ICRP) Publication 40 (ICRP 1984b) and ICRP Publication 60 series (ICRP 1991a). Development of the PAGs was based on three important principles, which also apply to the selection of any protective action before or during an incident:

- Prevent acute effects.
- Reduce risk of chronic effects.
- Require optimization to balance protection with other important factors and ensure that actions taken cause more benefit than harm.

Application

Protective actions may be recommended for a wide range of incidents, but generally would be utilized for incidents involving relatively significant releases of radionuclides. Examples of radiological incidents that have potential for significant releases are: a fire in a major facility such as a fuel manufacturing plant; an accident at a federal weapons complex facility; an accident at a commercial nuclear power plant; an act of radiological or nuclear terrorism; or a transportation accident involving a release of radioactive material. Each type of incident would pose a unique threat to public health, and must be planned for and managed accordingly. Thus, emergency response planning for a given facility, or potential scenario, must consider the radionuclides that may be involved; the nature of release dynamics; the timing of notification, response, and protective action implementation; and the feasibility of executing a particular protective action.

Officials responsible for emergency planning and policies should assess radiological and nuclear facilities, and potential scenarios that could lead to significant releases of radioactive materials, and use these principles and the accompanying PAGs and protective actions to perform emergency planning and exercises in advance. A PAG is defined as "the projected dose to reference man[1], or other defined individual, from a release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended." A protective action is a recommended action associated with a PAG. Protective actions are those actions that have the effect of reducing or avoiding radiation dose. Examples include evacuating an area, sheltering-in-place within a protective structure, administering potassium iodide (KI) as a supplemental action, or acquiring an alternate source of drinking water.

The 1992 PAG Manual was written to accommodate the worst release scenario deemed likely at the time - a major accident at a commercial nuclear power plant (NPP) resulting in significant offsite[2] release of radioactive material. Certain characteristics typify NPPs, including: fixed locations at which an accident might occur; a known suite of radionuclides on site that are dominated by short-lived radioisotopes; tight regulatory controls and requirements, skilled operational personnel who plan for and exercise emergency response; state and local involvement in emergency planning; well-developed and zoned emergency evacuation plans and routes, and advance notice (generally hours to days) prior to accidental release of radioactive material into the environment. Therefore, the 1992 PAG Manual provided radiation dose-based PAG values for decision makers for various exposure pathways (such as whole body, skin dose, and food ingestion), and protective actions that were adapted to some extent toward the mix of radionuclides released in NPP incidents, and to the operational environment of a commercial NPP.

Since then, new radiological and nuclear scenarios involving terrorist use of a radiological dispersal device (RDD) or an improvised nuclear device (IND) have gained priority status in radiological emergency response planning. An IND is a crude, yield-producing nuclear weapon fabricated from diverted fissile material. An RDD is a device or mechanism that is intended to spread radioactive material from the detonation of conventional explosives or other means. These types of incidents may occur anywhere with likely no warning.

In 1992, EPA conducted a symposium entitled "Implementing Protective Actions for Radiological Incidents at Other Than Nuclear Power Reactors," to evaluate PAGs for incidents other than NPP accidents and concluded that the PAGs could be applied to all radiological incidents (EPA 1992a). In 2003, DHS tasked an interagency working group to address these new threats as well as long-term cleanup issues. Evaluation by the working group of the threat posed by these and other potential incidents, including transportation and nuclear fuel processing incidents, resulted in a consensus guidance entitled "Application of Protective Action Guides for Radiological Dispersal Device (RDD) and Improvised Nuclear Device (IND) Incidents." The working group concluded that the PAGs and protective actions are applicable to all radiological incidents (DHS 2006). For purposes of this Manual, a radiological incident is defined as "an event or a series of events, whether deliberate or accidental, leading to the release or potential release into the environment of radioactive materials in sufficient quantity to warrant consideration of protective actions." The definition includes acts of nuclear or radiological terrorism, but not nuclear war.

[1] Reference man: A person with the anatomical and physiological characteristics of an average individual that is used in calculations assessing internal dose (also may be called "standard man").

[2] "Site" and "offsite" in this Manual refer to locations where the radiological incident occurs and are not limited to facility-type incidents.

PAGs must accommodate all facilities and circumstances potentially confronting emergency managers, including those that might occur at unpredictable locations, and those that occur with little or no warning. For example, an explosion or fire grants little or no warning and communities are likely to be caught by surprise, as would likely be the case in the event of an RDD or IND attack. Unpredictable locations make advance planning difficult. Sudden release of radioactivity into the environment leaves little time for officials to analyze options, and some protective actions, such as evacuation, may lead to greater net harm. Advance planning on the part of government officials and emergency responders for such cases is critical.

Protective Action Guides and Protective Actions for Radiological Incidents

Radiological emergencies, as defined in the 1992 PAG Manual, are divided into three incident phases for purposes of planning, preparation and response. The incident phases are defined as follows:

- Early Phase - "The period beginning at the projected (or actual) initiation of a release and extending to a few days later, when deposition of airborne materials has ceased and limited, but sufficient information has become available to permit decisions about additional or longer term protection." This phase may last from hours to days.

- Intermediate Phase – "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 phase and late phase and may last from weeks to months.

- Late Phase – "The period beginning when recovery actions designed to reduce radiation levels in the environment to acceptable levels are commenced, and ending when all recovery actions have been completed." This phase may extend from months to years.

PAGs and their associated protective actions are applicable for both the early and intermediate phases. Chapter 6 of this Manual includes a process for establishing cleanup levels for the late phase, rather than pre-establishing numerical PAGs.

Table A provides an overview of exposure routes, various protective actions, and other activities based on the phase of the incident. The table shows which exposure pathways are of concern in the earliest time frames and how the exposure pathways change over time after the incident. Sheltering-in-place and evacuation are the principal protective actions in the early phase. These actions are meant to avoid inhalation of gases or particulates in an atmospheric plume because, during this phase, consumption of contaminated food is generally not a priority issue. Administration of prophylactic drugs may be employed depending on the specific radionuclides released; in particular, potassium iodide (KI) (also called stable iodine) may be administered as a supplementary protective action in incidents involving the release of radioactive iodines, such as during NPP incidents. Some protective actions may begin prior to release of radioactive material in cases in which advance notice is possible.

	Early	Intermediate	Late
EXPOSURE ROUTE			
Direct Plume			
Inhalation Plume Material			
Contamination of Skin and Clothes			
Ground Shine (deposited material)			
Inhalation of Resuspended Material			
Ingestion of Contaminated Water			
Ingestion of Contaminated Food			
PROTECTIVE MEASURES			
Evacuation			
Sheltering			
Control of Access to the Public			
Administration of Prophylactic Drugs			
Decontamination of Persons			
Decontamination of Land and Property			
Relocation			
Food Controls			
Water Controls			
Livestock and Animal Protection			
Waste Control			
Refinement of Access Control			
Release of Personal Property			
Release of Real Property			
Re-entry of Non-emergency Workforce			
Re-entry to Homes			

Table A. Relationship between Exposure Routes, Activities and Time Frames for Effects[3] (DHS 2006)

PAGs are found in Table B with the principal associated protective actions. The PAGs are not meant to be applied as strict numeric criteria, but rather as guidelines to be considered alongside incident-specific factors. PAGs are for use only under emergency circumstances and imply relatively short time periods during which exposures would occur.

The PAGs and corresponding protective actions for the early and intermediate phases as found in the 1992 PAG Manual remain unchanged, except that the PAG for administration of stable iodine has been lowered from 25 rem (250 mSv) adult thyroid dose to 5 rem (50 mSv) child thyroid dose. Additionally, this update provides a new PAG for drinking water and new guidance for cleanup during the late phase. The late phase, however, constitutes remediation and environmental restoration of the affected area and thus is not appropriate for a PAG. No set values for late phase cleanup can be derived in advance given the enormous breadth of potential consequences and site-specific factors that must be considered in establishing cleanup objectives. Rather, risk- or dose-based cleanup values must be established on an incident-specific basis (or the site would be cleaned up under existing regulatory authority of a responsible government agency).

[3] For some activities, protective actions may be taken before release occurs. This would be the case if authorities have forewarning about a potential RDD/IND incident.

Table B. Protective Action Guides for Radiological Incidents

Phase	Protective Action Recommendation	Protective Action Guide
Early	Sheltering-in-place or evacuation of the public[a]	1 to 5 rem projected dose[b]
	Administration of prophylactic drugs - potassium iodide[c,d]	5 rem projected dose to child thyroid[e]
	Limit emergency worker exposure	5 rem (or greater under exceptional circumstances)[f]
Intermediate	Relocation of the public	2 rem projected dose first year. Subsequent years, 0.5rem/y projected dose[b]
	Food interdiction	0.5 rem/y projected dose, or 5 rem/y to any individual organ or tissue, whichever is limiting
	Drinking water interdiction	0.5 rem/y projected dose
	Limit emergency worker exposure	5 rem/y[b]
Late	Final site clean up and restoration	Site-specific optimization

[a]Should normally begin at 1 rem; take whichever action (or combination of actions) that results in the lowest exposure for the majority of the population. Sheltering may begin at lower levels if advantageous.
[b]Total Effective Dose Equivalent (TEDE) - the projected sum of the effective dose equivalent from external radiation exposure (i.e., groundshine) and the committed effective dose equivalent from inhaled radioactive material
[c]Provides thyroid protection from radioactive iodines only.
[d]For other information on radiological prophylactics and treatment, refer to www.fda.gov/cder/drugprepare/default.htm, www.bt.cdc.gov/radiation, or www.orau.gov/reacts.
[e]Committed Dose Equivalent (CDE). FDA understands that a KI administration program that sets different projected thyroid radioactive exposure thresholds for treatment of different population groups may be logistically impractical to implement during a radiological emergency. If emergency planners reach this conclusion, FDA recommends that KI be administered to both children and adults at the lowest intervention threshold (i.e., > 5 rem (50 mSv) projected internal thyroid exposure in children) (FDA 2001).
[f]In cases when radiation control options are not available, or, due to the magnitude of the incident, are not sufficient, doses to emergency workers above 5 rem (50 mSv) may be unavoidable and are generally approved by competent authority. For further discussion see Chapter 2, Section 2.4.4.

Early Phase PAGs and Protective Actions

The early phase is characterized by little or no data on actual releases to the environment and may rely upon crude estimates of airborne releases. Victims are triaged in the early phase. Decision time frames are short and preparation is critical to make prudent decisions when data is lacking or insufficient. Prompt, effective communication with the public, such as an order to shelter-in-place, is another major challenge. Officials should plan for rapid broadcast and dissemination of protective action orders to the public.

The principal PAG for the early phase is a projected dose of 1 to 5 rem (10 to 50 mSv) TEDE;[4] protective actions would normally be initiated at 1 rem (10 mSv). The principal associated protective actions are evacuation and sheltering-in-place. In cases where radioiodine may have been released, administration of the radioprotectant KI as a supplementary protective action should be considered if the committed dose equivalent exceeds 5 rem (50 mSv) to the *child* thyroid. This PAG is reduced from previous guidance. The lower dose, proposed by FDA, is for protection of children based on studies of Chernobyl exposure data. Decontamination is another protective action that may be utilized in the early phase and may include washing of contaminated individuals, changing out of contaminated clothing, and surficial decontamination of critical areas and objects. Individuals should also be instructed to cover breathing ways (nose and mouth) with available filtering material when airborne radionuclides may be present.

The decision to evacuate must weigh the anticipated radiation dose to individuals in the affected population against feasibility of evacuation within a determined time frame. For example, evacuating a population of 50,000 to avoid or reduce radiation dose to that population carries with it a statistical risk of injury or death associated with the

[4] Total Effective Dose Equivalent (TEDE) – the sum of effective dose equivalent from external radiation exposure and the committed effective dose equivalent from inhaled and ingested radioactive material.

evacuation. Evacuation also takes a given amount of time. In the case of an NPP accident the necessary time may be available to allow for orderly and relatively safe evacuation. However, when an incident occurs suddenly, as in the case of a fire or an RDD or IND in a dense urban area, evacuating a large group of people may increase the radiation dose to those people if they are caught within the plume or cannot escape a high dose rate outdoor area. Sheltering-in-place may be warranted for situations when evacuation poses undue risks or results in higher exposures than shelter-in-place.

Limits of exposure for emergency workers are also recommended. These remain unchanged from the 1992 PAG Manual. Responsible officials must use judgment when doses exceeding the OSHA annual limit of 5 rem (50 mSv) may be incurred and advise workers of the risks involved when doses approach 25 rem (250 mSv). There is no dose limit recommended for emergency workers performing life-saving activities. Further discussion of worker dose guidelines can be found in Chapter 2 of the revised PAG Manual.

Intermediate Phase PAGs and Protective Actions

The intermediate phase begins when the source is under control and field data become available. Site stabilization and radiological characterization occur, as well as prompt removal and/or decontamination of highly radioactive "hot-spots." Intermediate phase activities are intended to reduce or avoid dose to the public, control worker exposures, control the spread of radioactive contamination, and prepare for long-term cleanup operations.

Intermediate phase PAGs cover doses received in the first and subsequent years. Decisions must be made concerning the acceptability of occupation of homes and businesses by the public. If radiation doses in an area are deemed too high, temporary relocation should be implemented. The PAG for relocation of the public is 2 rem (20 mSv) in the first year and 0.5 rem (5 mSv) in any subsequent year (the intermediate phase dose does not include ingestion of food and water, which have separate provisions).

Keeping below the 0.5 rem (5 mSv) PAG for out years – the second year and beyond – may be achieved through allowing for the decay of shorter half-life radioisotopes (as in the case of an NPP accident), through decontamination efforts, or through other means of controlling public exposures (such as limiting access to certain areas). In the case of an RDD, in which a longer half-life radioisotope would likely be utilized, reductions in dose may prove difficult to achieve short of full-scale site restoration. If out-year doses are estimated to remain above 0.5 rem (5 mSv) relocation should be considered.

EPA recommends a PAG for the consumption of drinking water of 0.5 rem (5 mSv) in the first year of exposure. If this PAG level is projected to be exceeded, the protective action is to obtain an alternate source of drinking water. In some cases, water treatment or other actions may help reduce radiation doses received via drinking water. While this PAG applies to all potential sources of drinking water, radiological and nuclear emergencies will generally affect surface water bodies, such as lakes, rivers, and reservoirs. Some may cause deeper ground water contamination, but this is less likely. The drinking water PAG is not intended to set an acceptable level of contamination in water nor is it intended to serve as a remediation level in water.

The PAGs for the consumption of food and animal feed come as a recommendation from FDA. The PAG for the consumption of food and animal feed is 0.5 rem (5 mSv) CDE in the first year of exposure, or 5 rem (50 mSv) to any particular organ or tissue (committed dose equivalent), whichever is more limiting. When food or animal feed becomes, or may become, accidentally contaminated to a level that can result in exposure to the public exceeding the PAG, protective actions should be considered. Simple protective actions include covering exposed products, moving animals to shelter, and providing protected feed and water to animals. Temporary embargoes on food and agricultural products may be necessary to prevent public consumption of potentially contaminated food.

Finally, during the intermediate phase, government officials may begin convening to discuss long-term cleanup and site restoration strategies. All actions taken during the early and intermediate phases should be considered with respect to the impact they may have on long-term remediation to avoid actions that will exacerbate or lengthen cleanup operations.

Late Phase Cleanup and Site Restoration

The guidance contained in Chapter 6 of this Manual is not intended to impact site cleanups under other statutory authorities such as EPA's Superfund program, the NRC's decommissioning program, or other federal or state cleanup programs.

The late phase involves the final cleanup of areas and property contaminated with radioactive material. Unlike the early and intermediate phases of an incident, decision makers will have more time and information during the late phase to allow for better data collection, stakeholder involvement, and options analysis. In this respect, the late phase is no longer a response to an "emergency" and is better viewed in terms of the objectives of site restoration and cleanup.

Because of the extremely broad range of potential impacts that may occur (i.e., ranging from light contamination of a small area to widespread destruction of a major metropolitan area in the case of an IND), a pre-established numeric guideline is not recommended as best serving the needs of decision makers in the late phase. Rather, a process should be used to determine the societal objectives for expected land uses, and the options and approaches available, for selecting the most acceptable criteria. For example, if the incident is of limited size, such that the impacted area is small, then it might reasonably be expected that a complete return to normal conditions can be achieved within a short period of time. However, if the impacted area is large, then achieving low cleanup levels for remediation of the entire area and/or maintaining existing land uses (without at least some restrictions) may not be practicable. Such a process of determining societal objectives may be called optimization.

Optimization is a process in which various potential actions to reduce radiation dose are evaluated, the benefits of each action are then compared to the detriments of the action. The optimization process selects actions for which the benefits outweigh the detriments. Optimization (broadly defined) is a concept that is common to many state, federal, and international risk management programs that address radionuclides and chemicals, although it is not always identified as such. Optimization is a flexible approach in which a variety of dose and/or risk benchmarks may be identified from state, federal, or other sources (i.e., national and international advisory organizations). These benchmarks may be useful for analysis of remediation options and cleanup levels may move up or down depending on the site-specific circumstances and balancing of other relevant factors.

Optimization activities are quantitative and qualitative assessments applied at each stage of site restoration decision making, from evaluation of remedial options to implementation of the chosen alternative. The evaluation of options for the late phase of recovery after an incident should balance all of the relevant factors. These factors may include: areas impacted (e.g., size, location relative to population); types of contamination (chemical, biological, and radiological); other hazards present; human health; public welfare; ecological risks; projected land use; preservation or destruction of places of historical, national, or regional significance; technical feasibility; wastes generated and disposal options and costs; costs and available resources to implement and maintain remedial options; potential adverse impacts (i.e., to human health, the environment, and the economy) of remedial options; long-term effectiveness; public acceptability, including local cultural sensitivities; and economic effects (i.e., tourism, business, and industry). The optimization process provides the best opportunity for decision makers to gain public confidence through the involvement of stakeholders. This process may begin during, and proceed parallel to, intermediate phase protective actions.

Chapter 1
Overview

1.1 Introduction

The U.S. Environmental Protection Agency (EPA) has developed this Manual to assist public officials in planning for emergency response to radiological incidents. In the context of this Manual, a radiological incident is defined as an event or a series of events, deliberate or accidental, leading to the release or potential release into the environment of radioactive materials in sufficient quantity to warrant consideration of protective actions. Among the more potentially serious radiological incidents are: a transportation accident involving spent nuclear fuel; a fire in a major facility such as a fuel manufacturing plant; an accident at a federal weapons complex facility; an accident at a commercial nuclear power plant; or an act of radiological or nuclear terrorism, including the use of a radiological dispersion device (RDD) or the detonation of a yield-producing improvised nuclear device (IND). This Manual provides radiological protection criteria intended for application to all radiological incidents requiring consideration of protective actions, other than nuclear war. This Manual is designed for the use of those in federal, state, and local government with responsibility for emergency response planning. This Manual also provides guidance for implementation of the radiological protection criteria.

To aid in the prevention of unnecessary exposure to radiation in the event of a radiological incident when the source of exposure of the public is not under control, the public usually can be protected only by some form of intervention that will disrupt normal living. Such intervention is termed a protective action. A protective action guide (PAG) is the projected numerical dose to reference man[5], or other defined individual, from an unplanned or terrorist-created release of radioactive material at which a specific protective action to reduce or avoid that dose is recommended. Examples include evacuation of an area, sheltering-in-place within a protective structure, or acquiring an alternate source of drinking water.

The objective of this Manual is to provide such PAG levels for the principal protective actions available to public officials during a radiological incident and to provide guidance for their use.

PAGs must accommodate all incidents and circumstances potentially confronting emergency response planners, including those that might occur at unpredictable locations and those that occur with little or no warning. For example, an explosion or fire grants little or no warning and communities are likely to be caught by surprise. This would also likely be the case in the event of an RDD or IND detonation. Unpredictable locations make advance planning challenging. A sudden release of radioactivity into the environment leaves little time for officials to analyze options prior to making decisions. Further, some protective actions, such as evacuation, may lead to greater net harm depending on incident-specific circumstances. Advance planning on the part of government officials and emergency responders for such cases is critical.

The decision to advise members of the public to take an action to protect themselves from radiation from a radiological incident involves a complex judgment in which the risk avoided by the protective action must be weighed in the context of the risks involved in taking the action. Furthermore, the decision may have to be made under emergency conditions, with little or no detailed information available. Therefore, considerable planning is necessary to reduce to a manageable level the complexity of decisions required to effectively protect the public at the time of an incident.

An objective of emergency planning is to simplify the choice of possible responses so that judgments are required only for viable and useful alternatives when an emergency occurs. During the planning process, it is possible to make some value judgments and to determine which responses are not required, which decisions can be made on the basis of prior judgments, and which judgments must be made during an actual emergency. From this process, it is then possible to devise emergency response plans that can be used to respond to the spectrum of hazardous situations that may develop.

For nuclear power plants (NPPs) and other fixed facilities, the main contribution to the protection of the public from accidental releases of radioactive material is provided by site selection, design, quality assurance in construction of the facility, engineered safety systems, and the competence of staff in safe operation and maintenance. To provide

[5] Reference man: a person with the anatomical and physiological characteristics of an average individual that is used in calculations assessing internal dose (also may be called "standard man").

adequate protection of the public health and safety, current Nuclear Regulatory Commission (NRC) regulations require conservatism in design, construction, testing, operation and maintenance of nuclear power plants. A defense-in-depth approach has been mandated for preventing accidents from happening and for mitigating their consequences. Siting in less populated areas is emphasized. Furthermore, emergency response capabilities are mandated to provide additional defense-in-depth protection to the surrounding population (NRC 1980). These measures can reduce both the probability and the magnitude of potential consequences of an accident. Despite these measures, the potential for occurrence of radiological incidents at NPPs and fixed facilities cannot be ignored. Accordingly, emergency response planning to mitigate the consequences of an incident is essential.

1.1.1 Applicability of PAGs and Protective Actions to Radiological Incidents

It should be noted that the 1992 PAG Manual was prepared to accommodate the worst release scenario deemed likely at the time – a major accident at a commercial NPP resulting in significant offsite release of radioactive material. Certain characteristics typify such incidents, including: fixed locations at which an accident might occur; a known suite of radionuclides on site that are dominated by short-lived radioisotopes; tight regulatory controls and requirements; skilled operational personnel that plan for and exercise emergency response; state and local involvement in emergency planning; well-developed and zoned emergency evacuation plans and routes; and substantial advance notice (generally hours to days) prior to the accidental release of radioactive material into the environment. The 1992 PAG Manual therefore provided radiation dose-based PAG values for decision makers by exposure pathway (such as whole body, skin dose, and food ingestion), and protective actions that are adapted to some extent toward the mix of radionuclides that would potentially be released in NPP incidents, and to the operational environment of a commercial NPP.

Recently, however, new radiological and nuclear scenarios involving terrorist use of an RDD or an IND have gained priority status in radiological emergency response planning. These types of incidents will likely occur with no warning, and may occur almost anywhere. One important goal of this revision is to apply the existing PAGs for use in RDD and IND scenarios. The late phase guidance in Chapter 6 of this Manual is not intended to impact site cleanups occurring under other statutory authorities such as EPA's Superfund program, NRC's decommissioning program, or other federal or state cleanup programs.

1.2 Radiological Incident Phases and Protective Actions

For planning purposes it is convenient to identify three incident time phases that are generally accepted as being common to all radiological incident sequences; within each, different considerations apply to most protective actions. These are termed the early, intermediate, and late phases. Although these phases cannot be represented by precise periods of time – and may overlap – they provide a useful framework for the considerations involved in emergency response planning.

1.2.1 The Early Phase

The early phase (also referred to as the emergency phase) is the period at the beginning of a radiological incident when immediate decisions for effective use of protective actions are required and must therefore, for most incidents, be based on the status of the incident site and the prognosis for worsening conditions. When available, predictions of radiological conditions in the environment based on an estimate of the source or actual environmental measurements may also be used. Nuclear facilities, for example, have continuous, real-time radioactive effluent monitoring capabilities to monitor radioactive material released to the environment, and may have a network of offsite measurement stations. Protective actions based on the PAGs may be preceded by precautionary actions during this period. In the case of a transportation accident or an RDD or an IND detonation, there might not be sufficient time for protective actions to be implemented to reduce immediate exposure. The early phase may last from hours to days.

1.2.2 The Intermediate Phase

The intermediate phase is the period beginning after the source and radiological releases have been brought under control and reliable environmental measurements are available for use as a basis for decisions on additional protective actions. It extends until these additional protective actions are terminated. This phase may overlap the early and late phase and may last from weeks to many months.

1.2.3 The Late Phase

The late phase (also referred to as the recovery phase) is the period beginning when recovery actions designed to reduce radiation levels in the environment to final acceptable levels are commenced, and ending when all recovery actions have been completed. This period may extend from months to years. It should be noted that during the late phase, no specific numerical PAG is provided.

1.2.4 Types of Protective Actions

The protective actions available to avoid or reduce radiation dose can be categorized as a function of exposure pathway and incident phase. Evacuation and sheltering-in-place (supplemented by bathing and changes of clothing), are the principal protective actions for use during the early phase to protect the public from exposure to direct radiation and inhalation from an airborne plume. It may also be appropriate to initiate protective action for the milk supply during this period, and, in cases in which emergency response plans include procedures for issuing stable iodine to reduce thyroid dose (FEMA 1985, DHS 2006), this may be an appropriate protective action for the early phase.

Some protective actions are not addressed by assignment of a PAG. For example, the control of access to areas is a protective action whose introduction is coupled with a decision to implement one of the other early or intermediate phase protective actions and does not have a separate PAG. Although the use of simple, ad hoc respiratory protection accessible to many emergency responders may be applicable for supplementary protection in some circumstances, this protective action is primarily for use by emergency workers.

There are two types of protective actions during the intermediate phase. First, relocation and decontamination are the principal protective actions for protection of the public from whole body external exposure due to deposited material and from inhalation of any resuspended radioactive particulate materials during the intermediate and late phases. It is assumed that decisions will be made during the intermediate phase and late phase concerning whether areas from which the public has been relocated will be decontaminated and reoccupied, or condemned and the occupants permanently relocated. The second major type of protective action during the intermediate phase encompasses restrictions on the use of contaminated food and water. This protective action, in particular, may overlap the early and late phases. It should be noted that, during the late phase, no specific numerical PAG is provided.

It is necessary to distinguish between evacuation and relocation with regard to incident phases. Evacuation is the urgent removal of people from an area to avoid or reduce high-level, short-term exposure, usually from the plume or deposited activity. Relocation, on the other hand, is the removal or continued exclusion of people (households) from contaminated areas to avoid chronic radiation exposure. In certain cases, some groups that were not previously evacuated will require relocation. Conditions may develop in which some groups who have been evacuated in an emergency may be allowed to return based on the relocation PAG, while others may be converted to relocation status.

1.3 Basis for Development of the Protective Action Guides

The former Federal Radiation Council (FRC), in a series of recommendations issued in the 1960s, introduced the concept of a PAG and issued guides for avoidance of exposure due to ingestion of Sr-89, Sr-90, Cs-137, and I-131. Those guides were developed for the case of worldwide atmospheric fallout from weapons testing and are appropriate for application to intake due to long-term contamination from such atmospheric releases. They were not developed for protective actions relevant to prompt exposure to an airborne release from a fixed commercial nuclear facility, or for threats of radiation from terrorist attacks using either an RDD or an IND. Guidance in this Manual thus does not supersede this previous FRC guidance, but provides additional guidance for a broader range of exposure pathways and situations.

Due to the wide variety of nuclear facilities and transportation operations with potential for accident, scenarios involving detonation of an RDD or IND, and the atmospheric releases that could subsequently occur, it is not practical to provide specific implementing guidance for all situations. Examples of the types of sources leading to atmospheric releases that this guidance may be applied to are commercial nuclear power facilities, uranium fuel cycle facilities, nuclear weapons facilities, radiopharmaceutical manufacturers and users, space vehicle launch and reentry, RDDs, and INDs. For many specific applications, however, it will be appropriate to develop and use implementing procedures that are designed for use on a case-by-case basis.

The protective action guidance provided in this Manual incorporates the concepts and guidance contained in the FRC Reports 5 and 7 (FRC 1964b and FRC 1965). One of these is that the decision to implement protective actions should be based on the projected dose that would be received if the protective actions were not implemented. However, since these reports were issued, considerable additional guidance has been developed on the subject of emergency response (ICRP 1984b, IAEA 1989). EPA considered the following three principles in establishing exposure levels for the PAGs:

- Prevent acute effects.
- Reduce risk of chronic effects.
- Require optimization to balance protection with other important factors and ensure that actions taken cause more benefit than harm.

The above principles apply to the determination of any PAG. Principles 1 and 3 have been proposed for use by the international community as essential bases for decisions to intervene during an incident and Principle 2 has been recognized as an appropriate additional consideration (IAEA 1989). Appendices C and E of this Manual apply these principles to the determination of PAG levels for evacuation and relocation. Although it is important during emergency planning to consider a range of source terms to assess the costs associated with their implementation, the PAGs are determined so as to be independent of the magnitude or type of radiological release.

1.4 Emergency Planning Zones for Nuclear Facilities

In the case of commercial NPPs and facilities, an emergency planning zone (EPZ)[6] is the area around the facility that is the focus of the facility's emergency planning efforts. For obvious reasons, the concept of an emergency planning zone does not apply to radiological incidents involving a transportation accident or terrorist event as it is not possible to predict where such incidents might occur. However, should one of these types of radiological emergencies occur, one of the first actions taken should be designation of an initial affected area based on the type of incident, the type of release and the environmental conditions, including wind speed and direction, that exist at the time.

The planning elements for developing radiological emergency response plans for radiological incidents at commercial nuclear power facilities are provided in a separate document NUREG-0654/FEMA-REP-1 (NRC 1980), which references the protective action guidance in previous versions of this Manual as the basis for emergency response. Planning elements for other types of radiological incidents should be developed using similar types of considerations. Similarly, guidance for nuclear power facilities regarding time frames for response, the types of releases to be considered, EPZs, and the potential effectiveness of various protective actions is provided in NUREG-0396 (NRC and EPA 1978). The size and shape of the recommended EPZs were only partially based on consideration of the numerical values of the PAGs. A principal additional basis was that the planning zone for evacuation and sheltering-in-place should be large enough to accommodate any urban and rural areas affected and involve the various organizations needed for emergency response. This consideration is appropriate for any facility requiring an emergency response plan involving offsite areas. The emergency detailed planning and preparedness activities required for the 10-mile radius EPZ provide a substantial basis for expansion of response efforts beyond 10 miles in the event that this would become necessary. Experience gained through emergency response exercises is then expected to provide an adequate basis for expanding the response to an actual incident to larger areas, if needed. It is also noted that the 10-mile (16.1 km) radius EPZ for the early phase is large enough to avoid exceeding the PAGs for the early phase at its boundary for low-consequence, nuclear reactor, core-melt incidents and to avoid early fatalities for such incidents. To summarize, the 10-mile EPZ was based primarily on the following considerations:

[6] EPZs established by NUREG-0654 are not applicable to naval nuclear propulsion plants. The largest naval reactors are rated at less than one-fifth of a large U.S. commercial NPP. Additionally, since reactor power is directly linked to propulsion requirements, naval nuclear propulsion plants typically operate at low power when the ship is close to shore when high speeds are not required and are normally shut down when in port. Therefore, less than about 1% of the radioactivity contained in a typical commercial NPP could be released from a naval nuclear propulsion plant, limiting the possible dose to the general public and the size of the area of potential concern. Therefore, there is no need for towns and cities to have special emergency response plans such as those required for cities near commercial nuclear power plants.

- Projected doses from the traditional design basis accidents would not exceed PAG levels outside the zone.
- Projected doses from most core melt sequences would not exceed PAG levels outside the zone.
- For the worst core melt sequences, immediate life threatening doses would generally not occur outside the zone.
- Detailed planning within 10 miles would provide a substantial base for expansion of response efforts in the event that this proved necessary.

The 50-mile (80.5 km) EPZ for ingestion pathways was selected to account for the proportionately higher doses via ingestion compared to inhalation and whole body external exposure pathways.

1.5 Implementation of PAGs and Protective Actions

PAG levels do not imply an acceptable level of risk for normal, nonemergency conditions. They also do not represent the boundary between safe and unsafe conditions. Rather, they are the approximate levels at which the associated protective actions should be considered. Furthermore, under emergency conditions, in addition to the protective actions specifically identified for implementation by a PAG, any other reasonable measures available should be taken to minimize radiation exposure of the general public and of emergency workers.

The sequence of events during the early phase includes evaluation of conditions at the location of the incident, notification of responsible authorities, prediction or evaluation of potential consequences to the general public, recommendations for action, and implementation of actions for the protection of the public. In the early phase of response, the time available to implement the most effective protective actions may be limited.

Immediately upon becoming aware that an incident has occurred that may result in exposure of the population, responsible authorities should make a preliminary evaluation to determine the nature and potential magnitude of the incident. This evaluation should determine whether conditions indicate a significant possibility of a major release and, to the extent feasible, determine potential exposure pathways, populations at risk, and projected doses. The incident evaluation and recommendations should then be presented to emergency response authorities for consideration and implementation. In the absence of recommendations for protective actions in specific areas from the official responsible for the source, the emergency plan should, where practicable, provide for protective action in potentially affected areas.

Contrary to taking protective actions based on very little information during the early phase, dose projections used to support protective actions decisions during the intermediate and late phases will be based on measurements of environmental radioactivity and dose models. Following relocation of the public from affected areas to protect them from exposure to deposited materials, in addition to several other types of data (listed in DHS 2006), it will also be necessary to compile radiological and cost-of-decontamination data to form the basis for radiation-protection decisions for recovery.

Chapter 2
Protective Action Guides for the Early Phase

2.1 Introduction

Rapid action may be required to protect members of the public in the event of an incident involving a large release of radioactive materials into the environment. This chapter identifies the levels of exposure to radiation at which such prompt protective action should be initiated, set forth as PAGs for the general population, as well as guidance for the implementation of corresponding protective actions. Guidance for limiting the exposure of emergency workers during such an incident is also provided. Guidance in this chapter applies to any type of radiological accident or other incident that can result in exposure of the public to an airborne release of radioactive materials. In particular, the objective of this chapter is to provide guidance for estimating projected doses from exposure to an airborne plume of radioactive material and for choosing and implementing protective actions deemed necessary.

In the case of an airborne release, the principal relevant protective actions for the early phase are evacuation or sheltering-in-place. These may be supplemented by additional actions such as washing and changing clothing or by using stable iodine to partially block uptake of radioiodines by the thyroid. The responsible state and/or local authorities will need to decide whether these supplemental protective actions are needed and, if so, where and when they should be implemented. These decisions will be based primarily on (a) actual releases as well as the potential for releases; (b) projected doses as a function of time at various locations in the environment; and (c) dose savings and risks associated with various protective actions.

Additionally, dose conversion factors (DCFs) and derived response levels (DRLs) are provided for radionuclides that are most likely to be important in an incident involving an airborne release of radioactive materials. A DRL is a level of radioactivity in an environmental medium that would be expected to produce a dose equal to its corresponding PAG. A DCF is any factor used to change an environmental measurement to dose in the units of concern. Depending on the exposure pathway, other factors besides the DCF may be required to convert an environmental measurement into a dose. DCFs and DRLs for radionuclides not listed may be developed from the sources referenced in the tables in this chapter. Best practice may dictate use of more up-to-date or incident-specific DCFs, DRLs, and computational methods. It may also be necessary to use DCFs or DRLs not published in this Manual. Therefore, this Manual should not be considered as the only source of DCFs and DRLs. EPA and the FRPCC recommend the use of the most current dosimetry system and include those models from International Commission on Radiological Protection (ICRP) Report 60 to Report 74 (ICRP 60+)[7] for comparison to the PAGs, but recognize many states may continue to use the older ICRP dosimetry model until these states update their regulations and guidance.

2.1.1 Applicability

The PAGs and protective actions presented in this chapter are expected to be used for planning purposes, specifically developing radiological emergency response plans and exercising those plans. During a real incident, as a result of characteristics of the incident and local conditions that cannot be anticipated, professional judgment will be required in their application. For example, a radiological incident could occur in which environmental conditions or other constraints make evacuation impracticable. In these situations, sheltering-in-place may be the protective action of choice. Each case will require judgments by those responsible for decisions on protective actions at the time of an incident.

PAGs and protective actions are intended for general use to protect all individuals of an exposed population. They should be applied equally to most members of the population to avoid social and family disruption and the complexity of implementing different protective actions for different groups under emergency conditions. However, there are some population groups that are at markedly different levels of risk from some protective actions – particularly evacuation. Evacuation at higher values is appropriate for a few groups for whom the risk associated with evacuation is exceptionally high (e.g., people not readily mobile). This Manual provides guidance for this as well.

Some incidents may occur under circumstances in which protective actions cannot be implemented prior to a release. Other incidents may involve only slow, small releases over an extended period so that the urgency is

[7] FDA Guidance, "Accidental Radioactive Contamination of Human Food and Animal Feeds: Recommendations to State and Local Agencies," contained in Chapter 5 of this Manual is based on ICRP 56 and NRPB.

reduced and protective action may be more appropriately treated as relocation (see Chapter 3) rather than as evacuation. Careful judgment will be required to decide whether or not to apply these protective actions for the early phase under such circumstances.

As stated in Chapter 1, PAGs do not imply an acceptable level of risk for normal, nonemergency conditions. Further, the PAGs provided in this chapter are not intended for use as criteria for the ingestion of contaminated food or water, for relocation, or for return to an area contaminated by radioactivity. Separate guidance is provided for each of these situations in Chapters 3, 4, 5, and 6 of this Manual.

2.1.2 Incident Phase

The phase addressed by this chapter is denoted the "early phase." For practical purposes, this is defined as the period beginning at either the projected or actual initiation of a release and extending up to a few days later, when deposition of airborne materials has ceased and enough information has become available to permit reliable decisions about the need for longer-term protection. During the early phase of an incident, doses may accrue from both airborne and deposited radioactive materials. For the purpose of planning, it will usually be convenient to assume that the early phase will last for four days – that is, that the duration of the primary release is less than four days, and that exposure to deposited materials after four days can be addressed through other protective actions, such as relocation, if warranted. (Because of the unique characteristics of some facilities or situations, different time periods may be more appropriate for planning purposes, with corresponding modification of the DCFs cited in Chapter 3).

2.1.3 Notification

For incidents for which little or no warning will be available, such as an RDD or IND, a transportation accident, satellite reentry, or an incident in a foreign country, notification is most likely to occur through local emergency response organizations and the National Response Center. In such cases projections of dose and recommendations to state and local officials for protective actions will be made by the Coordinating Agency in consultation with DHS under the National Response Plan (DHS 2004) with support from the Advisory Team for the Environment, Food, and Health.

For incidents at nuclear facilities, the nuclear facility operator or other designated individual will provide the first notification to state and/or local authorities that a radiological emergency is occurring and will continue to provide follow-up notifications to inform the authorities of changes in the facility status, radioactive material releases, or changes in dose assessments or meteorological conditions. In the case of an incident with the potential for offsite consequences, notification of state and local response organizations by a facility operator should include recommendations, based on plant conditions, for early evacuation and/or sheltering-in-place in predesignated areas. Early estimates of the various components of projected doses to the population at the site boundary, as well as at more distant locations, along with estimated time frames, should be made as soon as the relevant source or release data become available. Emergency response planners should make arrangements with the facility operator to assure that this information will be made available on a timely basis and that dose projections will be provided in units that can be directly compared to the PAGs.

2.1.4 Initial Response and Sequence of Subsequent Actions during the Early Phase

Following notification of an atmospheric release of radioactive material, the protective actions that may be required are those that protect the population from inhalation of radioactive materials in the plume, from exposure to gamma radiation from the plume, and from short-term exposure to radioactive materials deposited on the ground. For releases containing significant amounts of alpha emitters, it may be necessary to include the dose from inhalation of resuspended radioactive material in developing dose projections for comparison to the PAG. For releases that contain a large amount of pure beta emitters, it may also be necessary to consider protective action to avoid doses to the skin from radioactive material deposited on the skin and clothing.

The early phase can further be divided into two periods: (1) the period immediately following the initiation of an incident (possibly before a release has even occurred), when little or no environmental data are available to confirm the magnitude of releases, and (2) the subsequent period, when environmental or source term measurements permit a more accurate assessment of projected doses.

During the first period, speed in completing such actions as evacuating, sheltering-in-place, and controlling access may be critical to minimizing exposure. Environmental measurements made during this period may have limited use because of the lack of availability of significant data and uncertainty about changes in environmental releases of radioactive material from their sources. In the case of an NPP incident, for example, the uncertainty might be due to changes in pressure and radionuclide concentrations within the structures from which the plume is being released. Therefore, it is advisable to initiate early protective actions in a predetermined manner that is related to facility conditions. This will normally be carried out based on recommendations provided by the facility operator. During the second period, when environmental levels are known, these actions can be adjusted as necessary.

Decisions to terminate existing protective actions should include, at a minimum, consideration of the status of the NPP or other release point and the PAGs for relocation. Withdrawal of protective actions from areas where they have already been implemented is usually not advisable during the early phase because of the potential for changing conditions and resultant confusion.

For various types of incidents, the sequence of actions may vary in details, depending on the specific emergency response plan. In general, the sequence and general reporting requirements will be the same.

2.1.5 NPP Emergency Planning Zones and the PAGs

For the purpose of identifying the size of the planning area needed to establish and test radiological emergency response plans, EPZs[8] are typically specified around NPPs and other nuclear facilities. There has been some confusion among emergency planners between these EPZs and the areas potentially affected by protective actions. It is not appropriate to use the maximum distance where a PAG might be exceeded as the basis for establishing the boundary of the EPZ for a facility. For example, the choice of EPZs for nuclear facilities has been based, primarily, on consideration of the area needed to assure an adequate planning basis for local response functions and the area in which acute health effects could occur.[9] These considerations will also be appropriate for use in selecting EPZs for most other nuclear facilities. However, since it will usually not be necessary to have offsite planning if PAGs cannot be exceeded offsite, EPZs need not be established for such cases.

2.1.6 Immediate Protective Action for NPPs and Other Nuclear Facilities

Guidance for developing emergency response plans for implementation of immediate protective actions for incidents at commercial NPPs is contained in NUREG-0654/FEMA-REP-1 (NRC 1980)[10]. Planning elements for incidents at other types of nuclear facilities should be developed using similar considerations. Information on the offsite consequences of incidents that can occur at commercial fuel cycle and material facilities licensed by the NRC can be found in NUREG-1140 (NRC 1988). The "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (NRC and EPA 1978) recommends that states designate an EPZ for protective action for plume exposure. Within this zone, an area should be predesignated for immediate response based on specified plant conditions prior to a release, or, given a release, prior to the availability of information on quantities of radioactive materials released. The shape of this area will depend on local topography, as well as political and other boundaries. Additional areas in the balance of the EPZ, particularly in the downwind direction, may also require evacuation or sheltering-in-place, as determined by dose projections. The size of these areas will be based on the potential magnitude of the release and on an angular spread determined by meteorological conditions and any other relevant factors.

The predesignated areas for immediate protective action may be reserved for use only in the most severe incidents and in cases when the facility operator cannot provide a quick estimate of projected dose based on actual releases. For lesser incidents, or if the facility operator is able to provide prompt offsite dose projections, the area for immediate protective action may be specified at the time of the incident in lieu of using a predesignated area.

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8 EPZs are not applicable to naval nuclear propulsion plants. See note 7 on page 1-4 for more information.

9 The development of EPZs for nuclear power facilities is discussed in the NUREG 0396. EPZs for these facilities have typically been chosen to have a radius of approximately 10 miles for planning evacuation and sheltering-in-place and a radius of approximately 50 miles for planning protection from ingestion of contaminated foods.

10 Immediate protective actions based on in-plant conditions are not applicable to naval nuclear propulsion plants. As discussed in note 7 on page 1-4, the possible dose and size of the area of potential concern is small for naval nuclear propulsion plants.

Such prompt offsite dose projections may be possible when the facility operator can estimate the potential off-site dose, based on information at the facility, using relationships developed during planning that relate abnormal plant conditions and meteorological conditions to potential offsite doses. After the release starts and the release rate is measurable and/or when plant conditions or measurements can be used to estimate the characteristics of the release and the release rate as a function of time, then these factors, along with atmospheric stability, wind speed, and wind direction, can be used to estimate integrated concentrations of radioactive materials as a function of location downwind. Although such projections are useful for initiating protective action, the accuracy of these methods for estimating projected dose will be uncertain prior to confirmatory field measurements because of unknown or uncertain factors affecting environmental pathways, inadequacies of computer modeling, and uncertainty in the data for release terms.

2.1.7 Immediate Protective Actions for RDDs and INDs

Acts of radiological and nuclear terrorism, on the other hand, may occur virtually anywhere. Major cities are potential targets of such incidents. The number of potential targets and the diverse circumstances make response planning for these incidents very difficult. An RDD or IND incident may be initiated without advance warning and the release would likely have a relatively short duration. If an explosive RDD is deployed without warning, there may be no time to take protective actions to reduce plume exposure. In the event of a covert dispersal, discovery or detection may not occur for days or weeks. If an IND explodes, there may only be time to make early phase protective action recommendations to protect against exposure from fallout in areas many miles downwind from the explosion (DHS 2006).

2.2 Exposure Pathways during the Early Phase

The PAG levels for members of the public specified in this chapter refer only to doses incurred during the early phase. These may include external gamma dose and beta dose to the skin from direct exposure to airborne materials and from deposited materials, and the committed dose to internal organs from inhalation of radioactive material. Exposure pathways that make only a small contribution (i.e., less than about 10%) to the dose incurred in the early phase need not be considered. For example, for NPP incidents, inhalation of resuspended particulate materials will generally fall into this category.

Individuals exposed to a plume may also be exposed to deposited material over longer periods of time via ingestion, direct external exposure, and inhalation pathways. Because it is usually not practicable at the time of an incident to project these long-term doses, and because different protective actions may be appropriate, these doses are not included in the dose specified in the PAGs for the early phase. Such doses are addressed by protective action guidance for the intermediate phase (see Chapter 3).

The first exposure pathway, direct exposure from an airborne release of radioactive material, will often be associated with exposure to an atmospheric plume of suspended radioactive material. The detailed content of such a plume will depend on the source of radiation involved and conditions of the incident. For example, in the case of an incident at an NPP, it will most commonly contain radioactive noble gases, but may also contain radioiodines and radioactive particulate materials. Many of these materials emit gamma radiation that can expose people nearby as the plume passes. In the case of some other types of incidents, particularly those involving releases of alpha emitting particulate materials, direct exposure to gamma radiation is not likely to be the most important pathway.

A second exposure pathway, inhalation, can occur during direct immersion in a radioactive atmospheric plume, in which case radioactive material is inhaled. Inhaled radioactive particulate materials, depending on their solubility in body fluids, may remain in the lungs or move via the bloodstream to other organs, prior to elimination from the body. Some radionuclides, once in the bloodstream, are concentrated in a single body organ resulting in only small amounts going to other organs. For example, inhalation of radioiodines will result in the movement of a significant fraction through the bloodstream to the thyroid gland.

As the passage of a radioactive plume containing particulate material and/or radioiodine progresses, some of these materials will deposit onto the ground and other surfaces, creating a third exposure pathway. Persons present after the plume has passed will receive exposure from gamma and beta radiation emitted from these deposited materials. As a result, the skin and clothes may also become contaminated, as is the case when particulate materials or radioiodines are present. When this occurs, internal body organs as well as the skin may be exposed. For a long-term exposure this groundshine exposure pathway can be more significant than external exposure due to immersion in

passing plume if large quantities of radioiodines or gamma-emitting particulate materials are contained in the release.

2.2.1 The Establishment of Exposure Patterns

Sufficient environmental measurements are unlikely to be available to project doses accurately during and immediately following the early response to a radiological incident. Doses must be projected using both initial environmental measurements or estimates of the source term and using atmospheric transport previously observed under or modeled using similar meteorological conditions. These projections are needed to determine whether protective actions should be implemented in additional areas during the early phase.

Source term measurements, or exposure rates or concentrations measured in the plume at a few selected locations, may be used to estimate the extent of the exposed area in a variety of ways, depending on the types of data and computation methods available. The most accurate method of projecting doses is through the use of an atmospheric diffusion and transport model that has been verified for use at the site in question or for similar site conditions. A variety of computer software can be used to estimate exposures in real time, or to extrapolate a series of previously-prepared isopleths for unit releases under various meteorological conditions. The latter can be adjusted for the estimated source magnitude or environmental measurements at a few locations during the incident. If the model projections have some semblance of consistency with environmental measurements, extrapolation to other distances and areas can be made with greater confidence. If projections using a sophisticated site-specific model are not available, a simple, but crude, method is to measure the plume centerline exposure rate[11] at ground level (measured at approximately 1 m height) at a known distance downwind of the release point and to calculate exposure rates at other downwind locations by assuming that the plume centerline exposure rate is a known function of the distance from the release point.

The following relationship can be used for this calculation:

$$D_2 = D_1 \, (R_1/R_2)^y$$

where D_1 and D_2 are exposure rates at the centerline of the plume at distances R_1 and R_2 from the release, respectively, and y is a constant that depends on atmospheric stability. For stability classes A and B, y = 2; for stability classes C and D, y = 1.5; and for stability classes E and F, y = 1. Classes A and B (unstable) occur with light winds and strong sunlight and classes E and F (stable) with light winds at night. Classes C and D generally occur with winds stronger than about 10 mph. This method of extrapolation is risky because the measurements available at the reference distance may be unrepresentative, especially if the plume is aloft and has a looping behavior. In the case of an elevated plume, the ground level concentration increases with distance from the source, and then decreases, whereas any high-energy gamma radiation from the overhead cloud continuously decreases with distance. For these reasons, this method of extrapolation will perform best for surface releases or if the point of measurement for an elevated release is sufficiently distant (usually more than 1 mile) from the point of release for the plume to have expanded to ground level. The accuracy of this method will be improved by the use of measurements from many locations averaged over time.

2.3 The PAGs and Protective Actions for the Early Phase: Evacuation, Sheltering-in-Place, and Administration of Potassium Iodide

The PAGs and corresponding protective actions for response during the early phase of an incident are summarized in Table 2-1. The PAG for evacuation (or, as an alternative in certain cases, sheltering-in-place) is expressed in terms of the projected sum of the effective dose equivalent from external radiation and the committed effective dose equivalent incurred from inhalation of radioactive materials from exposure and intake during the early phase. (Further references to dose to members of the public in this chapter refer to this definition unless otherwise specified). PAGs are also specified in terms of committed dose equivalent to the thyroid and dose equivalent to the

[11] The centerline exposure rate can be determined by traversing the plume at a point sufficiently far downwind that it has stabilized (usually more than one mile from the release point) while taking continuous exposure rate measurements.

skin. The PAG for the administration of KI as a supplementary protective action is specified in terms of the committed dose equivalent to the thyroid from radioiodines.

Table 2-1. PAGs and Protective Actions for the Early Phase of a Radiological Incident

Protective Action Response	PAG (projected dose)	Comments
Sheltering-in-place or evacuation of the public[a]	1 to 5 rem[b]	Evacuation (or, for some situations, sheltering-in-place[a]) should normally be initiated at 1 rem. Further guidance is provided in Section 2.3.1
Supplementary administration of prophylactic drugs – KI[c,d]	5 rem projected dose to child thyroid[e]	May require approval of state medical officials (or in accordance with established emergency plans)

[a]Should normally begin at 1 rem; take whichever action (or combination of actions) that results in the lowest exposure for the majority of the population. Sheltering may begin at lower levels if advantageous.
[b]Total Effective Dose Equivalent (TEDE) (See Section 2.3.2).
[c]Provides thyroid protection from radioactive iodines only.
[d]For other information on radiological prophylactics and treatment, refer to www.fda.gov/cder/drugprepare/default.htm, www.bt.cdc.gov/radiation or www.orau.gov/reacts.
[e]Committed dose equivalent (CDE) (See Section 2.3.2).

2.3.1 Evacuation and Sheltering-in-Place

The basis for the PAGs for the early phase is given in Appendix C of this Manual. In summary, this analysis indicates that evacuation of the public will usually be justified when the projected dose to an individual is 1 rem (10 mSv). This conclusion is based primarily on EPA's determination concerning acceptable levels of risk of effects on public health from radiation exposure in an emergency situation. The analysis also shows that at this radiation dose, the risk avoided is usually much greater than the risk from evacuation itself. However, EPA recognizes the uncertainties associated with quantifying risk associated with these levels of radiation exposure, as well as the variability of risk associated with evacuation under differing conditions.

The primary objective of evacuation is to move individuals away from the path of the plume, thereby avoiding exposure to airborne or deposited radioactive material. Evacuation, if completed before plume arrival, can be 100% effective in avoiding radiation exposure.

The emergency planning process for radiological incidents should include provisions for evacuation of special needs populations, such as persons whose mobility may be impaired, persons without personal transportation, children in schools and child care facilities, and persons in institutions.

A 2005 report by the NRC (NRC 2005) evaluated 50 incidents of public evacuation involving 1,000 or more people. The evacuations studied were in response to natural disasters, technological hazards, and malevolent acts occurring between January 1, 1990 and June 30, 2003. The report indicated that the following factors affecting notification of the public were statistically significant for an efficient evacuation:

- Public familiarity with alerting methods.
- Door-to-door notification.

The NRC report also indicated that public awareness of the hazard, evacuation procedures, and alerting methods in particular were often cited as contributing to the efficiency and effectiveness of evacuation. Findings of the NRC report also indicated that many communities are making improvements to response capabilities. These improvements include modernizing communication systems, conducting transportation analyses to improve traffic flow, improving local education awareness, and developing interagency and cross-boundary coordination plans.

Other general conclusions regarding evacuation that may be useful for planning purposes are summarized below:

- Advance planning is essential to identify potential problems that may occur in an evacuation (Hans et al. 1975). NRC case study interviewees of the 2005 report cited the following aspects of planning as contributing to efficiency and effectiveness of evacuation (NRC 2005):

    - High level of cooperation among agencies;
    - Use of multiple forms of emergency communications;
    - Community familiarity with alerting methods;
    - Community communication; and
    - Well-trained emergency responders.

- Evacuation costs are highly location-dependent and usually will not be a deterrent to carrying out an evacuation (Hans et al. 1975).

- Large or small population groups can be evacuated effectively with minimal risk of injury or death (Hans et al. 1975). In the NRC report, only six of the 50 cases studied involved deaths from the hazard, and of those six, only one case involved death from the evacuation itself (NRC 2005).

- The risk of injury or death to individual evacuees from transportation does not change as a function of the number of persons evacuated and can be conservatively estimated using National Highway Safety Council (NHSC) statistics for motor vehicle accidents (DOT 2004). Further, of the 50 evacuation cases evaluated by the NRC study, only four reported traffic accidents during evacuation (NRC 2005). Anecdotal information suggests that the evacuation risks will be lower than the NHSC statistics indicate.

Some judgment will be necessary when considering the types of protective actions to be implemented, and at what levels, in an emergency situation. Although the PAG for evacuation or sheltering-in-place is expressed as a range of 1-5 rem (10-50 mSv), it is emphasized that evacuation of members of the general population should be initiated for most incidents at a projected dose of 1 rem (10 mSv). It should be recognized that doses to some individuals may exceed 1 rem (10 mSv), even if protective actions are initiated within this guidance. It is also possible that conditions may exist at specific facilities or during potential scenarios that might warrant consideration of values other than those recommended for general use here.

Sheltering-in-place may be preferable to evacuation as a protective action in some situations. Due to the higher risk associated with evacuation of some special groups in the population (e.g., those who are not readily mobile), sheltering-in-place may be the preferred alternative for such groups as a protective action at projected doses of up to 5 rem (50 mSv). In addition, under unusually hazardous environmental circumstances, such as high levels of airborne particulates or physically dangerous conditions, use of sheltering-in-place at projected doses up to 5 rem (50 mSv) to the general population (and up to 10 rem (100 mSv) to special groups) may become justified. Sheltering-in-place may also provide protection equal to or greater than evacuation due to the nature of the source term and/or in the presence of time-specific or other site-specific conditions. Illustrative examples of situations or groups for which evacuation may not be appropriate at 1 rem (10 mSv) include:

- the presence of severe weather;
- competing disasters;
- institutionalized persons who are not readily mobile; and
- local physical factors that impede evacuation.

Evacuation will seldom be justified at less than 1 rem (10 mSv). The examples described above regarding selection of the most appropriate protective action are intended to be illustrative and not exhaustive. In general, sheltering-in-place should be preferred to evacuation whenever it provides equal or greater protection.

No specific minimum PAG is established for initiation of sheltering-in-place. Sheltering-in-place is a low-cost, low-risk protective action that can provide protection with an efficiency ranging from zero to almost 100%, depending

on the circumstances. It can also be particularly useful to assure that a population is positioned so that, if the need arises, communication with the population can be carried out expeditiously. For the above reasons, planners and decision makers should consider implementing sheltering-in-place at projected doses below 1 rem (10 mSv); however, implementing protective actions for projected doses at very low levels would not be reasonable (e.g., below 0.1 rem (1 mSv)). This guidance should not be construed as establishing an additional lower level PAG for sheltering-in-place.

Analyses for some hypothesized incidents, such as short-term releases, show that sheltering-in-place in residences and other buildings can be highly effective at reducing dose, may provide adequate protection, and may be more effective than evacuation when evacuation cannot be completed before plume arrival (DOE 1990). However, reliance on large dose reduction factors as a result of sheltering-in-place should be accompanied by cautious examination of possible failure mechanisms, and except in very unusual circumstances, should never be relied upon at projected doses greater than 10 rem (100 mSv). Such analyses should be based on realistic or "best estimate" dose models and include unavoidable dose incurred during evacuation. Sheltering-in-place and evacuation are discussed in greater detail in Section 2.4.

2.3.2 Thyroid Protection during the Early Phase

Following only those radiological incidents in which radioactive iodine is released, the thyroid is at a disproportionately high risk for induction of nonfatal cancer and nodules compared to other internal organs. As such, additional guidance to limit the risk of these effects is provided in this section (see footnote to Table 2-1). In addition, effective dose (the quantity used to express the PAG) encompasses only the risk of fatal cancer from irradiation of organs within the body and does not include dose to skin.

The use of stable iodine to protect against uptake of inhaled radioiodine by the thyroid is recognized as an effective supplement to evacuation for situations involving radioiodine releases when evacuation cannot be implemented or exposure could occur during evacuation. Stable iodine is most effective when administered immediately prior to exposure to radioiodine. However, significant blockage of the thyroid dose can be provided by administration of KI within 1 or 2 hours after uptake of radioiodine.

It should be noted that RDDs are not very likely to contain radioiodine, and thus KI would not be effective. The administration of other prophylactic drugs should be evaluated on a case-by-case basis dependent on the nature of the event and the radioisotopes involved (DHS 2006).

Changing of clothing is recommended primarily to provide protection from beta radiation from radioiodines and particulate materials deposited on the skin or clothing. Calculations indicate that dose to skin should seldom, if ever, be a controlling pathway. However, it is a good radiation protection practice to recommend these actions, even for alpha-emitting radioactive materials, as soon as practical for persons significantly exposed to a contaminating plume (i.e., when the projected dose from inhalation would have justified evacuation of the public under normal conditions).

FDA recently updated its guidance on the use of KI as a thyroid blocking agent during radiological emergencies (FDA 2001 and FDA 2002). FDA based these new dose recommendations on a review of the thyroid cancer data from the Chernobyl reactor accident of April 1986 and the experience of Poland in administering KI following the Chernobyl release (FDA 2001).

FDA recommends the following:

- Children 0-18 years of age: Administer KI when the projected radiation dose to the thyroid is 5 rem (50 mSv) or greater.
- Pregnant and lactating women: Administer KI when the projected radiation dose to the thyroid is 5 rem (50 mSv) or greater.
- Adults up to 40 years of age: Administer KI when the projected radiation dose to the thyroid is 10 rem (100 mSv) or greater.

- Adults over 40 years of age: Administer KI when the projected radiation dose to the thyroid is over 500 rem (5 Sv) in order to prevent hypothyroidism.

It should be noted that adults over 40 need to take KI only in the case of a projected large internal radiation dose to the thyroid (>500 rem) to prevent hypothyroidism which, if it occurred, would lead to lifelong dependence on thyroid hormone replacement therapy. As a rule, individuals with known allergy to KI or with pre-existing thyroid disease (e.g., Graves' disease, thyroid nodules, Hashimoto's thyroiditis) that might predispose them to adverse reactions should avoid KI. It is most likely these individuals will be adults who have little or no risk of developing thyroid cancer from radioactive exposure to the thyroid and who may in these cases incur substantial risks from taking KI. Thyroid irradiation in older adults (i.e., over 40 years of age) is associated with an extremely low incidence of cancer (FDA 2001).

However, FDA understands that a KI administration program that sets different projected thyroid radioactive exposure thresholds for treatment of different population groups may be logistically impractical to implement during a radiological emergency. If emergency planners reach this conclusion, FDA recommends that KI be administered to both children and adults at the lowest intervention threshold (i.e., >5 rem (50 mSv) projected internal thyroid exposure in children) (FDA 2001).

The protective effect of a single dose of KI lasts approximately 24 hours. It should be administered daily until the risk of significant exposure to radioiodine (either by inhalation or ingestion) no longer exists. Some persons may not be able to tolerate KI and pregnant and lactating women should not be given repeated doses except during severe contamination/exposure events. In such a case, neonates should be monitored for thyroid function. Individuals in this category should be given priority in regards to other protective measures such as sheltering-in-place, evacuation, and food supply control.

With the knowledge that radioiodines pass into breast milk, pediatricians should caution lactating mothers not to breastfeed their infants after potential exposure to the release of radioiodines unless no alternative infant food is available. If a lack of alternative infant food necessitates that a lactating mother must continue to breastfeed her infant, then repeat dosing to the mother with KI should be avoided. However, if the mother must continue to breastfeed her infant during an event with ongoing exposure to radioiodines, then an additional dose or doses of KI to the mother may be necessary.

Once the plume has passed and protective actions such as evacuation and/or sheltering-in-place are implemented, exposure to radioiodine should be limited through the implementation of food control measures and the administration of KI should be suspended. Food control measures include providing the public with non-contaminated food supplies while awaiting the eventual radioactive decay of contaminated food. As a result of radioactive decay, grain products and canned milk and vegetables from sources affected by radioactive fallout will not present a risk from radioiodine if they have been stored for weeks to months after production. There is no reason to continue the administration of KI once the plume has passed and food control measures have been put into place (see Chapters 3, 4 and 5 for additional information on these and other intermediate phase protective actions).

2.4 Implementing the Protective Actions for the Early Phase: Sheltering-in-Place and Evacuation

This section provides guidance for implementing the principal protective actions (evacuation and sheltering-in-place) for protection against the various exposure pathways resulting from an airborne plume during the early phase. Sheltering-in-place refers to the use of a readily available structure that will provide protection from exposure to an airborne plume. Evacuation refers to the movement of individuals away from the path of the plume.

Evacuation and sheltering-in-place provide different levels of dose reduction from the principal exposure pathways (inhalation of radioactive material and direct gamma exposure from the plume or from material deposited on surfaces). The effectiveness of evacuation will depend on many factors, such as how rapidly it can be implemented and the nature of the incident. For incidents where the principal source of dose is inhalation, evacuation could increase exposure if it is implemented during the passage of a short-term plume, since vehicles used to transport evacuees provide little protection against exposure (DOE 1990). Sheltering-in-place, which in most cases can be

almost immediately implemented, varies in effectiveness depending upon the type of release, the shelter available, the duration of the plume passage, and climatic conditions.

Studies have been conducted to evaluate sheltering-in-place (EPA 1978a) and evacuation (Hans et al. 1975) as protective actions for incidents at nuclear power facilities. Reference EPA 1978b suggests one method for evaluating and comparing the benefits of these two actions. This requires collecting planning information before and data following an incident, and using calculations and graphical means to evaluate whether evacuation, sheltering-in-place, or a combination of sheltering-in-place followed by evacuation should be recommended at different locations. Because of the many interacting variables, authorities may be forced to choose between making decisions during the planning phase, based on assumed data that may be grossly inaccurate, or using a time-consuming more comprehensive process after the incident when data may be available. In the former situation, the decision may not have a sound basis, whereas in the latter, the decision may come too late to be useful.

The recommended approach is to use planning information for making early decisions. The planned response should then be modified following the incident only if timely detailed information is available to support such modifications.

The planner should first compile the necessary information about the EPZ around the facility. For the case of nuclear power reactors, some of this information is described in NUREG-0654/FEMA-REP-1 (NRC 1980). For urban area emergency planners, much of the same information can and should be gathered to enhance preparedness. It should include identifying the population distribution, the effectiveness of residences and other structures for sheltering-in-place, institutions containing population groups that require special consideration, evacuation routes, logical boundaries for evacuation zones, transportation systems, communications systems, and special problem areas. In addition, the planner should identify the information that may be available following an incident, such as environmental monitoring data, meteorological conditions, and plant conditions. The planner should identify key data or information that would justify specific protective actions. The planner should also acknowledge the special conditions that would counter indicate the use of particular protective actions. For example, it may be appropriate to discourage the use of sheltering-in-place during periods of extreme temperatures that could lead to stress or injury.

The following sections discuss key factors that affect the choice between evacuation and sheltering-in-place.

2.4.1 Evacuation

The primary objective of evacuation is to avoid exposure to airborne or deposited radioactive material by moving individuals away from the path of the plume. Evacuation, if completed before plume arrival, can be 100% effective in avoiding future exposure. Even if evacuation coincides with or follows plume passage, a large reduction of exposure may be possible.

2.4.2 Sheltering-in-Place

Sheltering-in-place refers here to the use of readily available structures for protection against exposure to an airborne plume. To some extent, sheltering-in-place decisions should be based on the types of radioactive materials released and the likely exposure pathway. For releases consisting primarily of noble gases, external gamma exposure will be the dominant pathway.

Sheltering-in-place may be an appropriate protective action because:

- it positions the public to receive additional instructions when the possibility of high enough doses to justify evacuation exists, though remains unlikely;
- it may provide protection equal to or greater than evacuation;
- it is less expensive and disruptive than evacuation;
- since it may be implemented rapidly, shelter-in-place may be the protective action of choice if rapid evacuation is impeded by a) severe environmental conditions, e.g., severe weather or floods; b) health constraints, e.g., patients and workers in hospitals and nursing homes; c) long mobilization times that

may be associated with certain individuals, such as industrial and farm workers, or prisoners and guards; or d) physical constraints to evacuation, i.e., inadequate roads;

- sheltering-in-place may be more effective against inhalation of radioactive particulates than against external gamma exposure, especially for short- term plumes; and

- sheltering-in-place is usually not appropriate for areas where high doses are projected or for exposure lasting longer than 2 complete air exchanges of the shelter.

The use of large structures, such as shopping centers, schools, churches, and commercial buildings, as collection points during evacuation mobilization will generally provide greater protection against gamma radiation than use of small structures.

As with evacuation, delay in taking shelter during plume passage will reduce the protection from exposure to radiation. The degree of protection provided by structures is governed by attenuation of gamma radiation by structural components (the mass of walls, ceilings, etc.) and by outside/inside air exchange rates.

If external dose from the plume or from deposited materials is the controlling criterion, shelter construction and shelter size are the most important considerations; ventilation control and filtering are less important. Although sheltering-in-place will reduce the gamma exposure rate from deposited materials, it is not a suitable protective action for this pathway for long-term exposure. The main factors that reduce whole body exposure are:

- wall materials and thickness and size of structure;

- number of stories overhead; and

- use of a central location within the structure.

If a major release of radioiodine or respirable particulate materials occurs, inhalation dose will be the controlling pathway. For releases consisting primarily of noble gases, external gamma exposure will be most important. However, when inhalation is the primary exposure pathway, consideration should be given to the following:

- Dose reduction factors for sheltering-in-place can be improved in several ways for the inhalation pathway, including reducing air exchange rates by sealing cracks and openings with cloth or weather stripping, tape, etc. Although the risk to health from the action could be a constraint (particularly for infants and the infirm), using towels or handkerchiefs as a mask to filter the inhaled air will reduce dose from inhalation.

- Following plume passage, people should open shelters to reduce airborne activity trapped inside, and they should leave high exposure areas as soon as possible after cloud passage to avoid exposure to deposited radioactive material.

2.4.3 General Guidance for Evacuation and Sheltering-in-Place

The process of evaluating, recommending, and implementing evacuation or sheltering-in-place for the public is far from an exact science, particularly in light of time constraints that may prevent thorough analysis at the time of an incident. Their effectiveness, however, can be improved considerably by planning and exercise in advance. When applicable, early decisions should be based on information collected from the EPZ during the planning phase and on information regarding conditions at the nuclear facility at the time of the incident. For transportation accidents, RDDs, INDs, and other incident scenarios for which EPZs are not practicable, best estimates of dose projections should be used for decisions between evacuation and sheltering-in-place.

The following is a summary of planning guidance for evacuation and sheltering-in-place:

1. Evacuation may be the only effective protective action close to the plume source.

2. Evacuation will provide total protection from any airborne release if it is completed before arrival of the plume.

3. Evacuation may increase exposure if carried out during the plume passage.

4. Evacuation is also appropriate for protection from groundshine in areas with high exposure rates from deposited materials when suitable shelter is not available.

5. Sheltering-in-place may be appropriate (when available) for areas not designated for immediate evacuation because:

   a. it may provide protection equal to or greater than evacuation for rapidly developing releases (e.g., RDDs) if followed by evacuation; and

   b. it positions the public to receive additional instructions.

6. Sheltering-in-place is usually not appropriate for areas where high doses are projected or for exposure lasting longer than two complete air exchanges of the shelter.

7. Since it may be implemented rapidly, sheltering-in-place may be the protective action of choice if rapid evacuation is impeded by: a) severe environmental conditions, e.g., severe weather or floods; b) health constraints, e.g., patients and workers in hospitals and nursing homes; c) long mobilization times that may be associated with certain individuals, such as industrial and farm workers, or prisoners and guards; or d) physical constraints to evacuation, e.g., inadequate roads.

8. If a major release of radioiodine or particulate materials occurs, inhalation dose may be the controlling criterion for protective actions. In this case:

   a. breathing air filtered through common household items (e.g., folded handkerchiefs or towels) may be of significant help; and

   b. after confirmation that the plume has passed, sheltering-in-place is no longer effective and may be harmful. Shelters should be opened to avoid airborne activity trapped inside, and persons should leave high exposure areas as soon as possible after cloud passage to avoid exposure to deposited radioactive material.

9. If dose from external gamma radiation is the controlling criterion, shelter construction and size are the most important considerations; ventilation control and filtering are less important. The main factors that reduce whole body external dose are a) wall thickness and size of structure; b) number of stories overhead; c) central location within the structure; and d) the height of the cloud with respect to the building.

2.4.4 Comparison with Previously-Recommended PAGs

Many emergency response plans have already been developed using previously-recommended PAGs that apply to the whole body from direct (gamma) radiation from the plume and to the thyroid from inhalation of radioiodines. For NPP incidents, the former PAG for whole body exposure provides public health protection comparable to that provided by the new PAG expressed in terms of effective dose equivalent. This is demonstrated in Table C-10 (Appendix C), which shows comparative doses for NPP fuel-melt incident sequences having a wide range of magnitudes. As stated in Section 2.3.2, base on revised guidance, the PAG for thyroid dose has been lowered from a projected dose level of 25 rem (250 mSv) adult thyroid dose to a projected thyroid dose level of 5 rem (50 mSv) child thyroid dose (FDA 2001). On the other hand, application of these PAGs to alpha emitting radionuclides leads to quite different derived response levels from those based on earlier health physics considerations because of new DCFs and the weighting factors assigned to the exposed organs (EPA 1999).

2.5 Radiation Protection for the Emergency Responder and Planning for Implementation of the Protective Action Guides[12]

The purpose of this section is to discuss the context for the PAGs and to provide guidance for their application, particularly for the protection of emergency responders. Response organizations need to develop plans and protocols that address radiation protection during a radiological incident and that ensure appropriate training for responders and decision makers. Although this section discusses some of the important issues and information that must be communicated, it is not intended to provide a comprehensive discussion of the topic. Other detailed reports on radiation risk, risk management decision making, training, and public communication should be consulted in the development of plans, protocols, and training materials. Organizations that have published such reports include the National Council on Radiation Protection and Measurements, ICRP, International Atomic Energy Agency (IAEA), the American Nuclear Society, and the Health Physics Society.

2.5.1 The Protective Action Guides and Operations Guidelines into Perspective

The recommendations in this report were developed to assist decision makers and responders in planning for radiological terrorist incidents using RDDs and INDs. Decisions regarding protective actions for workers and the public during such incidents are risk management decisions and the recommendations in this section are provided in that context. In all cases, all practical and reasonable means should be used to reduce or eliminate exposures that are not necessary to protect public health and welfare.

2.5.2 The Difference between PAGs for Emergencies and Other Operations

Worker and public protection guidance and standards for normal operations are typically developed through risk management approaches and are implemented in federal and state regulations (e.g., 10 CFR part 20; 10 CFR part 835; 29 CFR 1910.1096). However, many factors or decision criteria differ during a radiological emergency versus normal operations. Some of the key decision criteria differences between emergency PAGs and typical occupational and public protection standards are shown in Table 2-2. Although there are times when implementation of standards or guidelines can cause or enhance other risks, these secondary risks normally can be controlled. Standards for normal operations provide a margin of safety that is greater than that in guidelines for emergency response because that margin can be provided in a manner that ensures no significant increase in public health risk or detriment to the public welfare. Currently, the development of standards and guidelines for normal operations is done in a manner that provides reasonable assurance that implementation of the standards will not cause more risk than it averts.

Table 2-2. Different Risk Management Considerations for Emergency and Normal Operations

Emergency	Normal Operations
An adversary may attempt to create conditions that will cause high radiation exposures, widespread contamination, and mass disruption.	Key elements to radiation protection are to contain radioactivity and confine access to it.
Key elements to radiation protection are to contain radioactivity and confine access to it.	There is adequate time to fully characterize situations and determine risks and mitigating measures.
Actions must be taken as soon as possible to minimize exposures even when information on the risks is incomplete.	Inaction or delays may increase costs but rarely results in consequences that cannot be mitigated.
During emergencies, the undesired consequences can be significant, uncontrollable, and unpredictable.	Consequences associated with implementation of the standard are well characterized, considered, and controlled so as not to be of concern from either a health or public welfare perspective.

During the early phase of an emergency response, however, tradeoffs are not only cost-related but may directly impact public health and welfare. It is difficult to ensure that implementation of recommendations does not result in more harm than good. Guidelines that prevent or restrict a responder's ability to provide medical assistance based on an uncertain cancer risk may result in loss of life of incident victims. If the PAGs delay firefighters' ability to control fires, resulting property damage can seriously affect overall public welfare or even cause an increase to health risks associated with the incident. The decision maker's use of public protection PAGs also must consider

[12] Source: Department of Homeland Security. Application of Protective Action Guides for Radiological Dispersal Device (RDD) and Improvised Nuclear Device (IND) Incidents. 71 Fed. Reg. 1 (Jan. 3, 2006).

secondary risks. Evacuation of the public could result in loss of life and injury as a result of the evacuation process that exceeds the increased public risk should the evacuations not occur. These and other considerations require that the PAGs and associated operational guides be developed so that decisions can appropriately consider risks, detriments, and costs associated with a radiological incident, as well as those associated with implementation of the protective action to, on balance, benefit the public welfare. Emergency response actions should be carried out following a careful consideration of both the benefits to be achieved by the ''rescue'' or response action (e.g., the significance of the outcome to individuals, populations, property, and the environment at risk considering their likely impaired status following an incident), and the potential for additional health impacts to those conducting the emergency response operation. That is, in making an emergency response decision, the potential for the success of the response/rescue operation and the significance of its benefits to the community should be balanced against the potential for rescuers to be exposed to new and significant health and safety risks. Actions should be based on balancing risks and benefits. Nothing in this guidance should be construed to imply that appropriate steps should not be taken to minimize dose to workers and the public, consistent with the "as low as reasonably achievable (ALARA)" principle applied to radiation protection activities in the United States. However, actions similarly should not restrict lifesaving or property-saving actions necessary for protection of public and public welfare.

2.5.3 Controlling Occupational Exposures and Doses to Emergency Responders

This section provides guidance for emergency responders concerning occupational doses of radiation during an emergency response. In many emergency situations, actual exposure of workers, including emergency responders, may be controlled to low doses when proper precautions are taken. However, it is important to recognize that conditions that exist during a radiological incident may limit the effectiveness of these precautions for some emergency responders. One of the major radiation protection controls used for normal operations is containment of the radioactive material. Another is to keep people away from the sources. However, during a radiological incident, use of these controls may not be possible. As a result, radiation exposures, particularly to emergency responders, may be unavoidable and may have the potential to exceed limits used for normal operations. Nonetheless, every reasonable effort should be made to control doses to levels that are as low as practicable.

2.5.4 Maintaining the ''As Low As Reasonably Achievable'' Principle

To minimize the risks from exposure to ionizing radiation, employers of emergency responders should prepare emergency response plans and protocols in advance to keep worker exposures as low as reasonably achievable. These protocols should include, to the extent they can be employed, the following health physics and industrial hygiene practices:

- Minimizing the time spent in the contaminated area (e.g., rotation of workers).
- Maintaining the maximum distance from sources of radiation.
- Shielding of the radiation source from the receptor.
- Tailoring of hazard controls to the work performed.
- Properly selecting and using respirators and other personal protective equipment (PPE) may be useful to prevent exposure to internally deposited radioactive materials (e.g., alpha and beta emitters).
- Using prophylactic medications, where medically appropriate, that either block the uptake or reduce the retention time of radioactive material in the body. The incident commander should be prepared to identify, to the extent possible, all hazardous conditions or substances and to perform appropriate site hazard analysis. Emergency management plans should include protocols to control worker exposures, establish exposure guidelines in advance, and outline procedures for worker protection. All activities should be performed in conjunction with emergency procedures that include provisions for exposure monitoring, worker training on the hazards involved in response operations and ways to control them, and medical monitoring.

2.5.5 Understanding Dose and Risk Relationships

Responders and incident commanders should understand the risks associated with radiation. PAG recommendations in this document provide a guideline level of 5 rem (50 mSv) for worker protection and alternative response worker guidelines [13] (see Table 2-3) for certain activities where exposures below 5 rem (50 mSv) cannot be maintained.

Table 2-3. Response Worker Guidelines

Total effective dose equivalent (TEDE) guideline	Activity	Condition
5 rem	All occupational exposures	All reasonably achievable actions have been taken to minimize dose.
10 rem[a]	Protecting valuable property necessary for public welfare (e.g., a power plant).	Exceeding 5 rem unavoidable and all appropriate actions taken to reduce dose. Monitoring available to project or measure dose.
25 rem[b]	Lifesaving or protection of large populations	Exceeding 5 rem unavoidable and all appropriate actions taken to reduce dose. Monitoring available to project or measure dose.

[a]For potential does >10 rem, special medical monitoring programs should be employed,and exposure should be tracked in terms of the unit of absorbed dose (rad) rather than TEDE (rem).
[b]In the case of a very large incident, such as an IND, incident commanders may need to consider raising the property and lifesaving response worker guidelines to prevent further loss of life and massive spread of destruction.

It is likely during most radiological incidents that the radiation control measures discussed above will be able to maintain doses below the 5 rem (50 mSv) occupational exposure PAG in almost all situations, including firefighting, general emergency response and transport to, and medical treatment of, contaminated victims at hospitals. However, in those situations in which victims are injured or trapped in high radiation areas or can only be reached via high radiation areas, exposure control options may be unavailable or insufficient, and doses above 5 rem (50 mSv) may be unavoidable. Response decisions allowing actions that could result in doses in excess of 5 rem (50 mSv) can only be made at the time of the incident, under consideration of the actual situation. In such situations, incident commanders and other responders need to understand the risk posed by such exposures in order to make informed decisions. The Response Worker Guidelines for life and property saving activities in Table 2-3 are provided to assist such decisions. The catastrophic event represented by an IND can cause other immediate widespread physical hazards such as firestorm and building instability; emergency intervention will be integral to preventing further loss of life and additional destruction. This intervention may result in increased exposure to emergency response personnel. Exceeding the Response Worker Guidelines in Table 2-3 in such an event may be unavoidable. Persons undertaking an emergency mission covered under the alternative occupational PAG levels should do so with full awareness of the sub-chronic and chronic risks involved, including knowledge of numerical estimates of the risk of delayed effects, and they should be given reasonable assurance that normal controls cannot be utilized to reduce doses below the general 5 rem occupational exposure PAG. The 25 rem (250 mSv) lifesaving Response Worker Guidelines provide assurance that exposures will not result in detrimental deterministic health effects (i.e., prompt or acute effects). If, due to extensive public health and welfare benefits (i.e., optimization considerations), response actions are deemed necessary that cause exposures that may exceed the 25 rem (250 mSv) alternative Response Worker Guideline, such response actions should only be taken with an understanding of the potential acute effects of radiation to the exposed responder (Table 2-4) and based on the determination that the benefits of the action clearly exceed the associated risks.

13 Alternative response worker guidelines are applicable only during emergency situations. They typically apply during the early phase of the emergency but may also be applicable in later phases under emergency situations such as a fire or a structure failure that puts life and property at risk. In addition to the obvious life saving situation, other examples of where the guidelines may be applicable include situations where it is necessary to access controls to prevent or mitigate explosions, fires or other catastrophic events. The alternative response worker guidelines are not applicable to normal restoration or cleanup actions.

Table 2-4. Acute Radiation Syndrome[a]

Feature or Illness	Effects of Whole Body Absorbed Dose from external radiation or internal absorption, by dose range in rad				
	0-100	100-200	200-600	600-800	>800
Nausea, Vomiting	None	5-50%	50-100%	75-100%	90-100%
Time of Onset		3-6 h	2-4 h	1-2 h	<1 h to minutes
Duration		<24 h	<24 h	<48 h	<48 h
Lymphocyte Count	Unaffected	Minimally Decreased	<1,000 at 24 h	<500 at 24 h	Decreases within hours
Central Nervous System Function	No Impairment	No Impairment	Cognitive impairment for 6-20 h	Cognitive impairment for > 20 h	Rapid incapacitation
Mortality	None	Minimal	Low with aggressive therapy	High	Very High: Significant neurological symptoms indicate lethal dose

[a]Prompt health effects with whole-body absorbed doses received within a few hours.
Source: Medical Management of Radiological Casualties, Second Edition, Armed Forces Radiobiology Research Institute. Bethesda, MD, April 2003.

The following paragraph is presented to help illustrate how certain toxicity information may be relevant in response decision making during emergencies. It is important to note that the approach used below to translate dose to risk in this discussion is a simplistic approach useful in developing rough estimates of risks for comparative purposes given limited data. However, other more realistic approaches are often used in assessing risks for risk management decisions (other than for emergencies) when more complete information about the contaminants and the potential for human exposure is available. These other approaches rely on radionuclide specific risk factors (e.g., Federal Guidance Report #13[14] and EPA Health Effects Assessment Summary Tables). The estimated risk of fatal cancer [15] for workers exposed to 10 rem (100 mSv) is 0.6% (6 cases per thousand exposed). Workers exposed to 25 rem (250 mSv) have an estimated risk of fatal cancer of 1.5% (15 cases per thousand exposed). Because of the latency period of cancer, younger workers face a larger risk of fatal cancer than older workers (for example, when exposed to 25 rem, 20 to 30-year-olds have a 9.1 per thousand risk of premature death, while 40 to 50-year-olds have a 5.3 per thousand risk of premature death).[16]

2.5.6 Incident Commanders and Responders Need to take Proper Training in Advance

When the 5 rem guideline is exceeded, workers should be provided the following:

- Medical follow-up.
- Training with respect to the risk associated with exposure to ionizing radiation.
- A thorough explanation of the latent risks associated with receiving exposures greater than 5 rem (50 mSv).

In addition, these PAGs represent dose constraint levels (e.g., when this level of dose is accumulated, the responder should not take part in the later stages of the response that may significantly increase their dose). It is assumed that doses acquired in response to a radiological incident would be ‘‘once in a lifetime’’ doses and that future radiological exposures would be substantially less. Incident commanders and responders need a thorough understanding of the worker exposure guidelines for radiological emergency response, including the associated risks

14 ‘‘Risks from Low-Level Environmental Exposure to Radionuclides,’’ Federal Guidance Report #13, U.S. Environmental Protection Agency, January 1998, EPA 402–R–97–014.

15 Risk per dose of a fatal cancer is assumed to be about 6×10^{-4} per rem. Cancer incidence is assumed to be about 7×10^{-4} per rem. (See Federal Guidance Report #13).

16 Federal Guidance Report #13. The numerical estimate of cancer risk presented above (from Federal Guidance #13) was obtained by linear extrapolation using the nominal risk estimates based on data from human exposures at high doses and high dose rates. Other methods of extrapolation to the low-dose region could yield higher or lower numerical estimates of cancer deaths. Studies of human populations exposed at low doses are inadequate to demonstrate the actual magnitude of risk at low doses (about 0.1 Sv or 10 rem and below). There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiological observation, and the possibility of no risk cannot be excluded.

and specific worker protection procedures. The reader is referred to the Federal Radiological Monitoring and Assessment Center (FRMAC) Radiological Emergency Response Health and Safety Manual (May 2001).

2.5.7 Occupational Standards

Under the provisions of the Occupational Safety and Health Act (OSHA), and equivalent statutes in the 26 states that operate OSHA-approved state plans, each employer is responsible for the health and safety of its employees. In accomplishing this, employers are expected to comply with the requirements of the Federal OSHA or state plan occupational safety and health standards applicable in the jurisdiction in which they are working. States with state plans enforce standards, under state law, which are ''at least as effective as'' Federal OSHA standards, and therefore may have more stringent or supplemental requirements. There are currently 22 states and jurisdictions operating complete state plans (covering both the private sector and state and local government employees including state and local emergency responders). Four of these state plans cover public (state and local government) employees only. Federal OSHA administers the safety and health program for the private sector in the remaining states and territories and also retains authority with regard to safety and health conditions for federal employees throughout the nation, but it does not have enforcement jurisdiction over state and local government employees. The primary occupational safety and health standard for emergency response is the Hazardous Waste Operations and Emergency Response (HAZWOPER) standard (29 CFR 1910.120). EPA has a Worker Protection (40 CFR 311) standard that applies the HAZWOPER standard to state and local workers in states that do not have their own occupational safety and health program.

For emergency response, the OSHA standard (among many other requirements) states that ''the individual in charge of the incident command system shall identify to the extent possible, all hazardous substances or conditions present and shall address as appropriate site analysis, use of engineering controls, maximum exposure limits, hazardous substance handling procedures, and use of any new technologies'' (29 CFR 1910.120(q)). As part of emergency preparedness activities, individuals authorized as incident commanders should receive the necessary training and planning prior to the incident, use the hazard information available, consult relevant standards, and apply all feasible and useful measures to minimize hazards to emergency responders.

OSHA's ionizing radiation standard (29 CFR 1910.1096), which may also apply in certain circumstances, limits quarterly dose [17] and includes other requirements such as monitoring, recordkeeping, training, and reporting.

The worker exposure levels are not PAGs but instead are regulatory limits that cannot be exceeded except under certain conditions. These occupational limits allow workers to receive radiation exposure during the course of performing their jobs. This limit offers the possibility that industrial and manufacturing facilities, critical infrastructures, and other business operations could be reopened without having to be cleaned up as long as they are in compliance with the 5 rem (50 mSv) dose limit and other OSHA requirements found in 29 CFR 1910.1096. Otherwise, the relocation PAGs could be used by decision makers to protect their citizens.

DOE employees and contractors, except for Naval Nuclear Propulsion Program (NNPP) employees and contractors are subject to DOE radiation protection regulations; requirements for worker protection from radiation exposure are contained in 10 CFR part 835. These requirements apply to all DOE employees and contractors that may be exposed to ionizing radiation as a result of their work for DOE, including work relating to emergency response activities. Section 835.3(d) indicates that nothing in the regulation ''shall be construed as limiting actions that may be necessary to protect health and safety.'' This clause is intended to recognize the fact that during emergencies, lifesaving or property-saving actions may necessitate actions that have the potential to cause doses in excess of the Department's radiation dose limits. Subpart N of part 835 provides direction for emergency exposure situations and indicates that:

- The risk of injury should be minimized.
- Actual and potential risks should be weighed against benefits of such actions causing exposures.

[17] 1.25 rem if cumulative lifetime dose is less than 5(n–18), where n is the worker's age at the last birthday, and adequate past and current exposure records are maintained to show exposures do not exceed the standard's radiation levels (29 CFR 1910.1096).

- No individual should be forced to perform a rescue action that involves substantial personal risk.

Individuals authorized to perform emergency actions that may result in exposures exceeding DOE dose limits should receive prior training and briefing on known or anticipated hazards. Under all circumstances, doses should be maintained as low as is reasonably achievable. Under DOE requirements, emergency response doses are not included with worker doses measured and calculated to demonstrate compliance with 10 CFR Part 835 dose limits. Requirements for radiation protection for NNPP employees and contractors are maintained in NNPP instructions. Requirements for the protection of those emergency workers are consistent with 10 CFR part 835 and this Manual. Requirements for the protection of NRC employees are covered by NRC Management Directive 10.131, ''Protection of NRC Employees Against Ionizing Radiation.'' Section VI, Guidance for Emergency Exposure Controls During Rescue and Recovery Activities, deals specifically with radiation exposure control during emergencies. For an IND incident, the radiological consequences could be so severe that many workers would be exposed in activities, such as emergency lifesaving functions, that would result in doses in excess of the 5 rem (50 mSv) limit for normal occupational activities.

2.6 Dose Projections

The PAGs set forth earlier in this chapter are specified in terms of the effective dose equivalent. This dose includes that incurred due to external gamma exposure of the whole body as well as the committed effective dose equivalent from inhaled radionuclides. Guidance is also provided on PAGs for the thyroid in terms of the committed dose equivalent to these organs. Further references to effective or organ dose equivalent refer to these two quantities, respectively. Methods for estimating projected dose exposure are discussed below. These require knowledge of, or assumptions for, the intensity and duration of exposure and make use of standard assumptions on the relation, for each radioisotope, between exposure and dose. Exposure and dose projections should be based on the best estimates available. The methods and models used here may be modified as necessary for specific sites for improved accuracy.

2.6.1 Dose Projection during the Early Phase

PAGs are expressed in terms of projected dose. However, in the early phase of an incident parameters other than projected dose may frequently provide a more appropriate basis for decisions to implement protective actions. When a facility is operating outside its design basis and a substantial release to the environment has started, or is imminent but has not yet occurred, data adequate to directly estimate the projected dose may not be available. For such cases, provision should be made during the planning stage for decisions to be made based on specific conditions at the source of a possible release that are relatable to ranges of anticipated offsite consequences. Emergency response plans for NPPs and facilities should make use of Emergency Action Levels (EALs), based on in-plant conditions, to trigger notification and recommendations to offsite officials to implement prompt evacuation or sheltering-in-place in specified areas in the absence of information on actual releases or environmental measurements.[18] Later, when these data become available, dose projections based on measurements may be used, in addition to plant conditions, as the basis for implementing further protective actions. (Exceptions may occur at sites with large exclusion areas where some field and source data may be available in sufficient time for protective action decisions to be based on environmental measurements). In the case of transportation accidents, an RDD or IND, or other incidents that are not related to a facility, it will often not be practicable to establish EALs.

The calculation of projected doses should be based on realistic dose models, to the extent practicable. Doses incurred prior to initiation of a protective action should not normally be included since the PAGs are based on dose that can be avoided by the protective action. Similarly, doses that might be received following the early phase should not be included for decisions on whether to evacuate or shelter. Doses that may be incurred from ingestion of food and water, long-term radiation exposure to deposited radioactive materials, or long-term inhalation of resuspended materials are chronic exposures for which neither emergency evacuation nor sheltering-in-place are appropriate protective actions. Separate PAGs relate the appropriate protective action decisions to those exposure pathways (Chapter 3). As noted earlier, the projection of doses in the early phase need, to include only those exposure pathways that contribute a significant fraction (i.e., more than about 10%) of the dose to an individual.

[18] Immediate protective actions based on in-plant conditions are not applicable to naval nuclear propulsion plants. See note 7 on page 1-4 and note 10 on page 2-3 for additional information. In addition, because of differences in design and operation, EALs based on plant condition are not applicable to naval nuclear propulsion plants.

In practical applications, dose projection will usually begin at the time of the anticipated (or actual) initiation of a release. For those situations where significant dose has already occurred prior to implementing protective action, the projected dose for comparison to a PAG should not include this prior dose.

2.6.2 Duration of Exposure

The projected dose for comparison to the early phase PAGs is normally calculated for exposure during the first 4 days following the projected (or actual) start of a release. The objective is to encompass the entire period of exposure to the plume and deposited material prior to implementation of any further, longer-term protective action such as relocation. For planning purposes, the 4-day period is chosen as the duration of exposure to deposited materials during the early phase because it is a reasonable estimate of the time necessary to make measurements, reach decisions, and prepare to implement further protective actions (such as relocation) if necessary. However, officials at the site at the time of the emergency may decide that a different time frame is more appropriate.

Protective actions are taken to avoid or reduce projected doses. Doses incurred before the start of the protective action being considered should not normally be included in evaluating the need for protective action. Similarly, doses that may be incurred at later times than those affected by the specific protective action should not be included. For example, doses that may be incurred through ingestion pathways or long-term exposure to deposited radioactive materials take place over a different, longer time period. Protective actions for such exposures should be based on guidance addressed in other chapters.

The projected dose from each radionuclide in a plume is proportional to the time-integrated concentration of the radionuclide in the plume at each location. This concentration will depend on the rate and the duration of the release and meteorological conditions. Release rates will vary with time, and this time-dependence cannot usually be predicted accurately. In the absence of more specific information, the release rate may be assumed to be constant.

Another factor affecting the estimation of projected dose is the duration of the plume at a particular location. For purposes of calculating projected dose from most pathways, exposure will start at a particular location when the plume arrives and end when the plume is no longer present, due either to an end to the release, or a change in wind direction. Exposure from one pathway (exposure to deposited materials) will continue for an extended period. Other factors such as the aerodynamic diameter and solubility of particles, shape of the plume, and terrain may also affect estimated dose and may be considered on a site- and/or source- specific basis.

Prediction of time frames for releases is difficult because of the wide range associated with the spectrum of potential incidents. Therefore, planners should consider the possible time periods between an initiating event and arrival of a plume, and the duration of releases in relation to the time needed to implement competing protective actions (i.e., evacuation and sheltering-in-place). Analyses of nuclear power reactors (NRC 1975) have shown that some incidents may take several days to develop to the point of a release while others may begin as early as 1.5 hours after an initiating event. Furthermore, the duration of a release may range from less than 1 hour to several days, with the major portion of the release usually occurring within the first day.

Radiological exposure rates from a plume are influenced by wind speed. The air concentration is approximately inversely related to the wind speed at the point of release. Concentrations are also affected by the turbulence of the air, which tends to increase with wind speed and sunlight, and by wandering of the plume, which is greater at the lower wind speeds. This results in higher concentrations generally being associated with low winds near the source and with moderate winds at larger distances. Higher wind speed also shortens the travel time of the plume. Planning information on time frames for releases from nuclear power facilities may be found in NRC and EPA 1978. Time frames for releases from other facilities will depend on the characteristics of the facility.

2.7 Dose Conversion Parameters and Derived Response Levels

This section provides dose conversion parameters (DCPs) and DRLs for those radionuclides likely to be important for responding to most types of incidents. These are supplemented by an example to demonstrate their application. The DCPs are useful when multiple radionuclides are involved because the total dose from a single exposure pathway will be the sum of the doses calculated for each radionuclide. The DRLs are surrogates for the PAGs and are directly usable for releases consisting primarily of a single nuclide, in which case the DRL can be compared directly to the measured or calculated concentration. (DRLs also can be used for multiple radionuclides by summing

the ratios of the environmental concentration of each nuclide to its respective DRL. To meet the PAG, this sum must be equal to or less than unity).

2.7.1 Procedures for Calculating DCP and DRL for Combined Exposure Pathways

This section provides information used in the development of the DCPs and DRLs in Tables 2-5 and 2-6. Three exposure pathways are included: whole body exposure to gamma radiation from the plume; inhalation from the plume; and whole body exposure to gamma radiation from deposited materials. They are each expressed in terms of the time-integrated air concentration so that they may be combined to yield a composite DCF for each radionuclide that reflects all three pathways. These data may be used to facilitate revising the DCFs in Tables 2-5 and 2-6 when more specific or technically improved assumptions are available, as well as to evaluate the relative importance of the individual pathways for specific radionuclide mixes.Early phase calculations presented in Chapter 2 are for use during the plume. Following plume passage, intermediate phase calculations in Chapter 3 should be used.

DCPs and DRLs for each of the three major exposure pathways for the early phase are provided. They are all expressed in terms of the time-integrated air concentration at the receptor so they can be conveniently summed over the three exposure pathways to obtain composite DRLs and DCPs for each radionuclide. These composite values are tabulated in Table 2-5 for effective dose and in Table 2-6 for equivalent thyroid dose from inhalation of radioiodines.

The tabulated DCPs and DRLs include assumptions on particle size, deposition velocity, the presence of short-lived daughters, and exposure duration as noted. The existence of more accurate data for individual radionuclides may justify modification of the DCPs and DRLs. The procedures described in this section for developing the DCPs and DRLs for individual exposure pathways may be used to assist with such modifications.

To apply Tables 2-5 and 2-6 to decisions on implementing PAG protective actions, one may use either the DCPs or DRLs. DCPs are used to calculate the projected composite dose for each radionuclide; these doses are then summed and compared to the PAG. The DRLs may be used by summing the ratios of the concentration of each radionuclide to its corresponding DRL. If the sum of the ratios exceeds unity, the corresponding protective action should be initiated.

The decision of whether evacuation is warranted at these levels is based on PAGs of 1 rem (10 mSv) for effective dose and 5 rem (50 mSv) for dose to the thyroid. To calculate the relevant dose, DCPs are provided in Table 2-5 for effective dose and Table 2-6 are used for equivalent thyroid dose from inhalation of radioiodine, respectively.

Table 2-5. Dose Conversion Parameters (DCPs) and Derived Response Levels (DRLs) for Combined[a] Exposure Pathways during the Early Phase of a Radiological Incident

				ICRP 60+ Calculated Values					1992 EPA Values		DCF Comparisons	
No.	Radionuclide	[b]Half-Life (d)	Branch Fraction	Table 2-7 (rem·cm^3 per h·μCi)	Table 2-8 (rem·cm^3 per h·μCi)	Table 2-9 (rem·cm^3 per h·μCi)	DCP Table 5.1 (rem·cm^3 per h·μCi)	DRL Table 5.1 (μCi·hr per cm^3)	DCP Table 5.1 (rem·cm^3 per h·μCi)	DRL Table 5.1 (μCi·h per cm^3)	DCPtotal / by 1992 DCPtotal	DRLtotal / by 1992 DRLtotal
1	Am-241	1.58E+05	-	8.96E+00	3.28E+08	6.78E+04	3.28E+08	3.05E-09	5.30E+08	1.90E-09	0.62	1.60
2	Ba-140/La-140	NA	NA	1.58E+03	2.38E+04	7.96E+03	3.33E+04	3.00E-05	NA	-	-	-
	Ba-140	1.27E+01	1.00E+00	1.07E+02	1.99E+04	6.47E+02	2.06E+04	4.85E-05	5.30E+03	1.90E-04	3.89	0.26
	La-140	1.68E+00	1.00E+00	1.48E+03	3.88E+03	7.31E+03	1.27E+04	7.89E-05	1.10E+04	8.80E-05	1.15	0.90
3	Ce-144/Pr-144/Pr-144m	NA	NA	4.54E+01	1.79E+05	7.17E+02	1.80E+05	5.55E-06	4.50E+05	2.20E-06	0.40	2.52
	Ce-144	2.84E+02	1.00E+00	1.01E+01	1.79E+05	1.06E+02	1.80E+05	5.57E-06	4.50E+05	2.20E-06	0.40	2.53
	Pr-144	1.20E-02	9.82E-01	3.52E+01	6.23E+01	6.11E+02	7.08E+02	1.41E-03	NA	-	-	-
	Pr-144m	5.00E-03	1.78E-02	2.93E+00	0.00E+00	3.93E+01	4.23E+01	2.37E-02	NA	-	-	-
	Pr-144	1.20E-02	9.99E-01	3.52E+01	6.23E+01	6.11E+02	7.08E+02	1.41E-03	NA	-	-	-
4	Cf-252	9.64E+02	-	4.83E-02	1.26E+08	2.60E+04	1.26E+08	7.92E-09	1.90E+08	5.30E-09	0.66	1.49
5	Cm-244	6.61E+03	-	4.52E-02	1.94E+08	4.00E+04	1.94E+08	5.15E-09	3.00E+08	3.40E-09	0.65	1.52
6	Co-60	1.93E+03	-	1.58E+03	1.05E+05	8.67E+03	1.15E+05	8.71E-06	2.70E+05	3.70E-06	0.43	2.36
7	Cs-134	7.53E+02	-	9.39E+02	6.94E+04	5.58E+03	7.60E+04	1.32E-05	6.30E+04	1.60E-05	1.21	0.82
8	Cs-136	1.31E+01	-	1.32E+03	9.46E+03	6.89E+03	1.77E+04	5.66E-05	1.80E+04	5.60E-05	0.98	1.01
9	Cs-137/Ba-137m	NA	NA	3.40E+02	1.33E+05	2.10E+03	1.36E+05	7.36E-06	4.10E+04	2.40E-05	3.31	0.31
	Cs-137	1.10E+04	1.00E+00	1.23E+00	1.33E+05	3.88E+01	1.33E+05	7.49E-06	NA	-	-	-
	Ba-137m	1.77E-03	9.46E-01	3.58E+02	0.00E+00	2.18E+03	2.54E+03	3.94E-04	NA	-	-	-
10	Gd-153	2.42E+02	-	4.14E+01	8.17E+03	3.47E+02	8.56E+03	1.17E-04	2.90E+04	3.40E-05	0.30	3.44
11	I-131	8.04E+00	-	2.25E+02	2.52E+04	1.16E+03	2.65E+04	3.77E-05	5.30E+04	1.90E-05	0.50	1.98
12	I-132	9.58E-02	-	1.40E+03	3.88E+02	2.87E+02	2.07E+03	4.83E-04	4.90E+03	2.00E-04	0.42	2.41
13	I-133	8.67E-01	-	3.67E+02	5.00E+03	6.98E+02	6.07E+03	1.65E-04	1.50E+04	6.80E-05	0.40	2.42
14	I-134	3.65E-02	-	1.62E+03	1.90E+02	1.25E+02	1.94E+03	5.16E-04	3.10E+03	3.30E-04	0.63	1.56
15	I-135/Xe-135m	NA	NA	1.04E+03	1.10E+03	5.74E+02	2.72E+03	3.68E-04	NA	-	-	-
	I-135	2.75E-01	1.00E+00	1.00E+03	1.10E+03	5.50E+02	2.65E+03	3.77E-04	8.10E+03	1.20E-04	0.33	3.14
	Xe-135m	1.06E-02	1.54E-01	2.53E+02	0.00E+00	1.57E+02	4.09E+02	2.44E-03	2.50E+02	4.10E-03	1.64	0.60
16	Ir-192	7.40E+01	-	4.80E+02	2.25E+04	2.88E+03	2.59E+04	3.86E-05	3.80E+04	2.70E-05	0.68	1.43

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Table 2-5. Dose Conversion Parameters (DCPs) and Derived Response Levels (DRLs) for Combined[a] Exposure Pathways during the Early Phase of a Radiological Incident (cont'd)

				ICRP 60+ Calculated Values					1992 EPA Values		DCF Comparisons	
No.	Radionuclide	[b]Half-Life (d)	Branch Fraction	Table 2-7 (rem·cm^3 per h·µCi)	Table 2-8 (rem·cm^3 per h·µCi)	Table 2-9 (rem·cm^3 per h·µCi)	DCP (rem·cm^3 per h·µCi)	DRL (µCi·h per cm^3)	DCP Table 5.1 (rem·cm^3 per h·µCi)	DRL Table 5.1 (µCi·h per cm^3)	DCPtotal / by 1992 DCPtotal	DRLtotal / by 1992 DRLtotal
17	Kr-87	5.30E-02	-	5.28E+02	NA	6.04E+01	5.88E+02	1.70E-03	5.10E+02	2.00E-03	1.15	0.85
18	Kr-88/Rb-88	NA	NA	1.73E+03	3.97E+03	3.98E+02	6.11E+03	1.64E-04	NA	-	-	-
	Kr-88	1.18E-01	1.00E+00	1.29E+03	3.88E+03	2.78E+02	5.45E+03	1.83E-04	1.30E+03	7.80E-04	4.19	0.24
	Rb-88	1.24E-02	1.00E+00	4.43E+02	9.40E+01	1.19E+02	6.56E+02	1.52E-03	5.20E+02	1.90E-03	1.26	0.80
19	La-140	1.68E+00	-	1.48E+03	3.88E+03	3.98E+03	9.33E+03	1.07E-04	1.10E+04	8.80E-05	0.85	1.22
20	Mo-99/Tc-99m	NA	NA	1.54E+02	3.44E+03	6.59E+02	4.25E+03	2.35E-04	NA	-	-	-
	Mo-99	2.75E+00	1.00E+00	9.30E+01	3.38E+03	4.23E+02	3.89E+03	2.57E-04	5.20E+03	1.90E-04	0.75	1.35
	Tc-99m	2.51E-01	8.76E-01	6.98E+01	6.84E+01	2.70E+02	4.09E+02	2.45E-03	1.70E+02	6.00E-03	2.40	0.41
21	Np-239	2.36E+00	-	9.24E+01	3.51E+03	3.42E+02	3.94E+03	2.54E-04	3.60E+03	2.80E-04	1.09	0.91
22	Pm-147	9.58E+02	-	1.15E-01	2.38E+04	5.00E+00	2.38E+04	4.21E-05	4.70E+04	2.10E-05	0.51	2.00
23	Pu-238	3.20E+04	-	4.66E-02	3.68E+08	7.58E+04	3.68E+08	2.72E-09	4.70E+08	2.10E-09	0.78	1.30
24	Pu-239	8.79E+06	-	4.63E-02	4.05E+08	8.36E+04	4.05E+08	2.47E-09	5.20E+08	1.90E-09	0.78	1.30
25	Ra-226/Rn-222…	NA	NA	1.11E+03	3.25E+07	1.31E+04	3.25E+07	3.08E-08	NA	-	-	-
	Ra-226	5.84E+05	1.00E+00	3.78E+00	3.24E+07	6.70E+03	3.24E+07	3.09E-08	1.00E+07	9.70E-08	3.24	0.32
	Rn-222	3.82E+00	1.00E+00	2.35E-01	0.00E+00	1.44E+00	1.67E+00	5.97E-01	NA	-	-	-
	Po-218	2.12E-03	1.00E+00	5.60E-03	0.00E+00	3.26E-02	3.82E-02	2.62E+01	NA	-	-	-
	Pb-214	1.86E-02	1.00E+00	1.45E+02	5.00E+04	9.14E+02	5.11E+04	1.96E-05	NA	-	-	-
	Bi-214	1.38E-02	1.00E+00	9.64E+02	5.24E+04	5.43E+03	5.88E+04	1.70E-05	NA	-	-	-
	Po-214	1.90E-09	1.00E+00	5.07E-02	0.00E+00	2.99E-01	3.49E-01	2.86E+00	NA	-	-	-
	At-218	2.31E-05	2.00E-04	1.29E+00	0.00E+00	1.37E+01	1.50E+01	6.67E-02	NA	-	-	-
	Bi-214	1.38E-02	1.00E+00	9.64E+02	5.24E+04	5.43E+03	5.88E+04	1.70E-05	NA	-	-	-
	Po-214	1.90E-09	1.00E+00	5.07E-02	0.00E+00	2.99E-01	3.49E-01	2.86E+00	NA	-	-	-
26	Ru-103/Rh-103m	NA	NA	2.77E+02	1.01E+04	1.64E+03	1.20E+04	8.36E-05	NA	-	-	-
	Ru-103	3.93E+01	1.00E+00	2.77E+02	1.00E+04	1.63E+03	1.20E+04	8.37E-05	1.30E+04	7.70E-05	0.92	1.09
	Rh-103m	3.90E-02	9.97E-01	8.01E-02	9.29E+00	3.22E+00	1.26E+01	7.94E-02	NA	-	-	-

Table 2-5. Dose Conversion Parameters (DCPs) and Derived Response Levels (DRLs) for Combined[a] Exposure Pathways during the Early Phase of a Radiological Incident (cont'd)

				ICRP 60+ Calculated Values					1992 EPA Values		DCF Comparisons	
No.	Radionuclide	[b]Half-Life (d)	Branch Fraction	Table 2-7 ($rem \cdot cm^3$ Per $h \cdot \mu Ci$)	Table 2-8 ($rem \cdot cm^3$ per $h \cdot \mu Ci$)	Table 2-9 ($rem \cdot cm^3$ per $h \cdot \mu Ci$)	DCP ($rem \cdot cm^3$ per $h \cdot \mu Ci$)	DRL ($\mu Ci \cdot h$ per cm^3)	DCP Table 5.1 ($rem \cdot cm^3$ per $h \cdot \mu Ci$)	DRL Table 5.1 ($\mu Ci \cdot h$ per cm^3)	DCPtotal / by 1992 DCPtotal	DRLtotal / by 1992 DRLtotal
27	Ru-106/Rh-106	NA	NA	1.41E+02	2.25E+05	1.34E+03	2.26E+05	4.42E-06	5.70E+05	1.70E-06	0.40	2.60
	Ru-106	3.68E+02	1.00E+00	0.00E+00	2.25E+05	4.62E+01	2.25E+05	4.45E-06	NA	-	-	-
	Rh-106	3.46E-04	1.00E+00	1.41E+02	0.00E+00	1.29E+03	1.43E+03	6.97E-04	NA	-	-	-
28	Sb-127/Te-127	NA	NA	4.19E+02	6.79E+03	1.84E+03	9.05E+03	1.10E-04	NA	-	-	-
	Sb-127	3.85E+00	1.00E+00	4.15E+02	6.40E+03	1.82E+03	8.63E+03	1.16E-04	9.50E+03	1.10E-04	0.91	1.05
	Te-127	3.90E-01	8.24E-01	4.44E+00	4.77E+02	2.77E+01	5.09E+02	1.97E-03	NA	-	-	-
29	Sb-129/Te-129	NA	NA	9.22E+02	9.55E+02	3.57E+02	2.23E+03	4.48E-04	NA	-	-	-
	Sb-129	1.80E-01	1.00E+00	8.92E+02	8.51E+02	3.35E+02	2.08E+03	4.81E-04	2.00E+03	5.00E-04	1.04	0.96
	Te-129	4.83E-02	7.75E-01	3.80E+01	1.34E+02	2.79E+01	2.00E+02	5.01E-03	1.40E+02	7.00E-03	1.43	0.72
30	Se-75	1.20E+02	-	2.23E+02	4.56E+03	1.34E+03	6.13E+03	1.63E-04	1.20E+04	8.30E-05	0.51	1.97
31	Sr-89	5.05E+01	-	5.81E+00	2.70E+04	2.57E+02	2.73E+04	3.66E-05	5.00E+04	2.00E-05	0.55	1.83
32	Sr-90/Y-90	NA	NA	1.18E+01	5.40E+05	5.32E+02	5.40E+05	1.85E-06	NA	-	-	-
	Sr-90	1.06E+04	1.00E+00	1.31E+00	5.34E+05	1.16E+02	5.35E+05	1.87E-06	1.60E+06	6.40E-07	0.33	2.92
	Y-90	2.67E+00	1.00E+00	1.05E+01	5.11E+03	4.15E+02	5.53E+03	1.81E-04	1.00E+04	9.90E-05	0.55	1.83
33	Sr-91/Y-91m	NA	NA	6.17E+02	1.41E+03	5.50E+02	2.58E+03	3.88E-04	NA	-	-	-
	Sr-91	3.96E-01	1.00E+00	4.35E+02	1.39E+03	3.91E+02	2.21E+03	4.51E-04	2.40E+03	4.20E-04	0.92	1.07
	Y-91m	3.45E-02	5.78E-01	3.15E+02	3.88E+01	2.74E+02	6.28E+02	1.59E-03	NA	-	-	-
34	Te-129m/Te-129	NA	NA	4.55E+01	2.70E+04	4.79E+02	2.76E+04	3.63E-05	NA	-	-	-
	Te-129m	3.36E+01	1.00E+00	2.07E+01	2.70E+04	2.12E+02	2.72E+04	3.68E-05	2.90E+04	3.50E-05	0.94	1.05
	Te-129	4.83E-02	6.50E-01	3.80E+01	1.34E+02	4.12E+02	5.84E+02	1.71E-03	1.40E+02	7.00E-03	4.17	0.24
35	[c]Te-131m/Te-131	NA	NA	9.28E+02	3.66E+03	2.18E+03	6.78E+03	1.48E-04	NA	-	-	-
	[c]Te-131m	1.25E+00	1.00E+00	8.71E+02	3.64E+03	2.03E+03	6.54E+03	1.53E-04	8.60E+03	1.20E-04	0.76	1.27
	Te-131	1.74E-02	2.22E-01	2.55E+02	9.70E+01	7.16E+02	1.07E+03	9.36E-04	NA	-	-	-
36	Te-132/I-132	NA	NA	1.52E+03	7.37E+03	6.12E+03	1.50E+04	6.66E-05	2.00E+04	5.00E-05	0.75	1.33
	Te-132	3.26E+00	1.00E+00	1.24E+02	6.98E+03	5.39E+02	7.64E+03	1.31E-04	1.20E+04	8.50E-05	0.64	1.54
	I-132	9.58E-02	1.00E+00	1.40E+03	3.88E+02	5.58E+03	7.36E+03	1.36E-04	4.90E+03	2.00E-04	1.50	0.68

Table 2-5. Dose Conversion Parameters (DCPs) and Derived Response Levels (DRLs) for Combined[a] Exposure Pathways during Early Phase of a Radiological Incident (cont'd)

				ICRP 60+ Calculated Values					1992 EPA Values		DCF Comparisons	
No.	Radionuclide	[b]Half-Life (d)	Branch Fraction	Table 2-7 (rem·cm^3 per h·µCi)	Table 2-8 (rem·cm^3 per h·µCi)	Table 2-9 (rem·cm^3 per h·µCi)	DCP (rem·cm^3 per h·µCi)	DRL (µCi·h per cm^3)	DCP Table 5.1 (rem·cm^3 per h·µCi)	DRL Table 5.1 (µCi·h per cm^3)	DCPtotal / by 1992 DCPtotal	DRLtotal / by 1992 DRLtotal
37	Tm-170	1.29E+02	-	4.88E+00	3.16E+04	1.05E+02	3.17E+04	3.15E-05	3.20E+04	3.20E-05	0.99	0.98
38	Xe-133	5.24E+00	-	1.77E+01	NA	1.16E+02	1.33E+02	7.51E-03	2.00E+01	5.00E-02	6.66	0.15
39	Xe-135	3.79E-01	-	1.46E+02	NA	1.29E+02	2.75E+02	3.64E-03	1.40E+02	7.00E-03	1.96	0.52
40	Xe-138	9.84E-03	-	7.29E+02	0.00E+00	1.43E+01	7.43E+02	1.35E-03	7.20E+02	1.40E-03	1.03	0.96
41	Y-91	5.85E+01	-	8.27E+00	3.04E+04	2.81E+02	3.07E+04	3.26E-05	5.90E+04	1.70E-05	0.52	1.92
42	Yb-169	3.20E+01	-	1.50E+02	1.01E+04	1.01E+03	1.13E+04	8.85E-05	1.10E+04	8.90E-05	1.03	0.99

[a]Table 2-7 gives Equivalent Dose from 1 hour submersion in passing plume, Table 2-8 gives Committed Effective Dose from inhalation while in the passing plume for 1 hour and Table 2-9 gives Equivalent Dose from 4 days from groundshine and Committed Effective Dose from inhalation of resuspended material over 4 d.

[b]Values from Turbo FRMAC 2.0, RFC 2 (DCFPAK, K. Eckerman).
[c]Te-131m values in this table are subject to change pending further development of new parent-daughter rules.

2.7.2 Dose Conversion Parameter Calculation for Table 2-5

The Table 2-5 DCP values for the combined exposure pathways (i.e., 1 hour of plume inhalation, 1 hour of plume submersion, 96 hours of groundshine and 96 hours of inhalation of resuspended material) are simply the summation of the corresponding values for each radionuclide from Table 2-7 (1 hour of plume submersion), Table 2-8 (1 hour of plume inhalation) and Table 2-9 (96 hours of groundshine and 96 hours of inhalation of resuspended material).

$$DCP_{Combined,E,i} = \sum_{i}^{P+D} \left(DCP_{Submersion,E,i} + DCP_{Inhalation,E,i} + DCP_{groundshine+inh,E,i} \right)$$

Where:

$\sum_{i}^{P+D}$ = represents the summation of values from the parent radionuclide (P) and any short-lived daughter radionuclide(s) (D);

$DCP_{Combined, E, i}$ = the Dose Conversion Parameter, value for the effective dose rate from all pathways (i.e., combined pathways including 1 h of plume inhalation, 1 h of plume submersion, 96 h of groundshine and 96 h of inhalation of resuspended material) from exposure to radionuclide *i* and any short-lived daughter radionuclide(s), rem•cm^3 /h•µCi;

$DCP_{Submersion, E, i}$ = Dose Conversion Parameter, value for the effective dose rate per unit activity from external exposure (i.e., 1 h external exposure from submersion in the plume) to radionuclide *i* and any short-lived daughter radionuclide(s) in the plume, rem·cm^3 / h·µCi, from Table 2-7;

$DCP_{Inhalation,E,i}$ = Dose Conversion Parameter, value for the committed effective dose rate per unit activity from exposure (i.e., 1 h plume inhalation) to radionuclide *i* and any short-lived daughter radionuclide(s), rem·cm^3 / h·µCi, from Table 2-8; and

$DCP_{groundshine+inh,\ E,\ i}$ = Dose Conversion Parameter, effective, value for the dose rate per unit activity from groundshine and the inhalation of resuspended material from radionuclide *i* and any short-lived daughter radionuclide(s) over the early phase time period and, rem·cm^3 / h·µCi, from Table 2-9.

2.7.3 Derived Response Level Calculation for Table 2-5

The DRL is calculated using the following equation:

$$DRL_{Combined,E,i} = \sum_{i}^{P+D}\left(\frac{PAG}{DCP_{Combined,E,i}}\right), \qquad \frac{\mu Ci \cdot h}{cm^3} = \frac{1\,rem}{rem \cdot cm^3 / h \cdot \mu Ci}$$

Where:

$\sum_{i}^{P+D}$ = represents the summation of values from the parent radionuclide (P) and any short-lived daughter radionuclide(s) (D);

$DRL_{Combined,E,\ i}$ = Derived Response Level (DRL), effective, value for the effective dose from all pathways (i.e., combined pathways including 1 h of plume inhalation, 1 h of plume submersion, 96 h of groundshine and 96 h of inhalation of resuspended material) from exposure to radionuclide *i* and any short-lived daughter radionuclide(s), µCi·h/cm^3;

PAG = EPA's Protective Action Guide (PAG), 1 rem for effective dose or 5 rem organ dose; and

$DCP_{Combined,\ E,\ i}$ = tThe Dose Conversion Parameter, value for the effective dose rate from all pathways (i.e., combined pathways including 1 h of plume inhalation, 1 h of plume submersion, 96 h of groundshine and 96 h of inhalation of resuspended material) from exposure to radionuclide *i* and any short-lived daughter radionuclide(s), rem•cm^3 /h•µCi.

Persons exposed to an airborne particulate plume will receive dose to skin from beta emitters in the plume as well as from those deposited on skin and clothing. Although it is possible to detect beta radiation, it is not practical, for purposes of decisions on evacuation and sheltering-in-place, to determine dose to skin by field measurement of the beta dose equivalent rate near the skin surface. Such doses are determined more practically through calculations based on time-integrated air concentration, an assumed deposition velocity, and an assumed time period between deposition and skin decontamination. For the purpose of evaluation the relative dose compared to the dose from external gamma exposure and inhalation, dose conversion factors were evaluated using a deposition velocity of 1 cm/s and an exposure time before decontamination of 12 hours. Using these conservative assumptions, it was determined that skin beta dose should seldom, if ever, be a controlling pathway during the early phase. Therefore, no DCPs or DRLs are listed for skin beta dose.

Table 2-6. DCPs and DRLs for Inhalation (Assume 1 hour Exposure Period), based on 5 rem Committed Equivalent Dose Summary: List of Most Restrictive DCPs and DRLs for all Age Groups [a]

					ICRP 60+ - Revised Values		1992 Values (Table 5.5)		DCF Comparisons	
No.	Radionuclide	[a]Half-Life (d)	$DCF_{Inh,\ thyroid}$ (Sv/Bq)	Age Group	$DCP_{Inh,\ thyroid}$ (rem·cm^3 per h·µCi)	$DRL_{Inh,\ thyroid}$ (µCi·h per cm^3)	EPA $DCP_{Inh,\ thyroid}$ (rem·cm^3 per h·µCi)	EPA $DRL_{Inh,\ thyroid}$ (µCi·h per cm^3)	$DCP_{Inh,\ thyroid}$ / by 1992 $DCP_{Inh,\ thyroid}$	$DRL_{Inh.\ Thyroid}$ / by 1992 $DRL_{Inh,\ thyroid}$

	I-125	60.1	2.25E-07	10-y-old	9.32E+05	5.36E-06	9.60E+05	5.20E-06	0.97	1.03
36	Te-132/I-132	3.26	2.89E-07	1-y-old	3.74E+05	1.34E-05	2.90E+05	1.80E-05	1.29	0.74
	I-129	5.73E+09	1.33E-06	10-y-ld	5.51E+06	9.07E-07	6.90E+06	7.20E-07	0.80	1.26
11	I-131	8.04	1.43E-06	1–y-old	1.85E+06	2.70E-06	1.30E+06	3.90E-06	1.42	0.69
12	I-132	0.0958	1.63E-08	1–y-old	2.11E+04	2.37E-04	7.70E+03	6.50E-04	2.74	0.36
13	I-133	0.867	3.51E-07	1-y-old	4.55E+05	1.10E-05	2.20E+05	2.30E-05	2.07	0.48
14	I-134	0.0365	3.08E-09	1-y-old	3.99E+03	1.25E-03	1.30E+03	3.90E-03	3.07	0.32
15	I-135	0.275	6.98E-08	1–y-ld	9.04E+04	5.53E-05	3.80E+04	1.30E-04	2.38	0.43

Table 2-6a. DCPs and DRLs for Inhalation (assume 1 hour Exposure Period), based on 5 rem Committed Equivalent Dose (Newborn) [a]

				ICRP 60 - Revised Values		1992 Values (Table 5.2)		DCF Comparisons	
No.	Radionuclide	[a]Half-Life (d)	$DCF_{Inh,\ thyroid}$ (Sv/Bq)	$DCP_{Inh,\ Ht}$ (rem·cm^3 per h·µCi)	$DRL_{Inh,\ HT}$ (µCi·h per cm^3)	EPA $DCP_{Inh,\ thyroid}$ (rem·cm^3 per h·µCi)	EPA $DRL_{Inh,\ thyroid}$ (µCi·h per cm^3)	DCPInh, thyroid / by 1992 DCPInh, thyroid	DRLInh. Thyroid / by 1992 DRLInh, thyroid
	I-125	60.1	4.07E-07	2.86E+05	1.75E-05	9.60E+05	5.20E-06	0.30	3.36
36	Te-132/I-132	3.26	3.56E-07	2.50E+05	2.00E-05	2.90E+05	1.80E-05	0.86	1.11
	I-129	5.73E+09	1.43E-06	1.01E+06	4.97E-06	6.90E+06	7.20E-07	0.15	6.91
11	I-131	8.04E+00	1.43E-06	1.01E+06	4.97E-06	1.30E+06	3.90E-06	0.77	1.28
12	I-132	9.58E-02	1.80E-08	1.27E+04	3.95E-04	7.70E+03	6.50E-04	1.64	0.61
13	I-133	8.67E-01	3.82E-07	2.69E+05	1.86E-05	2.20E+05	2.30E-05	1.22	0.81
14	I-134	3.65E-02	3.40E-09	2.39E+03	2.09E-03	1.30E+03	3.90E-03	1.84	0.54
15	I-135	2.75E-01	7.67E-08	5.39E+04	9.27E-05	3.80E+04	1.30E-04	1.42	0.71

Table 2-6b. DCPs and DRLs for Inhalation (assume 1 hour exposure period), based on 5 rem Committed Equivalent Dose (1-Year-Old Child) [a]

				ICRP 60 - Revised Values		1992 Values (Table 5.2)		DCF Comparisons	
No.	Radionuclide	[b]Half-Life (d)	$DCF_{Inh, thyroid}$ (Sv/Bq)	$DCP_{Inh, Ht}$ (rem·cm^3 per h·µCi)	$DRL_{Inh, HT}$ (µCi·h per cm^3)	EPA $DCP_{Inh, thyroid}$ (rem·cm^3 per h·µCi)	EPA $DRL_{Inh, thyroid}$ (µCi·h per cm^3)	DCPInh, thyroid / by 1992 DCPInh, thyroid	DRLInh. Thyroid / by 1992 DRLInh, thyroid
	I-125	60.1	4.60E-07	5.96E+05	8.39E-06	9.60E+05	5.20E-06	0.62	1.61
36	Te-132/I-132	3.26	2.89E-07	3.74E+05	1.34E-05	2.90E+05	1.80E-05	1.29	0.74
	I-129	5.73E+09	1.72E-06	2.23E+06	2.24E-06	6.90E+06	7.20E-07	0.32	3.12
11	I-131	8.04E+00	1.43E-06	1.85E+06	2.70E-06	1.30E+06	3.90E-06	1.42	0.69
12	I-132	9.58E-02	1.63E-08	2.11E+04	2.37E-04	7.70E+03	6.50E-04	2.74	0.36
13	I-133	8.67E-01	3.51E-07	4.55E+05	1.10E-05	2.20E+05	2.30E-05	2.07	0.48
14	I-134	3.65E-02	3.08E-09	3.99E+03	1.25E-03	1.30E+03	3.90E-03	3.07	0.32
15	I-135	2.75E-01	6.98E-08	9.04E+04	5.53E-05	3.80E+04	1.30E-04	2.38	0.43

Table 2-6c. DCPs and DRLs for Inhalation (assume 1 hour exposure period), based on 5 rem Committed Equivalent Dose (5-Year-Old Child) [a]

				ICRP 60 - Revised Values		1992 Values (Table 5.2)		DCF Comparisons	
No.	Radionuclide	[b]Half-Life (d)	$DCF_{Inh, thyroid}$ (Sv/Bq)	$DCP_{Inh, Ht}$ (rem·cm^3 per h·µCi)	$DRL_{Inh, HT}$ (µCi·h per cm^3)	EPA $DCP_{Inh, thyroid}$ (rem·cm^3 per h·µCi)	EPA $DRL_{Inh, thyroid}$ (µCi·h per cm^3)	DCPInh, thyroid / by 1992 DCPInh, thyroid	DRLInh. Thyroid / by 1992 DRLInh, thyroid
	I-125	60.1	2.95E-07	6.21E+05	8.05E-06	9.60E+05	5.20E-06	0.65	1.55
36	Te-132/I-132	3.26	1.37E-07	2.88E+05	1.73E-05	2.90E+05	1.80E-05	0.99	0.96
	I-129	5.73E+09	1.22E-06	2.57E+06	1.95E-06	6.90E+06	7.20E-07	0.37	2.70
11	I-131	8.04E+00	7.29E-07	1.53E+06	3.26E-06	1.30E+06	3.90E-06	1.18	0.84
12	I-132	9.58E-02	7.64E-09	1.61E+04	3.11E-04	7.70E+03	6.50E-04	2.09	0.48
13	I-133	8.67E-01	1.64E-07	3.45E+05	1.45E-05	2.20E+05	2.30E-05	1.57	0.63
14	I-134	3.65E-02	1.44E-09	3.03E+03	1.65E-03	1.30E+03	3.90E-03	2.33	0.42
15	I-135	2.75E-01	3.26E-08	6.86E+04	7.29E-05	3.80E+04	1.30E-04	1.81	0.56

Table 2-6d. DCPs and DRLs for Inhalation (Assume 1 Hour Exposure Period), based on 5 rem Committed Equivalent Dose (10-Year-Old Child) [a]

				ICRP 60 - Revised Values		1992 Values (Table 5.2)		DCF Comparisons	
No.	Radionuclide	[b]Half-Life (d)	$DCF_{Inh, thyroid}$ (Sv/Bq)	$DCP_{Inh, Ht}$ (rem·cm^3 per h·µCi)	$DRL_{Inh, HT}$ (µCi·h per cm^3)	EPA $DCP_{Inh, thyroid}$ (rem·cm^3 per h·µCi)	EPA $DRL_{Inh, thyroid}$ (µCi·h per cm^3)	DCPInh, thyroid / by 1992 DCPInh, thyroid	DRLInh. Thyroid / by 1992 DRLInh, thyroid
	I-125	60.1	2.25E-07	9.32E+05	5.36E-06	9.60E+05	5.20E-06	0.97	1.03
36	Te-132/I-132	3.26	6.13E-08	2.54E+05	1.97E-05	2.90E+05	1.80E-05	0.88	1.09
	I-129	5.73E+09	1.33E-06	5.51E+06	9.07E-07	6.90E+06	7.20E-07	0.80	1.26
11	I-131	8.04E+00	3.70E-07	1.53E+06	3.26E-06	1.30E+06	3.90E-06	1.18	0.84
12	I-132	9.58E-02	3.43E-09	1.42E+04	3.52E-04	7.70E+03	6.50E-04	1.85	0.54
13	I-133	8.67E-01	7.38E-08	3.06E+05	1.63E-05	2.20E+05	2.30E-05	1.39	0.71
14	I-134	3.65E-02	6.48E-10	2.69E+03	1.86E-03	1.30E+03	3.90E-03	2.07	0.48
15	I-135	2.75E-01	1.46E-08	6.05E+04	8.26E-05	3.80E+04	1.30E-04	1.59	0.64

Table 2-6e. DCPs and DRLs for Inhalation (Assume 1 Hour Exposure Period), based on 5 rem Committed Equivalent Dose (15-Year-Old Child) [a]

				ICRP 60 - Revised Values		1992 Values (Table 5.2)		DCF Comparisons	
No.	Radionuclide	[b]Half-Life (d)	$DCF_{Inh, thyroid}$ (Sv/Bq)	$DCP_{Inh, Ht}$ (rem·cm^3 per h·µCi)	$DRL_{Inh, HT}$ (µCi·h per cm^3)	EPA $DCP_{Inh, thyroid}$ (rem·cm^3 per h·µCi)	EPA $DRL_{Inh, thyroid}$ (µCi·h per cm^3)	DCPInh, thyroid / by 1992 DCPInh, thyroid	DRLInh. Thyroid / by 1992 DRLInh, thyroid
	I-125	60.1	1.45E-07	7.40E+05	6.75E-06	9.60E+05	5.20E-06	0.77	1.30
36	Te-132/I-132	3.26	3.82E-08	1.95E+05	2.56E-05	2.90E+05	1.80E-05	0.67	1.42
	I-129	5.73E+09	9.11E-07	4.65E+06	1.07E-06	6.90E+06	7.20E-07	0.67	1.49
11	I-131	8.04E+00	2.23E-07	1.14E+06	4.39E-06	1.30E+06	3.90E-06	0.88	1.13
12	I-132	9.58E-02	2.08E-09	1.06E+04	4.71E-04	7.70E+03	6.50E-04	1.38	0.72
13	I-133	8.67E-01	4.39E-08	2.24E+05	2.23E-05	2.20E+05	2.30E-05	1.02	0.97
14	I-134	3.65E-02	3.94E-10	2.01E+03	2.49E-03	1.30E+03	3.90E-03	1.55	0.64
15	I-135	2.75E-01	8.80E-09	4.49E+04	1.11E-04	3.80E+04	1.30E-04	1.18	0.86

Table 2-6f. DCPs and DRLs for Inhalation (Assume 1 Hour Exposure Period), based on 5 rem Committed Equivalent Dose (Adult Male)[a]

				ICRP 60 - Revised Values		1992 Values (Table 5.2)		DCF Comparisons	
No.	Radionuclide	[b]Half-Life (d)	$DCF_{Inh,\ thyroid}$ (Sv/Bq)	$DCP_{Inh,\ Ht}$ (rem·cm^3 per h·µCi)	$DRL_{Inh,\ HT}$ (µCi·h per cm^3)	EPA $DCP_{Inh,\ thyroid}$ (rem·cm^3 per h·µCi)	EPA $DRL_{inh,\ thyroid}$ (µCi·h per cm^3)	DCPInh, thyroid / by 1992 DCPInh, thyroid	DRLInh. Thyroid / by 1992 DRLInh, thyroid
	I-125	60.1	1.04E-07	5.77E+05	8.66E-06	9.60E+05	5.20E-06	0.60	1.67
36	Te-132/I-132	3.26	2.50E-08	1.39E+05	3.60E-05	2.90E+05	1.80E-05	0.48	2.00
	I-129	5.73E+09	7.16E-07	3.97E+06	1.26E-06	6.90E+06	7.20E-07	0.58	1.75
11	I-131	8.04E+00	1.47E-07	8.16E+05	6.13E-06	1.30E+06	3.90E-06	0.63	1.57
12	I-132	9.58E-02	1.36E-09	7.55E+03	6.62E-04	7.70E+03	6.50E-04	0.98	1.02
13	I-133	8.67E-01	2.84E-08	1.58E+05	3.17E-05	2.20E+05	2.30E-05	0.72	1.38
14	I-134	3.65E-02	2.59E-10	1.44E+03	3.48E-03	1.30E+03	3.90E-03	1.11	0.89
15	I-135	2.75E-01	5.76E-09	3.20E+04	1.56E-04	3.80E+04	1.30E-04	0.84	1.20

[a] The EPA PAG Manual assessments are generally based upon the ICRP activity-weighted, average-hourly breathing rates (BR) in an attempt to account for the various in-door and out-door activities (e.g., sleep, rest/sitting, light and heavy activity) engaged in throughout the day. The only time the PAG Manual does not use the activity-weighted, average hourly BR is when the receptor is assumed to be in the plume. Receptors in the plume are assumed to have a light-activity BR = 1.2 m^3/h because it is assumed they are attempting to move out of the plume. The evacuating receptor is assumed to breathe at this elevated rate for a 1 hour duration.
[b] Values from Turbo FRMAC 2.0, RFC 2 (DCFPAK, K. Eckerman)

2.7.4 Dose Conversion Parameter Calculation for Table 2-6

The DCPs in Table 2-6 are calculated using the same method described in Section 2.9 for Table 2-8, with the following modifications:

- The calculations are carried out only for those radionuclides with the potential to produce significant thyroid doses.
- The calculations are carried out for receptors of the various age groups (i.e., infant/newborn, 1-year-old, 5-year-old, 10-year-old, 15-year-old, and adult male) specified by ICRP 60 (ICRP 1991a),
- The calculations are carried out using age-specific, ICRP 66 breathing rates (BR) for the receptors (ICRP 1994).
- The EPA PAG Manual assessments are generally based upon the ICRP activity-weighted, average hourly BR in an attempt to account for the various indoor and outdoor activities (e.g., sleep, rest/sitting, light and heavy activity) engaged in throughout the day. The only time the PAG Manual does not use the activity-weighted, average hourly BR is when the receptor is assumed to be in the plume. Receptors in the plume are assumed to have a light-activity BR = 1.2 m^3/h because it is assumed they are attempting to move out of the plume. The evacuating receptor is assumed to breathe at this elevated rate for a 1-hour duration.
- The calculations are carried out using age-specific, ICRP 60+ inhalation dose conversion factors for the thyroid (not effective dose) ($DCF_{Inh,thy}$) for the receptors.
- The summary version of Table 2-6 only includes the $DCP_{Inh,thy}$ values for the most restrictive age group for each radionuclide.

2.7.5 Derived Response Level Calculation for Table 2-6

The DRLs in Table 2-6 are calculated using the same method described above for Table 2-8, with the following modifications:

- The DRLs are based on a PAG of 5 rem to the thyroid.
- The summary version of Table 2-6 only includes the $DRL_{Inh,thy}$ values for the most restrictive age group for each radionuclide.

NOTE: the most restrictive age group for Te-132/I-132, I-131, I-132, I-133, I-134, and I-135 is the 1-year-old and the most restrictive age group for I-125 and I-129 is the 10-year-old.

2.8 External Exposure to Gamma Radiation from the Plume

Table 2-7 provides DCPs and DRLs for external exposure to gamma radiation due to immersion in contaminated air. The values for gamma radiation will provide conservative estimates for exposure to an overhead plume. They are derived under the assumption that the plume is correctly approximated by a semi-infinite source.

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Table 2-7. DCFs and DRLs for External Exposure Due to Submersion in Contaminated Air (Assume 1 hour Exposure Period) - ICRP 60+

					ICRP 60+ - SNL Revised Values		1992 Values (Table 5.3)		DCF Comparisons	
No.	Radionuclide	[a]Half-Life (d)	[1]Branch Fraction	$^{a}DC_{Submersion, E}$ ($Sv \cdot m^3$ per $s \cdot Bq$)	$DCP_{Submersion, E}$ ($rem \cdot cm^3$ per $h \cdot \mu Ci$)	$DRL_{Submersion, E}$ ($\mu Ci \cdot h$ per cm^3)	EPA $DCF_{Submersion, E}$ ($rem \cdot cm^3$ per $h \cdot \mu Ci$)	EPA $DRL_{Submersion, E}$ ($\mu Ci \cdot h$ per cm^3)	$DCP_{Submersion, E}$ / by 1992 DCF_{Inh}	$DRL_{Submersion, E}$ / by 1992 DCF_{Inh}
1	Am-241	1.58E+05	-	6.74E-16	8.96E+00	1.12E-01	1.10E+01	9.20E-02	0.81	1.21
2	Ba-140/La-140	NA	NA	NA	1.58E+03	6.31E-04	NA	NA	NA	NA
	Ba-140	1.27E+01	1.00E+00	8.07E-15	1.07E+02	9.32E-03	1.10E+02	9.30E-03	0.98	1.00
	La-140	1.68E+00	1.00E+00	1.11E-13	1.48E+03	6.77E-04	1.40E+03	7.10E-04	1.05	0.95
3	Ce144/Pr144/Pr144m	NA	NA	NA	4.54E+01	2.07E-02	3.10E+01	3.20E-02	1.47	0.65
	Ce-144	2.84E+02	1.00E+00	7.63E-16	1.01E+01	9.85E-02	1.00E+01	9.70E-02	1.01	1.02
	Pr-144	1.20E-02	9.82E-01	2.65E-15	3.52E+01	2.84E-02	NA	NA	NA	NA
	Pr-144m	5.00E-03	1.78E-02	2.20E-16	2.93E+00	3.42E-01	NA	NA	NA	NA
	Pr-144	1.20E-02	9.99E-01	2.65E-15	3.52E+01	2.84E-02	NA	NA	NA	NA
4	Cf-252	9.64E+02	-	3.63E-18	4.83E-02	2.07E+01	4.30E-02	2.30E+01	1.12	0.90
5	Cm-244	6.61E+03	-	3.40E-18	4.52E-02	2.21E+01	4.80E-02	2.10E+01	0.94	1.05
6	Co-60	1.93E+03	-	1.19E-13	1.58E+03	6.32E-04	1.50E+03	6.70E-04	1.06	0.94
7	Cs-134	7.53E+02	-	7.06E-14	9.39E+02	1.06E-03	9.10E+02	1.10E-03	1.03	0.97
8	Cs-136	1.31E+01	-	9.94E-14	1.32E+03	7.56E-04	1.30E+03	7.80E-04	1.02	0.97
9	Cs-137/Ba-137m	NA	NA	NA	3.40E+02	2.94E-03	3.50E+02	2.90E-03	0.97	1.02
	Cs-137	1.10E+04	1.00E+00	9.28E-17	1.23E+00	8.10E-01	NA	NA	NA	NA
	Ba-137m	1.77E-03	9.46E-01	2.69E-14	3.58E+02	2.80E-03	NA	NA	NA	NA
10	Gd-153	2.42E+02	-	3.11E-15	4.14E+01	2.42E-02	5.10E+01	2.00E-02	0.81	1.21
11	I-131	8.04E+00	-	1.69E-14	2.25E+02	4.45E-03	2.20E+02	4.60E-03	1.02	0.97
12	I-132	9.58E-02	-	1.05E-13	1.40E+03	7.16E-04	1.40E+03	7.40E-04	1.00	0.97
13	I-133	8.67E-01	-	2.76E-14	3.67E+02	2.72E-03	3.50E+02	2.90E-03	1.05	0.94
14	I-134	3.65E-02	-	1.22E-13	1.62E+03	6.16E-04	1.60E+03	6.40E-04	1.01	0.96
15	I-135/Xe135m	NA	NA	NA	1.04E+03	9.60E-04	NA	NA	NA	NA
	I-135	2.75E-01	1.00E+00	7.54E-14	1.00E+03	9.97E-04	9.50E+02	1.10E-03	1.06	0.91

Table 2-7. DCFs and DRLs for External Exposure Due to Submersion in Contaminated Air (assume 1 hour Exposure Period) - ICRP 60+ (cont'd)

					ICRP 60+ - SNL Revised Values		1992 Values (Table 5.3)		DCF Comparisons	
No.	Radionuclide	[a]Half-Life (d)	[1]Branch Fraction	[1]$DC_{Submersion, E}$ ($Sv \cdot m^3$ per $s \cdot Bq$)	$DCP_{Submersion, E}$ ($rem \cdot cm^3$ per $h \cdot \mu Ci$)	$DRL_{Submersion, E}$ ($\mu Ci \cdot h$ per cm^3)	EPA $DCF_{Submersion, E}$ ($rem \cdot cm^3$ per $h \cdot \mu Ci$)	EPA $DRL_{Submersion, E}$ ($\mu Ci \cdot h$ per cm^3)	$DCP_{Submersion, E}$ / by 1992 DCF_{Inh}	$DRL_{Submersion, E}$ / by 1992 DCF_{Inh}
	Xe-135m	1.06E-02	1.54E-01	1.90E-14	2.53E+02	3.96E-03	NA	NA	NA	NA
16	Ir-192	7.40E+01	-	3.61E-14	4.80E+02	2.08E-03	4.70E+02	2.10E-03	1.02	0.99
17	Kr-87	5.30E-02	-	3.97E-14	5.28E+02	1.89E-03	5.10E+02	2.00E-03	1.04	0.95
18	Kr-88/Rb-88	NA	NA	NA	1.73E+03	5.77E-04	NA	NA	NA	NA
	Kr-88	1.18E-01	1.00E+00	9.71E-14	1.29E+03	7.74E-04	1.30E+03	7.80E-04	0.99	0.99
	Rb-88	1.24E-02	1.00E+00	3.33E-14	4.43E+02	2.26E-03	4.10E+02	2.50E-03	1.08	0.90
19	La-140	1.68E+00	-	1.11E-13	1.48E+03	6.77E-04	1.40E+03	7.10E-04	1.05	0.95
20	Mo-99/Tc-99m	NA	NA	NA	1.54E+02	6.49E-03	NA	NA	NA	NA
	Mo-99	2.75E+00	1.00E+00	6.99E-15	9.30E+01	1.08E-02	9.10E+01	1.10E-02	1.02	0.98
	Tc-99m	2.51E-01	8.76E-01	5.25E-15	6.98E+01	1.43E-02	7.60E+01	1.30E-02	0.92	1.10
21	Np-239	2.36E+00	-	6.95E-15	9.24E+01	1.08E-02	9.60E+01	1.00E-02	0.96	1.08
22	Pm-147	9.58E+02	-	8.67E-18	1.15E-01	8.67E+00	2.10E-03	4.80E+02	54.91	0.02
23	Pu-238	3.20E+04	-	3.50E-18	4.66E-02	2.15E+01	5.00E-02	2.00E+01	0.93	1.07
24	Pu-239	8.79E+06	-	3.48E-18	4.63E-02	2.16E+01	4.70E-02	2.10E+01	0.98	1.03
25	Ra-226/Rn-222…	NA	NA	NA	1.11E+03	8.98E-04	NA	NA	NA	NA
	Ra-226	5.84E+05	1.00E+00	2.84E-16	3.78E+00	2.65E-01	3.90E+00	2.60E-01	0.97	1.02
	Rn-222	3.82E+00	1.00E+00	1.77E-17	2.35E-01	4.25E+00	NA	NA	NA	NA
	Po-218	2.12E-03	1.00E+00	4.21E-19	5.60E-03	1.79E+02	NA	NA	NA	NA
	Pb-214	1.86E-02	1.00E+00	1.09E-14	1.45E+02	6.90E-03	NA	NA	NA	NA
	Bi-214	1.38E-02	1.00E+00	7.25E-14	9.64E+02	1.04E-03	NA	NA	NA	NA
	Po-214	1.90E-09	1.00E+00	3.81E-18	5.07E-02	1.97E+01	NA	NA	NA	NA
	At-218	2.31E-05	2.00E-04	9.71E-17	1.29E+00	7.74E-01	NA	NA	NA	NA
	Bi-214	1.38E-02	1.00E+00	7.25E-14	9.64E+02	1.04E-03	NA	NA	NA	NA
	Po-214	1.90E-09	1.00E+00	3.81E-18	5.07E-02	1.97E+01	NA	NA	NA	NA

Table 2-7. DCFs and DRLs for External Exposure Due to Submersion in Contaminated Air (Assume 1 hour Exposure Period) - ICRP 60+ (cont'd)

No.	Radionuclide	[a]Half-Life (d)	[a]Branch Fraction	[a]$DC_{Submersion, E}$ (Sv · m^3 per (s · Bq)	ICRP 60+ - SNL Revised Values		1992 Values (Table 5.3)		DCF Comparisons	
					$DCP_{Submersion, E}$ (rem·cm^3 per h·μCi)	$DRL_{Submersion, E}$ (μCi·h per cm^3)	EPA $DCF_{Submersion, E}$ (rem·cm^3 per h·μCi)	EPA $DRL_{Submersion, E}$ (μCi·h per cm^3)	$DCP_{Submersion, E}$ / by 1992 DCF_{Inh}	$DRL_{Submersion, E}$ / by 1992 DCF_{Inh}
26	Ru-103/Rh-103m	NA	NA	NA	2.77E+02	3.61E-03	NA	NA	NA	NA
	Ru-103	3.93E+01	1.00E+00	2.08E-14	2.77E+02	3.61E-03	2.80E+02	3.60E-03	0.99	1.00
	Rh-103m	3.90E-02	9.97E-01	6.02E-18	8.01E-02	1.25E+01	NA	NA	NA	NA
27	Ru-106/Rh-106	NA	NA	NA	1.41E+02	7.09E-03	1.20E+02	8.40E-03	1.17	0.84
	Ru-106	3.68E+02	1.00E+00	0.00E+00	0.00E+00	NA	NA	NA	NA	NA
	Rh-106	3.46E-04	1.00E+00	1.06E-14	1.41E+02	7.09E-03	NA	NA	NA	NA
28	Sb-127/Te-127	NA	NA	NA	4.19E+02	2.39E-03	NA	NA	NA	NA
	Sb-127	3.85E+00	1.00E+00	3.12E-14	4.15E+02	2.41E-03	3.90E+02	2.60E-03	1.06	0.93
	Te-127	3.90E-01	8.24E-01	3.34E-16	4.44E+00	2.25E-01	NA	NA	NA	NA
29	Sb-129/Te-129	NA	NA	NA	9.22E+02	1.08E-03	NA	NA	NA	NA
	Sb-129	1.80E-01	1.00E+00	6.71E-14	8.92E+02	1.12E-03	8.60E+02	1.20E-03	1.04	0.93
	Te-129	4.83E-02	7.75E-01	2.86E-15	3.80E+01	2.63E-02	3.10E+01	3.20E-02	1.23	0.82
30	Se-75	1.20E+02	-	1.68E-14	2.23E+02	4.48E-03	2.30E+02	4.40E-03	0.97	1.02
31	Sr-89	5.05E+01	-	4.37E-16	5.81E+00	1.72E-01	8.20E-02	1.20E+01	70.88	0.01
32	Sr-90/Y-90	NA	NA	NA	1.18E+01	8.45E-02	NA	NA	NA	NA
	Sr-90	1.06E+04	1.00E+00	9.83E-17	1.31E+00	7.65E-01	0.00E+00	0.00E+00	NA	NA
	Y-90	2.67E+00	1.00E+00	7.92E-16	1.05E+01	9.49E-02	0.00E+00	0.00E+00	NA	NA
33	Sr-91/Y-91m	NA	NA	NA	6.17E+02	1.62E-03	NA	NA	NA	NA
	Sr-91	3.96E-01	1.00E+00	3.27E-14	4.35E+02	2.30E-03	4.10E+02	2.40E-03	1.06	0.96
	Y-91m	3.45E-02	5.78E-01	2.37E-14	3.15E+02	3.17E-03	NA	NA	NA	NA
34	Te-129m/Te-129	NA	NA	NA	4.55E+01	2.20E-02	NA	NA	NA	NA
	Te-129m	3.36E+01	1.00E+00	1.56E-15	2.07E+01	4.82E-02	2.00E+01	5.10E-02	1.04	0.95
	Te-129	4.83E-02	6.50E-01	2.86E-15	3.80E+01	2.63E-02	3.10E+01	3.20E-02	1.23	0.82
35	[b]Te-131m/Te-131	NA	NA	NA	9.28E+02	1.08E-03	NA	NA	NA	NA

Table 2-7. DCFs and DRLs for External Exposure Due to Submersion in Contaminated Air (Assume 1 hour Exposure Period) - ICRP 60+ (cont'd)

					ICRP 60+ - SNL Revised Values		1992 Values (Table 5.3)		DCF Comparisons	
No.	Radionuclide	[a]Half-Life (d)	[a]Branch Fraction	[a]$DC_{Submersion, E}$ (Sv · m^3 per (s · Bq)	$DCP_{Submersion, E}$ (rem·cm^3 per h·μCi)	$DRL_{Submersion, E}$ (μCi·h per cm^3)	EPA $DCF_{Submersion, E}$ (rem·cm^3 per h·μCi)	EPA $DRL_{Submersion, E}$ (μCi·h per cm^3)	$DCP_{Submersion, E}$ / by 1992 DCF_{Inh}	$DRL_{Submersion, E}$ / by 1992 DCF_{Inh}
	[b]Te-131m	1.25E+00	1.00E+00	6.55E-14	8.71E+02	1.15E-03	8.50E+02	1.20E-03	1.02	0.96
	Te-131	1.74E-02	2.22E-01	1.92E-14	2.55E+02	3.92E-03	NA	NA	NA	NA
36	Te-132/I-132	NA	NA	NA	1.52E+03	6.58E-04	NA	NA	NA	NA
	Te-132	3.26E+00	1.00E+00	9.32E-15	1.24E+02	8.07E-03	1.20E+02	8.00E-03	1.03	1.01
	I-132	9.58E-02	1.00E+00	1.05E-13	1.40E+03	7.16E-04	1.40E+03	7.40E-04	1.00	0.97
37	Tm-170	1.29E+02	-	3.67E-16	4.88E+00	2.05E-01	2.70E+00	3.80E-01	1.81	0.54
38	Xe-133	5.24E+00	-	1.33E-15	1.77E+01	5.65E-02	2.00E+01	5.00E-02	0.88	1.13
39	Xe-135	3.79E-01	-	1.10E-14	1.46E+02	6.84E-03	1.40E+02	7.00E-03	1.05	0.98
40	Xe-138	9.84E-03	-	5.48E-14	7.29E+02	1.37E-03	7.10E+02	1.40E-03	1.03	0.98
41	Y-91	5.85E+01	-	6.22E-16	8.27E+00	1.21E-01	2.10E+00	4.70E-01	3.94	0.26
42	Yb-169	3.20E+01	-	1.13E-14	1.50E+02	6.65E-03	1.60E+02	6.10E-03	0.94	1.09

[a]Values from Turbo FRMAC 2.0, RFC 2 (DCFPAK, K. Eckerman)
[b]Te-131m values in this table are subject to change pending further development of new parent-daughter rules.

2.8.1 Dose Conversion Parameter Calculation for Table 2-7

The DCP for the effective dose received from external exposure while submersed in the passing plume is calculated as shown below. It is assumed that the receptor is submersed in the passing plume for 1 hour.

It should be noted that it is assumed that short-lived daughter radionuclides are in secular equilibrium with the parent radionuclide throughout the exposure period and the external dose from any daughter radionuclides is added to parent's dose to derive the total $DCP_{Submersion, E}$.

$$DCP_{Submersion,E,i} = \sum_{i}^{P+D} \left(DC_{Submersion,E,i} * CF \right)$$

$$\frac{rem \cdot cm^3}{h \cdot \mu Ci} = \frac{Sv \cdot m3}{s.\ Bq} * \frac{rem \cdot cm^3 / h \cdot Ci}{Sv \cdot m^3 / s \cdot Bq}$$

Where:

$\sum_{i}^{P+D}$ = represents the summation of values from the parent radionuclide (P) and all short-lived daughter radionuclides (D);

$DCP_{Submersion,\ E,\ i}$ = Dose Conversion Parameter, value for the effective dose rate per unit activity from external exposure (i.e., 1 h external exposure from submersion in the plume) to radionuclide *i* and any short-lived daughter radionuclide(s) in the plume, rem·cm^3 / h·µCi;

$DC_{Submersion,E,\ i}$ = Dose Coefficient, value for the effective dose from external exposure to radionuclide *i* from submersion in contaminated plume, Sv/s per Bq/m^3, (values from ICRP 60+ dosimetry models, DCFPAK, 2006); and

CF = unit conversion factor, 1.33E+16 rem·cm^3/h·µCi per Sv·m^3/s·Bq.

$$\frac{1.33E16 \frac{rem / cm^3}{h / \mu Ci}}{\frac{Sv / m^3}{s / Bq}} = \frac{Sv / m^3}{s / Bq} * \frac{100\ rem}{Sv} * \frac{Bq}{dps} * \frac{3.7E4 dps}{\mu Ci} * \frac{3.6E3 s}{hr} * \frac{(100 cm)^3}{m^3}$$

2.8.2 Derived Response Level Calculation for Table 2-7

The DRL for external exposure for 1 hour while submersed in the passing plume is calculated using the following equation:

$$DRL_{Submersion,E,i} = \sum_{i}^{P+D} \left(\frac{PAG}{DCP_{Submersion,E,i}} \right), \qquad \frac{\mu Ci \cdot h}{cm^3} = \frac{1 rem}{rem \cdot cm^3 / h \cdot \mu Ci}$$

Where:

$\sum_{i}^{P+D}$ = represents the summation of values from the parent radionuclide (P) and any short-lived daughter radionuclide(s) (D);

$DRL_{Submersion,E,\ i}$ = Derived Response Level (DRL), value for the effective dose from external exposure (i.e., 1 h external exposure from submersion in the plume) to radionuclide *i* and any short-lived daughter radionuclide(s), µCi·h/cm^3;

PAG = EPA's Protective Action Guide (PAG), 1 rem for effective dose or 5 rem organ dose; and

$DCP_{Submersion, E, i}$ = Dose Conversion Parameter, value for the effective dose rate per unit activity from external exposure (i.e., 1 h external exposure from submersion in the plume) to radionuclide *i* and any short-lived daughter radionuclide(s) in the plume, rem·cm^3 / h·µCi.

2.9 Inhalation from the Plume

Table 2-8 provides DCFs and DRLs for CEDE due to inhalation of an airborne plume of radioactive particulate materials. It is assumed that the radionuclides are in the chemical and physical form that yields the highest dose, and that the particle size is 1 micrometer (µm) mean aerodynamic diameter. For other chemical and physical forms of practical interest, the doses may differ, but in general only by a small factor. If the chemical and/or physical form (i.e. solubility class or particle size) is known or can be predicted, the DCFs for inhalation should be adjusted as appropriate.

The dose factors used to develop the DCFs in Table 2-8 of this Manual are ICRP 60 values given in DCFPAK 2006. The breathing rates used to develop the DCFs in Table 2-8 of this Manual are ICRP 66 values. Although the DCFs for some radionuclides would be slightly higher for children, the conservatism in the PAGs and procedures for their application provide an adequate margin for safety. The advantage of using a single source of current data for the development of DCFs for these and any other relevant radionuclides is also a consideration in the selection of this database for use in emergency response applications.

The units given in Table 2.1 of EPA 1988 are converted to the units in Table 2-7 of this Manual, using a breathing rate 9.2E+5 cm^3 x h^{-1}, by the factor:

$$Sv \times Bq^{-1} \times 4.4E + 12 = \text{rem per } \mu Ci \times cm^{-3} \times h$$

The DRLs are simply the reciprocal of the DCF.

Table 2-8. DCFs and DRLs for Inhalation (Assume 1 Hour Exposure Period), Based on 1 rem Committed Effective Dose - ICRP 60+.

						ICRP 60+ - SNL Revised Values		1992 Values (Table 5.4)			DCF Comparisons	
No.	Radionuclide	[a]Half-Life (d)	[a]Branch Fraction	[a]DCF_{Inh} (Sv/Bq)	[a]ICRP 66 Inhalation Class	$DCP_{Inh, E}$ (rem·cm^3 per h·µCi)	$DRL_{Inh, E}$ (µCi·h per cm^3)	EPA $DCF_{Inh, E}$ (rem·cm^3 per h·µCi)	EPA $DRL_{Inh, E}$ (µCi·h per cm^3)	Lung Clearance Class	$DCP_{Inh, E}$ / by 1992 $DCF_{Inh, E}$	$DRL_{Inh, E}$ / by 1992 $DRL_{Inh, E}$
1	Am-241	1.58E+05	-	9.64E-05	F	3.28E+08	3.05E-09	5.30E+08	1.90E-09	W	0.62	1.60
2	Ba-140/La-140	NA	NA	NA	NA	2.38E+04	4.21E-05	NA	NA	NA	NA	NA
	Ba-140	1.27E+01	1.00E+00	5.84E-09	S	1.99E+04	5.03E-05	4.50E+03	2.20E-04	D	4.42	0.23
	La-140	1.68E+00	1.00E+00	1.14E-09	S	3.88E+03	2.58E-04	5.80E+03	1.70E-04	W	0.67	1.52
3	Ce144/Pr144/Pr144m	NA	NA	NA	NA	1.79E+05	5.57E-06	4.50E+05	2.20E-06	Y	0.40	2.53
	Ce-144	2.84E+02	1.00E+00	5.27E-08	S	1.79E+05	5.57E-06	4.50E+05	2.20E-06	Y	0.40	2.53

	Pr-144	1.20E-02	9.82E-01	1.83E-11	S	6.23E+01	1.61E-02	NA	NA	NA	NA	NA
	Pr-144m	5.00E-03	1.78E-02	0.00E+00	NA	0.00E+00	0.00E+00	NA	NA	NA	NA	NA
	Pr-144	1.20E-02	9.99E-01	1.83E-11	S	6.23E+01	1.61E-02	NA	NA	NA	NA	NA
4	Cf-252	9.64E+02	-	3.71E-05	F	1.26E+08	7.92E-09	1.90E+08	5.30E-09	Y	0.66	1.49
5	Cm-244	6.61E+03	-	5.70E-05	F	1.94E+08	5.15E-09	3.00E+08	3.40E-09	W	0.65	1.52
6	Co-60	1.93E+03	-	3.07E-08	S	1.05E+05	9.57E-06	2.60E+05	3.80E-06	Y	0.40	2.52
7	Cs-134	7.53E+02	-	2.04E-08	S	6.94E+04	1.44E-05	5.60E+04	1.80E-05	D	1.24	0.80
8	Cs-136	1.31E+01	-	2.78E-09	S	9.46E+03	1.06E-04	8.80E+03	1.10E-04	D	1.08	0.96
9	Cs-137/Ba-137m	NA	NA	NA	NA	1.33E+05	7.49E-06	3.80E+04	2.60E-05	D	3.51	0.29
	Cs-137	1.10E+04	1.00E+00	3.92E-08	S	1.33E+05	7.49E-06	NA	NA	NA	NA	NA
	Ba-137m	1.77E-03	9.46E-01	0.00E+00	NA	0.00E+00	0.00E+00	NA	NA	NA	NA	NA
10	Gd-153	2.42E+02	-	2.40E-09	S	8.17E+03	1.22E-04	2.90E+04	3.50E-05	D	0.28	3.50
11	I-131	8.04E+00	-	7.39E-09	F	2.52E+04	3.98E-05	3.90E+04	2.50E-05	D	0.65	1.59
12	I-132	9.58E-02	-	1.14E-10	M	3.88E+02	2.58E-03	4.60E+02	2.20E-03	D	0.84	1.17
13	I-133	8.67E-01	-	1.47E-09	F	5.00E+03	2.00E-04	7.00E+03	1.40E-04	D	0.71	1.43
14	I-134	3.65E-02	-	5.59E-11	S	1.90E+02	5.26E-03	1.60E+02	6.30E-03	D	1.19	0.83
15	I-135/Xe135m	NA	NA	NA	NA	1.10E+03	9.10E-04	NA	NA	NA	NA	NA
	I-135	2.75E-01	1.00E+00	3.23E-10	F	1.10E+03	9.10E-04	1.50E+03	6.80E-04	D	0.73	1.34

Table 2-8. DCFs and DRLs for Inhalation (Assume 1 Hour Exposure Period), Based on 1 rem Committed Effective Dose - ICRP 60+ (cont'd)

						ICRP 60+ - SNL Revised Values		1992 Values (Table 5.4)			DCF Comparisons	
No.	Radionuclide	[a]Half-Life (d)	[a]Branch Fraction	[a]DCF_{Inh} (Sv/Bq)	[a]ICRP 66 Inhalation Class	$DCP_{Inh, E}$ (rem·cm^3 per h·µCi)	$DRL_{Inh, E}$ (µCi·h per cm^3)	EPA $DCF_{Inh, E}$ (rem·cm^3 per h·µCi)	EPA $DRL_{Inh, E}$ (µCi·h per cm^3)	Lung Clearance Class	$DCP_{Inh, E}$ / by 1992 $DCF_{Inh, E}$	$DRL_{Inh, E}$ / by 1992 $DRL_{Inh, E}$
	Xe-135m	1.06E-02	1.54E-01	0.00E+00	NA	0.00E+00	0.00E+00	NA	NA	NA	NA	NA
16	Ir-192	7.40E+01	-	6.62E-09	S	2.25E+04	4.44E-05	3.40E+04	3.00E-05	Y	0.66	1.48
17	Kr-87	5.30E-02	-	0.00E+00	NA	NA	NA	NA	NA	NA	NA	NA
18	Kr-88/Rb-88	NA	NA	NA	NA	3.97E+03	2.52E-04	NA	NA	NA	NA	NA
	Kr-88	1.18E-01	1.00E+00	0.00E+00	NA	3.88E+03	2.58E-04	NA	NA	NA	NA	NA
	Rb-88	1.24E-02	1.00E+00	2.76E-11	S	9.40E+01	1.06E-02	1.00E+02	1.00E-02	D	0.94	1.06
19	La-140	1.68E+00	-	1.14E-09	S	3.88E+03	2.58E-04	5.80E+03	1.70E-04	W	0.67	1.52
20	Mo-99/Tc-99m	NA	NA	NA	NA	3.44E+03	2.91E-04	NA	NA	NA	NA	NA
	Mo-99	2.75E+00	1.00E+00	9.92E-10	S	3.38E+03	2.96E-04	4.80E+03	2.10E-04	Y	0.70	1.41
	Tc-99m	2.51E-01	8.76E-01	2.01E-11	S	6.84E+01	1.46E-02	3.90E+01	2.60E-02	D	1.75	0.56
21	Np-239	2.36E+00	-	1.03E-09	S	3.51E+03	2.85E-04	3.00E+03	3.30E-04	W	1.17	0.86
22	Pm-147	9.58E+02	-	6.98E-09	F	2.38E+04	4.21E-05	4.70E+04	2.10E-05	Y	0.51	2.00
23	Pu-238	3.20E+04	-	1.08E-04	F	3.68E+08	2.72E-09	4.70E+08	2.10E-09	W	0.78	1.30
24	Pu-239	8.79E+06	-	1.19E-04	F	4.05E+08	2.47E-09	5.20E+08	1.90E-09	W	0.78	1.30
25	Ra-226/Rn-222…	NA	NA	NA	NA	3.25E+07	3.08E-08	NA	NA	NA	NA	NA
	Ra-226	5.84E+05	1.00E+00	9.51E-06	S	3.24E+07	3.09E-08	1.00E+07	9.70E-08	W	3.24	0.32
	Rn-222	3.82E+00	1.00E+00	0.00E+00	NA	0.00E+00	0.00E+00	NA	NA	NA	NA	NA
	Po-218	2.12E-03	1.00E+00	0.00E+00	NA	0.00E+00	0.00E+00	NA	NA	NA	NA	NA
	Pb-214	1.86E-02	1.00E+00	1.47E-08	S	5.00E+04	2.00E-05	NA	NA	NA	NA	NA
	Bi-214	1.38E-02	1.00E+00	1.54E-08	S	5.24E+04	1.91E-05	NA	NA	NA	NA	NA
	Po-214	1.90E-09	1.00E+00	0.00E+00	NA	0.00E+00	0.00E+00	NA	NA	NA	NA	NA
	At-218	2.31E-05	2.00E-04	0.00E+00	NA	0.00E+00	0.00E+00	NA	NA	NA	NA	NA
	Bi-214	1.38E-02	1.00E+00	1.54E-08	S	5.24E+04	1.91E-05	NA	NA	NA	NA	NA
	Po-214	1.90E-09	1.00E+00	0.00E+00	NA	0.00E+00	0.00E+00	NA	NA	NA	NA	NA

Table 2-8. DCFs and DRLs for Inhalation (Assume 1 Hour Exposure period), based on 1 rem Committed Effective Dose - ICRP 60+ (cont'd)

						ICRP 60+ - SNL Revised Values		1992 Values (Table 5.4)			DCF Comparisons	
No.	Radionuclide	[a]Half-Life (d)	[a]Branch Fraction	[a]DCF_{Inh} (Sv/Bq)	[1]ICRP 66 Inhalation Class	$DCP_{Inh, E}$ (rem·cm^3 per h·µCi)	$DRL_{Inh, E}$ (µCi·h per cm^3)	EPA $DCF_{Inh, E}$ (rem·cm^3 per h·µCi)	EPA $DRL_{Inh, E}$ (µCi·h per cm^3)	Lung Clearance Class	$DCP_{Inh, E}$ / by 1992 $DCF_{Inh, E}$	$DRL_{Inh, E}$ / by 1992 $DRL_{Inh, E}$
26	Ru-103/Rh-103m	NA	NA	NA	NA	1.01E+04	9.95E-05	NA	NA	NA	NA	NA
	Ru-103	3.93E+01	1.00E+00	2.95E-09	S	1.00E+04	9.96E-05	1.10E+04	9.30E-05	Y	0.91	1.07
	Rh-103m	3.90E-02	9.97E-01	2.73E-12	S	9.29E+00	1.08E-01	NA	NA	NA	NA	NA
27	Ru-106/Rh-106	NA	NA	NA	NA	2.25E+05	4.45E-06	5.70E+05	1.70E-06	Y	0.39	2.62
	Ru-106	3.68E+02	1.00E+00	6.60E-08	S	2.25E+05	4.45E-06	NA	NA	NA	NA	NA
	Rh-106	3.46E-04	1.00E+00	0.00E+00	NA	0.00E+00	NA	NA	NA	NA	NA	NA
28	Sb-127/Te-127	NA	NA	NA	NA	6.79E+03	1.47E-04	NA	NA	NA	NA	NA
	Sb-127	3.85E+00	1.00E+00	1.88E-09	S	6.40E+03	1.56E-04	7.20E+03	1.40E-04	W	0.89	1.12
	Te-127	3.90E-01	8.24E-01	1.40E-10	S	4.77E+02	2.10E-03	NA	NA	NA	NA	NA
29	Sb-129/Te-129	NA	NA	NA	NA	9.55E+02	1.05E-03	NA	NA	NA	NA	NA
	Sb-129	1.80E-01	1.00E+00	2.50E-10	S	8.51E+02	1.18E-03	7.70E+02	1.30E-03	W	1.11	0.90
	Te-129	4.83E-02	7.75E-01	3.93E-11	S	1.34E+02	7.48E-03	1.10E+02	9.30E-03	D	1.22	0.80
30	Se-75	1.20E+02	-	1.34E-09	S	4.56E+03	2.19E-04	1.00E+04	9.80E-05	W	0.46	2.24
31	Sr-89	5.05E+01	-	7.94E-09	S	2.70E+04	3.70E-05	5.00E+04	2.00E-05	Y	0.54	1.85
32	Sr-90/Y-90	NA	NA	NA	NA	5.40E+05	1.85E-06	NA	NA	NA	NA	NA
	Sr-90	1.06E+04	1.00E+00	1.57E-07	S	5.34E+05	1.87E-06	1.60E+06	6.40E-07	Y	0.33	2.92
	Y-90	2.67E+00	1.00E+00	1.50E-09	S	5.11E+03	1.96E-04	1.00E+04	9.90E-05	Y	0.51	1.98
33	Sr-91/Y-91m	NA	NA	NA	NA	1.41E+03	7.09E-04	NA	NA	NA	NA	NA
	Sr-91	3.96E-01	1.00E+00	4.08E-10	S	1.39E+03	7.20E-04	2.00E+03	5.00E-04	Y	0.69	1.44
	Y-91m	3.45E-02	5.78E-01	1.14E-11	S	3.88E+01	2.58E-02	NA	NA	NA	NA	NA
34	Te-129m/Te-129	NA	NA	NA	NA	2.70E+04	3.70E-05	NA	NA	NA	NA	NA
	Te-129m	3.36E+01	1.00E+00	7.92E-09	S	2.70E+04	3.71E-05	2.90E+04	3.50E-05	W	0.93	1.06
	Te-129	4.83E-02	6.50E-01	3.93E-11	S	1.34E+02	7.48E-03	1.10E+02	9.30E-03	D	1.22	0.80
35	[b]Te-131m/Te-131	NA	NA	NA	NA	3.66E+03	2.73E-04	NA	NA	NA	NA	NA
	[b]Te-131m	1.25E+00	1.00E+00	1.07E-09	M	3.64E+03	2.75E-04	7.70E+03	1.30E-04	W	0.47	2.11
	Te-131	1.74E-02	2.22E-01	2.85E-11	M	9.70E+01	1.03E-02	NA	NA	NA	NA	NA
36	Te-132/I-132	NA	NA	NA	NA	7.37E+03	1.36E-04	1.20E+04	8.50E-05	W	0.61	1.60

Table 2-8. DCFs and DRLs for Inhalation (Assume 1 Hour Exposure Period), Based on 1 rem Committed Effective Dose - ICRP 60+ (cont'd)

						ICRP 60+ - SNL Revised Values		1992 Values (Table 5.4)			DCF Comparisons	
No.	Radionuclide	[a]Half-Life (d)	[a]Branch Fraction	[a]DCF_{inh} (Sv/Bq)	[a]ICRP 66 Inhalation Class	$DCP_{inh,E}$ (rem·cm^3 per h·µCi)	$DRL_{inh,E}$ (µCi·h per cm^3)	EPA $DCF_{inh,E}$ (rem·cm^3 per h·µCi)	EPA $DRL_{inh,E}$ (µCi·h per cm^3)	Lung Clearance Class	$DCP_{inh,E}$ / by 1992 $DCF_{inh,E}$	$DRL_{inh,E}$ / by 1992 $DRL_{inh,E}$
	Te-132	3.26E+00	1.00E+00	2.05E-09	M	6.98E+03	1.43E-04	1.10E+04	8.80E-05	W	0.63	1.63
	I-132	9.58E-02	1.00E+00	1.14E-10	M	3.88E+02	2.58E-03	4.60E+02	2.20E-03	D	0.84	1.17
37	Tm-170	1.29E+02	-	9.29E-09	S	3.16E+04	3.16E-05	3.20E+04	3.20E-05	W	0.99	0.99
38	Xe-133	5.24E+00	-	0.00E+00	NA	NA	NA	NA	NA	NA	NA	NA
39	Xe-135	3.79E-01	-	0.00E+00	NA	NA	NA	NA	NA	NA	NA	NA
40	Xe-138	9.84E-03	-	0.00E+00	NA	0.00E+00	NA	NA	NA	NA	NA	NA
41	Y-91	5.85E+01	-	8.93E-09	S	3.04E+04	3.29E-05	5.90E+04	1.70E-05	Y	0.52	1.94
42	Yb-169	3.20E+01	-	2.98E-09	S	1.01E+04	9.86E-05	9.70E+03	1.00E-04	Y	1.05	0.99

[a]Values from Turbo FRMAC 2.0, RFC 2 (DCFPAK, K. Eckerman)
[b]Te-131m values in this table are subject to change pending further development of new parent-daughter rules.

2.9.1 Dose Conversion Parameter Calculation for Table 2-8

The DCP, effective, for inhalation is calculated as shown below. It is assumed that the receptor is exposed to the passing plume for 1 h.

$$DCP_{Inh,E,i} = \sum_{i}^{P+D}\left(BR * DCF_{Inh,E,i} * CF \right)$$

$$\frac{rem \cdot cm^3}{h \cdot \mu Ci} = \frac{cm^3}{h} * \frac{Sv}{Bq} * \frac{rem \cdot Bq}{Sv \cdot \mu Ci}$$

Where:

$\sum_{i}^{P+D}$ = represents the summation of values from the parent radionuclide (P) and all short-lived daughter radionuclides (D);

$DCP_{Inhalation,E,i}$ = Dose Conversion Parameter, value for the committed effective dose rate per unit activity from exposure (i.e., 1 h plume inhalation) to radionuclide *i* and any short-lived daughter radionuclide(s), rem·cm^3 / h·µCi;

BR = breathing rate, 9.20E+05 cm^3/h, activity-weighted, average-hourly breathing rate for an adult (from ICRP 66, Table B.16.B);

$DCF_{Inh,E,i}$ = inhalation dose conversion factor, value for the committed effective dose from the inhalation of radionuclide *i* (most restrictive lung clearance class) (values from ICRP 60+ dosimetry models, DCFPAK, 2006), Sv/Bq; and

CF = unit conversion factor, value to convert Sv/Bq to rem/µCi, 3.70E+06 rem/µCi per Bq/Sv.

$$\frac{3.7E6\, rem/\mu Ci}{Sv/Bq} = \frac{Sv}{Bq} * \frac{10^2\ rem}{Sv} * \frac{Bq}{dps} * \frac{3.7E4\, dps}{Bq}$$

2.9.2 Derived Response Level Calculation for Table 2-8

The DRL for 1 hour of inhalation in the passing plume is calculated using the following equation:

$$DRL_{Inhalation,E,i} = \sum_{i}^{P+D}\left(\frac{PAG}{DCP_{Inhalation,E,i}} \right), \qquad \frac{\mu Ci \cdot h}{cm^3} = \frac{1\, rem}{rem \cdot cm^3/h \cdot \mu Ci}$$

Where:

$\sum_{i}^{P+D}$ = represents the summation of values from the parent radionuclide (P) and any short-lived daughter radionuclide(s) (D);

$DRL_{Inhalation,E,i}$ = the Derived Response Level (DRL), value based on the committed effective dose for exposure (i.e., 1 h plume inhalation) to radionuclide *i* and any short-lived daughter radionuclide(s) in the plume, µCi·h/cm^3;

PAG = EPA's PAG, 1 rem for effective dose; and

$DCP_{Inhalation,E,i}$ = the Dose Conversion Parameter, value for the committed effective dose rate per unit activity from exposure (i.e., 1 h plume inhalation) to radionuclide *i* and any short-lived daughter radionuclide(s), rem·cm^3 / h·µCi.

2.10 External Dose from Deposited Materials

Table 2-9 provides DCFs and DRLs for four-day exposure to gamma radiation from selected radionuclides following deposition of particulate materials on the ground from a plume. The deposition velocity for all radionuclides (assumed to be 0.1 cm/s) could vary widely depending on the physical and chemical characteristics of the deposited material and the surface and meteorological conditions. In the case of precipitation, the amount of deposition (and thus the DCFs for this exposure pathway) will be much higher. To account for the in-growth of short-lived daughters in deposited materials after measurements are made, the tabulated values include their contribution to dose over the assumed four-day period of exposure. Because the deposition velocity can be much lower or higher than assumed in developing the DCFs for deposited materials, decision makers are cautioned to pay particular attention to actual measurements of gamma exposure from deposited materials for evacuation decisions after plume passage.

The objective is to calculate DCFs for single radionuclides in terms of effective dose equivalent from four days of exposure to gamma radiation from deposited radioactive materials. In order to be able to sum the DCFs with those for other exposure pathways, the DCF is expressed in terms of dose per unit time-integrated air concentration where the deposition from the plume is assumed to occur at approximately the beginning of the incident.

Because of large uncertainties in the assumptions for deposition, air concentrations are an inadequate basis for the decisions on the need to decontaminate individuals. Field measurements should be used for this (See Chapter 3, Section 3.7.3). It should be noted that, even in situations where skin beta dose might exceed 50 rem (5 Sv), evacuation would not usually be the appropriate protective action because skin decontamination and clothing changes are easily available and effective. However, evacuation would usually already be justified in these situations due to dose from inhalation during plume passage.

Table 2-9. Calculation for Early Phase (First 96 Hours)

											ICRP 60+ - Revised Values		1992 Values
No.	Radionuclide	[a]Half-Life (d)	[a]Branch Fraction	[a]$ExDC_{ground}$ (Sv·m^2 per s·Bq)	[a]CRP (h)	[a]KP (h/cm)	[a]DCF_{Inh} (Sv/Bq)	[a]ICRP 66 Inhalation Class	$DCF_{groundshine}$ (rem·cm^3 per h·µCi)	DCF_{Inh} (rem·cm^3 per h·µCi)	$DCP_{groundshine+Inh}$ (rem·cm^3 per h·µCi)	$DRL_{groundshine+Inh}$ (µCi·h per cm^3)	EPA $DCF_{groundshine}$ (rem·cm^3 per h·µCi)
1	Am-241	1.58E+05	-	2.33E-17	95.9	5.73E-07	9.64E-05	F	8.77E+01	6.77E+04	6.78E+04	1.48E-05	1.20E+02
2	Ba-140/La-140	NA	NA						7.95E+03	4.53E+00	7.96E+03	1.26E-04	NA
	Ba-140	1.27E+01	1.00E+00	1.90E-16	86.2	5.30E-07	5.84E-09	S	6.43E+02	3.79E+00	6.47E+02	1.55E-03	7.00E+02
	La-140	1.68E+00	1.00E+00	2.16E-15	86.2	5.30E-07	1.14E-09	S	7.31E+03	7.40E-01	7.31E+03	1.37E-04	4.10E+03
3	Ce144/Pr144/Pr144m	NA	NA						6.80E+02	3.69E+01	7.17E+02	1.39E-03	2.00E+02
	Ce-144	2.84E+02	1.00E+00	1.84E-17	95.4	5.71E-07	5.27E-08	S	6.89E+01	3.69E+01	1.06E+02	9.45E-03	8.50E+01
	Pr-144	1.20E-02	9.82E-01	1.63E-16	95.4	5.71E-07	1.83E-11	S	6.11E+02	1.28E-02	6.11E+02	1.64E-03	NA
	Pr-144m	5.00E-03	1.78E-02	1.05E-17	95.4	5.71E-07	0.00E+00	NA	3.93E+01	0.00E+00	3.93E+01	2.54E-02	NA
	Pr-144	1.20E-02	9.99E-01	1.63E-16	95.4	5.71E-07	1.83E-11	S	6.11E+02	1.28E-02	6.11E+02	1.64E-03	NA
4	Cf-252	9.64E+02	-	5.24E-19	95.8	5.72E-07	3.71E-05	F	1.97E+00	2.60E+04	2.60E+04	3.85E-05	2.50E+00
5	Cm-244	6.61E+03	-	6.44E-19	95.9	5.73E-07	5.70E-05	F	2.42E+00	4.00E+04	4.00E+04	2.50E-05	3.30E+00
6	Co-60	1.93E+03	-	2.30E-15	95.8	5.72E-07	3.07E-08	S	8.65E+03	2.15E+01	8.67E+03	1.15E-04	8.90E+03
7	Cs-134	7.53E+02	-	1.48E-15	95.7	5.72E-07	2.04E-08	S	5.56E+03	1.43E+01	5.58E+03	1.79E-04	6.20E+03
8	Cs-136	1.31E+01	-	2.03E-15	86.4	5.31E-07	2.78E-09	S	6.89E+03	1.81E+00	6.89E+03	1.45E-04	7.60E+03
9	Cs-137/Ba-137m	NA	NA						2.07E+03	2.75E+01	2.10E+03	4.76E-04	2.40E+03
	Cs-137	1.10E+04	1.00E+00	2.99E-18	95.9	5.73E-07	3.92E-08	S	1.13E+01	2.75E+01	3.88E+01	2.58E-02	NA
	Ba-137m	1.77E-03	9.46E-01	5.79E-16	95.9	5.73E-07	0.00E+00	NA	2.18E+03	0.00E+00	2.18E+03	4.59E-04	NA
10	Gd-153	2.42E+02	-	9.22E-17	95.4	5.70E-07	2.40E-09	S	3.45E+02	1.68E+00	3.47E+02	2.88E-03	5.00E+02
11	^{3}I-131	8.04E+00	-	3.64E-16	81.1	5.07E-07	7.39E-09	F	1.16E+03	4.59E+00	1.16E+03	8.59E-04	1.30E+04
12	I-132	9.58E-02	-	2.20E-15	3.32	3.32E-08	1.14E-10	M	2.87E+02	4.64E-03	2.87E+02	3.49E-03	3.10E+03

13	I-133	8.67E-01	-	6.17E-16	28.8	2.37E-07	1.47E-09	F	6.98E+02	4.27E-01	6.98E+02	1.43E-03	7.30E+03
14	^{3}I-134	3.65E-02	-	2.53E-15	1.26	1.26E-08	5.59E-11	S	1.25E+02	8.63E-04	1.25E+02	7.99E-03	1.30E+03
15	I-135/Xe135m	NA	NA						5.74E+02	3.70E-02	5.74E+02	1.74E-03	NA
	I-135	2.75E-01	1.00E+00	1.47E-15	9.53	9.35E-08	3.23E-10	F	5.50E+02	3.70E-02	5.50E+02	1.82E-03	5.70E+03
	Xe-135m	1.06E-02	1.54E-01	4.19E-16	9.53	9.35E-08	0.00E+00	NA	1.57E+02	0.00E+00	1.57E+02	6.38E-03	NA
16	^{3}Ir-192	7.40E+01	-	7.77E-16	94.1	5.65E-07	6.62E-09	S	2.87E+03	4.58E+00	2.88E+03	3.48E-04	3.40E+03
17	Kr-87	5.30E-02	-	8.40E-16	1.83	1.83E-08	0.00E+00	NA	6.04E+01	0.00E+00	6.04E+01	1.66E-02	NA
18	Kr-88/Rb-88	NA	NA						3.98E+02	1.39E-03	3.98E+02	2.51E-03	NA
	Kr-88	1.18E-01	1.00E+00	1.73E-15	4.1	4.10E-08	0.00E+00	NA	2.78E+02	0.00E+00	2.78E+02	3.59E-03	NA

Table 2-9. Calculation for Early Phase (First 96 Hours) (cont'd)

No.	Radionuclide	[a]Half-Life (d)	[a]Branch Fraction	[a]ExDC$_{ground}$ (Sv·m^2 per s·Bq)	[a]CRP (h)	[a]KP (h/cm)	[a]DCF$_{Inh}$ (Sv/Bq)	[a]ICRP 66 Inhalation Class	DCF$_{groundshine}$ (rem·cm^3 per h·µCi)	DCF$_{Inh}$ (rem·cm^3 per h·µCi)	ICRP 60+ - Revised Values: DCP$_{groundshine+Inh}$ (rem·cm^3 per h·µCi)	ICRP 60+ - Revised Values: DRL$_{groundshine+Inh}$ (µCi·h per cm^3)	1992 Values (T: EPA DCF$_{groundshine}$ (rem·cm^3 per h·µCi)	D
	Rb-88	1.24E-02	1.00E+00	7.41E-16	4.1	4.10E-08	2.76E-11	S	1.19E+02	1.39E-03	1.19E+02	8.38E-03	1.00E+01	
19	La-140	1.68E+00	-	2.16E-15	46.9	3.41E-07	1.14E-09	S	3.98E+03	4.76E-01	3.98E+03	2.51E-04	4.10E+03	
20	Mo-99/Tc-99m	NA	NA						6.59E+02	5.06E-01	6.59E+02	1.52E-03	NA	
	Mo-99	2.75E+00	1.00E+00	1.78E-16	60.4	4.09E-07	9.92E-10	S	4.22E+02	4.97E-01	4.23E+02	2.37E-03	4.00E+02	
	Tc-99m	2.51E-01	8.76E-01	1.14E-16	60.4	4.09E-07	2.01E-11	S	2.70E+02	1.01E-02	2.70E+02	3.70E-03	5.30E+01	
21	Np-239	2.36E+00	-	1.54E-16	56.4	3.89E-07	1.03E-09	S	3.41E+02	4.91E-01	3.42E+02	2.93E-03	4.50E+02	
22	Pm-147	9.58E+02	-	2.80E-20	95.8	5.72E-07	6.98E-09	F	1.05E-01	4.89E+00	5.00E+00	2.00E-01	1.60E-02	
23	Pu-238	3.20E+04	-	6.26E-19	95.9	5.73E-07	1.08E-04	F	2.36E+00	7.58E+04	7.58E+04	1.32E-05	3.40E+00	
24	Pu-239	8.79E+06	-	2.84E-19	95.9	5.73E-07	1.19E-04	F	1.07E+00	8.36E+04	8.36E+04	1.20E-05	1.50E+00	
25	Ra-226/Rn-222…	NA	NA						6.35E+03	6.70E+03	1.31E+04	7.66E-05	NA	
	Ra-226	5.84E+05	1.00E+00	6.11E-18	95.9	5.73E-07	9.51E-06	S	2.30E+01	6.68E+03	6.70E+03	1.49E-04	3.00E+01	
	Rn-222	3.82E+00	1.00E+00	3.82E-19	95.9	5.73E-07	0.00E+00	NA	1.44E+00	0.00E+00	1.44E+00	6.95E-01	NA	
	Po-218	2.12E-03	1.00E+00	8.66E-21	95.9	5.73E-07	0.00E+00	NA	3.26E-02	0.00E+00	3.26E-02	3.07E+01	NA	
	Pb-214	1.86E-02	1.00E+00	2.40E-16	95.9	5.73E-07	1.47E-08	S	9.04E+02	1.03E+01	9.14E+02	1.09E-03	NA	
	Bi-214	1.38E-02	1.00E+00	1.44E-15	95.9	5.73E-07	1.54E-08	S	5.42E+03	1.08E+01	5.43E+03	1.84E-04	NA	
	Po-214	1.90E-09	1.00E+00	7.93E-20	95.9	5.73E-07	0.00E+00	NA	2.99E-01	0.00E+00	2.99E-01	3.35E+00	NA	
	At-218	2.31E-05	2.00E-04	3.64E-18	95.9	5.73E-07	0.00E+00	NA	1.37E+01	0.00E+00	1.37E+01	7.30E-02	NA	
	Bi-214	1.38E-02	1.00E+00	1.44E-15	95.9	5.73E-07	1.54E-08	S	5.42E+03	1.08E+01	5.43E+03	1.84E-04	NA	
	Po-214	1.90E-09	1.00E+00	7.93E-20	95.9	5.73E-07	0.00E+00	NA	2.99E-01	0.00E+00	2.99E-01	3.35E+00	NA	
26	Ru-103/Rh-103m	NA	NA						1.64E+03	2.02E+00	1.64E+03	6.11E-04	NA	
	Ru-103	3.93E+01	1.00E+00	4.49E-16	92.6	5.58E-07	2.95E-09	S	1.63E+03	2.02E+00	1.63E+03	6.12E-04	1.90E+03	
	Rh-103m	3.90E-02	9.97E-01	8.86E-19	92.6	5.58E-07	2.73E-12	S	3.22E+00	1.87E-03	3.22E+00	3.10E-01	NA	
27	Ru-106/Rh-106	NA	NA						1.29E+03	4.62E+01	1.34E+03	7.46E-04	8.30E+02	
	Ru-106	3.68E+02	1.00E+00	0.00E+00	95.5	5.71E-07	6.60E-08	S	0.00E+00	4.62E+01	4.62E+01	2.17E-02	NA	
	Rh-106	3.46E-04	1.00E+00	3.45E-16	95.5	5.71E-07	0.00E+00	NA	1.29E+03	0.00E+00	1.29E+03	7.73E-04	NA	
28	Sb-127/Te-127	NA	NA						1.84E+03	1.10E+00	1.84E+03	5.44E-04	NA	
	Sb-127	3.85E+00	1.00E+00	6.76E-16	68.4	4.48E-07	1.88E-09	S	1.82E+03	1.03E+00	1.82E+03	5.51E-04	1.90E+03	
	Te-127	3.90E-01	8.24E-01	1.03E-17	68.4	4.48E-07	1.40E-10	S	2.77E+01	7.69E-02	2.77E+01	3.61E-02	NA	
29	Sb-129/Te-129	NA	NA						3.57E+02	2.13E-02	3.57E+02	2.80E-03	NA	

Table 2-9. Calculation for Early Phase (First 96 Hours) (cont'd)

No.	Radionuclide	[a]Half-Life (d)	[a]Branch Fraction	[a]ExDC$_{ground}$ (Sv·m^2 per s·Bq)	[a]CRP (h)	[a]KP (h/cm)	[a]DCF$_{Inh}$ (Sv/Bq)	[a]ICRP 66 Inhalation Class	DCF$_{groundshine}$ (rem·cm^3 per h·µCi)	DCF$_{Inh}$ (rem·cm^3 per h·µCi)	ICRP 60+ - Revised Values: DCP$_{groundshine+Inh}$ (rem·cm^3 per h·µCi)	ICRP 60+ - Revised Values: DRL$_{groundshine+Inh}$ (µCi·h per cm^3)	1992 Values (T: EPA DCF$_{groundshine}$ (rem·cm^3 per h·µCi)	D
	Sb-129	1.80E-01	1.00E+00	1.37E-15	6.23	6.21E-08	2.50E-10	S	3.35E+02	1.90E-02	3.35E+02	2.98E-03	3.70E+02	
	Te-129	4.83E-02	7.75E-01	1.14E-16	6.23	6.21E-08	3.93E-11	S	2.79E+01	2.99E-03	2.79E+01	3.59E-02	3.90E+00	
30	Se-75	1.20E+02	-	3.61E-16	94.8	5.68E-07	1.34E-09	S	1.34E+03	9.33E-01	1.34E+03	7.44E-04	1.70E+03	
31	Sr-89	5.05E+01	-	6.86E-17	93.3	5.61E-07	7.94E-09	S	2.51E+02	5.46E+00	2.57E+02	3.89E-03	5.20E-01	
32	Sr-90/Y-90	NA	NA						4.20E+02	1.11E+02	5.32E+02	1.88E-03	NA	
	Sr-90	1.06E+04	1.00E+00	1.64E-18	95.9	5.73E-07	1.57E-07	S	6.17E+00	1.10E+02	1.16E+02	8.59E-03	NA	
	Y-90	2.67E+00	1.00E+00	1.10E-16	95.9	5.73E-07	1.50E-09	S	4.14E+02	1.05E+00	4.15E+02	2.41E-03	NA	

33	Sr-91/Y-91m	NA	NA						5.50E+02	6.60E-02	5.50E+02	1.82E-03	NA
	Sr-91	3.96E-01	1.00E+00	7.27E-16	13.7	1.30E-07	4.08E-10	S	3.91E+02	6.50E-02	3.91E+02	2.56E-03	3.80E+02
	Y-91m	3.45E-02	5.78E-01	5.10E-16	13.7	1.30E-07	1.14E-11	S	2.74E+02	1.82E-03	2.74E+02	3.65E-03	NA
34	Te-129m/Te-129	NA	NA						4.74E+02	5.41E+00	4.79E+02	2.09E-03	NA
	Te-129m	3.36E+01	1.00E+00	5.70E-17	92.1	5.56E-07	7.92E-09	S	2.06E+02	5.40E+00	2.12E+02	4.73E-03	1.40E+02
	Te-129	4.83E-02	6.50E-01	1.14E-16	92.1	5.56E-07	3.93E-11	S	4.12E+02	2.68E-02	4.12E+02	2.43E-03	3.90E+00
35	[d]Te-131m/Te-131	NA	NA						2.18E+03	3.89E-01	2.18E+03	4.58E-04	NA
	[d]Te-131m	1.25E+00	1.00E+00	1.34E-15	38.5	2.95E-07	1.07E-09	M	2.03E+03	3.87E-01	2.03E+03	4.94E-04	3.50E+01
	Te-131	1.74E-02	2.22E-01	4.74E-16	38.5	2.95E-07	2.85E-11	M	7.16E+02	1.03E-02	7.16E+02	1.40E-03	NA
36	Te-132/I-132	NA	NA						6.12E+03	1.14E+00	6.12E+03	1.63E-04	6.70E+03
	Te-132	3.26E+00	1.00E+00	2.12E-16	64.6	4.30E-07	2.05E-09	M	5.38E+02	1.08E+00	5.39E+02	1.86E-03	6.60E+02
	I-132	9.58E-02	1.00E+00	2.20E-15	64.6	4.30E-07	1.14E-10	M	5.58E+03	6.01E-02	5.58E+03	1.79E-04	3.10E+03
37	Tm-170	1.29E+02	-	2.64E-17	94.9	5.68E-07	9.29E-09	S	9.84E+01	6.47E+00	1.05E+02	9.54E-03	2.40E+01
38	Xe-133	5.24E+00	-	3.95E-17	74.5	4.76E-07	0.00E+00	NA	1.16E+02	0.00E+00	1.16E+02	8.66E-03	NA
39	Xe-135	3.79E-01	-	2.50E-16	13.1	1.25E-07	0.00E+00	NA	1.29E+02	0.00E+00	1.29E+02	7.78E-03	NA
40	Xe-138	9.84E-03	-	1.07E-15	0.341	3.41E-09	0.00E+00	NA	1.43E+01	0.00E+00	1.43E+01	6.98E-02	NA
41	Y-91	5.85E+01	-	7.46E-17	93.7	5.63E-07	8.93E-09	S	2.74E+02	6.16E+00	2.81E+02	3.56E-03	1.30E+01
42	Yb-169	3.20E+01	-	2.78E-16	91.9	5.55E-07	2.98E-09	S	1.00E+03	2.03E+00	1.01E+03	9.95E-04	1.30E+03

[a]Values from Turbo FRMAC 2.0, RFC 2 (DCFPAK, K. Eckerman), CRP and KP values only apply to the early phase (first 96 h)
[b]This column compares the ICRP 60 calculated DCF that includes the dose contribution from groundshine and inhalation of resuspended material to the 1992 EPA PAG Manual DCF (ICRP 30) that includes only the dose from only groundshine.
[c]Value listed as the EPA's DRL value is not calculated by this spreadsheet, but is as listed in the 1992 EPA PAG Manual.
[d]Te-131m values in this table are subject to change pending further development of new parent-daughter rules.

2.10.1 Dose Conversion Parameter Calculation for Table 2-9

The Table 2-9 DCP ($DCP_{groundshine+inh,\ E}$) is calculated using the equation below. The DCP includes the dose from external exposure to groundshine and the inhalation of resuspended material over the early phase (i.e., 0-96 hours following release).

$$DCP_{groundshine+inh,E,i} = \sum_{i}^{P+D}\left[\left(V_g * ExDC_{ground,E,i} * CF_1 * GRF * CRP_i\right) + \left(V_g * KP_i * BR * DCF_{Inh,E,i} * CF_2\right)\right]$$

$$\frac{rem \cdot cm^3}{h \cdot \mu Ci} = \left(\frac{cm}{h} * \frac{Sv \cdot m^2}{s \cdot Bq} * \frac{rem \cdot cm2 \,/\, h \cdot \mu Ci}{Sv \cdot m^2 \,/\, s \cdot Bq} * unitless * \frac{e^{-hr/hr}}{1/h}\right) + \left(\frac{cm}{h} * \frac{h}{cm} * \frac{cm^3}{h} * \frac{Sv}{Bq} * \frac{rem/\mu Ci}{Sv/Bq}\right)$$

Where:

$\sum_{i}^{P+D}$ = Represents the summation of values from the parent radionuclide (p) and any short-lived daughter radionuclide(s) (D);

$DCP_{groundshine+inh,\ E,\ i}$ = Dose Conversion Parameter, effective, value for the dose rate per unit activity from groundshine and the inhalation of resuspended material from radionuclide *i* and any short-lived daughter radionuclide(s) over the Early Phase time period and, rem·cm^3 / h·µCi;

V_g = deposition velocity for all radionuclides, 360 cm/h (particulates are assumed), (based on Shemel, 1980 and per agreement with EPA and NRC),

$ExDC_{ground,E,i}$ = External Dose Coefficient, the effective dose rate from the external exposure to radionuclide *i* per unit activity deposited on the ground, Sv·m^2/s·Bq, (values from ICRP 60+ dosimetry models, DCFPAK, 2006); and

CF_1 = Unit Conversion Factor, converts Sv·m^2/s·Bq to rem·cm^2/h·µCi, 1.33E+14 rem·cm^2/h·µCi per Sv·m^2/s·Bq.

$$\frac{1.33E14\ rem \cdot cm^2 \,/\, h \cdot \mu Ci}{Sv \cdot m^2 \,/\, s \cdot Bq} = \frac{Sv \cdot m^2}{s \cdot Bq} * \frac{10^2\ rem}{Sv} * \frac{\left(10^2\ cm^2\right)^2}{m^2} * \frac{3.6E3s}{h} * \frac{Bq}{dps} * \frac{3.7E4\ dps}{Bq}$$

GRF = Ground Roughness Factor, a unitless constant (0.82) that compensates for the fact that the external exposure is not coming from an infinite flat plane (HPS, 2002);

CRP_i = Combined Removal Parameter, value that adjusts the external (groundshine) dose from radionuclide *i* for radioactive decay and weathering effects that decrease the groundshine dose over the early phase time period, (value from Turbo FRMAC 2.0, RFC 2, see Appendix D for details of calculation method);

KP_i = Resuspension Parameter value that adjusts the airborne radioactivity level of radionuclide *i* for radioactive decay and resuspension (value from Turbo FRMAC 2.0, RFC 2, see below for details of calculation method), h/cm,

BR = breathing rate, 9.20E+05 cm^3/h, activity-weighted, average-hourly breathing rate for an adult (Table B.16.B, ICRP 1994);

$DCF_{Inh,E,i}$ = Inhalation Dose Conversion Factor, effective, value for the effective dose per unit activity from inhalation of radionuclide *i* and for the most restrictive lung clearance class, Sv/Bq, (values from ICRP 60+ dosimetry models, DCFPAK, 2006); and

CF_2 = unit conversion factor (CF) converting Sv/Bq to rem/µCi, 3.70E+06 rem/µCi per Bq/Sv.

$$\frac{3.7E6\,rem/\mu Ci}{Sv/Bq} = \frac{Sv}{Bq} * \frac{10^2\,rem}{Sv} * \frac{Bq}{dps}\frac{3.7E4\,dps}{Bq}$$

Further calculations for calculating DCP ($DCP_{groundshine+inh,\,E}$) are provided in Appendix D of this Manual.

2.10.2 Derived Response Level Calculation for Table 2-9

The DRL is calculated using the following equation:

$$DRL_{Groundshine+Inh,E} = \sum_{i}^{P+D}\left(\frac{PAG}{DCP_{Groundshine+Inh,E,i}}\right),$$

$$\frac{\mu Ci \cdot h}{cm^3} = \frac{1\,rem}{rem \cdot cm^3/h \cdot \mu Ci}$$

Where:

$\sum_{i}^{P+D}$ = represents the summation of values from the parent radionuclide (P) and any short-lived daughter radionuclide(s) (D);

$DRL_{Groundshin,Inh,E}$ = Derived Response Level (DRL), effective, for exposure (i.e., groundshine and inhalation of resuspended material) to the radionuclide and any short-lived daughter radionuclide(s), µCi·h/cm^3;

PAG = EPA's PAG, 1 rem for effective dose or 5 rem organ (e.g., thyroid) dose; and

$DCP_{groundshine+inh,E,i}$ = Dose Conversion Parameter, effective, value for the dose rate per unit activity from groundshine and the inhalation of resuspended material from radionuclide *i* and any short-lived daughter radionuclide(s) over the Early Phase time period and, rem·cm^3 / h·µCi.

Chapter 3
Protective Action Guides for the Intermediate Phase

3.1 Introduction

Following the early phase of a radiological incident it may be necessary to temporarily relocate residents from areas where extensive deposition of radioactive materials has occurred until decontamination has taken place. This chapter identifies PAG levels of dose for the intermediate phase as well as guidance for the implementation of corresponding protective actions. The guidance provided in this chapter is intended for use by state and local officials in developing their radiological emergency response plans to protect the public from exposure to radiation from deposited radioactive materials. Due to the wide variety of types of radiological incidents and radionuclide releases that could occur, it is not practical to provide implementing guidance for all situations.

The PAG level for relocation specified in this chapter refers only to estimates of doses due to exposure during the first year after the incident. Exposure pathways include external exposure to radiation from deposited radioactivity and inhalation of resuspended materials. The 0.5 rem/y (5 mSv/y) for doses subsequent to the first year PAG level can be found in Section 3.10. Meeting the 0.5 rem annual dose after the first year may not be easy for the radionuclides typically associated with RDDs and this may drive relocating more people earlier to meet the 0.5 rem/year rather then relocating them after they have been exposed to 2 rem (20 mSv) in the first year. Protective action guidance for ingestion exposure pathways, which also apply during the intermediate phase, is discussed separately in Chapters 4 and 5.

As stated previously in Chapters 1 and 2, PAG levels are expressed in terms of the projected doses above which specified protective actions are warranted. PAGs should be considered administratively required only for use in planning, i.e., in developing radiological emergency response plans. During an incident, due to unanticipated local conditions and constraints, professional judgment by responsible officials will be required in their application. Situations can be envisaged in which contamination from a radiological incident occurs at a site or time at which relocation of the public, based on the recommended PAGs, may be impracticable. Conversely, under some conditions, relocation may be quite practicable at projected dose estimates below the PAGs. These situations require judgments by those responsible for protective action decisions at the time of the incident.

In the case of deposition of radioactive materials (considered to be the radiation exposure source most likely of concern during the intermediate phase), the major relevant protective action is relocation. Persons not relocated (i.e., those in less contaminated areas) may reduce their dose through the application of simple decontamination techniques and by spending more time than usual in low exposure rate areas (e.g., indoors). The PAG level for relocation applies to doses that can be avoided by taking action to relocate members of the public. Doses already incurred prior to evacuation are not included.

Contrary to the situation during the early phase of a radiological incident, when decisions usually must be made and implemented quickly by state and local officials before federal assistance is available, many decisions and actions during the intermediate phase can be delayed until federal resources are present, as described in the Nuclear/Radiological Incident Annex of the National Response Plan (DHS 2004).

At the time of decisions on relocation and early decontamination, sheltering-in-place and evacuation should have already been completed to protect the public from exposure to the airborne plume and from high exposure rates from deposited materials (groundshine). These protective actions may have been implemented prior to verification of the path of the plume and therefore some persons may have been evacuated from areas where actual doses are much lower than were projected. Others who were in the path of the plume may have been sheltered or not protected at all. During the intermediate phase of the response, persons must be relocated from areas where the projected dose (based on soil sample analyses and/or other more realistic calculations) exceeds the PAG for relocation, and other actions taken to reduce doses to persons who are not relocated from contaminated areas.

3.1.1 Applicability

Similar to the guidance provided for the early phase in Chapter 2, the PAGs and corresponding protective actions provided in this chapter are expected to be used for planning and development of emergency response plans, and the

exercise of those plans. They should not be misconstrued to indicate acceptable levels of exposure during normal, nonemergency conditions. For more information on the intended use of the PAGs, see Section 1.5.

3.1.2 The Intermediate Phase

The phase addressed by this chapter is denoted the "intermediate phase." This phase, for practical purposes, is defined as the period beginning after the source and releases have been brought under control and environmental measurements (e.g., aerial monitoring, field sampling) 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; however, for purposes of dose projection, it is assumed to last for 1 year. Prior to this period, protective actions will have been taken based upon the guidance for the early phase as presented in Chapter 2. It is assumed that decisions will be made during the intermediate phase concerning whether particular areas or properties from which persons have been evacuated will be decontaminated and reoccupied or condemned and the occupants permanently relocated. These actions will be carried out during the late or recovery phase, discussed in Chapter 6.

3.2 Exposure Pathways during the Intermediate Phase

During the intermediate phase, the principal exposure pathways for the public occupying areas contaminated with deposited radioactive materials are expected to be exposure of the whole body to external gamma radiation from deposited radioactive materials (groundshine) and internal exposure from the inhalation of the resuspended materials. For reactor and RDD/IND incidents, external gamma radiation is expected to be the dominant source.

Almost invariably, relocation decisions will be based on doses from the above pathways. However, in rare cases in which food or drinking water is contaminated to levels above the PAG for ingestion, and its withdrawal from use will create a risk of starvation or dehydration greater than that from the radiation dose, the dose from ingestion should be added to the dose from the above pathways. PAGs and protective actions related specifically to the withdrawal of contaminated food and water from use are discussed in Chapters 4 and 5.

Other potentially significant exposure pathways include exposure to beta radiation from surface contamination and direct ingestion of contaminated soil. These pathways are not expected to be controlling for NPP incidents (Aaberg 1989).

3.3 The PAGs and Protective Actions for the Intermediate Phase: Relocation and Dose Reduction

The principal protective actions for reducing exposure of the public to deposited radioactive materials are relocation, decontamination, shielding, time limits on exposure, and control of the spread of surface contamination. Relocation is the most effective, and, usually, the most disruptive. It is therefore only applied when the dose is sufficiently high to warrant it. The other protective actions are generally applied to reduce exposure of persons who are not relocated, or who return from evacuation status to areas that received lower levels of deposited radioactivity. This chapter provides guidance for translating radiological conditions in the environment into projected dose that provides the basis for decisions on the appropriate protective actions.

PAGs for protection from deposited radioactivity during the intermediate phase are summarized in Table 3-1. The basis for these values is presented in detail in Appendix E. In summary, relocation is warranted when the projected sum of the dose equivalent from external gamma radiation and the committed effective dose equivalent from inhalation of resuspended radionuclides exceeds 2 rem (20 mSv) in the first year.

In addition to dose reduction provided by partial occupancy in homes and other structures, people who are not relocated can reduce their doses by the application of various techniques. Dose reduction efforts can range from the simple processes of scrubbing and/or flushing surfaces, removal and disposal of small spots of soil found to be highly contaminated (e.g., from settlement of water), and spending more time than usual in lower exposure rate areas (e.g., indoors), to the difficult and time consuming processes of removal, disposal, and replacement of contaminated surfaces. It is anticipated that simple processes would be most appropriate to reduce exposure rates for persons living in contaminated areas outside the relocation area. Many of these can be carried out by the residents with support from officials for monitoring, guidance on appropriate actions, and disposal. The more difficult processes will usually be appropriate for recovery of areas from which the population is relocated.

Large areas may have to be restricted under these PAGs. As the affected land area that would have to be restricted under this PAG increases, protective actions will become more difficult and costly to implement, especially in densely populated areas. For situations in which implementation becomes impracticable or impossible (e.g., large city), informed judgment must be exercised to assure priority of protection for individuals in areas having the highest exposure rates.

Table 3-1. Protective Action Guides for Exposure to Deposited Radioactivity during the Intermediate Phase of a Radiological Incident

Protective Action Recommendation	PAG (First-Year Projected Dose)[a]	Comments
Relocate the general population.[b]	≥ 2 rem	
Apply simple dose reduction techniques.[c]	<2 rem	These protective actions should be taken to reduce doses to as low as practicable levels.

[a]TEDE (TEDE - the projected sum of the effective dose equivalent from external radiation exposure (i.e., groundshine) and the committed effective dose equivalent from inhaled radioactive material). Projected dose refers to the dose that would be received in the absence of shielding from structures or the application of dose reduction techniques. These PAGs may not provide adequate protection from some long-lived radionuclides (see Section 3.8).
[b]People previously evacuated from areas outside the relocation zone defined by this PAG may return to occupy their residences. Cases involving relocation of persons at high risk from such action (e.g., patients under intensive care) may be evaluated individually.
[c]Simple dose reduction techniques include scrubbing and/or flushing hard surfaces, minor removal of soil from spots where radioactive materials have concentrated and spending more time than usual indoors or in other low exposure rate areas.

3.3.1 The Population Affected

The PAG for relocation are intended for use in establishing the boundary of a relocation area within an area that has been subjected to deposition of radioactive materials. During their development, consideration was given to the higher risk of effects on health to children and fetuses from radiation dose and the higher risk to some other population groups for relocation. To avoid the complexity of implementing separate protective actions for individual members of the population, the relocation PAG is established at a level that will provide adequate protection for the general population.

Contamination below the PAG may extend beyond the relocation area. Monitoring and simple dose reduction efforts are recommended in this area to reduce doses to the extent practical. Such actions (whether performed by authorities or by members of the public themselves) are a good practice and are consistent with the principle of reducing radiation exposure. Such actions are unlikely to be practical in areas where projected avoided dose is less than 10%. Persons residing in contaminated areas outside the relocation area will be at some risk from radiation dose. Therefore, guidance on the reduction of dose during the first year to residents outside this zone is also provided. Due to the high cost of relocation, it is more practical to reduce dose in this population group by early application of simple, low-impact protective actions other than by relocation.

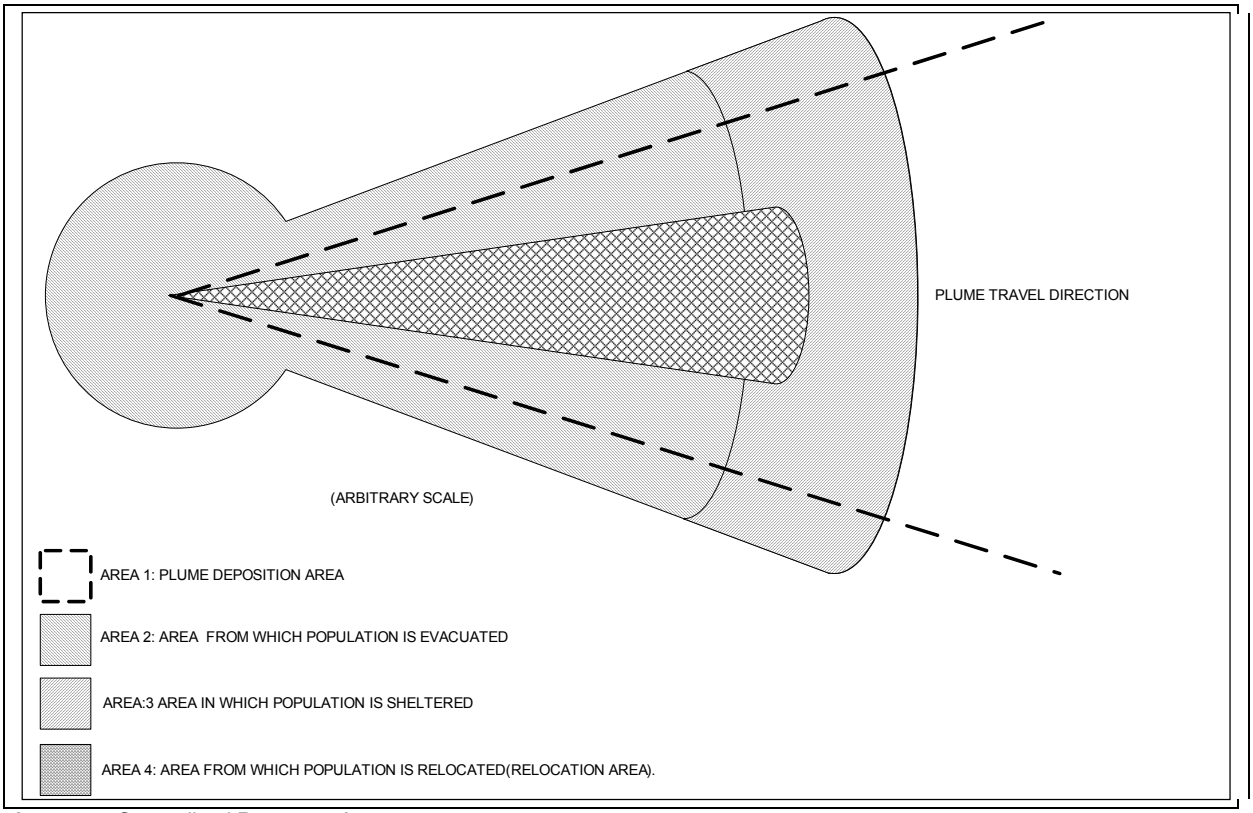

Figure 3-1. Generalized Response Areas

3.3.2 Areas Involved

Figure 3-1 provides a generalized example of the different areas[19] and population groups to be dealt with. The plume deposition area is assumed to be represented by Area 1. In reality, variations in meteorological conditions would almost certainly produce a more complicated shape, but the same principles would apply.

In situations in which notification is available prior to a release of radioactive materials, such as an NPP accident, persons will already have been evacuated from Area 2 and sheltered in Area 3. Persons who have been evacuated from or sheltered in Areas 2 and 3, respectively, as precautionary actions for protection from the plume, but whose homes are outside the plume deposition area (Area 1), may return to their homes as soon as environmental monitoring verifies the boundary of the area that received deposition (Area 1).

Area 4 is designated a relocation area and is defined as the area where projected doses are equal to or greater than the relocation PAG. Persons residing just outside the boundary of the relocation area may receive a dose near the PAG for relocation if decontamination or other dose reduction efforts are not implemented.

Area 1, with the exception of the relocation area, represents the area of contamination that may continue to be occupied by the general public during the intermediate phase. Nevertheless, there will be contamination levels in this area that will require continued monitoring and dose reduction efforts other than relocation.

The relative positions of the boundaries shown in Figure 3-1 depend on areas evacuated and sheltered and the radiological and meteorological characteristics of the release. For example, Area 4 (the relocation area) could fall entirely inside Area 2 (area evacuated), so that the only persons to be relocated would be those residing in Area 4 who were either missed in the evacuation process or who, because of the high risk for their evacuation, had remained sheltered during plume passage.

[19] Emergency planning zones do not apply to naval nuclear-powered warships.

3.3.3 Priorities

In most cases protective actions during the intermediate phase will be carried out over a relatively long period of time (e.g., months). It is therefore useful to consider what priorities are appropriate. Further, for situations where the affected area is so large that it is impractical to relocate all of the public, especially from areas exceeding the PAGs by only a small amount, priorities are needed for protective actions. The following priorities are appropriate:

- As a first priority, protect all persons from doses that could cause acute health effects from all exposure pathways, including previous exposure to the plume.

- Recommend the application of simple decontamination techniques and that persons remain indoors as much as possible to reduce exposure rates.

3.4 Sequence of Events

Following passage of the airborne plume, several tasks, as shown in Figure 3-2, must be accomplished simultaneously to provide for timely protection of the public. The decisions on the early task of relocating persons from high exposure rate areas must be based on exposure rate measurements and dose analyses. It is expected that monitoring and dose assessment will be an ongoing process, with priority given to the areas with the highest exposure rates. The general sequence of events is itemized below, but the time frames will overlap, as demonstrated in Figure 3-2.

1. Based on environmental data, determine the areas where the projected 1-year dose will exceed 2 rem (20 mSv) and relocate persons from those areas, with priority given for persons in the highest exposure rate areas.

2. Allow previously evacuated persons to return as quickly as possible to their residences if they were evacuated from areas where field gamma measurements indicate that exposure rates are near normal background levels (not in excess of twice (2x) the normal background in the area before the incident). If, however, areas of high deposition are found to be near areas with low deposition such that resuspended activity could drift into the occupied areas, establish a buffer zone to restrict residential use until the situation is analyzed and dose projections are confirmed to validate that the buffer zone is no longer needed.

3. Determine the location of the isodose-rate line (Section 3.4.1) corresponding to the relocation PAG, establish the boundary of the relocation area, and relocate any persons who still reside within the zone. Also, assign any evacuees who reside within the relocation area to relocation status. Evacuated persons whose residence is in the area between the boundary of the plume deposition and the boundary to the relocation area may return gradually as confidence is gained to do so through dose projections for the area.

4. Evaluate the dose reduction effectiveness of simple decontamination techniques and of sheltering-in-place in response to exposure due to partial occupancy of residences and workplaces. Results of these evaluations may influence recommendations for reducing exposure rates for persons who are not relocated from areas near, but outside, the relocation area.

5. Establish a mechanism for controlling access to and egress from the relocation area. Typically this would be accomplished through control points at roadway accesses to the relocation area.

6. Establish monitoring and decontamination stations to support control of the relocation area.

7. Implement simple decontamination techniques in contaminated areas outside the relocation area, with priorities for areas with higher exposure rates.

8. Collect data needed to establish long-term radiation protection criteria for recovery and data to determine the effectiveness of various decontamination or other recovery techniques.

9. Begin operations to recover contaminated property in the relocation area.

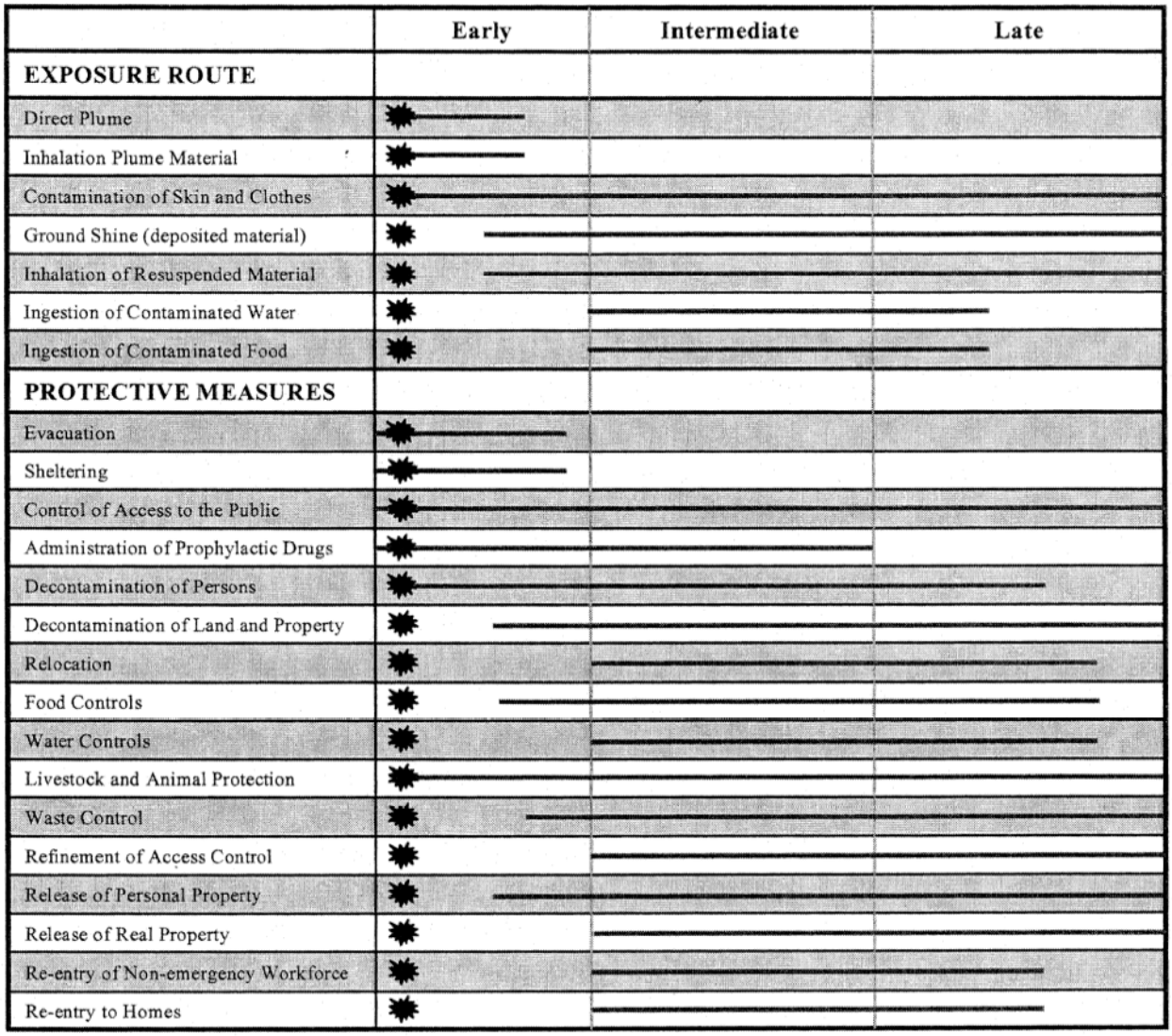

Figure 3-2. Potential Time Frame of Response to a Radiological Incident (DHS 2006).

3.4.1 Establishment of Isodose-Rate Lines

As soon as federal or other assistance is available for aerial and ground monitoring, a concentrated effort should begin to establish isodose-rate lines on maps and the identification of boundaries to the relocation area. Planning for this effort should include the development of standard maps that can be used by all of the involved monitoring and dose assessment organizations to record monitoring data.

Aerial monitoring (e.g., the Department of Energy Aerial Measuring System (AMS)) can be used to collect data for establishing general patterns of radiation exposure rates from deposited radioactive materials and may form the primary basis for the development of dose lines out to the limit of aerial detectability. Initially during the early phase, the detectability is limited to just exposure rate changes of a few times natural background levels. However, later during the intermediate phase more sensitive measurements detect levels of radioactivity contributing just a small fraction to the natural background. Periodic air sample measurements will also be needed to verify the contribution to dose from inhalation of resuspended materials.

Gamma exposure rates measured at 1 m (approximately 3.28 feet) will no doubt vary as a function of the location of the measurement within a very small area. This could be caused by different deposition rates for different types of surfaces (e.g., smooth surfaces versus heavy vegetation). Rinsing or precipitation could also reduce levels in some areas and raise levels in others where runoff settles. In general, where exposure rates vary within designated areas, dose projections for people residing within these areas should be estimated using an appropriate average exposure rate.

Measurements made at 1 m to project whole-body dose from gamma radiation should be made with instruments of the "closed window" type so as to avoid the detection of beta radiation. Although beta exposure will contribute to skin dose, its contribution to the overall risk of health effects from the radionuclides expected to be associated with reactor incidents should not be controlling in comparison to the whole body gamma dose (Aaberg 1989).

3.4.2 Dose Projection

The primary dose of interest for reactor incidents is the sum of the effective dose equivalent from external exposure and the committed effective dose equivalent from inhalation. The exposure periods of interest are the first year and second year after the incident (see Section D.6). However, dispersals of alpha-emitting material must be monitored carefully. Examples include nuclear weapons incidents or alpha RDDs. In these cases inhalation of resuspended material is likely to be the dominant dose pathway. It is possible that there will be little or no associate gamma radiation or beta activity. Not only are alpha survey instruments required but use of proper measurement techniques is critical.

Calculation of the projected gamma dose from measurements will require knowledge of the principal radionuclides contributing to exposure and their relative abundances. Information on these radiological characteristics can be compiled either through the use of portable gamma spectrometers or by radionuclide analysis of environmental samples. Several measurement locations may be required to determine whether any selective radionuclide deposition occurred as a function of meteorology, surface type, distance from the point of release, or other factors. In accordance with the Nuclear/Radiological Incident Annex to the National Response Plan (DHS 2004), DOE, EPA, and other federal agencies have the capability to assist state officials in performing environmental measurements, including determination of radiological characteristics of deposited materials.

The gamma exposure rate may decrease rapidly if deposited materials include a significant fraction of short-lived radionuclides. Therefore, the relationship between instantaneous exposure rate and projected first- and second-year annual doses will change as a function of time and these relationships must be established for the particular mix of deposited radioactive materials present at the time of the gamma exposure rate measurement. Over time, residual doses resulting from an RDD or IND will depend largely upon the half-lives of the radionuclides involved and could potentially remain significant over many years. It should be noted that natural attenuation as well as nuclear decay can affect long term dose assessments.

For incidents involving releases from NPPs, gamma radiation from deposited radioactive materials is expected to be the principal exposure pathway, as noted above. Other pathways should also be evaluated, and their contributions considered, if significant. For example, any time alpha emitting radionuclides are involved the inhalation of resuspended material must be considered. Similarly, the skin beta dose may be important for particulates deposited or transferred to the skin, as may be the case for an RDD comprised of Sr-90. Sections 3.4.3 and 3.7 provide methods for evaluating the projected dose from whole body external exposure (3.4.3) and from inhalation of resuspended particulate material (3.7), based on environmental information.

Exposure from ingestion of food and water is normally considered independently of decisions for relocation and decontamination (see Chapters 4 and 5). In rare instances, however, where withdrawal of food and/or water from use would, in itself, create a health risk, relocation may be an appropriate protective action for protection from exposure via ingestion. In this case, the committed effective dose equivalent from ingestion should be added to the projected dose from other exposure pathways for decisions on relocation.

3.4.3 Projected External Gamma Dose

Projected whole body external gamma doses at 1 m height at particular locations during the first year and second year after the incident are the parameters of interest. The environmental information available for calculating these

doses is expected to be the current gamma exposure rate at 1 m height and the relative abundance of each radionuclide contributing significantly to that exposure rate. Calculation models are available for predicting future exposure rates as a function of time with consideration of radioactive decay and weathering. The weathering model was developed using data from the Chernobyl nuclear power plant accident (HPS 2002); the gamma exposure rate at 1 m is addressed in reference DOE 1988.

Following the incident, measurements should be conducted to determine the dose reduction factors associated with simple, rapid, decontamination techniques so that these factors can also be applied to calculating dose to persons who are not relocated. However, these factors should not be included in calculating projected dose for decisions on relocation.

Relocation decisions can generally be made on the basis of the year projected dose. However, projected doses during the second year are needed for decisions on the need for other protective actions for persons who are not relocated. DCFs are therefore needed for converting environmental measurements to projected dose during the first year and second year following the incident. Of the two types of environmental measurements that can be made to project whole body external gamma dose, gamma exposure rate in air is the easiest to make and is the most directly linked to gamma dose rate. However, analyses of a few environmental samples (particularly, soil samples) must be coupled with gamma exposure rate to properly project decreasing dose rates.

Table 3-3. Total Dose Parameters for Surface Deposition Corrected for Radioactive Decay and Weathering

No.	Radio-Nuclide	[a]Half-Life (d)	Branch Fraction	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExDC (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExXC (mR/hr per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExDF (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExXF (mR/h per pCi/m^2)	Early Phase [a,d]DRL (µCi/m^2)	Early Phase [b]TDP_Dp (mrem per pCi/m^2)	Year One [a,d]DRL (µCi/m^2)	Year One [b]TDP_Dp (mrem per pCi/m^2)	Year Two [a,d]DRL (µCi/m^2)	Year Two [b]TDP_Dp (mrem per pCi/m^2)
1	Am-241	1.58E+05	-	3.10E-10	4.43E-10	2.54E-10	3.63E-10	5.29E+01	1.89E-05	3.53E+01	5.67E-05	6.91E+01	7.24E-06
2	Ba-140/La-140	NA	NA	3.13E-08	4.48E-08	2.57E-08	3.67E-08	4.52E+02	2.21E-06	1.78E+02	1.12E-05	2.22E+10	2.25E-14
	Ba-140	1.27E+01	1.00E+00	2.53E-09	3.61E-09	2.07E-09	2.96E-09	-	-	-	-	-	-
	La-140	1.68E+00	1.00E+00	2.88E-08	4.11E-08	2.36E-08	3.37E-08	see La-140 listed separately (as parent) below					
3	Ce144/Pr144/Pr144m	NA	NA	2.42E-09	3.45E-09	1.98E-09	2.83E-09	5.01E+03	2.00E-07	1.87E+02	1.07E-05	1.33E+02	3.77E-06
	Ce-144	2.84E+02	1.00E+00	2.45E-10	3.50E-10	2.01E-10	2.87E-10	-	-	-	-	-	-
	Pr-144	1.20E-02	9.82E-01	2.17E-09	3.10E-09	1.78E-09	2.54E-09	-	-	-	-	-	-
	Pr-144m	5.00E-03	1.78E-02	1.40E-10	2.00E-10	1.15E-10	1.64E-10	-	-	-	-	-	-
	Pr-144	1.20E-02	9.99E-01	2.17E-09	3.10E-09	1.78E-09	2.54E-09	-	-	-	-	-	-
4	Cf-252	9.64E+02	-	6.98E-12	9.97E-12	5.72E-12	8.18E-12	1.38E+02	7.25E-06	9.85E+01	2.03E-05	3.39E+02	1.47E-06
5	Cm-244	6.61E+03	-	8.58E-12	1.23E-11	7.04E-12	1.01E-11	8.95E+01	1.12E-05	6.22E+01	3.22E-05	1.60E+02	3.13E-06
6	Co-60	1.93E+03	-	3.06E-08	4.37E-08	2.51E-08	3.58E-08	4.14E+02	2.42E-06	1.06E+01	1.89E-04	3.49E+00	1.43E-04
7	Cs-134	7.53E+02	-	1.97E-08	2.81E-08	1.62E-08	2.31E-08	6.44E+02	1.55E-06	1.81E+01	1.10E-04	7.34E+00	6.81E-05
8	Cs-136	1.31E+01	-	2.70E-08	3.86E-08	2.21E-08	3.16E-08	5.22E+02	1.92E-06	2.01E+02	9.95E-06	1.45E+10	3.45E-14
9	Cs-137/Ba-137m	NA	NA	7.33E-09	1.05E-08	6.01E-09	8.59E-09	1.71E+03	5.85E-07	4.19E+01	4.77E-05	1.24E+01	4.03E-05
	Cs-137	1.10E+04	1.00E+00	3.98E-11	5.69E-11	3.26E-11	4.66E-11	-	-	-	-	-	-
	Ba-137m	1.77E-03	9.46E-01	7.71E-09	1.10E-08	6.32E-09	9.03E-09	-	-	-	-	-	-
10	Gd-153	2.42E+02	-	1.23E-09	1.76E-09	1.01E-09	1.44E-09	1.04E+04	9.62E-08	3.93E+02	5.09E-06	3.25E+02	1.54E-06
11	I-131	8.04E+00	-	4.85E-09	6.93E-09	3.98E-09	5.68E-09	3.09E+03	3.24E-07	1.81E+03	1.10E-06	2.49E+16	2.01E-20
12	I-132	9.58E-02	-	2.93E-08	4.19E-08	2.40E-08	3.43E-08	1.25E+04	8.00E-08	2.51E+04	7.97E-08	NA	NA
13	I-133	8.67E-01	-	8.22E-09	1.17E-08	6.74E-09	9.63E-09	5.15E+03	1.94E-07	9.89E+03	2.02E-07	1.76E+130	2.84E-134
14	I-134	3.65E-02	-	3.37E-08	4.81E-08	2.76E-08	3.95E-08	2.86E+04	3.50E-08	5.72E+04	3.50E-08	NA	NA
15	I-135/Xe135m	NA	NA	2.05E-08	2.92E-08	1.68E-08	2.40E-08	6.26E+03	1.60E-07	1.25E+04	1.60E-07	NA	NA
	I-135	2.75E-01	1.00E+00	1.96E-08	2.80E-08	1.61E-08	2.30E-08	-	-	-	-	-	-
	Xe-135m	1.06E-02	1.54E-01	5.58E-09	7.97E-09	4.58E-09	6.54E-09	-	-	-	-	-	-

Table 3-3. Total Dose Parameters for Surface Deposition Corrected for Radioactive Decay and Weathering (continued)

No.	Radio-Nuclide	[a]Half-Life (d)	Branch Fraction	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExDC (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExXC (mR/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExDF (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExXF (mR/h per pCi/m^2)	Early Phase [a,d]DRL (µCi/m^2)	Early Phase [b]TDP_Dp (mrem per pCi/m^2)	Year One [a,d]DRL (µCi/m^2)	Year One [b]TDP_Dp (mrem per pCi/m^2)	Year Two [a,d]DRL (µCi/m^2)	Year Two [b]TDP_Dp (mrem per pCi/m^2)
16	Ir-192	7.40E+01	-	1.04E-08	1.49E-08	8.53E-09	1.22E-08	1.25E+03	8.00E-07	9.95E+01	2.01E-05	8.90E+02	5.62E-07
17	Kr-87	5.30E-02	-	1.12E-08	1.60E-08	9.18E-09	1.31E-08	5.94E+04	1.68E-08	1.19E+05	1.68E-08	NA	NA
18	Kr-88/Rb-88	NA	NA	3.29E-08	4.70E-08	2.70E-08	3.85E-08	9.04E+03	1.11E-07	1.81E+04	1.10E-07	NA	NA
	Kr-88	1.18E-01	1.00E+00	2.30E-08	3.29E-08	1.89E-08	2.69E-08	-	-	-	-	-	-
	Rb-88	1.24E-02	1.00E+00	9.87E-09	1.41E-08	8.09E-09	1.16E-08	-	-	-	-	-	-
19	La-140	1.68E+00	-	2.88E-08	4.11E-08	2.36E-08	3.37E-08	9.03E+02	1.11E-06	1.46E+03	1.37E-06	1.31E+68	3.82E-72
20	Mo-99/Tc-99m	NA	NA	3.70E-09	5.29E-09	3.04E-09	4.34E-09	5.45E+03	1.83E-07	6.93E+03	2.89E-07	1.85E+43	2.70E-47
	Mo-99	2.75E+00	1.00E+00	2.37E-09	3.39E-09	1.94E-09	2.78E-09	-	-	-	-	-	-
	Tc-99m	2.51E-01	8.76E-01	1.52E-09	2.17E-09	1.25E-09	1.78E-09	9.25E+04	1.08E-08	1.85E+05	1.08E-08	NA	NA
21	Np-239	2.36E+00	-	2.05E-09	2.93E-09	1.68E-09	2.40E-09	1.05E+04	9.52E-08	1.46E+04	1.37E-07	1.96E+50	2.55E-54
22	Pm-147	9.58E+02	-	3.73E-13	5.33E-13	3.06E-13	4.37E-13	7.17E+05	1.39E-09	3.34E+05	5.99E-09	2.93E+05	1.71E-09
23	Pu-238	3.20E+04	-	8.34E-12	1.19E-11	6.84E-12	9.77E-12	4.73E+01	2.11E-05	3.27E+01	6.12E-05	8.16E+01	6.13E-06
24	Pu-239	8.79E+06	-	3.78E-12	5.40E-12	3.10E-12	4.43E-12	4.29E+01	2.33E-05	2.97E+01	6.73E-05	7.36E+01	6.79E-06
25	Ra-226/Rn-222…	NA	NA	2.25E-08	3.21E-08	1.84E-08	2.63E-08	2.76E+02	3.62E-06	1.30E+01	1.53E-04	3.91E+00	1.28E-04
	Ra-226	5.84E+05	1.00E+00	8.14E-11	1.16E-10	6.67E-11	9.54E-11	-	-	-	-	-	-
	Rn-222	3.82E+00	1.00E+00	5.09E-12	7.27E-12	4.17E-12	5.96E-12	-	-	-	-	-	-
	Po-218	2.12E-03	1.00E+00	1.15E-13	1.64E-13	9.43E-14	1.35E-13	-	-	-	-	-	-
	Pb-214	1.86E-02	1.00E+00	3.20E-09	4.57E-09	2.62E-09	3.75E-09	-	-	-	-	-	-
	Bi-214	1.38E-02	1.00E+00	1.92E-08	2.74E-08	1.57E-08	2.25E-08	-	-	-	-	-	-
	Po-214	1.90E-09	1.00E+00	1.06E-12	1.51E-12	8.69E-13	1.24E-12	-	-	-	-	-	-
	At-218	2.31E-05	2.00E-04	4.85E-11	6.93E-11	3.98E-11	5.68E-11	-	-	-	-	-	-
	Bi-214	1.38E-02	1.00E+00	1.92E-08	2.74E-08	1.57E-08	2.25E-08	-	-	-	-	-	-
	Po-214	1.90E-09	1.00E+00	1.06E-12	1.51E-12	8.69E-13	1.24E-12	-	-	-	-	-	-
26	Ru-103/Rh-103m	NA	NA	5.99E-09	8.56E-09	4.91E-09	7.02E-09	2.19E+03	4.57E-07	3.08E+02	6.49E-06	5.69E+04	8.79E-09
	Ru-103	3.93E+01	1.00E+00	5.98E-09	8.54E-09	4.90E-09	7.01E-09	-	-	-	-	-	-

Table 3-3. Total Dose Parameters for Surface Deposition Corrected for Radioactive Decay and Weathering (continued)

No.	Radio-Nuclide	[a]Half-Life (d)	Branch Fraction	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExDC (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExXC (mR/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExDF (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExXF (mR/h per pCi/m^2)	Early Phase [a,d]DRL (µCi/m^2)	Early Phase [b]TDP_Dp (mrem per pCi/m^2)	Year One [a,d]DRL (µCi/m^2)	Year One [b]TDP_Dp (mrem per pCi/m^2)	Year Two [a,d]DRL (µCi/m^2)	Year Two [b]TDP_Dp (mrem per pCi/m^2)
	Rh-103m	3.90E-02	9.97E-01	1.18E-11	1.69E-11	9.68E-12	1.38E-11	-	-	-	-	-	-
27	Ru-106/Rh-106	NA	NA	4.60E-09	6.57E-09	3.77E-09	5.39E-09	2.68E+03	3.73E-07	9.04E+01	2.21E-05	5.23E+01	9.56E-06
	Ru-106	3.68E+02	1.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	-	-	-	-	-	-
	Rh-106	3.46E-04	1.00E+00	4.60E-09	6.57E-09	3.77E-09	5.39E-09	-	-	-	-	-	-
28	Sb-127/Te-127	NA	NA	9.12E-09	1.30E-08	7.48E-09	1.07E-08	1.95E+03	5.13E-07	2.01E+03	9.95E-07	2.06E+31	2.43E-35
	Sb-127	3.85E+00	1.00E+00	9.01E-09	1.29E-08	7.39E-09	1.06E-08	-	-	-	-	-	-
	Te-127	3.90E-01	8.24E-01	1.37E-10	1.96E-10	1.12E-10	1.60E-10	-	-	-	-	-	-
29	Sb-129/Te-129	NA	NA	1.95E-08	2.78E-08	1.60E-08	2.28E-08	1.01E+04	9.90E-08	2.01E+04	9.95E-08	NA	NA
	Sb-129	1.80E-01	1.00E+00	1.83E-08	2.61E-08	1.50E-08	2.14E-08	-	-	-	-	-	-
	Te-129	4.83E-02	7.75E-01	1.52E-09	2.17E-09	1.25E-09	1.78E-09	-	-	-	-	-	-
30	Se-75	1.20E+02	-	4.81E-09	6.87E-09	3.94E-09	5.63E-09	2.67E+03	3.75E-07	1.48E+02	1.35E-05	3.56E+02	1.40E-06
31	Sr-89	5.05E+01	-	9.14E-10	1.31E-09	7.49E-10	1.07E-09	1.40E+04	7.14E-08	1.59E+03	1.26E-06	7.00E+04	7.14E-09
32	Sr-90/Y-90	NA	NA	1.49E-09	2.13E-09	1.22E-09	1.75E-09	6.76E+03	1.48E-07	2.05E+02	9.76E-06	6.13E+01	8.16E-06
	Sr-90	1.06E+04	1.00E+00	2.19E-11	3.13E-11	1.80E-11	2.57E-11	-	-	-	-	-	-
	Y-90	2.67E+00	1.00E+00	1.47E-09	2.10E-09	1.21E-09	1.72E-09	1.39E+04	7.19E-08	1.80E+04	1.11E-07	8.54E+44	5.85E-49
33	Sr-91/Y-91m	NA	NA	1.36E-08	1.94E-08	1.12E-08	1.59E-08	6.54E+03	1.53E-07	1.31E+04	1.53E-07	NA	NA
	Sr-91	3.96E-01	1.00E+00	9.69E-09	1.38E-08	7.95E-09	1.14E-08	-	-	-	-	-	-
	Y-91m	3.45E-02	5.78E-01	6.79E-09	9.70E-09	5.57E-09	7.95E-09	-	-	-	-	-	-
34	Te-129m/Te-129	NA	NA	1.75E-09	2.50E-09	1.43E-09	2.05E-09	7.50E+03	1.33E-07	1.23E+03	1.63E-06	6.75E+05	7.41E-10
	Te-129m	3.36E+01	1.00E+00	7.59E-10	1.08E-09	6.22E-10	8.89E-10	-	-	-	-	-	-
	Te-129	4.83E-02	6.50E-01	1.52E-09	2.17E-09	1.25E-09	1.78E-09	4.80E+05	2.08E-09	9.59E+05	2.09E-09	NA	NA
35	[d]Te-131m/Te-131	NA	NA	1.93E-08	2.76E-08	1.58E-08	2.26E-08	1.64E+03	6.10E-07	2.93E+03	6.83E-07	6.90E+90	7.25E-95
	[d]Te-131m	1.25E+00	1.00E+00	1.79E-08	2.56E-08	1.47E-08	2.10E-08	-	-	-	-	-	-
	Te-131	1.74E-02	2.22E-01	6.32E-09	9.03E-09	5.18E-09	7.40E-09	3.21E+05	3.12E-09	6.42E+05	3.12E-09	NA	NA
36	Te-132/I-132	NA	NA	3.21E-08	4.59E-08	2.63E-08	3.76E-08	5.87E+02	1.70E-06	6.74E+02	2.97E-06	1.05E+36	4.76E-40

Table 3-3. Total Dose Parameters for Surface Deposition Corrected for Radioactive Decay and Weathering (continued)

				[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF		[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF		Early Phase		Year One		Year Two	
No.	Radio-Nuclide	[a]Half-Life (d)	Branch Fraction	ExDC (mrem/h per pCi/m^2)	ExXC (mR/h per pCi/m^2)	ExDF (mrem/h per pCi/m^2)	ExXF (mR/h per pCi/m^2)	[a,d]DRL (µCi/m^2)	[b]TDP_Dp (mrem per pCi/m^2)	[a,d]DRL (µCi/m^2)	[b]TDP_Dp (mrem per pCi/m^2)	[a,d]DRL (µCi/m^2)	[b]TDP_Dp (mrem per pCi/m^2)
	Te-132	3.26E+00	1.00E+00	2.82E-09	4.03E-09	2.31E-09	3.30E-09	-	-	-	-	-	-
	I-132	9.58E-02	1.00E+00	2.93E-08	4.19E-08	2.40E-08	3.43E-08	-	-	-	-	-	-
37	Tm-170	1.29E+02	-	3.52E-10	5.03E-10	2.89E-10	4.12E-10	3.43E+04	2.92E-08	1.92E+03	1.04E-06	4.02E+03	1.24E-07
38	Xe-133	5.24E+00	-	5.26E-10	7.51E-10	4.31E-10	6.16E-10	3.11E+04	3.22E-08	2.56E+04	7.81E-08	6.74E+24	7.42E-29
39	Xe-135	3.79E-01	-	3.33E-09	4.76E-09	2.73E-09	3.90E-09	2.79E+04	3.58E-08	5.59E+04	3.58E-08	NA	NA
40	Xe-138	9.84E-03	-	1.43E-08	2.04E-08	1.17E-08	1.68E-08	2.51E+05	3.98E-09	5.02E+05	3.98E-09	NA	NA
41	Y-91	5.85E+01	-	9.94E-10	1.42E-09	8.15E-10	1.16E-09	1.28E+04	7.81E-08	1.27E+03	1.57E-06	2.83E+04	1.77E-08
42	Yb-169	3.20E+01	-	3.70E-09	5.29E-09	3.03E-09	4.33E-09	3.58E+03	2.79E-07	6.08E+02	3.29E-06	4.85E+05	1.03E-09

[a]Values from Turbo FRMAC 2.0, RFC 2, based on ICRP 60+ dosimetry model (DCFPAK, K. Eckerman).
[b]TDP (Total Dose Parameter), includes Equivalent Dose from groundshine and Committed Effective Dose from inhalation of resuspended material, corrected for radioactive decay and weathering.
[c]Values from Turbo FRMAC 2.0, RFC 2, based on ICRP 60+ dosimetry model.
[d]Te-131m values in this table are subject to change pending further development of new parent-daughter rules.

Table 3-4. Total Dose Parameters Corrected for Radioactive Decay (Not Corrected for Weathering)

No.	Radio-Nuclide	[a]Half-Life (d)	[a]Branch Fraction	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExDC (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExXC (mR/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExDF (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExXF (mR/h per pCi/m^2)	Early Phase [a,d]DRL (µCi/m^2)	Early Phase [b]TDP_Dp (mrem per pCi/m^2)	Year One [a,d]DRL (µCi/m^2)	Year One [b]TDP_Dp (mrem per pCi/m^2)	Year Two [a,d]DRL (µCi/m^2)	Year Two [b]TDP_Dp (mrem per pCi/m^2)
1	Am-241	1.58E+05	-	3.10E-10	4.43E-10	2.54E-10	3.63E-10	5.29E+01	1.89E-05	3.52E+01	5.68E-05	6.49E+01	7.70E-06
2	Ba-140/La-140	NA	NA	3.13E-08	4.48E-08	2.57E-08	3.67E-08	4.51E+02	2.22E-06	1.77E+02	1.13E-05	1.86E+10	2.69E-14
	Ba-140	1.27E+01	1.00E+00	2.53E-09	3.61E-09	2.07E-09	2.96E-09	-	-	-	-	-	-
	La-140	1.68E+00	1.00E+00	2.88E-08	4.11E-08	2.36E-08	3.37E-08	see La-140 listed separately (as parent) below					
3	Ce144/Pr144/Pr144m	NA	NA	2.42E-09	3.45E-09	1.98E-09	2.83E-09	5.01E+03	2.00E-07	1.73E+02	1.15E-05	1.06E+02	4.73E-06
	Ce-144	2.84E+02	1.00E+00	2.45E-10	3.50E-10	2.01E-10	2.87E-10	-	-	-	-	-	-
	Pr-144	1.20E-02	9.82E-01	2.17E-09	3.10E-09	1.78E-09	2.54E-09	-	-	-	-	-	-
	Pr-144m	5.00E-03	1.78E-02	1.40E-10	2.00E-10	1.15E-10	1.64E-10	-	-	-	-	-	-
	Pr-144	1.20E-02	9.99E-01	2.17E-09	3.10E-09	1.78E-09	2.54E-09	-	-	-	-	-	-
4	Cf-252	9.64E+02	-	6.98E-12	9.97E-12	5.72E-12	8.18E-12	1.38E+02	7.25E-06	9.85E+01	2.03E-05	3.37E+02	1.48E-06
5	Cm-244	6.61E+03	-	8.58E-12	1.23E-11	7.04E-12	1.01E-11	8.95E+01	1.12E-05	6.22E+01	3.22E-05	1.60E+02	3.13E-06
6	Co-60	1.93E+03	-	3.06E-08	4.37E-08	2.51E-08	3.58E-08	4.14E+02	2.42E-06	9.70E+00	2.06E-04	2.76E+00	1.81E-04
7	Cs-134	7.53E+02	-	1.97E-08	2.81E-08	1.62E-08	2.31E-08	6.44E+02	1.55E-06	1.66E+01	1.20E-04	5.82E+00	8.59E-05
8	Cs-136	1.31E+01	-	2.70E-08	3.86E-08	2.21E-08	3.16E-08	5.21E+02	1.92E-06	1.99E+02	1.01E-05	1.21E+10	4.13E-14
9	Cs-137/Ba-137m	NA	NA	7.33E-09	1.05E-08	6.01E-09	8.59E-09	1.71E+03	5.85E-07	3.84E+01	5.21E-05	9.82E+00	5.09E-05
	Cs-137	1.10E+04	1.00E+00	3.98E-11	5.69E-11	3.26E-11	4.66E-11	-	-	-	-	-	-
	Ba-137m	1.77E-03	9.46E-01	7.71E-09	1.10E-08	6.32E-09	9.03E-09	-	-	-	-	-	-
10	Gd-153	2.42E+02	-	1.23E-09	1.76E-09	1.01E-09	1.44E-09	1.04E+04	9.62E-08	3.65E+02	5.48E-06	2.60E+02	1.92E-06
11	I-131	8.04E+00	-	4.85E-09	6.93E-09	3.98E-09	5.68E-09	3.08E+03	3.25E-07	1.80E+03	1.11E-06	2.09E+16	2.39E-20
12	I-132	9.58E-02	-	2.93E-08	4.19E-08	2.40E-08	3.43E-08	1.25E+04	8.00E-08	2.51E+04	7.97E-08	NA	NA
13	I-133	8.67E-01	-	8.22E-09	1.17E-08	6.74E-09	9.63E-09	5.15E+03	1.94E-07	9.88E+03	2.02E-07	1.49E+130	3.36E-134
14	I-134	3.65E-02	-	3.37E-08	4.81E-08	2.76E-08	3.95E-08	2.86E+04	3.50E-08	5.72E+04	3.50E-08	NA	NA
15	I-135/Xe135m	NA	NA	2.05E-08	2.92E-08	1.68E-08	2.40E-08	6.25E+03	1.60E-07	1.25E+04	1.60E-07	NA	NA
	I-135	2.75E-01	1.00E+00	1.96E-08	2.80E-08	1.61E-08	2.30E-08	-	-	-	-	-	-
	Xe-135m	1.06E-02	1.54E-01	5.58E-09	7.97E-09	4.58E-09	6.54E-09	-	-	-	-	-	-

Table 3-4. Total Dose Parameters Corrected for Radioactive Decay (Not Corrected for Weathering) (continued)

No.	Radio-Nuclide	[a]Half-Life (d)	[a]Branch Fraction	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExDC (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExXC (mR/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExDF (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExXF (mR/h per pCi/m^2)	Early Phase [a,d]DRL (µCi/m^2)	Early Phase [b]TDP_Dp (mrem per pCi/m^2)	Year One [a,d]DRL (µCi/m^2)	Year One [b]TDP_Dp (mrem per pCi/m^2)	Year Two [a,d]DRL (µCi/m^2)	Year Two [b]TDP_Dp (mrem per pCi/m^2)
16	Ir-192	7.40E+01	-	1.04E-08	1.49E-08	8.53E-09	1.22E-08	1.25E+03	8.00E-07	9.50E+01	2.11E-05	7.25E+02	6.90E-07
17	Kr-87	5.30E-02	-	1.12E-08	1.60E-08	9.18E-09	1.31E-08	5.94E+04	1.68E-08	1.19E+05	1.68E-08	NA	NA
18	Kr-88/Rb-88	NA	NA	3.29E-08	4.70E-08	2.70E-08	3.85E-08	9.04E+03	1.11E-07	1.81E+04	1.10E-07	NA	NA
	Kr-88	1.18E-01	1.00E+00	2.30E-08	3.29E-08	1.89E-08	2.69E-08	-	-	-	-	-	-
	Rb-88	1.24E-02	1.00E+00	9.87E-09	1.41E-08	8.09E-09	1.16E-08	-	-	-	-	-	-
19	La-140	1.68E+00	-	2.88E-08	4.11E-08	2.36E-08	3.37E-08	9.02E+02	1.11E-06	1.46E+03	1.37E-06	1.10E+68	4.55E-72
20	Mo-99/Tc-99m	NA	NA	3.70E-09	5.29E-09	3.04E-09	4.34E-09	5.44E+03	1.84E-07	6.92E+03	2.89E-07	1.56E+43	3.21E-47
	Mo-99	2.75E+00	1.00E+00	2.37E-09	3.39E-09	1.94E-09	2.78E-09	-	-	-	-	-	-
	Tc-99m	2.51E-01	8.76E-01	1.52E-09	2.17E-09	1.25E-09	1.78E-09	9.24E+04	1.08E-08	1.85E+05	1.08E-08	NA	NA
21	Np-239	2.36E+00	-	2.05E-09	2.93E-09	1.68E-09	2.40E-09	1.05E+04	9.52E-08	1.46E+04	1.37E-07	1.65E+50	3.03E-54
22	Pm-147	9.58E+02	-	3.73E-13	5.33E-13	3.06E-13	4.37E-13	7.17E+05	1.39E-09	3.24E+05	6.17E-09	2.40E+05	2.08E-09
23	Pu-238	3.20E+04	-	8.34E-12	1.19E-11	6.84E-12	9.77E-12	4.73E+01	2.11E-05	3.27E+01	6.12E-05	8.15E+01	6.13E-06
24	Pu-239	8.79E+06	-	3.78E-12	5.40E-12	3.10E-12	4.43E-12	4.29E+01	2.33E-05	2.97E+01	6.73E-05	7.35E+01	6.80E-06
25	Ra-226/Rn-222…	NA	NA	2.25E-08	3.21E-08	1.84E-08	2.63E-08	2.76E+02	3.63E-06	1.20E+01	1.67E-04	3.09E+00	1.62E-04
	Ra-226	5.84E+05	1.00E+00	8.14E-11	1.16E-10	6.67E-11	9.54E-11	-	-	-	-	-	-
	Rn-222	3.82E+00	1.00E+00	5.09E-12	7.27E-12	4.17E-12	5.96E-12	-	-	-	-	-	-
	Po-218	2.12E-03	1.00E+00	1.15E-13	1.64E-13	9.43E-14	1.35E-13	-	-	-	-	-	-
	Pb-214	1.86E-02	1.00E+00	3.20E-09	4.57E-09	2.62E-09	3.75E-09	-	-	-	-	-	-
	Bi-214	1.38E-02	1.00E+00	1.92E-08	2.74E-08	1.57E-08	2.25E-08	-	-	-	-	-	-
	Po-214	1.90E-09	1.00E+00	1.06E-12	1.51E-12	8.69E-13	1.24E-12	-	-	-	-	-	-
	At-218	2.31E-05	2.00E-04	4.85E-11	6.93E-11	3.98E-11	5.68E-11	-	-	-	-	-	-
	Bi-214	1.38E-02	1.00E+00	1.92E-08	2.74E-08	1.57E-08	2.25E-08	-	-	-	-	-	-
	Po-214	1.90E-09	1.00E+00	1.06E-12	1.51E-12	8.69E-13	1.24E-12	-	-	-	-	-	-
26	Ru-103/Rh-103m	NA	NA	5.99E-09	8.56E-09	4.91E-09	7.02E-09	2.19E+03	4.57E-07	3.00E+02	6.67E-06	4.70E+04	1.06E-08

Table 3-4. Total Dose Parameters Corrected for Radioactive Decay (Not Corrected for Weathering) (continued)

No.	Radio-Nuclide	[a]Half-Life (d)	[a]Branch Fraction	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExDC (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExXC (mR/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExDF (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExXF (mR/h per pCi/m^2)	Early Phase [a,d]DRL (µCi/m^2)	Early Phase [b]TDP_Dp (mrem per pCi/m^2)	Year One [a,d]DRL (µCi/m^2)	Year One [b]TDP_Dp (mrem per pCi/m^2)	Year Two [a,d]DRL (µCi/m^2)	Year Two [b]TDP_Dp (mrem per pCi/m^2)
	Ru-103	3.93E+01	1.00E+00	5.98E-09	8.54E-09	4.90E-09	7.01E-09	-	-	-	-	-	-
	Rh-103m	3.90E-02	9.97E-01	1.18E-11	1.69E-11	9.68E-12	1.38E-11	-	-	-	-	-	-
27	Ru-106/Rh-106	NA	NA	4.60E-09	6.57E-09	3.77E-09	5.39E-09	2.68E+03	3.73E-07	8.36E+01	2.39E-05	4.16E+01	1.20E-05
	Ru-106	3.68E+02	1.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	-	-	-	-	-	-
	Rh-106	3.46E-04	1.00E+00	4.60E-09	6.57E-09	3.77E-09	5.39E-09	-	-	-	-	-	-
28	Sb-127/Te-127	NA	NA	9.12E-09	1.30E-08	7.48E-09	1.07E-08	1.95E+03	5.13E-07	2.01E+03	9.95E-07	1.74E+31	2.87E-35
	Sb-127	3.85E+00	1.00E+00	9.01E-09	1.29E-08	7.39E-09	1.06E-08	-	-	-	-	-	-
	Te-127	3.90E-01	8.24E-01	1.37E-10	1.96E-10	1.12E-10	1.60E-10	-	-	-	-	-	-
29	Sb-129/Te-129	NA	NA	1.95E-08	2.78E-08	1.60E-08	2.28E-08	1.01E+04	9.90E-08	2.01E+04	9.95E-08	NA	NA
	Sb-129	1.80E-01	1.00E+00	1.83E-08	2.61E-08	1.50E-08	2.14E-08	-	-	-	-	-	-
	Te-129	4.83E-02	7.75E-01	1.52E-09	2.17E-09	1.25E-09	1.78E-09	-	-	-	-	-	-
30	Se-75	1.20E+02	-	4.81E-09	6.87E-09	3.94E-09	5.63E-09	2.67E+03	3.75E-07	1.39E+02	1.44E-05	2.87E+02	1.74E-06
31	Sr-89	5.05E+01	-	9.14E-10	1.31E-09	7.49E-10	1.07E-09	1.40E+04	7.14E-08	1.53E+03	1.31E-06	5.76E+04	8.68E-09
32	Sr-90/Y-90	NA	NA	1.49E-09	2.13E-09	1.22E-09	1.75E-09	6.75E+03	1.48E-07	1.88E+02	1.06E-05	4.85E+01	1.03E-05
	Sr-90	1.06E+04	1.00E+00	2.19E-11	3.13E-11	1.80E-11	2.57E-11	-	-	-	-	-	-
	Y-90	2.67E+00	1.00E+00	1.47E-09	2.10E-09	1.21E-09	1.72E-09	1.39E+04	7.19E-08	1.80E+04	1.11E-07	7.20E+44	6.94E-49
33	Sr-91/Y-91m	NA	NA	1.36E-08	1.94E-08	1.12E-08	1.59E-08	6.54E+03	1.53E-07	1.31E+04	1.53E-07	NA	NA
	Sr-91	3.96E-01	1.00E+00	9.69E-09	1.38E-08	7.95E-09	1.14E-08	-	-	-	-	-	-
	Y-91m	3.45E-02	5.78E-01	6.79E-09	9.70E-09	5.57E-09	7.95E-09	-	-	-	-	-	-
34	Te-129m/Te-129	NA	NA	1.75E-09	2.50E-09	1.43E-09	2.05E-09	7.49E+03	1.34E-07	1.20E+03	1.67E-06	5.59E+05	8.94E-10
	Te-129m	3.36E+01	1.00E+00	7.59E-10	1.08E-09	6.22E-10	8.89E-10	-	-	-	-	-	-
	Te-129	4.83E-02	6.50E-01	1.52E-09	2.17E-09	1.25E-09	1.78E-09	4.80E+05	2.08E-09	9.59E+05	2.09E-09	NA	NA
35	[d]Te-131m/Te-131	NA	NA	1.93E-08	2.76E-08	1.58E-08	2.26E-08	1.64E+03	6.10E-07	2.93E+03	6.83E-07	5.82E+90	8.59E-95
	[d]Te-131m	1.25E+00	1.00E+00	1.79E-08	2.56E-08	1.47E-08	2.10E-08	-	-	-	-	-	-
	Te-131	1.74E-02	2.22E-01	6.32E-09	9.03E-09	5.18E-09	7.40E-09	3.21E+05	3.12E-09	6.42E+05	3.12E-09	NA	NA

Table 3-4. Total Dose Parameters Corrected for Radioactive Decay (Not Corrected for Weathering) (continued)

No.	Radio-Nuclide	[a]Half-Life (d)	[a]Branch Fraction	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExDC (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Uncorrected for GRF ExXC (mR/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExDF (mrem/h per pCi/m^2)	[c]Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF ExXF (mR/h per pCi/m^2)	Early Phase [a,d]DRL (µCi/m^2)	Early Phase [b]TDP_Dp (mrem per pCi/m^2)	Year One [a,d]DRL (µCi/m^2)	Year One [b]TDP_Dp (mrem per pCi/m^2)	Year Two [a,d]DRL (µCi/m^2)	Year Two [b]TDP_Dp (mrem per pCi/m^2)
36	Te-132/I-132	NA	NA	3.21E-08	4.59E-08	2.63E-08	3.76E-08	5.87E+02	1.70E-06	6.73E+02	2.97E-06	8.86E+35	5.64E-40
	Te-132	3.26E+00	1.00E+00	2.82E-09	4.03E-09	2.31E-09	3.30E-09	-	-	-	-	-	-
	I-132	9.58E-02	1.00E+00	2.93E-08	4.19E-08	2.40E-08	3.43E-08	-	-	-	-	-	-
37	Tm-170	1.29E+02	-	3.52E-10	5.03E-10	2.89E-10	4.12E-10	3.42E+04	2.92E-08	1.80E+03	1.11E-06	3.24E+03	1.54E-07
38	Xe-133	5.24E+00	-	5.26E-10	7.51E-10	4.31E-10	6.16E-10	3.11E+04	3.22E-08	2.55E+04	7.84E-08	5.67E+24	8.82E-29
39	Xe-135	3.79E-01	-	3.33E-09	4.76E-09	2.73E-09	3.90E-09	2.79E+04	3.58E-08	5.58E+04	3.58E-08	NA	NA
40	Xe-138	9.84E-03	-	1.43E-08	2.04E-08	1.17E-08	1.68E-08	2.51E+05	3.98E-09	5.02E+05	3.98E-09	NA	NA
41	Y-91	5.85E+01	-	9.94E-10	1.42E-09	8.15E-10	1.16E-09	1.28E+04	7.81E-08	1.22E+03	1.64E-06	2.32E+04	2.16E-08
42	Yb-169	3.20E+01	-	3.70E-09	5.29E-09	3.03E-09	4.33E-09	3.57E+03	2.80E-07	5.94E+02	3.37E-06	4.02E+05	1.24E-09

[a]Values from Turbo FRMAC 2.0, RFC 2, based on ICRP 60+ dosimetry model (DCFPAK, K. Eckerman).
[b]TDP (Total Dose Parameter), includes Equivalent Dose from groundshine and Committed Effective Dose from inhalation of resuspended material, corrected for radioactive decay.
[c]Values from Turbo FRMAC 2.0, RFC 2, based on ICRP 60+ dosimetry model.
[d]Te-131m values in this table are subject to change pending further development of new parent-daughter rules.

3.5 Method Used for Calculation of the Deposition Derived Response Levels and Table 3-4 Values

This section describes how the Deposition DRL (Dp_DRL or DRL) for the "marker" radionuclide is calculated. A DRL is a level of radioactivity in an environmental medium that would be expected to produce a dose equal to its corresponding PAG. The "marker" radionuclide concept is used to eliminate the need to separately measure the concentration of every radionuclide in the environmental medium because this process is very laborious. Generally a gamma-emitting radionuclide that can easily be identified in the field or laboratory is chosen as the "marker" radionuclide. The "marker" radionuclide DRL specifies the level of radioactivity (µCi/m^2) of the "marker" radionuclide at which the dose from all radionuclides in the mixture (i.e., all parent radionuclides and short-lived daughter radionuclides) will equal the corresponding PAG. The Dp_DRL includes the external dose from material deposited on the ground (i.e., groundshine) and the dose from the inhalation of resuspended material.

3.5.1 Final Calculation for Deposition Derived Response Level

3.5.1.1 Final Calculation for Deposition Derived Response Level for Marker Radionuclide in a Radionuclide Mixture

The formula below is used to calculate the Dp_DRL for the "marker" radionuclide. The Dp_DRL varies with the radionuclide mixture, the radionuclide chosen as the "marker," the time phase under consideration and the applicable PAG.

$$DRL\_Dp_{i,TP} = \frac{PAG_{TP} * Dp_i}{Dp\_TDP\_Dp_{E,i,TP}}, \qquad \frac{\mu Ci}{m^2} = \frac{mrem * \frac{\mu Ci}{m^2}}{mrem}$$

Where:

$Dp\_DRL_{i,TP}$ = Deposition Derived Response Level, the level of activity of "marker" radionuclide *i* at which the dose from all radionuclides in mixture would result in a dose equal to the PAG for the time phase (TP) under consideration, μCi/m^2;

Dp_i = Deposition, the radioactivity level of "marker" radionuclide *i* per unit area of ground, μCi/m^2;

PAG = EPA's Protective Action Guide for the time phase under consideration, mrem;

$Dp\_TDP\_Dp_{E,i,TP}$ = Deposition Total Dose Parameter for Surface Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, from the deposited radioactivity level of all parent radionuclide(s) *i* and any short-lived daughters, mrem; and

TP = Time Phase, the period of time (i.e., Early Phase, 1st-y, 2nd-y,) over which the assessment is performed.

3.5.1.2 Final Calculation for Deposition Derived Response Level for a Parent Radionuclide and Any Short-lived Daughters in Secular Equilibrium

$$DRL\_Dp_{i,TP} = \frac{PAG_{TP}}{TDP\_Dp_{E,i,TP}}, \qquad \frac{\mu Ci}{m^2} = \frac{mrem}{\frac{mrem}{\mu Ci / m^2}}$$

Where:

$Dp\_DRL_{i,TP}$ = Deposition Derived Response Level, the level of parent radionuclide *i*, and its short-lived daughter radionuclides in secular equilibrium, at which the dose from all radionuclides result in a dose equal to the PAG for the time phase (TP) under consideration, μCi/m^2;

PAG = EPA's Protective Action Guide for the time phase under consideration, mrem;

$TDP\_Dp_{E,i,TP}$ = Total Dose Parameter for Surface Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground, mrem·m^2/μCi; and

TP = Time Phase, the period of time (i.e., early phase, 1st-y, 2nd-y) over which the assessment is performed.

3.5.2 Calculation of Internal Dose Component of the Dp_DRL

3.5.2.1 Effective Dose Parameter Calculation

The equation below is used to calculate the committed effective dose component from inhaling the resuspended radioactivity over the time period under consideration.

$$EDP_{\text{inh, E, } i, TP} = CDF_{\text{inh, E, i}} * KP_{i, TP},$$

$$\frac{mrem \cdot m^2}{\mu Ci} = \frac{mrem \cdot m^3}{\mu Ci \cdot s} * \frac{s}{m}$$

Where:

$EDP_{inh, E, i, TP}$ = Effective Dose Parameter, the committed effective dose received per unit activity of radionuclide *i* deposited on the ground from the inhalation of the resuspended radionuclide over the time phase under consideration, mrem·m^2/µCi;

$CDF_{inh, E, i}$ = Committed Dose Factor, the committed effective dose rate from breathing (at a specified rate) air contaminated with a unit activity of radionuclide *i*, mrem·m^3/ s·µCi; and

$KP_{i, TP}$ = Resuspension Parameter, value that adjusts the airborne radioactivity level of radionuclide *i* for radioactive decay and resuspension over the time phase under consideration, (value from Turbo FRMAC 2.0, RFC 2), s/m.

3.5.2.2 Committed Dose Factor Calculation

$$CDF_{\text{inh, E, i}} = BR * IDCF_{\text{inh, E, i}} * CF, \qquad \frac{mrem \cdot m^3}{s \cdot \mu Ci} = \frac{m^3}{s} * \frac{Sv}{Bq} * \frac{mrem \cdot Bq}{\mu Ci \cdot Sv}$$

Where:

$CDF_{inh, E, i}$ = Committed Dose Factor, the committed effective dose rate from breathing (at a specified rate) air contaminated with a unit activity of radionuclide *i*, mrem·m^3/ s·µCi;

BR = Breathing Rate, the activity-weighted, average hourly breathed air in per unit time by an adult male (ICRP, 1994, Table B.16B), 2.56E-04 m^3/s;

IDCF = Inhalation Dose Conversion Factor, the committed effective dose conversion factor for radionuclide *i*, (values from ICRP 60+ dosimetry models, DCFPAK, 2006), mrem/µCi; and

CF = Unit Conversion Factor, converts Sv/Bq to mrem/µCi, 3.7E+09 mrem/µCi per Sv/Bq.

$$\frac{3.7E9\, mrem / \mu Ci}{Sv / Bq} = \frac{Sv}{Bq} * \frac{10^5\, mrem}{Sv} * \frac{Bq}{dps} * \frac{3.7E4\, dps}{Bq}$$

3.5.2.3 Resuspension Parameter Calculation

The Resuspension Parameter (KP) adjusts the inhalation dose for radioactive decay and the time-dependent resuspension factor (K) that occurs over the time period under consideration. The KP integral below does not have an exact solution when "K" is in a time-dependent form. Therefore, the integral cannot be solved analytically and must be solved using a software program that capable of numerical integration. It should be noted that the K model described below may not be appropriate for the environmental conditions existing in the area under investigation. An alternate K model may be substituted, with EPA approval, if the alternate model can be shown to more accurately model the resuspension in the area under investigation.

$$KP_{i,TP} = \int_{T_1}^{T_2} K * e^{(-\lambda_i * T)} dT \,, \qquad \frac{s}{m} = \int_{s}^{s} \frac{1}{m} * e^{\left(\frac{-\frac{1}{s} * s}{\frac{1}{s}}\right)}$$

Where:

$KP_{i, TP}$ = Resuspension Parameter, value that adjusts the inhalation dose from radionuclide *i* for radioactive decay and the time-dependent resuspension factor (K) over the time period under consideration, s/m;

T_1 = the start of the Time Phase (integration period) under consideration, s;

T_2 = the end of the Time Phase (integration period) under consideration, s;

K = Resuspension Factor, value based on the time-varying formula from NCRP Report No. 129, *Recommended Screening Limits for Contaminated Surface Soil and Review of Factors Relevant to Site-Specific Studies*, (NCRP 1999), m^{-1}

- K = 1.00E-06 m^{-1} for t <1 day or
- K = 1.00E-06 m^{-1}/t for t > 1 and ≤1,000 days or,
- K = 1.00E-09 m^{-1} for t >1000 day.

λ_i = Decay constant for radionuclide *i*, s^{-1}.

3.5.3 Calculation of External Dose Component of the Dp_DRL

The equation below is used to calculate the external dose component received from the radionuclides deposited on the ground over the time period under consideration.

3.5.3.1 External Dose Parameter Calculation

The external dose parameter for deposition (ExDP_Dp) gives the effective dose from groundshine per unit activity deposited on the ground over the time period under consideration and adjusted for the ground roughness factor (GRF).

$$\mathrm{ExDP\_Dp}_{ground,E,i,TP} = CRP_{i,TP} * ExDF_{ground,E,i} \,, \qquad \left(\frac{mrem \cdot m^2}{\mu Ci}\right) = (s) * \left(\frac{mrem \cdot m^2}{s \cdot \mu Ci}\right)$$

Where:

$ExDP\_Dp_{ground, E, i, TP}$ = External Dose Parameter for Deposition, the effective dose from groundshine per unit activity deposited on the ground from radionuclide *i* over the time phase under consideration and adjusted for the GRF, mrem·m^2/ μCi;

$CRP_{i, TP}$ = Combined Removal Parameter, value that adjusts the external (groundshine) dose from radionuclide *i* for radioactive decay and weathering effects which decrease the groundshine dose over the time phase under consideration, s, (see Section D.1 in Appendix D); and

$ExDF_{ground, E, i}$ = External Dose Factor for Deposition, the effective dose rate from the external exposure to radionuclide *i* per unit activity deposited on the ground and adjusted for the ground roughness factor, mrem·m^2/ s·μCi.

3.5.4 Total Dose Parameter for Surface Deposition Calculation

The total dose parameter for surface deposition (TDP_S) is the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground.

$$TDP\_Dp_{E,i,TP} = EDP_{inh,E,TP,i} + ExDP\_Dp_{ground,E,i,TP},$$

$$\frac{mrem \cdot m^2}{\mu Ci} = \frac{mrem \cdot m^2}{\mu Ci} + \frac{mrem \cdot m^2}{\mu Ci}$$

Where:

$TDP\_Dp_{E, i, TP}$ = Total Dose Parameter for Surface Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground, mrem·m^2/μCi;

$EDP_{inh, E, i, TP}$ = Effective Dose Parameter, the committed effective dose received from the inhalation of the resuspended radionuclide *i* over the time phase under consideration and per unit of radioactivity of radionuclide *i* deposited on the ground, mrem·m^2/μCi; and

$ExDP\_Dp_{ground, E, i, TP}$ = External Dose Parameter for Deposition, the groundshine dose received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground and adjusted for the ground roughness factor, mrem·m^2/μCi.

3.5.5 Deposition Total Dose Parameter Calculation

The deposition total dose parameter for deposition (Dp_TDP_DP) is the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, from the deposited radioactivity level of all parent radionuclide(s) and any short-lived daughter radionuclides.

$$Dp\_TDP\_Dp_{E,i,TP} = \sum_{i}^{P+D} \left(Dp_i * TDP\_Dp_{E,i,TP} \right)$$

Where:

$\sum_{i}^{P+D}$ = Represents the summation of values from all parent (P) and short-lived daughter (D) radionuclide(s);

$Dp\_TDP\_Dp_{E,\,i,\,TP}$ = Deposition Total Dose Parameter for Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, from the deposited radioactivity level of all parent radionuclide(s) and any short-lived daughters, mrem;

Dp_i = Deposition, the radioactivity level of radionuclide *i* per unit area of ground, μCi/m^2; and

$TDP\_Dp_{E,\,i,\,TP}$ = Total Dose Parameter for Surface Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground, mrem·m^2/μCi.

3.5.6 Example Calculation of Deposition Derived Response Level (Dp_DRL) for a Radionuclide Mixture

Given the radionuclide mixture in Table 3-5 and considering weathering effects, calculate the deposition DRL for a marker radionuclide for the first-year time phase. Generally a radionuclide that is easily detected in the field or in the laboratory is chosen as the marker radionuclide which is used to represent the entire mixture. For this mixture, either I-131 or Cs-137 could be chosen as the marker radionuclide because they have prominent gamma signatures that allow them to be individually detected in a mixture of radionuclides. For this example, Cs-137 is chosen as the marker radionuclide for the DRL calculation. Follow the steps below to calculate the Dp_DRL for the mixture over the first-year time phase using Cs-137 as the marker radionuclide.

1. Multiply the relative activity level of each parent radionuclide (Dp_i) in the mixture by its corresponding total dose parameter for surface deposition (TDP_Dp) for the time phase under consideration and then sum the products to derive the deposition total dose parameter for surface deposition (Dp_TDP_Dp) for the radionuclide mixture.

$$\mathrm{Dp\_TDP\_Dp}_{E,\,i,TP} = \sum_{i}^{P+D}\left(Dp_i * \mathrm{TDP\_Dp}_{E,\,i,TP}\right)$$

Where:

$\sum_{i}^{P+D}$ = Represents the summation of values from all parent (P) and short-lived daughter (D) radionuclide(s);

$Dp\_TDP\_Dp_{E,\,i,\,TP}$ = Deposition Total Dose Parameter for Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, from the deposited radioactivity level of all parent radionuclide(s) and any short-lived daughters, mrem;

Dp_i = Deposition, the radioactivity level of radionuclide *i* per unit area of ground, μCi/m^2, (from Table 3-5); and

$TDP\_Dp_{E,\,i,\,TP}$ = Total Dose Parameter, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground, (from Table 3-3), mrem·m^2/μCi.

$$\mathrm{Dp\_TDP\_Dp}_{E,mix,1st-y} = \sum_{i}^{P+D}\left[\begin{array}{l}\left(Dp_{I131} * \mathrm{TDP\_Dp}_{E,I131,1st-y}\right)+\left(Dp_{Te132} * \mathrm{TDP\_Dp}_{E,Te132,1st-y}\right)+\\ \left(Dp_{Ru103} * \mathrm{TDP\_Dp}_{E,Ru103,1st-y}\right)+\left(Dp_{Ru106} * \mathrm{TDP\_Dp}_{E,Ru106,1st-y}\right)+\\ \left(Dp_{Cs134} * \mathrm{TDP\_Dp}_{E,Cs-134,1st-y}\right)+\left(Dp_{Cs137} * \mathrm{TDP\_Dp}_{E,Cs-137,1st-y}\right)\end{array}\right]$$

$$\mathrm{Dp\_TDP\_Dp}_{E,mix,1st-y} = \sum_{i}^{P+D}\left[\begin{array}{l}\left(200\ pCi * \frac{1.10E-06\ pCi}{pCi/m^2}\right)+\left(3600\ pCi * \frac{2.97E-06\ pCi}{pCi/m^2}\right)+\\ \left(220\ pCi * \frac{2.21E-05\ pCi}{pCi/m^2}\right)+\left(50\ pCi * \frac{2.21E-05\ pCi}{pCi/m^2}\right)+\\ \left(68\ pCi * \frac{1.10E-04\ pCi}{pCi/m^2}\right)+\left(44.4\ pCi * \frac{4.77E-05\ pCi}{pCi/m^2}\right)\end{array}\right]$$

$$\mathrm{Dp\_TDP\_Dp}_{E,mix,1st-y} = 2.32E-02\ mrem$$

2. Calculate the Dp_DRL for the first-year time phase for Cs-137 as the marker radionuclide using the following equation.

$$DP\_DRL_{Cs-137,1st-y} = \frac{PAG_{1st}-y * Dp_{Cs-137}}{Dp\_TDP\_Dp_{E,mix,1st-y}} = \frac{2000mrem * \frac{44.4\,pCi}{m^2}}{2.31E-02mrem} = \frac{3.84E+06\,pCi}{m^2}$$

Where:

$\mathrm{Dp\_DRL}_{\text{Cs-137, 1st-y}}$ = Deposition Derived Response Level, the level of activity of "marker" radionuclide *i* at which the dose from all radionuclides in mixture would result in a dose equal to the PAG for the time phase (TP) under consideration, µCi/m^2;

Dp_{Cs137} = Deposition, the radioactivity level of "marker" radionuclide *i* per unit area of ground, (from Table 3-5), µCi/m^2;

PAG_{1st-y} = EPA's Protective Action Guide for the time phase under consideration, 2000 mrem; and

$Dp\_TDP\_Dp_{E,\ mix,\ 1st\text{-}y}$ = Deposition Total Dose Parameter for Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, from the deposited radioactivity level of all parent radionuclide(s) *i* and any short-lived daughters, 2.31E-06 mrem (from Step 1).

Therefore, the Dp_DRL for Cs-137 that is equal to PAG of 2000 mrem for the first-year time phase is 3.84E+06 pCi/m^2. Areas with Cs-137 deposition activities greater to this value exceed the PAG of 2000 mrem.

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Table 3-5. Calculation of Marker Radionuclide (Cs-137) DRL for a Radionuclide Mixture Based on Measured Isotopic Concentrations - Corrected for the Groundroughness Factor, Radioactive Decay and Weathering Effects

					Early Phase		Year One		Year Two	
Radio-Nuclide	Half-Life (d)	Branch Fraction	Measured Sample Activity (Dp_i) (pCi Sample)	[4]Estimated Sample Activity (Dp_i) (pCi Sample)	TDP_S (mrem per pCi/m^2)	Dpi * TDP_S (mrem)	TDP_S (mrem per pCi/m^2)	Dpi * TDP_S (mrem)	TDP_S (mrem per pCi/m^2)	Dpi * TDP_S (mrem)
I-131	8.04	-	2.60E+02		3.24E-07	8.41E-05	1.10E-06	2.87E-04	2.01E-20	5.22E-18
Te-132/I-132					1.70E-06	6.13E-03	2.97E-06	1.07E-02	4.76E-40	1.71E-36
Te-132	3.26	1	3.60E+03							
I-132	9.58E-02	1	3.60E+03							
Ru-103/Rh-103m					4.57E-07	1.00E-04	6.49E-06	1.43E-03	8.79E-09	1.93E-06
Ru-103	3.93E+01	1	2.20E+02							
Rh-103m	3.90E-02	0.997		2.19E+02						
Ru-106/Rh-106					3.73E-07	1.87E-05	2.21E-05	1.11E-03	9.56E-06	4.78E-04
Ru-106	3.68E+02	1		5.00E+01						
Rh-106	3.46E-04	1	5.00E+01							
Cs-134	7.53E+02	-	6.80E+01		1.55E-06	1.06E-04	1.10E-04	7.51E-03	6.81E-05	4.63E-03
Cs-137/Ba-137m					5.85E-07	2.60E-05	4.77E-05	2.12E-03	4.03E-05	1.79E-03
Cs-137	1.10E+04	1		4.44E+01						
Ba-137m	1.77E-03	0.946	4.20E+01							
				$Dp\_TDP\_S_{mix}$ =		6.47E-03		2.31E-02		6.90E-03
				DRL_{Cs137} =		6.86E+06		3.84E+06		3.22E+06

[1]The data in this table are only examples to demonstrate a calculational process. The results should not be used in prediction of relationships that would exist following a nuclear incident.
[2]Values are based on ICRP 60+ and are corrected for the ground roughness factor (GRF), radioactive decay and weathering effects (WCF).
[3]Exposure rate at height of 1 m above ground and at time of deposition and are corrected for Ground Rougness Factor (GRF).
[4]Activity of non-gamma emitting or unmeasured radionuclides inferred from parent/daughter relationships. Short-lived daughters are assumed to be in secular equilibrium with parent

Table 3-5a. [a,b]Example Calculation of Activity-Weighted Total Dose Parameters for Exposure Rate for a Radionuclide Mixture Based on Measured Isotopic Concentrations - Corrected for the Groundroughness Factor, Radioactive Decay and Weathering Effects

Radio-Nuclide	[c]Half-Life (d)	Branch Fraction	[c]Initial Exposure Rate, ExXC (mR/h per pCi/m^2)	Measured Sample Activity (Dp_i) (pCi Sample)	[d]Estimated Sample Activity (Dp_i) (pCi Sample)	[e]Calculated Exposure Rate at 1 m, ExXR (mR/h)	Early Phase		Year One		Year Two	
							TDP_XR at 1 m (mrem per mR/h)	TEDE (mrem)	TDP_XR at 1 m (mrem per mR/h)	TEDE (mrem)	TDP_XR at 1 m (mrem per mR/h)	TEDE (mrem)
I-131	8.04	-	5.68E-09	2.60E+02		1.48E-06	5.70E+01	8.41E-05	1.94E+02	2.87E-04	3.53E-12	5.22E-18
Te-132/I-132							4.53E+01	6.13E-03	7.89E+01	1.07E-02	1.27E-32	1.71E-36
Te-132	3.26	1	3.30E-09	3.60E+03		1.19E-05						
I-132	9.58E-02	1	3.43E-08	3.60E+03		1.24E-04						
Ru-103/Rh-103m							6.51E+01	1.00E-04	9.25E+02	1.43E-03	1.25E+00	1.93E-06
Ru-103	3.93E+01	1	7.01E-09	2.20E+02		1.54E-06						
Rh-103m	3.90E-02	0.997	1.38E-11		2.19E+02	3.03E-09						
Ru-106/Rh-106							6.92E+01	1.87E-05	4.11E+03	1.11E-03	1.77E+03	4.78E-04
Ru-106	3.68E+02	1	0.00E+00		5.00E+01	0.00E+00						
Rh-106	3.46E-04	1	5.39E-09	5.00E+01		2.69E-07						
Cs-134	7.53E+02	-	2.31E-08	6.80E+01		1.57E-06	6.73E+01	1.06E-04	4.79E+03	7.51E-03	2.95E+03	4.63E-03
Cs-137/Ba-137m							6.81E+01	2.60E-05	5.56E+03	2.12E-03	4.69E+03	1.79E-03
Cs-137	1.10E+04	1	4.66E-11		4.44E+01	2.07E-09						
Ba-137m	1.77E-03	0.946	9.03E-09	4.20E+01		3.79E-07						
					Mixture Totals ($ExXR_{mix}$) =	1.41E-04		6.47E-03		2.31E-02		6.90E-03
					TDP_XR for Mixture ($TDP\_XR_{mixture}$) =		4.60E+01		1.64E+02		4.91E+01	

[a]The data in this table are only examples to demonstrate a calculational process. The results should not be used in prediction of relationships that would exist following a nuclear incident.
[b]Values are based on ICRP 60+ and are corrected for the ground roughness factor (GRF), radioactive decay and weathering effects (WCF).
[c]Exposure rate at height of 1 m above ground and at time of deposition and are corrected for Ground Roughness Factor (GRF).
[d]Activity of non-gamma emitting or unmeasured radionuclides inferred from parent/daughter relationships. Short-lived daughters are assumed to be in secular equilibrium with parent radionuclides
[e]Values corrected for Ground Roughness Factor (GRF).

Table 3-5b. [a,b]Example Calculation of the Activity-Weighted Total Dose Parameters for Exposure Rate for a Radionuclide Mixture Based on Measured Isotopic Concentrations - Corrected for the Groundroughness Factor and Radioactive Decay; not Corrected for Weathering Effects

Radio-Nuclide	[c]Half-Life (d)	Branch Fraction	[c]Initial Exposure Rate, ExXC (mR/h per pCi/m^2)	Measured Sample Activity (Dp_i) (pCi Sample)	[d]Estimated Sample Activity (Dp_i) (pCi Sample)	[e]Calculated Exposure Rate at 1 m, ExXR (mR/h)	Early Phase		Year One		Year Two	
							TDP_XR at 1 m (mrem per mR/h)	TEDE (mrem)	TDP_XR at 1 m (mrem per mR/h)	TEDE (mrem)	TDP_XR at 1 m (mrem per mR/h)	TEDE (mrem)
I-131	8.04	-	5.68E-09	2.60E+02		1.48E-06	5.71E+01	8.44E-05	1.96E+02	2.89E-04	4.21E-12	6.22E-18
Te-132/I-132							4.53E+01	6.13E-03	7.89E+01	1.07E-02	1.27E-32	1.71E-36
Te-132	3.26	1	3.30E-09	3.60E+03		1.19E-05						
I-132	9.58E-02	1	3.43E-08	3.60E+03		1.24E-04						
Ru-103/Rh-103m							6.51E+01	1.00E-04	9.50E+02	1.47E-03	1.52E+00	2.34E-06
Ru-103	3.93E+01	1	7.01E-09	2.20E+02		1.54E-06						
Rh-103m	3.90E-02	0.997	1.38E-11		2.19E+02	3.03E-09						
Ru-106/Rh-106							6.92E+01	1.87E-05	4.44E+03	1.20E-03	2.23E+03	6.01E-04
Ru-106	3.68E+02	1	0.00E+00		5.00E+01	0.00E+00						
Rh-106	3.46E-04	1	5.39E-09	5.00E+01		2.69E-07						
Cs-134	7.53E+02	-	2.31E-08	6.80E+01		1.57E-06	6.73E+01	1.06E-04	5.22E+03	8.19E-03	3.72E+03	5.84E-03
Cs-137/Ba-137m							6.81E+01	2.60E-05	6.06E+03	2.31E-03	5.93E+03	2.26E-03
Cs-137	1.10E+04	1	4.66E-11		4.44E+01	2.07E-09						
Ba-137m	1.77E-03	0.946	9.03E-09	4.20E+01		3.79E-07						
					Mixture Totals =	1.41E-04		6.47E-03		2.41E-02		8.71E-03
					TDP_XR for Mixture ($TDP\_XR_{mixture}$) =		4.60E+01		1.72E+02		6.19E+01	

[a]The data in this table are only examples to demonstrate a calculational process. The results should not be used in prediction of relationships that would exist following a nuclear incident.
[b]Values are based on ICRP 60+ and are corrected for the ground roughness factor (GRF), radioactive decay and weathering effects (WCF).
[c]Exposure rate at height of 1 m above ground and at time of deposition and are corrected for Ground Roughness Factor (GRF).
[d]Activity of non-gamma emitting or unmeasured radionuclides inferred from parent/daughter relationships. Short-lived daughters are assumed to be in secular equilibrium with parent radionuclides
[e]Values corrected for Ground Roughness Factor (GRF).

Table 3-5c: Calculation of Total Dose Parameters for Exposure Rate (TDP_X) - Corrected for the Groundroughness Factor, Radioactive Decay and Weathering Effects

No.	Radio-Nuclide	[a]Half-Life (d)	Branch Fraction	Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF: ExDF (mrem/h per pCi/m^2)	ExXF (mR/h per pCi/m^2)	Early Phase: [a]DRL (µCi/m^2)	Early Phase: [b]TDP_X (mrem per mR/h)	Year One: [a]DRL (µCi/m^2)	Year One: [b]TDP_X (mrem per mR/h)	Year Two: [a]DRL (µCi/m^2)	Year Two: [2]TDP_X (mrem per mR/h)
1	Am-241	1.58E+05	-	2.54E-10	3.63E-10	5.29E+01	5.21E+04	3.53E+01	1.56E+05	6.91E+01	1.99E+04
2	Ba-140/La-140	NA	NA	2.57E-08	3.67E-08	4.52E+02	6.03E+01	1.78E+02	3.06E+02	2.22E+10	6.14E-07
	Ba-140	1.27E+01	1.00E+00	2.07E-09	2.96E-09	-	-	-	-	-	-
	La-140	1.68E+00	1.00E+00	2.36E-08	3.37E-08	see La-140 listed separately (as parent) below					
3	Ce144/Pr144/Pr144m	NA	NA	1.95E-09	2.79E-09	5.09E+03	7.05E+01	1.90E+02	3.78E+03	1.35E+02	1.33E+03
	Ce-144	2.84E+02	1.00E+00	2.01E-10	2.87E-10	-	-	-	-	-	-
	Pr-144	1.20E-02	9.82E-01	1.78E-09	2.54E-09	-	-	-	-	-	-
	Pr-144m	5.00E-03	1.78E-02	1.15E-10	1.64E-10	-	-	-	-	-	-
4	Cf-252	9.64E+02	-	5.72E-12	8.18E-12	1.38E+02	8.86E+05	9.85E+01	2.48E+06	3.39E+02	1.80E+05
5	Cm-244	6.61E+03	-	7.04E-12	1.01E-11	8.95E+01	1.11E+06	6.22E+01	3.20E+06	1.60E+02	3.11E+05
6	Co-60	1.93E+03	-	2.51E-08	3.58E-08	4.14E+02	6.74E+01	1.06E+01	5.26E+03	3.49E+00	4.00E+03
7	Cs-134	7.53E+02	-	1.62E-08	2.31E-08	6.44E+02	6.73E+01	1.81E+01	4.79E+03	7.34E+00	2.95E+03
8	Cs-136	1.31E+01	-	2.21E-08	3.16E-08	5.22E+02	6.06E+01	2.01E+02	3.15E+02	1.45E+10	1.09E-06
9	Cs-137/Ba-137m	NA	NA	6.01E-09	8.59E-09	1.71E+03	6.81E+01	4.19E+01	5.56E+03	1.24E+01	4.69E+03
	Cs-137	1.10E+04	1.00E+00	3.26E-11	4.66E-11	-	-	-	-	-	-
	Ba-137m	1.77E-03	9.46E-01	6.32E-09	9.03E-09	-	-	-	-	-	-
10	Gd-153	2.42E+02	-	1.01E-09	1.44E-09	1.04E+04	6.67E+01	3.93E+02	3.53E+03	3.25E+02	1.07E+03
11	I-131	8.04E+00	-	3.98E-09	5.68E-09	3.09E+03	5.70E+01	1.81E+03	1.94E+02	2.49E+16	3.53E-12
12	I-132	9.58E-02	-	2.40E-08	3.43E-08	1.25E+04	2.33E+00	2.51E+04	2.32E+00	NA	NA
13	I-133	8.67E-01	-	6.74E-09	9.63E-09	5.15E+03	2.02E+01	9.89E+03	2.10E+01	1.76E+130	2.95E-126
14	I-134	3.65E-02	-	2.76E-08	3.95E-08	2.86E+04	8.86E-01	5.72E+04	8.86E-01	NA	NA
15	I-135/Xe135m	NA	NA	1.68E-08	2.40E-08	6.26E+03	6.67E+00	1.25E+04	6.68E+00	NA	NA
	I-135	2.75E-01	1.00E+00	1.61E-08	2.30E-08	-	-	-	-	-	-
	Xe-135m	1.06E-02	1.54E-01	4.58E-09	6.54E-09	-	-	-	-	-	-
16	Ir-192	7.40E+01	-	8.53E-09	1.22E-08	1.25E+03	6.57E+01	9.95E+01	1.65E+03	8.90E+02	4.61E+01
17	Kr-87	5.30E-02	-	9.18E-09	1.31E-08	5.94E+04	1.28E+00	1.19E+05	1.28E+00	NA	NA

Table 3-5c: Calculation of Total Dose Parameters for Exposure Rate (TDP_X) - Corrected for the Groundroughness Factor, Radioactive Decay and Weathering Effects

No.	Radio-Nuclide	[a]Half-Life (d)	Branch Fraction	Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF: EDR (mrem/h per pCi/m^2)	Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF: ExXF (mR/h per pCi/m^2)	Early Phase: [a]DRL (µCi/m^2)	Early Phase: [b]TDP_X (mrem per mR/h)	Year One: [a]DRL (µCi/m^2)	Year One: [b]TDP_X (mrem per mR/h)	Year Two: [a]DRL (µCi/m^2)	Year Two: [2]TDP_X (mrem per mR/h)
18	Kr-88/Rb-88	NA	NA	2.70E-08	3.85E-08	9.04E+03	2.87E+00	1.81E+04	2.87E+00	NA	NA
	Kr-88	1.18E-01	1.00E+00	1.89E-08	2.69E-08	-	-	-	-	-	-
	Rb-88	1.24E-02	1.00E+00	8.09E-09	1.16E-08	-	-	-	-	-	-
19	La-140	1.68E+00	-	2.36E-08	3.37E-08	9.03E+02	3.28E+01	1.46E+03	4.06E+01	1.31E+68	1.13E-64
20	Mo-99/Tc-99m	NA	NA	3.04E-09	4.34E-09	5.45E+03	4.23E+01	6.93E+03	6.66E+01	1.85E+43	6.23E-39
	Mo-99	2.75E+00	1.00E+00	1.94E-09	2.78E-09	-	-	-	-	-	-
	Tc-99m	2.51E-01	8.76E-01	1.25E-09	1.78E-09	9.25E+04	6.07E+00	1.85E+05	6.07E+00	NA	NA
21	Np-239	2.36E+00	-	1.68E-09	2.40E-09	1.05E+04	3.97E+01	1.46E+04	5.70E+01	1.96E+50	1.06E-45
22	Pm-147	9.58E+02	-	3.06E-13	4.37E-13	7.17E+05	3.19E+03	3.34E+05	1.37E+04	2.93E+05	3.91E+03
23	Pu-238	3.20E+04	-	6.84E-12	9.77E-12	4.73E+01	2.16E+06	3.27E+01	6.26E+06	8.16E+01	6.27E+05
24	Pu-239	8.79E+06	-	3.10E-12	4.43E-12	4.29E+01	5.26E+06	2.97E+01	1.52E+07	7.36E+01	1.53E+06
25	Ra-226/Rn-222…	NA	NA	7.10E-11	1.01E-10	5.35E+02	1.84E+04	3.36E+02	5.87E+04	4.84E+02	1.02E+04
	Ra-226	5.84E+05	1.00E+00	6.67E-11	9.54E-11	-	-	-	-	-	-
	Rn-222	3.82E+00	1.00E+00	4.17E-12	5.96E-12	-	-	-	-	-	-
	Po-218	2.12E-03	1.00E+00	9.43E-14	1.35E-13	-	-	-	-	-	-
	At-218	2.31E-05	2.00E-04	3.98E-11	5.68E-11	-	-	-	-	-	-
26	Ru-103/Rh-103m	NA	NA	4.91E-09	7.02E-09	2.19E+03	6.51E+01	3.08E+02	9.25E+02	5.69E+04	1.25E+00
	Ru-103	3.93E+01	1.00E+00	4.90E-09	7.01E-09	-	-	-	-	-	-
	Rh-103m	3.90E-02	9.97E-01	9.68E-12	1.38E-11	-	-	-	-	-	-
27	Ru-106/Rh-106	NA	NA	3.77E-09	5.39E-09	2.68E+03	6.92E+01	9.04E+01	4.11E+03	5.23E+01	1.77E+03
	Ru-106	3.68E+02	1.00E+00	0.00E+00	0.00E+00	-	-	-	-	-	-
	Rh-106	3.46E-04	1.00E+00	3.77E-09	5.39E-09	-	-	-	-	-	-
28	Sb-127/Te-127	NA	NA	7.48E-09	1.07E-08	1.95E+03	4.80E+01	2.01E+03	9.31E+01	2.06E+31	2.27E-27
	Sb-127	3.85E+00	1.00E+00	7.39E-09	1.06E-08	-	-	-	-	-	-
	Te-127	3.90E-01	8.24E-01	1.12E-10	1.60E-10	-	-	-	-	-	-
29	Sb-129/Te-129	NA	NA	1.60E-08	2.28E-08	1.01E+04	4.34E+00	2.01E+04	4.36E+00	NA	NA

Table 3-5c: Calculation of Total Dose Parameters for Exposure Rate (TDP_X) - Corrected for the Groundroughness Factor, Radioactive Decay and Weathering Effects

No.	Radio-Nuclide	[a]Half-Life (d)	Branch Fraction	Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF: EDR (mrem/h per pCi/m^2)	Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF: ExXF (mR/h per pCi/m^2)	Early Phase: [a]DRL (µCi/m^2)	Early Phase: [b]TDP_X (mrem per mR/h)	Year One: [a]DRL (µCi/m^2)	Year One: [b]TDP_X (mrem per mR/h)	Year Two: [a]DRL (µCi/m^2)	Year Two: [2]TDP_X (mrem per mR/h)
	Sb-129	1.80E-01	1.00E+00	1.50E-08	2.14E-08	-	-	-	-	-	-
	Te-129	4.83E-02	7.75E-01	1.25E-09	1.78E-09	-	-	-	-	-	-
30	Se-75	1.20E+02	-	3.94E-09	5.63E-09	2.67E+03	6.65E+01	1.48E+02	2.40E+03	3.56E+02	2.49E+02
31	Sr-89	5.05E+01	-	7.49E-10	1.07E-09	1.40E+04	7.14E-08	1.59E+03	1.26E-06	7.00E+04	7.14E-09
32	Sr-90/Y-90	NA	NA	1.22E-09	1.75E-09	6.76E+03	8.46E+01	2.05E+02	5.58E+03	6.13E+01	4.67E+03
	Sr-90	1.06E+04	1.00E+00	1.80E-11	2.57E-11	-	-	-	-	-	-
	Y-90	2.67E+00	1.00E+00	1.21E-09	1.72E-09	1.39E+04	4.18E+01	1.80E+04	6.45E+01	8.54E+44	3.40E-40
33	Sr-91/Y-91m	NA	NA	1.12E-08	1.59E-08	6.54E+03	9.59E+00	1.31E+04	9.57E+00	NA	NA
	Sr-91	3.96E-01	1.00E+00	7.95E-09	1.14E-08	-	-	-	-	-	-
	Y-91m	3.45E-02	5.78E-01	5.57E-09	7.95E-09	-	-	-	-	-	-
34	Te-129m/Te-129	NA	NA	1.43E-09	2.05E-09	7.50E+03	6.52E+01	1.23E+03	7.95E+02	6.75E+05	3.62E-01
	Te-129m	3.36E+01	1.00E+00	6.22E-10	8.89E-10	-	-	-	-	-	-
	Te-129	4.83E-02	6.50E-01	1.25E-09	1.78E-09	4.80E+05	1.17E+00	9.59E+05	1.17E+00	NA	NA
35	[c]Te-131m/Te-131	NA	NA	1.58E-08	2.26E-08	1.64E+03	2.70E+01	2.93E+03	3.02E+01	6.90E+90	3.20E-87
	[c]Te-131m	1.25E+00	1.00E+00	1.47E-08	2.10E-08	-	-	-	-	-	-
	Te-131	1.74E-02	2.22E-01	5.18E-09	7.40E-09	3.21E+05	4.21E-01	6.42E+05	4.21E-01	NA	NA
36	Te-132/I-132	NA	NA	2.63E-08	3.76E-08	5.87E+02	4.53E+01	6.74E+02	7.89E+01	1.05E+36	1.27E-32
	Te-132	3.26E+00	1.00E+00	2.31E-09	3.30E-09	-	-	-	-	-	-
	I-132	9.58E-02	1.00E+00	2.40E-08	3.43E-08	-	-	-	-	-	-
37	Tm-170	1.29E+02	-	2.89E-10	4.12E-10	3.43E+04	7.07E+01	1.92E+03	2.53E+03	4.02E+03	3.02E+02
38	Xe-133	5.24E+00	-	4.31E-10	6.16E-10	3.11E+04	5.22E+01	2.56E+04	1.27E+02	6.74E+24	1.20E-19
39	Xe-135	3.79E-01	-	2.73E-09	3.90E-09	2.79E+04	9.19E+00	5.59E+04	9.17E+00	NA	NA
40	Xe-138	9.84E-03	-	1.17E-08	1.68E-08	2.51E+05	2.38E-01	5.02E+05	2.38E-01	NA	NA
41	Y-91	5.85E+01	-	8.15E-10	1.16E-09	1.28E+04	6.71E+01	1.27E+03	1.35E+03	2.83E+04	1.52E+01
42	Yb-169	3.20E+01	-	3.03E-09	4.33E-09	3.58E+03	6.44E+01	6.08E+02	7.59E+02	4.85E+05	2.38E-01

Table 3-5c: Calculation of Total Dose Parameters for Exposure Rate (TDP_X) - Corrected for the Groundroughness Factor, Radioactive Decay and Weathering Effects

						Early Phase		Year One		Year Two	
				Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF		[a]DRL	[b]TDP_X	[a]DRL	[b]TDP_X	[a]DRL	[2]TDP_X
No.	Radio-Nuclide	[a]Half-Life (d)	Branch Fraction	ExDF (mrem/h per pCi/m^2)	ExXF (mR/h per pCi/m^2)	(μCi/m^2)	(mrem per mR/h)	(μCi/m^2)	(mrem per mR/h)	(μCi/m^2)	(mrem per mR/h)

[a]FRMAC 2.0, RFC 2, based on ICRP 60+ dosimetry model (DCFPAK, K. Eckerman).
[b]TDP_X include Equivalent Dose from groundshine and Committed Effective Dose from inhalation of resuspended material, corrected for radioactive decay and weathering.
[c]Te-131m values in this table are subject to change pending further development of new parent-daughter rules.

Table 3-5d. Calculation of Total Dose Parameters for Exposure Rate (TDP_X) - Corrected for the Groundroughness Factor and Radioactive Decay; not Corrected for Weathering Effects

No.	Radio-Nuclide	[a]Half-Life (d)	[a]Branch Fraction	Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF: EnDR (mrem/hr per pCi/m^2)	Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF: ExXF (mR/h per pCi/m^2)	Early Phase [a]DRL (µCi/m^2)	Early Phase [b]TDP_X (mrem per mR/hr)	Year One [a]DRL (µCi/m^2)	Year One [b]TDP_X (mrem per mR/h)	Year Two [a]DRL (µCi/m^2)	Year Two [b]TDP_X (mrem per mR/h)
1	Am-241	1.58E+05	-	2.54E-10	3.63E-10	5.29E+01	5.21E+04	3.52E+01	1.56E+05	6.49E+01	2.12E+04
2	Ba-140/La-140	NA	NA	2.57E-08	3.67E-08	4.51E+02	6.04E+01	1.77E+02	3.08E+02	1.86E+10	7.32E-07
	Ba-140	1.27E+01	1.00E+00	2.07E-09	2.96E-09	-	-	-	-	-	-
	La-140	1.68E+00	1.00E+00	2.36E-08	3.37E-08	see La-140 listed separately (as parent) below		-	-	-	-
3	Ce144/Pr144/Pr144m	NA	NA	1.95E-09	2.79E-09	5.08E+03	7.07E+01	1.76E+02	4.08E+03	1.07E+02	1.68E+03
	Ce-144	2.84E+02	1.00E+00	2.01E-10	2.87E-10	-	-	-	-	-	-
	Pr-144	1.20E-02	9.82E-01	1.78E-09	2.54E-09	-	-	-	-	-	-
	Pr-144m	5.00E-03	1.78E-02	1.15E-10	1.64E-10	-	-	-	-	-	-
4	Cf-252	9.64E+02	-	5.72E-12	8.18E-12	1.38E+02	8.86E+05	9.85E+01	2.48E+06	3.37E+02	1.81E+05
5	Cm-244	6.61E+03	-	7.04E-12	1.01E-11	8.95E+01	1.11E+06	6.22E+01	3.20E+06	1.60E+02	3.11E+05
6	Co-60	1.93E+03	-	2.51E-08	3.58E-08	4.14E+02	6.74E+01	9.70E+00	5.75E+03	2.76E+00	5.05E+03
7	Cs-134	7.53E+02	-	1.62E-08	2.31E-08	6.44E+02	6.73E+01	1.66E+01	5.22E+03	5.82E+00	3.72E+03
8	Cs-136	1.31E+01	-	2.21E-08	3.16E-08	5.21E+02	6.07E+01	1.99E+02	3.18E+02	1.21E+10	1.31E-06
9	Cs-137/Ba-137m	NA	NA	6.01E-09	8.59E-09	1.71E+03	6.81E+01	3.84E+01	6.06E+03	9.82E+00	5.93E+03
	Cs-137	1.10E+04	1.00E+00	3.26E-11	4.66E-11	-	-	-	-	-	-
	Ba-137m	1.77E-03	9.46E-01	6.32E-09	9.03E-09	-	-	-	-	-	-
10	Gd-153	2.42E+02	-	1.01E-09	1.44E-09	1.04E+04	6.67E+01	3.65E+02	3.80E+03	2.60E+02	1.33E+03
11	I-131	8.04E+00	-	3.98E-09	5.68E-09	3.08E+03	5.71E+01	1.80E+03	1.96E+02	2.09E+16	4.21E-12
12	I-132	9.58E-02	-	2.40E-08	3.43E-08	1.25E+04	2.33E+00	2.51E+04	2.32E+00	NA	NA
13	I-133	8.67E-01	-	6.74E-09	9.63E-09	5.15E+03	2.02E+01	9.88E+03	2.10E+01	1.49E+130	3.48E-126
14	I-134	3.65E-02	-	2.76E-08	3.95E-08	2.86E+04	8.86E-01	5.72E+04	8.86E-01	NA	NA
15	I-135/Xe135m	NA	NA	1.68E-08	2.40E-08	6.25E+03	6.68E+00	1.25E+04	6.68E+00	NA	NA
	I-135	2.75E-01	1.00E+00	1.61E-08	2.30E-08	-	-	-	-	-	-
	Xe-135m	1.06E-02	1.54E-01	4.58E-09	6.54E-09	-	-	-	-	-	-
16	Ir-192	7.40E+01	-	8.53E-09	1.22E-08	1.25E+03	6.57E+01	9.50E+01	1.73E+03	7.25E+02	5.66E+01
17	Kr-87	5.30E-02	-	9.18E-09	1.31E-08	5.94E+04	1.28E+00	1.19E+05	1.28E+00	NA	NA

Table 3-5d. Calculation of Total Dose Parameters for Exposure Rate (TDP_X) - Corrected for the Groundroughness Factor and Radioactive Decay; not Corrected for Weathering Effects

No.	Radio-Nuclide	[a]Half-Life (d)	[a]Branch Fraction	Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF: ExDF (mrem/hr per pCi/m^2)	ExXF (mR/h per pCi/m^2)	Early Phase: [a]DRL (µCi/m^2)	Early Phase: [b]TDP_X (mrem per mR/hr)	Year One: [a]DRL (µCi/m^2)	Year One: [b]TDP_X (mrem per mR/h)	Year Two: [a]DRL (µCi/m^2)	Year Two: [b]TDP_X (mrem per mR/h)
18	Kr-88/Rb-88	NA	NA	2.70E-08	3.85E-08	9.04E+03	2.87E+00	1.81E+04	2.87E+00	NA	NA
	Kr-88	1.18E-01	1.00E+00	1.89E-08	2.69E-08	-	-	-	-	-	-
	Rb-88	1.24E-02	1.00E+00	8.09E-09	1.16E-08	-	-	-	-	-	-
19	La-140	1.68E+00	-	2.36E-08	3.37E-08	9.02E+02	3.29E+01	1.46E+03	4.06E+01	1.10E+68	1.35E-64
20	Mo-99/Tc-99m	NA	NA	3.04E-09	4.34E-09	5.44E+03	4.24E+01	6.92E+03	6.67E+01	1.56E+43	7.39E-39
	Mo-99	2.75E+00	1.00E+00	1.94E-09	2.78E-09	-	-	-	-	-	-
	Tc-99m	2.51E-01	8.76E-01	1.25E-09	1.78E-09	9.24E+04	6.08E+00	1.85E+05	6.07E+00	NA	NA
21	Np-239	2.36E+00	-	1.68E-09	2.40E-09	1.05E+04	3.97E+01	1.46E+04	5.70E+01	1.65E+50	1.26E-45
22	Pm-147	9.58E+02	-	3.06E-13	4.37E-13	7.17E+05	3.19E+03	3.24E+05	1.41E+04	2.40E+05	4.77E+03
23	Pu-238	3.20E+04	-	6.84E-12	9.77E-12	4.73E+01	2.16E+06	3.27E+01	6.26E+06	8.15E+01	6.28E+05
24	Pu-239	8.79E+06	-	3.10E-12	4.43E-12	4.29E+01	5.26E+06	2.97E+01	1.52E+07	7.35E+01	1.54E+06
25	Ra-226/Rn-222…	NA	NA	7.10E-11	1.01E-10	5.35E+02	1.84E+04	3.33E+02	5.92E+04	4.30E+02	1.15E+04
	Ra-226	5.84E+05	1.00E+00	6.67E-11	9.54E-11	-	-	-	-	-	-
	Rn-222	3.82E+00	1.00E+00	4.17E-12	5.96E-12	-	-	-	-	-	-
	Po-218	2.12E-03	1.00E+00	9.43E-14	1.35E-13	-	-	-	-	-	-
	At-218	2.31E-05	2.00E-04	3.98E-11	5.68E-11	-	-	-	-	-	-
26	Ru-103/Rh-103m	NA	NA	4.91E-09	7.02E-09	2.19E+03	6.51E+01	3.00E+02	9.50E+02	4.70E+04	1.52E+00
	Ru-103	3.93E+01	1.00E+00	4.90E-09	7.01E-09	-	-	-	-	-	-
	Rh-103m	3.90E-02	9.97E-01	9.68E-12	1.38E-11	-	-	-	-	-	-
27	Ru-106/Rh-106	NA	NA	3.77E-09	5.39E-09	2.68E+03	6.92E+01	8.36E+01	4.44E+03	4.16E+01	2.23E+03
	Ru-106	3.68E+02	1.00E+00	0.00E+00	0.00E+00	-	-	-	-	-	-
	Rh-106	3.46E-04	1.00E+00	3.77E-09	5.39E-09	-	-	-	-	-	-
28	Sb-127/Te-127	NA	NA	7.48E-09	1.07E-08	1.95E+03	4.80E+01	2.01E+03	9.31E+01	1.74E+31	2.69E-27
	Sb-127	3.85E+00	1.00E+00	7.39E-09	1.06E-08	-	-	-	-	-	-
	Te-127	3.90E-01	8.24E-01	1.12E-10	1.60E-10	-	-	-	-	-	-
29	Sb-129/Te-129	NA	NA	1.60E-08	2.28E-08	1.01E+04	4.34E+00	2.01E+04	4.36E+00	NA	NA
	Sb-129	1.80E-01	1.00E+00	1.50E-08	2.14E-08	-	-	-	-	-	-

Table 3-5d. Calculation of Total Dose Parameters for Exposure Rate (TDP_X) - Corrected for the Groundroughness Factor and Radioactive Decay; not Corrected for Weathering Effects

No.	Radio-Nuclide	[a]Half-Life (d)	[a]Branch Fraction	Initial Dose and Exposure Rates at 1 m Above Ground Surface - Corrected for GRF: ExDF (mrem/hr per pCi/m^2)	ExXF (mR/h per pCi/m^2)	Early Phase: [a]DRL (µCi/m^2)	Early Phase: [b]TDP_X (mrem per mR/hr)	Year One: [a]DRL (µCi/m^2)	Year One: [b]TDP_X (mrem per mR/h)	Year Two: [a]DRL (µCi/m^2)	Year Two: [b]TDP_X (mrem per mR/h)
	Te-129	4.83E-02	7.75E-01	1.25E-09	1.78E-09	-	-	-	-	-	-
30	Se-75	1.20E+02	-	3.94E-09	5.63E-09	2.67E+03	6.65E+01	1.39E+02	2.55E+03	2.87E+02	3.09E+02
31	Sr-89	5.05E+01	-	7.49E-10	1.07E-09	1.40E+04	7.14E-08	1.53E+03	1.31E-06	5.76E+04	8.68E-09
32	Sr-90/Y-90	NA	NA	1.22E-09	1.75E-09	6.75E+03	8.48E+01	1.88E+02	6.09E+03	4.85E+01	5.90E+03
	Sr-90	1.06E+04	1.00E+00	1.80E-11	2.57E-11	-	-	-	-	-	-
	Y-90	2.67E+00	1.00E+00	1.21E-09	1.72E-09	1.39E+04	4.18E+01	1.80E+04	6.45E+01	7.20E+44	4.03E-40
33	Sr-91/Y-91m	NA	NA	1.12E-08	1.59E-08	6.54E+03	9.59E+00	1.31E+04	9.57E+00	NA	NA
	Sr-91	3.96E-01	1.00E+00	7.95E-09	1.14E-08	-	-	-	-	-	-
	Y-91m	3.45E-02	5.78E-01	5.57E-09	7.95E-09	-	-	-	-	-	-
34	Te-129m/Te-129	NA	NA	1.43E-09	2.05E-09	7.49E+03	6.52E+01	1.20E+03	8.14E+02	5.59E+05	4.37E-01
	Te-129m	3.36E+01	1.00E+00	6.22E-10	8.89E-10	-	-	-	-	-	-
	Te-129	4.83E-02	6.50E-01	1.25E-09	1.78E-09	4.80E+05	1.17E+00	9.59E+05	1.17E+00	NA	NA
35	Te-131m/Te-131	NA	NA	1.58E-08	2.26E-08	1.64E+03	2.70E+01	2.93E+03	3.02E+01	5.82E+90	3.80E-87
	Te-131m	1.25E+00	1.00E+00	1.47E-08	2.10E-08	-	-	-	-	-	-
	Te-131	1.74E-02	2.22E-01	5.18E-09	7.40E-09	3.21E+05	4.21E-01	6.42E+05	4.21E-01	NA	NA
36	Te-132/I-132	NA	NA	2.63E-08	3.76E-08	5.87E+02	4.53E+01	6.73E+02	7.90E+01	8.86E+35	1.50E-32
	Te-132	3.26E+00	1.00E+00	2.31E-09	3.30E-09	-	-	-	-	-	-
	I-132	9.58E-02	1.00E+00	2.40E-08	3.43E-08	-	-	-	-	-	-
37	Tm-170	1.29E+02	-	2.89E-10	4.12E-10	3.42E+04	7.09E+01	1.80E+03	2.69E+03	3.24E+03	3.74E+02
38	Xe-133	5.24E+00	-	4.31E-10	6.16E-10	3.11E+04	5.22E+01	2.55E+04	1.27E+02	5.67E+24	1.43E-19
39	Xe-135	3.79E-01	-	2.73E-09	3.90E-09	2.79E+04	9.19E+00	5.58E+04	9.19E+00	NA	NA
40	Xe-138	9.84E-03	-	1.17E-08	1.68E-08	2.51E+05	2.38E-01	5.02E+05	2.38E-01	NA	NA
41	Y-91	5.85E+01	-	8.15E-10	1.16E-09	1.28E+04	6.71E+01	1.22E+03	1.41E+03	2.32E+04	1.85E+01
42	Yb-169	3.20E+01	-	3.03E-09	4.33E-09	3.57E+03	6.46E+01	5.94E+02	7.77E+02	4.02E+05	2.87E-01

[1]Turbo FRMAC 2.0, RFC 2, based on ICRP 60+ dosimetry model (DCFPAK, K. Eckerman).
[2]TDP_X include Equivalent Dose from groundshine and Committed Effective Dose from inhalation of resuspended material, corrected for radioactive decay and weathering.
[c]Te-131m values in this table are subject to change pending further development of new parent-daughter rules.

3.6 Calculation of Skin Dose from Groundshine and Contamination

3.6.1 Calculation of Skin Dose from Groundshine and Contamination - Groundshine Dose Adjusted for Radioactive Decay and Weathering

This section specifies the method to adjust the total skin equivalent dose for both radioactive decay and weathering effects. The total skin dose conversion factors estimate the total skin equivalent dose (HTotal, skin) that is received over the first-year period from materials deposited on the ground. Table 3-6a (including correction for weathering effects) and Table 3-6b (does not include correction for weathering effects) provide the dose conversion factors for the total skin equivalent dose (Hskin) that is received over the first-year period from materials deposited on the ground. The total skin equivalent dose includes dose contributions from groundshine and from material deposited directly on the skin surface (i.e., skin contamination).

3.6.1.1 Calculation of Skin Dose from Groundshine - Groundshine Dose Adjusted for Radioactive Decay and Weathering

The equation below is used to estimate the skin equivalent dose received from groundshine over the first-year period after deposition. It is assumed that the receptor is exposed to ground shine for 2080 hours during the first-year period (*Evaluation of Skin and Ingestion Exposure Pathways*, EPA 1989b). It is further assumed the receptor is not exposed contiguously over the exposure period, but is exposed intermittently for some period of time (e.g., 2080 hours) during the year. Therefore, it is necessary to use the Average Combined Removal Parameter (AvCRP) integrated over the time period of interest, rather than the CRP for the time period of interest, to estimate the groundshine dose over the first year.

$$H_{groundshine,skin,i,TP} = ExDC_{groundshine,skin,i} * CF_1 * GRF * AvCRP_{i,TP} * (AXP\_Dp_{groundshine} * CF_2)$$

$$H_{groundshine,skin,i} = \frac{Sv \cdot m2}{s \cdot Bq} * \frac{1.17E17 \; mrem \cdot m^2 \,/\, y \cdot \mu Ci}{Sv \cdot m^2 \,/\, s \cdot Bq} * unitless * unitless * \left(h * \frac{y}{h} \right) = mrem \cdot m^2 \,/\, \mu Ci$$

Where:

$H_{groundshine,skin,i,TP}$ = Equivalent dose to the skin from groundshine per unity activity of radionuclide *i* deposited on the ground and over the 1st-y time period, mrem·m^2/μCi,

$ExDC_{groundshine,skin,i}$ = External Dose Coefficient, value for the external equivalent dose to the skin per unity activity of radionuclide *i* deposited on the ground, Sv·m^2/s·Bq, (values from ICRP 60+ dosimetry models (DCFPAK, 2006),

CF_1 = Unit Conversion Factor, 1.17E+17 mrem·m^2/y·μCi per Sv·m^2/s·Bq,

$$\frac{1.17E17 \; mrem \cdot m^2 \,/\, y \cdot \mu Ci}{Sv \cdot m^2 \,/\, s \cdot Bq} = \frac{Sv \cdot m^2}{s \cdot Bq} * \frac{10^5 \; mrem}{Sv} * \frac{3.15E7 s}{y} * \frac{Bq}{dps} * \frac{3.7E4 dps}{\mu Ci}$$

GRF = Ground Roughness Factor, a unitless constant (0.82) that compensates for the fact that the external exposure is not coming from an infinite flat plane (HPS 2002),

$AvCRP_{i, TP}$ = Average Combined Removal Parameter, value that adjusts the external (groundshine) dose from radionuclide *i* for radioactive decay and weathering effects which decrease the groundshine dose over the time phase under consideration (see below for details of calculation method), h,

CF_2 = Unit Conversion Factor to convert the hours to years, 1.14E-04 y/h, and

$AXP\_Dp_{groundshine}$ = Annual Exposure Parameter for Deposition, the period of time over the 1st y during which receptor is assumed to be exposed to groundshine, 2080 h, (p. 12, EPA 1989b).

Calculation of Average Combined Removal Parameter (AvCRP)

The AvCRP is integrated over the time period of interest (i.e., first year) and is calculated using the following equation:

$$AvCRP_{i,TP} = \frac{\int_{T_1}^{T_2} \left(EffXP_{i,TP} * WF_{i,TP} \right) * CF}{\int_{T_1}^{T_2} dT} = \frac{CRP_{i,TP} * CF}{T_2 - T_1}$$

$$unitless = \frac{h * \frac{y}{h}}{y}$$

Where:

$AvCRP_{i, TP}$ = Average Combined Removal Parameter, value that represents the average adjustment for the external dose (groundshine) from radionuclide *i* for radioactive decay and weathering over the time period of interest, h;

$CRP_{TP, i}$ = Combined Removal Parameter, the value which adjusts the external (groundshine) dose from radionuclide *i* for radioactive decay and weathering, h, (see Appendix D, Section D.3, for details);

CF = Unit Conversion Factor to convert the hours to years, 1.14E-04 y/h;

T_1 = Time of the beginning of the integration period, 0 y; and

T_2 = Time of the end of the integration period, 1 y.

Table 3.5c provides the total dose parameter for exposure rate (TDP_XR) values for radionuclides, in units of mrem per mR/h. TDP_XR values include the effective dose from groundshine and committed effective dose from the inhalation of resuspended material, and are adjusted for the ground roughness, radioactive decay and weathering effects. Table 3.5d provides the TDP_XR values for radionuclides, in units of mrem per mR/hr, that are adjusted only for the ground roughness and radioactive decay (i.e., weathering effects are not considered). TDP_XR values are provided for the EPA time phases (i.e., early, first year,

and second year). As described below, TDP_XR values can be used to estimate the total effective dose (TED) that an adult receptor would receive over each of the three time phases, based on the initial exposure rate reading at 1 m above the surface.

3.6.1.2 Calculation of Skin Dose from Skin Contamination over the First Year

The method below is used to estimate the skin equivalent dose that is received over the first year from radioactive contamination deposited on the surface of the skin. It is assumed that the receptor has contamination on their skin for 800 hours during the first year after deposition (EPA 1989b). It is also assumed that the fraction of the contamination on the surface (e.g., ground) that is on the receptor's skin is 0.0625 for the maximally exposed receptor and 0.00625 for the average exposed receptor (EPA 1989b). Finally using the logic presented in Section 3.6.1.1, the AvCRP and not the CRP must be used to adjust the material that is available to be deposited on the skin for radioactive decay and weathering effects.

$$H_{contam,skin,i,TP} = DRCF_{contam,skin,i,TP} * CF_1 * SSF_{contam} * AvCRP_{i,TP} * (AXP\_C_{contam} * CF)$$

$$H_{contam,i,TP} = \frac{Sv \cdot cm^2}{y \cdot Bq} * \frac{3.70E5 \; mrem \cdot m^2 \,/\, y \cdot \mu Ci}{Sv \cdot cm^2 \,/\, y \cdot Bq} * unitless * unitless * \left(h * \frac{y}{h} \right) = mrem \cdot m^2 \,/\, \mu Ci$$

Where:

$H_{contam,skin,i,\,TP}$ = Equivalent dose to the skin per unit activity of radionuclide *i* deposited on the external surface of the skin (contamination) and over the 1st-y time period, mrem·m^2/μCi;

$DRCF_{contam,skin,i,TP}$ = Dose-Rate Conversion Factor, value the represents the equivalent dose to the skin from the electrons from radionuclide *i* deposited on the external surface of the skin and over the 1st-y time period (Health Physics 53(2), p. 135-141, 1987), Sv·cm^2/y·Bq;

CF_1 = Unit Conversion Factor, 3.70E+05 mrem / μCi/m^2 per Sv / Bq/cm^2;

SSF_{contam} = Surface to Skin Factor for contamination, the fraction of the contamination on the surface (e.g., ground) that is deposited on the skin, 0.00625 or 0.0625 (unitless) (p. 32, EPA 1989b);

$AvCRP_{i,\,TP}$ = Average Combined Removal Parameter, value that represents the average adjustment for the external dose (groundshine) from radionuclide *i* for radioactive decay and weathering over the time period of interest (see Section 3.6.1.1 for details of calculation method), h;

CF = Unit Conversion Factor to convert the hours to years, 1.14E-04 y/h; and

AXP_C_{contam} = Annual Exposure Parameter for Contamination on the Skin, the period of time over the first year during which receptor is assumed to have contamination on the surface of their skin, 800 h, (p. 11, EPA 1989b).

3.6.1.3 Calculation of Total Skin Dose Conversion Factor from Ground Shine and Skin Contamination over the First Year

The DCFs for the total skin equivalent dose that is received over the first-year period from materials deposited on the ground is derived by summing the equivalent doses from groundshine and contamination on the surface of the skin.

$$H_{Total,skin,i,TP} = H_{groundshine,skin,i,TP} + H_{contamination,skin,i,TP}$$

$$mrem \cdot m^2 / \mu Ci = mrem \cdot m^2 / \mu Ci + mrem \cdot m^2 / \mu Ci$$

Where:

$H_{Total, skin,i, TP}$ = Total Equivalent dose to the skin per unit activity of radionuclide *i* deposited on the surface (e.g., ground) and over the 1st-y time period, mrem·m^2/μCi;

$H_{groundshine,skin,i,TP}$ = Equivalent dose to the skin from groundshine per unity activity of radionuclide *i* deposited on the ground and over the 1st-y time period, mrem·m^2/μCi; and

$H_{contam,skin,i, TP}$ = Equivalent dose to the skin from radioactive material deposited on the skin (contamination) per unit activity of radionuclide *i* deposited on the surface (ground) and over the 1st-y time period, mrem·m^2/μCi.

3.6.2 Calculation of Skin Dose from Groundshine and Contamination – Dose Adjusted Only for Radioactive Decay (i.e., Weathering Factor not Applied)

If desired the weathering factor (WF) can be ignored when calculating the total equivalent doses ($D_{Total, skin,i, TP}$) for Table 3-6b. To ignore the WF and to adjust the $D_{Total, skin,i, TP}$ for only radioactive decay, substitute the Effective Exposure Period (EffXP) (See Section D.3.2 of Appendix D) for the combined removal parameter (CRP) to calculate the average effective exposure period (AvEffXP) instead of the average combined removal parameter (AvCRP).

The AvEffXP is integrated over the time period of interest (i.e., first year) and is calculated using the following equation:

$$AvEffXP_{i,TP} = \frac{\int_{T_1}^{T_2} \left(EffXP_{i,TP} \right) * CF}{\int_{T_1}^{T_2} dT} = \frac{\int_{T_1}^{T_2} e^{(-T * \lambda_i)} dT * CF}{\int_{T_1}^{T_2} dT} = \frac{\left(\frac{e^{(-T_2 * \lambda_i)} - e^{(-T_1 * \lambda_i)}}{-\lambda_i} \right) * CF}{T_2 - T_1}$$

$$unitless = \frac{h * \frac{y}{h}}{y}$$

Where:

$AvEffXP_{i, TP}$ = Average Effective Exposure Period term integrated over the time period of interest (i.e., 1st-y), value that represents the average radioactive decay adjustment of the groundshine dose from radionuclide *i* over the time period of interest, unitless;

$EffXP_{i, TP}$ = Effective Exposure Period, value that adjusts the groundshine dose from radionuclide *i* for radioactive decay that occurs over the time period under consideration, h;

λ_i = Decay constant for radionuclide *i*, h^{-1};

CF = Unit Conversion Factor to convert the hours to years, 1.14E-04 y/h;

T_1 = Time of the beginning of the integration period, 0 y; and

T_2 = Time of the end of the integration period, 1 y.

3.7 Calculating Total Dose Parameter for Exposure Rate (TDP_XR) Values

TDP_XR values are provided for the EPA time phases (i.e., early, first year, second year). As described below, TDP_XR values can be used to estimate the total dose, effective (TDE) that an adult receptor would receive over each of the three time phases, based on the initial exposure rate reading at 1 m above the surface.

3.7.1 Method Used to calculate TDP_XR Values for a Deposited Radionuclide and any Short-Lived Daughter Radionuclides in Secular Equilibrium

The following method is used to calculate TDP_XR values.

$$TDP\_XR_{E,i,TP} = \frac{PAG_{E,TP}}{DRL\_Dp_{E,i,TP}} * \frac{1}{ExXF_{ground,\ E,i}}, \quad \frac{mrem}{mR/h} = \frac{mrem}{\mu Ci/m^2} * \frac{\mu Ci/m^2}{mR/h}$$

Where:

$TDP\_XR_{E, i, TP}$ = Total Dose Parameter for Exposure Rate, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of exposure rate (at a height of 1 m) of radionuclide *i*, and any short-lived daughters, mrem per mR/h;

PAG = EPA's Protective Action Guide for the time phase under consideration, mrem;

$Dp\_DRL_{E, i, TP}$ = Deposition Derived response Level, the level of activity of "marker" radionuclide *i* at which the dose from radionuclide *i*, and any short-lived daughter radionuclides, would result in a dose equal to the PAG for the time phase under consideration, (as calculated above or from Table 3-3 or 3-4 depending on whether or not weathering adjustment is desired), $\mu Ci/m^2$;

$ExXF_{ground, E, i}$ = External Exposure Factor (effective), defined below; and

TP = Time Phase, the period of time (i.e., early phase, 1st-y, 2nd-y) over which the assessment is performed.

$$ExXF_{ground,E,i} = \sum_{i}^{P+D}\left(\frac{ExDC_{E,i} * CF_1}{DXCF} * GRF\right)$$

Where:

$\sum_{i}^{P+D}$ = Represents the summation of values from the parent radionuclide (P) and any short-lived daughter radionuclide(s) (D);

$ExXF_{ground, E, i}$ = External Exposure Factor (effective), the exposure rate (adjusted for ground roughness) due to radionuclide *i*, and any short-lived daughters, per unit activity deposited on the ground, mR/h per μCi/m^2;

$ExDC_{ground, E,i}$ = External Dose Coefficient (effective), the effective dose rate from the external exposure to radionuclide *i* per unit activity deposited on the ground, Sv·m^2/s·Bq, (values from ICRP 60+ dosimetry models (DCFPAK, 2006));

DXCF = Dose to Exposure Conversion Factor, 0.7 mrem/mR (EPA, 1992, p. 7-11),

CF_1 = Unit Conversion Factor, 1.33E+13 mrem·m^2/y·μCi per Sv·m^2/s·Bq,

$$\frac{1.33E13\,mrem\cdot m^2 \big/ h\cdot\mu Ci}{Sv\cdot m^2 \big/ s\cdot Bq} = \frac{Sv\cdot m^2}{s\cdot Bq} * \frac{10^5\,mrem}{Sv} * \frac{3600\,s}{h} * \frac{Bq}{dps} * \frac{3.7E4\,dps}{\mu Ci}$$; and

GRF = Ground Roughness Factor, a unitless constant (0.82) that compensates for the fact that the external exposure is not coming from an infinite flat plane (HPS 2002).

3.7.2 Example of Using TDP_XR to Estimate the Dose from a Deposited Radionuclide and any Short-Lived Daughter Radionuclides in Secular Equilibrium

The following equation can be used to estimate the TDE that an adult receptor would receive over each of the three time phases, based on the initial exposure rate reading at 1 m above the surface.

$$TDE_{i,TP} = ExXR_i * TDP\_XR_{E,i,TP} \quad , \quad mrem = \frac{mR}{h} * \frac{mrem}{mR \big/ h}$$

Where:

$TDE_{i, TP}$ = Total Dose, Effective, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended radionuclide *i*, and any short-lived daughter radidonuclides, over the time phase of interest, mrem;

$ExXR_i$ = External exposure rate, initial exposure rate of the radionuclide *i* deposited on the ground that is adjusted for the GRF and measured at a height of 1 m, mR/h; and

$TDP\_XR_{E, i, TP}$ = Total Dose Parameter for Exposure Rate, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of exposure rate (at a height of 1 m) of radionuclide *i*, and any short-lived daughters, mrem per mR/h.

Method Used to Calculate Activity-Weighted TDP_XR Values- Dose Projection-

Table 3-5a and 3-5b provide the activity-weighted total dose parameter for exposure rate (TDP_XR) values for a radionuclide mixture which are based on the measured isotopic concentrations in samples. The TDP_XR values are for material which has been deposited on the surface (ground) and include the effective dose from groundshine and the committed effective dose from the inhalation of resuspended material. The TDP_XR values do not include the in-plume dose (i.e., inhalation, air submersion). The TDP_XR values in Table 3-5a are adjusted for the ground roughness factor (GRF), radioactive decay and weathering effects (WF). The TDP_XR values in Table 3-5b are adjusted for the GRF and radioactive decay, but are not adjusted for weathering effects.

The following steps can be used to develop $TDCP_X$ values for a radionuclide mixture and to calculate projected future doses from gamma exposure rate measurements for individual radionuclides and for radionuclide mixtures.

1. Using spectral analysis of gamma emissions from an environmental sample of deposited radioactivity, determine the relative abundance of the principal gamma emitting radionuclides. It may be necessary to analyze uniform samples/measurements from several different locations to determine if the relative concentration of each radionuclide remains constant. The results of the various samples/measurements must be identically expressed as the activity of each radionuclide per surface are (e.g., pCi/m^2). As necessary, include the radioactivity of daughter radionuclides that are in secular equilibrium with their parent radionuclide(s), making sure to adjust the daughter activities for the branching fraction.

2. Multiply the measured or inferred activity (Dp_i) of each radionuclide from Step 1 by the corresponding external exposure coefficient (ExXC) (mR/h per pCi/m^2), from Table 3-3, which has been adjusted for the GRF to determine each radionuclide's contribution to the total exposure rate (mR/h) of the mixture. Sum the results for each radionuclide to obtain the total exposure rate from the deposited radionuclide mixture.

$$ExXR_{mix} = \sum_{i}^{P+D}\left(Dp_i * ExXC_i\right), \qquad \frac{mR}{h} = \frac{pCi}{m^2} * \frac{mR/h}{pCi/m^2}$$

Where:

$\sum_{i}^{P+D}$ = Represents the summation of values from all parent radionuclides (P) and any short-lived daughters (D) in secular equilibrium in the mixture;

$ExXR_{mix}$ = External exposure rate of the mixture, activity-weighted initial exposure rate of the radionuclide mixture deposited on the ground that is adjusted for the GRF and measured at a height of 1 m, mR/h;

Dp_i = Deposited activity concentration, the measured or estimated activity of radionuclide *i*, pCi/m^2; and

$ExXC_i$ = External exposure coefficient, the external exposure coefficient of radionuclide *i* deposited on the ground that is adjusted for the GRF and measured at a height of 1 m, mR/h per pCi/m^2.

3. Multiply the initial external exposure rate (ExXR) of each radionuclide in the mixture by the radionuclide's TDP_XR for the time phase of interest (early, first year, second year) to determine the total dose, effective (TDE) that would be received by a receptor that remains in the contaminated area over the time period of interest. Sum the TDE from the individual radionuclides to determine the TDE from the deposited radionuclide mixture.

$$TDE_{mix,TP} = \sum_{i}^{P+D} (ExXR_i * TDP\_XR_{E,i,TP}) \quad , \quad mrem = \frac{mR}{h} * \frac{mrem}{mR/h}$$

Where:

$\sum_{i}^{P+D}$ = Represents the summation of values from all parent radionuclides (P) and any short-lived daughters (D) in secular equilibrium in the mixture;

$TDE_{mix,\,TP}$ = Total Dose, Effective, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended radionuclide *i*, and any short-lived daughter radionuclides, over the time phase of interest, mrem;

$ExXR_i$ = External exposure rate, initial exposure rate of the radionuclide *i* deposited on the ground that is adjusted for the GRF and measured at a height of 1 m, mR/h; and

$TDP\_XR_{E,\,i,\,TP}$ = Total Dose Parameter for Exposure Rate, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of initial exposure rate (at a height of 1 m) of radionuclide *i*, and any short-lived daughters, mrem per mR/h.

4. Divide the total dose, effective for the mixture ($TDE_{mix,\,TP}$) for the time phase of interest, from Step 3, by the activity-weighted initial exposure rate of the radionuclide mixture ($ExXR_{mix}$), from Step 2, to determine the total dose parameter for external exposure from the radionuclide mixture (TDP_XR_{mix}) for the time phase of interest.

$$TDP\_XR_{mix,TP} = \frac{TED_{mix,TP}}{ExXR_{mix}} \quad , \quad \frac{mrem}{mR/h} = \frac{mrem}{mR/h}$$

Table 3-5a and 3-5b provide the TDP_XR_i values over each of the three time phases for each radionuclide, and any short-lived daughters in secular equilibrium, in the radionuclide mixture. Table 3-5a and 3-5b also provide the TDP_XR_{mix} values for the entire radionuclide mixture over each of the three time phases.

3.7.3 Example Calculation of the Total Dose Parameter for Exposure rate (TDP_XR) for a Mixture

Considering weathering effects, calculate the TDP_XR_{mix} for the mixture in Table 3-5a for the first-year time phase. Column 7 of Table 3-5a indicates that the initial total exposure rate of the radionuclide mixture is 1.41E-4 mR/h and Column 11 indicates this initial exposure rate equates to a projected dose of 2.31E-2 mrem over the first year. Therefore, the first-year TDP_XR for the mixture is:

$$TDP\_XR_{mix,1st-y} = \frac{TDE_{mix,1st-y}}{ExXR_{mix}} = \frac{2.31E-2mrem}{1.41E-4 mr/h} = \frac{163.8mrem}{mR/h}$$

Where:

$TDP\_XR_{mix,\ 1st\text{-}y}$ = Total Dose Parameter for Exposure Rate, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the 1st-y time phase, per unit of initial exposure rate (at a height of 1 m) of radionuclide *i*, and any short-lived daughters, mrem per mR/h;

$TDE_{mix,\ 1st\text{-}y}$ = Total Dose, Effective, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended radionuclide *i*, and any short-lived daughter radionuclides, over the 1st-y time phase of interest, (from Table 3-5a) mrem; and

$ExXR_{mix}$ = External exposure rate, initial exposure rate of the radionuclide mixture deposited on the ground that is adjusted for the GRF and measured at a height of 1 m, (from Table 3-5a) mR/h.

Therefore for this mixture, the projected dose over the first-year time phase is 164 mrem for each mR/h measured at the beginning of the period.

3.7.4 Example Calculation of the Total Effective Dose (TED) for a Mixture

Any measured exposure rate at 1 m above the deposited mixture can be multiplied by the TDP_XR for the time phase of interest to estimate the dose over that time phase. Based on the mixture in Table 3-5a and considering weathering effects, estimate the receptor's dose over the first-year time phase if the initial exposure rate at the point of interest is measured to be 12 mR/h.

$$TDE_{mix,1st-y} = ExXR * TDP\_XR_{mix,1st-y} = \frac{12mR}{h} * \frac{163.8mrem}{mR/h} = 1966mrem$$

Where:

All terms are as described above.

3.7.5 Example Calculation of the Derived Response Level for Exposure Rate (DRL_XR) for a Mixture

The DRL for the mixture can be expressed in terms of the initial exposure rate. Considering weathering effects, determine the derived response level for the mixture in Table 3-5a for the first-year time phase.

$$DRL\_XR_{mix,1st-y} = \frac{PAG_{1st-y}}{TDP\_X_{mix,1st-y}} = \frac{2000mrem}{\frac{163.8mrem}{mR/h}} = \frac{12.2mR}{h}$$

Where:

$DRL\_XR_{mix,\ 1st\text{-}y}$ = Derived Response Level for Exposure Rate for the mixture over the 1st-y time phase, mR/h;

$PAG_{1st\text{-}Y}$ = Applicable PAG for the 1st-y time phase; and

$TDP\_XR_{mix,\ 1st\text{-}y}$ = Total Dose Parameter for Exposure Rate, the sum of the effective dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the 1st-y time phase, per unit of initial exposure rate (at a height of 1 m) of radionuclide *i*, and any short-lived daughters, (from Table 3-5a) mrem per mR/h.

Therefore, an initial exposure rate reading of 12.2 mR/h represents the boundary for the 2 rem (20 mSv) relocation PAG.

3.8 Applying the Protective Action Guides for Relocation

Establishing the boundary of a relocation area may result in 3 different types of actions:

1. Persons who, based on the guidance described in Chapter 2, have already been evacuated from an area that is now designated as a relocation area must be assigned relocation status.
2. Persons not previously evacuated but who reside inside the relocation area should relocate.
3. Persons who normally reside outside the relocation area, but were previously evacuated, may return. A gradual return is recommended, as discussed in Chapter 6.

Small adjustments to the boundary of the relocation area established based on the PAG may be justified on the basis of difficulty or ease of implementation. For example, the use of a convenient natural boundary could be a logical reason for adjustment of the relocation area. However, such decisions should be supported by demonstration that exposure rates to persons not relocated can be promptly reduced by methods other than relocation to meet the PAG, as well as the longer dose objectives addressed in Section 3.2.1.

The relocation PAG applies principally to personal residences but may impact other facilities as well. For example, it could impact work locations, hospitals, and park lands as well as the use of highways and other transportation facilities. For each type of facility, the occupancy time of individuals should be taken into account to determine the criteria for using a facility or area. It might be necessary to avoid continuous use of homes in an area because radiation levels are too high. However, a factory or office building in the same area could be used because occupancy times are shorter. Similarly, a highway could be used at higher contamination levels because the exposure time of highway users would be considerably less than the time spent at home.

3.8.1 Exposure Limits for Persons Reentering the Relocation Area

Individuals permitted to reenter a relocation area to work, or for other justified reasons, will require protection from radiation. Such individuals should enter the relocation area under controlled conditions in accordance with dose limitations and other procedures for control of occupationally exposed workers (EPA

1987). Ongoing doses received by these individuals from living in a contaminated area outside the relocation area need not be included as part of this dose limitation applicable to workers. In addition, dose received previously from the plume and associated with groundshine, during the early phase of the radiological incident, need not be considered

After the relocation area is established, persons will need to reenter for a variety of reasons, including recovery activities, retrieval of property, security patrol, operation of vital services, and, in some cases, care and feeding of farm and other animals. It may be possible to quickly decontaminate access ways to vital institutions and businesses in certain areas so that they can be occupied by adults either for living (i.e., institutions such as nursing homes, and hospitals) or for employment. Clearance of these areas for such occupancy will require dose reduction to comply with occupational exposure limits (EPA 1987). Dose projections should include both external exposure from deposited material and inhalation of resuspended deposited material for the duration of the planned exposure. Persons working in areas inside the emergency relocation area should operate under the controlled conditions normally established for occupational exposure (EPA 1987).

Table 3-6a. Calculation of Total Skin Dose from Contamination on the Ground (Correcting for Radioactive Decay and Weathering Effects)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[1] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRF (mrem/y per µCi/m^2)	[2]CRP for first year (corrects for rad. decay & WCF) (h)	[4]External Skin DCF (corrected for GRCF, rad. decay & WCF) (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[3] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[5]Electron dose-rate factors for skin contam. contamination corrected for WCF & decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
Co-58	NA	1.14E-15	1.33E+02	1.09E+02	2.28E+03	6.73E+00	2.80E-03	1.04E+03	6.48E+00	1.54E-01	6.89E+00	6.89E-06		1.20E-01	1.20E-07	57.39
Co-60	NA	2.76E-15	3.22E+02	2.64E+02	7.54E+03	5.39E+01	9.90E-03	3.66E+03	2.29E+01	1.80E+00	5.57E+01	5.57E-05		4.20E-01	4.20E-07	132.64
Se-75	NA	4.76E-16	5.55E+01	4.55E+01	3.44E+03	4.24E+00	8.40E-04	3.11E+02	1.94E+00	6.97E-02	4.31E+00	4.31E-06		NA	NA	NA
Rb-86	NA	7.72E-15	9.00E+02	7.38E+02	6.37E+02	1.27E+01	2.00E-02	7.40E+03	4.63E+01	3.07E-01	1.30E+01	1.30E-05		6.30E+01	6.30E-05	0.21
Kr-87	NA	1.35E-14	1.57E+03	1.29E+03	1.83E+00	6.40E-02	No data available	0.00E+00	0.00E+00	0.00E+00	6.40E-02	6.40E-08		NA	NA	NA
Kr-88	Na	4.43E-15	5.16E+02	4.23E+02	4.10E+00	4.71E-02	No data available	0.00E+00	0.00E+00	0.00E+00	4.71E-02	4.71E-08		NA	NA	NA
Sr-89	NA	6.66E-15	7.76E+02	6.37E+02	1.68E+03	2.90E+01	2.00E-02	7.40E+03	4.63E+01	8.10E-01	2.98E+01	2.98E-05		1.50E+02	1.50E-04	0.20
Sr-90	NA	1.40E-16	1.63E+01	1.34E+01	7.94E+03	2.88E+00	1.60E-02	5.92E+03	3.70E+01	3.06E+00	5.94E+00	5.94E-06		1.20E+01	1.20E-05	0.50
Y-90	1	1.05E-14	1.22E+03	1.00E+03	7.94E+03	2.16E+02	2.10E-02	7.77E+03	4.86E+01	4.02E+00	2.20E+02	2.20E-04				
Sr90/Y90	NA												2.26E-04	NA	NA	NA
Y-90, parent	NA	1.05E-14	1.22E+03	1.00E+03	9.21E+01	2.51E+00	2.10E-02	7.77E+03	4.86E+01	4.66E-02	2.55E+00	2.55E-06		2.20E+02	2.20E-04	0.01
Sr-91	NA	7.53E-15	8.78E+02	7.20E+02	1.37E+01	2.67E-01	2.00E-02	7.40E+03	4.63E+01	6.61E-03	2.74E-01	2.74E-07		NA	NA	NA
Y-91	NA	6.92E-15	8.07E+02	6.61E+02	1.92E+03	3.44E+01	2.00E-02	7.40E+03	4.63E+01	9.26E-01	3.53E+01	3.53E-05		1.60E+02	1.60E-04	0.22
Zr-95	NA	8.91E-16	1.04E+02	8.52E+01	2.08E+03	4.80E+00	1.20E-02	4.44E+03	2.78E+01	6.02E-01	5.40E+00	5.40E-06		7.20E-01	7.20E-07	7.50
Nb-95	0.993	9.05E-16	1.05E+02	8.65E+01	2.08E+03	4.88E+00	2.30E-03	8.51E+02	5.32E+00	1.15E-01	4.99E+00	4.99E-06		NA	NA	NA
Nb-95m	6.98E-03	1.09E-16	1.27E+01	1.04E+01	2.08E+03	5.87E-01	1.60E-02	5.92E+03	3.70E+01	8.02E-01	1.39E+00	1.39E-06		NA	NA	NA
Nb-95	1	9.05E-16	1.05E+02	8.65E+01	2.08E+03	4.88E+00	2.30E-03	8.51E+02	5.32E+00	1.15E-01	4.99E+00	4.99E-06		NA	NA	NA
Zr95/Nb95/Nb95m/Nb95													1.04E-05	NA	NA	NA
Nb-95, parent		9.05E-16	1.05E+02	8.65E+01	1.19E+03	2.79E+00	2.30E-03	8.51E+02	5.32E+00	6.60E-02	2.86E+00	2.86E-06		6.10E-01	6.10E-07	4.68

Table 3-6a. Calculation of Total Skin Dose from Contamination on the Ground (Correcting for Radioactive Decay and Weathering Effects) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[1] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRF (mrem/y per µCi/m^2)	[2]CRP for first year (corrects for rad. decay & WCF) (h)	[4]External Skin DCF (corrected for GRCF, rad. decay & WCF) (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[3] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[5]Electron dose-rate factors for skin contam. contamination corrected for WCF & decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
Mo-99	NA	3.76E-15	4.38E+02	3.59E+02	9.50E+01	9.25E-01	1.90E-02	7.03E+03	4.39E+01	4.35E-02	9.69E-01	9.69E-07		4.40E+00	4.40E-06	0.22
Tc-99m	0.876	1.44E-16	1.68E+01	1.38E+01	9.50E+01	3.54E-02	2.10E-03	7.77E+02	4.86E+00	4.81E-03	4.02E-02	4.02E-08				
Tc99	1	Note: Tc99 not considered because its T1/2= 2.13E5 y and is too long to be in equilibrium														
Tc99	0.124	Note: Tc99 not considered because its T1/2= 2.13E5 y and is too long to be in equilibrium														
Mo99/Tc99m													1.00E-06	NA	NA	NA
Tc-99m, parent		1.44E-16	1.68E+01	1.38E+01	8.68E+00	3.24E-03	2.10E-03	7.77E+02	4.86E+00	4.39E-04	3.68E-03	3.68E-09		7.70E-03	7.70E-09	0.48
Rh-105	NA	1.76E-16	2.05E+01	1.68E+01	5.10E+01	2.33E-02	1.30E-02	4.81E+03	3.01E+01	1.60E-02	3.92E-02	3.92E-08		6.50E-02	6.50E-08	0.60
Ru-103	NA	6.16E-16	7.18E+01	5.89E+01	1.32E+03	2.11E+00	5.80E-03	2.15E+03	1.34E+01	1.85E-01	2.29E+00	2.29E-06		6.80E-01	6.80E-07	3.37
Ru-106	NA	0.00E+00	0.00E+00	0.00E+00	5.86E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		NA	NA	NA
Rh-106	1	1.42E-14	1.66E+03	1.36E+03	5.86E+03	2.16E+02	2.20E-02	8.14E+03	5.09E+01	3.11E+00	2.19E+02	2.19E-04		NA	NA	NA
Ru106/Rh106	NA												2.19E-04	6.40E-01	6.40E-07	341.67
Rh-106, parent		1.42E-14	1.66E+03	1.36E+03	1.20E-02	4.41E-04	2.20E-02	8.14E+03	5.09E+01	6.36E-06	4.48E-04	4.48E-10		NA	NA	NA
Sb-127	NA	2.85E-15	3.32E+02	2.72E+02	1.33E+02	9.82E-01	1.80E-02	6.66E+03	4.16E+01	5.77E-02	1.04E+00	1.04E-06		3.40E+00	3.40E-06	0.31
Te-127	0.824	5.40E-16	6.29E+01	5.16E+01	1.33E+02	1.86E-01	1.60E-02	5.92E+03	3.70E+01	5.13E-02	2.37E-01	2.37E-07				
Sb127/Te127													1.24E-06	NA	NA	NA
Te-127, parent	NA	5.40E-16	6.29E+01	5.16E+01	1.35E+01	1.89E-02	1.60E-02	5.92E+03	3.70E+01	5.21E-03	2.41E-02	2.41E-08		1.00E+00	1.00E-06	0.02
Te-127m	NA	5.20E-17	6.06E+00	4.97E+00	3.21E+03	4.32E-01	4.70E-03	1.74E+03	1.09E+01	3.64E-01	7.96E-01	7.96E-07		7.80E-01	7.80E-07	1.02

Table 3-6a. Calculation of Total Skin Dose from Contamination on the Ground (Correcting for Radioactive Decay and Weathering Effects) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRF (mrem/y per µCi/m^2)	[b]CRP for first year (corrects for rad. decay & WCF) (hr)	[d]External Skin DCF (corrected for GRCF, rad. decay & WCF) (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron dose-rate factors for skin contam. contamination corrected for WCF & decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
Te-127	0.976	5.40E-16	6.29E+01	5.16E+01	3.21E+03	4.49E+00	1.60E-02	5.92E+03	3.70E+01	1.24E+00	5.73E+00	5.73E-06		NA	NA	NA
Te127m/Te127													6.39E-06	NA	NA	NA
Sb-129	NA	5.10E-15	5.94E+02	4.87E+02	6.23E+00	8.23E-02	1.70E-02	6.29E+03	3.93E+01	2.55E-03	8.49E-02	8.49E-08		NA	NA	NA
Te-129	0.775	5.74E-15	6.69E+02	5.49E+02	6.23E+00	9.26E-02	2.00E-02	7.40E+03	4.63E+01	3.00E-03	9.56E-02	9.56E-08		NA	NA	NA
Sb129/Te129													1.59E-07	NA	NA	NA
Te-129m	NA	2.27E-15	2.65E+02	2.17E+02	1.13E+03	6.64E+00	1.30E-02	4.81E+03	3.01E+01	3.54E-01	7.00E+00	7.00E-06		3.40E+01	3.40E-05	0.21
Te-129	0.65	5.74E-15	6.69E+02	5.49E+02	1.13E+03	1.68E+01	2.00E-02	7.40E+03	4.63E+01	5.45E-01	1.73E+01	1.73E-05		NA	NA	NA
I-129	0.35	Note: I-129 not considered because its T1/2 = 1.57E7 y and is too long to be in equilibrium														
Te129m/Te129	NA												1.83E-05	NA	NA	NA
Te-129, parent		5.74E-15	6.69E+02	5.49E+02	1.67E+00	2.48E-02	2.00E-02	7.40E+03	4.63E+01	8.05E-04	2.56E-02	2.56E-08		5.00E-01	5.00E-07	0.05
Te-131m	NA	2.20E-15	2.56E+02	2.10E+02	4.32E+01	2.46E-01	1.50E-02	5.55E+03	3.47E+01	1.56E-02	2.62E-01	2.62E-07		2.90E-01	2.90E-07	0.90
I-131	0.778	Note: I-131 not considered because its T1/2 = 8.04 d and is longer than Te131m T1/2 of 1.25 d.														
Xe-131m	1.11E-02	Note: Xe131m not considered because it is I131 progeny (and its T1/2 = 11.9 d and is longer than Te131m T1/2 of 1.25 d.														
Te-131	0.222	8.36E-15	9.74E+02	7.99E+02	4.32E+01	9.36E-01	2.30E-02	8.51E+03	5.32E+01	2.40E-02	9.60E-01	9.60E-07				
I-131	1	Note: I-131 not considered because its T1/2 = 8.04 d and is longer than Te131m T1/2 of 1.25 d.														
Xe-131m	1.11E-02	Note: Xe-131m not considered because it is I-131 progeny (and its T1/2 = 11.9 d and is longer than Te-131m[e] T1/2 of 1.25 d.														
[e]Te131m/Te131													4.75E-07	NA	NA	NA
Te-132	NA	2.99E-16	3.48E+01	2.86E+01	1.13E+02	8.75E-02	7.00E-03	2.59E+03	1.62E+01	1.91E-02	1.07E-01	1.07E-07		5.40E-03	5.40E-09	19.74

Table 3-6a. Calculation of Total Skin Dose from Contamination on the Ground (Correcting for Radioactive Decay and Weathering Effects) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRF (mrem/y per µCi/m^2)	[b]CRP for first year (corrects for rad. decay & WCF) (h)	[d]External Skin DCF (corrected for GRCF, rad. decay & WCF) (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron dose-rate factors for skin contam. contamination corrected for WCF & decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
I-132	1	7.54E-15	8.79E+02	7.21E+02	1.13E+02	2.21E+00	1.90E-02	7.03E+03	4.39E+01	5.18E-02	2.26E+00	2.26E-06		NA	NA	NA
Te132/I132	NA												2.37E-06	NA	NA	NA
I-132, parent		7.54E-15	8.79E+02	7.21E+02	3.32E+00	6.48E-02	1.90E-02	7.03E+03	4.39E+01	1.52E-03	6.64E-02	6.64E-08		5.00E+01	5.00E-05	0.00
I-131	NA	6.43E-16	7.49E+01	6.15E+01	2.77E+02	4.61E-01	1.50E-02	5.55E+03	3.47E+01	1.00E-01	5.62E-01	5.62E-07		8.50E-01	8.50E-07	0.66
I-133	NA	4.55E-15	5.30E+02	4.35E+02	3.00E+01	3.54E-01	1.90E-02	7.03E+03	4.39E+01	1.37E-02	3.67E-01	3.67E-07		NA	NA	NA
[f]Xe-133	NA	6.93E-17	8.08E+00	6.62E+00	1.81E+02	3.25E-02	No data available	0.00E+00	0.00E+00	0.00E+00	3.25E-02	3.25E-08		NA	NA	NA
I-134	NA	9.85E-15	1.15E+03	9.41E+02	1.26E+00	3.22E-02	2.00E-02	7.40E+03	4.63E+01	6.08E-04	3.28E-02	3.28E-08		NA	NA	NA
I-135	NA	4.83E-15	5.63E+02	4.62E+02	9.53E+00	1.19E-01	1.80E-02	6.66E+03	4.16E+01	4.14E-03	1.23E-01	1.23E-07		NA	NA	NA
Cs-134	NA	2.17E-15	2.53E+02	2.07E+02	6.85E+03	3.85E+01	1.20E-02	4.44E+03	2.78E+01	1.98E+00	4.05E+01	4.05E-05		2.60E+01	2.60E-05	1.56
[f]Xe-135	NA	2.09E-15	2.44E+02	2.00E+02	1.31E+01	7.09E-02	No data available	0.00E+00	0.00E+00	0.00E+00	7.09E-02	7.09E-08		NA	NA	NA
Cs-136	NA	2.54E-15	2.96E+02	2.43E+02	4.49E+02	2.95E+00	1.30E-02	4.81E+03	3.01E+01	1.41E-01	3.10E+00	3.10E-06		1.40E-01	1.40E-07	22.11
Cs-137	NA	2.75E-16	3.21E+01	2.63E+01	7.94E+03	5.66E+00	1.40E-02	5.18E+03	3.24E+01	2.68E+00	8.34E+00	8.34E-06		NA	NA	NA
Ba-137m	0.946	1.65E-15	1.92E+02	1.58E+02	7.94E+03	3.39E+01	2.10E-03	7.77E+02	4.86E+00	4.02E-01	3.43E+01	3.43E-05		NA	NA	NA
Cs-137/Ba-137m													4.08E-05	2.10E+01	2.10E-05	1.94
Ba-137m, parent		1.65E-15	1.92E+02	1.58E+02	6.14E-02	2.62E-04	2.10E-03	7.77E+02	4.86E+00	3.11E-06	2.66E-04	2.66E-10		NA	NA	NA
[f]Xe-138	NA	7.65E-15	8.92E+02	7.31E+02	3.41E-01	6.76E-03	No data available				6.76E-03	6.76E-09		NA	NA	NA
Ba-140	NA	1.95E-15	2.27E+02	1.86E+02	4.37E+02	2.21E+00	1.70E-02	6.29E+03	3.93E+01	1.79E-01	2.39E+00	2.39E-06		9.10E+00	9.10E-06	0.26
La-140	1	8.24E-15	9.60E+02	7.88E+02	4.37E+02	9.33E+00	2.00E-02	7.40E+03	4.63E+01	2.11E-01	9.54E+00	9.54E-06		NA	NA	NA

Table 3-6a. Calculation of Total Skin Dose from Contamination on the Ground (Correcting for Radioactive Decay and Weathering Effects) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRF (mrem/y per µCi/m^2)	[b]CRP for first year (corrects for rad. decay & WCF) (h)	[d]External Skin DCF (corrected for GRCF, rad. decay & WCF) (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron dose-rate factors for skin contam. contamination corrected for WCF & decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
Ba-140/La-140													1.19E-05	NA	NA	NA
La-140	NA	8.24E-15	9.60E+02	7.88E+02	5.80E+01	1.24E+00	2.00E-02	7.40E+03	4.63E+01	2.80E-02	1.27E+00	1.27E-06		1.20E+01	1.20E-05	0.11
Ce-141	NA	1.32E-16	1.54E+01	1.26E+01	1.10E+03	3.76E-01	1.70E-02	6.29E+03	3.93E+01	4.51E-01	8.27E-01	8.27E-07		6.60E-01	6.60E-07	1.25
Ce-143	NA	3.99E-15	4.65E+02	3.81E+02	4.76E+01	4.92E-01	1.90E-02	7.03E+03	4.39E+01	2.18E-02	5.14E-01	5.14E-07		2.30E+00	2.30E-06	0.22
Ce-144	NA	2.61E-17	3.04E+00	2.49E+00	5.38E+03	3.64E-01	8.90E-03	3.29E+03	2.06E+01	1.15E+00	1.52E+00	1.52E-06		NA	NA	NA
Pr-144	9.82E-01	1.27E-14	1.48E+03	1.21E+03	5.38E+03	1.77E+02	2.20E-02	8.14E+03	5.09E+01	2.85E+00	1.80E+02	1.80E-04		NA	NA	NA
Pr-144m	1.78E-02	2.67E-17	3.11E+00	2.55E+00	5.38E+03	3.72E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.72E-01	3.72E-07		NA	NA	NA
Pr-144	9.99E-01	1.27E-14	1.48E+03	1.21E+03	5.38E+03	1.77E+02	2.20E-02	8.14E+03	5.09E+01	2.85E+00	1.80E+02	1.80E-04		NA	NA	NA
Ce-144/Pr-144/Pr-144m													1.81E-04	8.70E-01	8.70E-07	208.43
Pr-143	NA	2.00E-15	2.33E+02	1.91E+02	4.65E+02	2.41E+00	1.80E-02	6.66E+03	4.16E+01	2.02E-01	2.61E+00	2.61E-06		1.30E+01	1.30E-05	0.20
Pm-147	NA	1.20E-19	1.40E-02	1.15E-02	7.08E+03	2.20E-03	5.40E-03	2.00E+03	1.25E+01	9.22E-01	9.24E-01	9.24E-07		NA	NA	NA
Nd-147	NA	1.10E-15	1.28E+02	1.05E+02	3.77E+02	1.07E+00	1.70E-02	6.29E+03	3.93E+01	1.55E-01	1.23E+00	1.23E-06		4.30E+00	4.30E-06	0.29
Gd-153	NA	1.41E-16	1.64E+01	1.35E+01	5.05E+03	1.84E+00	1.10E-03	4.07E+02	2.54E+00	1.34E-01	1.98E+00	1.98E-06		NA	NA	NA
Yb-169	AN	3.66E-16	4.27E+01	3.50E+01	1.08E+03	1.02E+00	8.80E-03	3.26E+03	2.04E+01	2.29E-01	1.25E+00	1.25E-06		NA	NA	NA
[f]Tm-170	NA	2.12E-15	2.47E+02	2.03E+02	3.60E+03	1.98E+01	No data available	0.00E+00	0.00E+00	0.00E+00	1.98E+01	1.98E-05		NA	NA	NA
Ir-192	NA	1.21E-15	1.41E+02	1.16E+02	2.37E+03	7.43E+00	1.70E-02	6.29E+03	3.93E+01	9.71E-01	8.40E+00	8.40E-06		NA	NA	NA
Ra-226	NA	8.12E-18	9.46E-01	7.76E-01	8.03E+03	1.69E-01	4.20E-04	1.55E+02	9.71E-01	8.13E-02	2.50E-01	2.50E-07		NA	NA	NA
Rn-222	1.00E+00	5.20E-19	6.06E-02	4.97E-02	8.03E+03	1.08E-02	No data available	0.00E+00	0.00E+00	0.00E+00	1.08E-02	1.08E-08		NA	NA	NA
Po-218	1.00E+00	1.17E-20	1.36E-03	1.12E-03	8.03E+03	2.43E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.43E-04	2.43E-10		NA	NA	NA

Table 3-6a. Calculation of Total Skin Dose from Contamination on the Ground (Correcting for Radioactive Decay and Weathering Effects) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRF (mrem/y per µCi/m^2)	[b]CRP for first year (corrects for rad. decay & WCF) (hr)	[d]External Skin DCF (corrected for GRCF, rad. decay & WCF) (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron Dose-rate factors for skin contam. contamination corrected for WCF & decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value divided by EPA Value
Pb-214	1.00E+00	9.10E-16	1.06E+02	8.70E+01	8.03E+03	1.89E+01	2.20E-02	8.14E+03	5.09E+01	4.26E+00	2.32E+01	2.32E-05		NA	NA	NA
Bi-214	1.00E+00	8.48E-15	9.88E+02	8.10E+02	8.03E+03	1.76E+02	2.00E-02	7.40E+03	4.63E+01	3.87E+00	1.80E+02	1.80E-04		NA	NA	NA
Po-214	1.00E+00	1.09E-19	1.27E-02	1.04E-02	8.03E+03	2.27E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.27E-03	2.27E-09		NA	NA	NA
At-218	2.00E-04	2.32E-17	2.70E+00	2.22E+00	8.03E+03	4.83E-01	No data available	0.00E+00	0.00E+00	0.00E+00	4.83E-01	4.83E-07		NA	NA	NA
Bi-214	1.00E+00	8.48E-15	9.88E+02	8.10E+02	8.03E+03	1.76E+02	2.00E-02	7.40E+03	4.63E+01	3.87E+00	1.80E+02	1.80E-04		NA	NA	NA
Po-214	1.00E+00	1.09E-19	1.27E-02	1.04E-02	8.03E+03	2.27E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.27E-03	2.27E-09		NA	NA	NA
[f]Ra-226/Rn-222/Po-218/At-218													2.04E-04	NA	NA	NA
Pu-238	NA	9.64E-18	1.12E+00	9.21E-01	8.00E+03	2.00E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.00E-01	2.00E-07		NA	NA	NA
Pu-239	NA	3.67E-18	4.28E-01	3.51E-01	8.03E+03	7.63E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.63E-02	7.63E-08		NA	NA	NA
Np-239	NA	2.63E-16	3.07E+01	2.51E+01	8.14E+01	5.55E-02	2.30E-02	8.51E+03	5.32E+01	4.51E-02	1.01E-01	1.01E-07		3.40E-02	3.40E-08	2.96
Am-241	NA	8.32E-17	9.70E+00	7.95E+00	8.02E+03	1.73E+00	2.20E-05	8.14E+00	5.09E-02	4.25E-03	1.73E+00	1.73E-06		4.60E-02	4.60E-08	37.67
Cm-244	NA	8.70E-18	1.01E+00	8.31E-01	7.88E+03	1.78E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.78E-01	1.78E-07		NA	NA	NA
Cf-252	NA	5.75E-18	6.70E-01	5.50E-01	7.09E+03	1.06E-01	4.80E-06	1.78E+00	1.11E-02	8.20E-04	1.06E-01	1.06E-07		NA	NA	NA

[a]From FGR Report #12, Table III.3 (EPA 1993)
[b]From Turbo FRMAC 2.0, RFC 2 (DCFPAK, K. Eckerman).
[c]From Kocher & Eckermann, Health Physics 53(2), p. 135 - 141, 1987)
[d]Assume exposure period to groundshine = 2080 h/y.
[e]Assume exposure period to contamination uniformly distributed on the skin = 800 h/y.
[f]Total does not include dose from contamination on the skin if Electron Dose-Rate Factors not available.
[e]Te-131m values in the tables are subject to change pending further development of new parent-daughter rules.

Table 3-6b. Calculation of Total Skin Dose from Contamination on the Ground (Correcting only for Radioactive Decay)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRCF (mrem/y per µCi/m^2)	First-year EffXP[b] from TF 2.0, 2 - only corrects for decay TF 2.0, 2 (h)	[d]External Skin DCF corrected for GRF, WCF & decay (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron dose-rate factors for skin contam. contamination corrected for WCF and decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value divided by EPA Value
Co-58	NA	1.14E-15	1.33E+02	1.09E+02	2.38E+03	7.03E+00	2.80E-03	1.04E+03	6.48E+00	1.61E-01	7.19E+00	7.19E-06		1.47E-01	1.47E-07	4.89E+01
Co-60	NA	2.76E-15	3.22E+02	2.64E+02	8.21E+03	5.87E+01	9.90E-03	3.66E+03	2.29E+01	1.96E+00	6.07E+01	6.07E-05		5.60E-01	5.60E-07	1.08E+02
Se-75	NA	4.76E-16	5.55E+01	4.55E+01	3.65E+03	4.50E+00	8.40E-04	3.11E+02	1.94E+00	7.39E-02	4.57E+00	4.57E-06		NA	NA	NA
Rb-86	NA	7.72E-15	9.00E+02	7.38E+02	6.46E+02	1.29E+01	2.00E-02	7.40E+03	4.63E+01	3.11E-01	1.32E+01	1.32E-05		6.70E+01	6.70E-05	1.97E-01
Kr-87	NA	1.35E-14	1.57E+03	1.29E+03	1.83E+00	6.40E-02	No data available	0.00E+00	0.00E+00	0.00E+00	6.40E-02	6.40E-08		NA	NA	NA
Kr-88	Na	4.43E-15	5.16E+02	4.23E+02	4.10E+00	4.71E-02	No data available	0.00E+00	0.00E+00	0.00E+00	4.71E-02	4.71E-08		NA	NA	NA
Sr-89	NA	6.66E-15	7.76E+02	6.37E+02	1.74E+03	3.00E+01	2.00E-02	7.40E+03	4.63E+01	8.39E-01	3.09E+01	3.09E-05		1.60E+02	1.60E-04	1.93E-01
Sr-90	NA	1.40E-16	1.63E+01	1.34E+01	8.66E+03	3.14E+00	1.60E-02	5.92E+03	3.70E+01	3.34E+00	6.48E+00	6.48E-06		1.70E+01	1.70E-05	3.81E-01
Y-90	1	1.05E-14	1.22E+03	1.00E+03	8.66E+03	2.36E+02	2.10E-02	7.77E+03	4.86E+01	4.38E+00	2.40E+02	2.40E-04				
Sr90/Y90	NA												2.46E-04	NA	NA	NA
Y-90, parent	NA	1.05E-14	1.22E+03	1.00E+03	9.23E+01	2.51E+00	2.10E-02	7.77E+03	4.86E+01	4.67E-02	2.56E+00	2.56E-06		2.90E+02	2.90E-04	8.82E-03
Sr-91	NA	7.53E-15	8.78E+02	7.20E+02	1.37E+01	2.67E-01	2.00E-02	7.40E+03	4.63E+01	6.61E-03	2.74E-01	2.74E-07		NA	NA	NA
Y-91	NA	6.92E-15	8.07E+02	6.61E+02	2.00E+03	3.59E+01	2.00E-02	7.40E+03	4.63E+01	9.64E-01	3.68E+01	3.68E-05		1.90E+02	1.90E-04	1.94E-01
Zr-95	NA	8.91E-16	1.04E+02	8.52E+01	2.17E+03	5.01E+00	1.20E-02	4.44E+03	2.78E+01	6.28E-01	5.64E+00	5.64E-06		8.30E-01	8.30E-07	6.79E+00
Nb-95	0.993	9.05E-16	1.05E+02	8.65E+01	2.17E+03	5.09E+00	2.30E-03	8.51E+02	5.32E+00	1.20E-01	5.21E+00	5.21E-06		NA	NA	NA
Nb-95m	6.98E-03	1.09E-16	1.27E+01	1.04E+01	2.17E+03	6.13E-01	1.60E-02	5.92E+03	3.70E+01	8.37E-01	1.45E+00	1.45E-06		NA	NA	NA
Nb-95	1	9.05E-16	1.05E+02	8.65E+01	2.17E+03	5.09E+00	2.30E-03	8.51E+02	5.32E+00	1.20E-01	5.21E+00	5.21E-06		NA	NA	NA
Zr-95/Nb-95/Nb-95m/Nb-95													1.09E-05	NA	NA	NA
Nb-95, parent		9.05E-16	1.05E+02	8.65E+01	1.22E+03	2.86E+00	2.30E-03	8.51E+02	5.32E+00	6.76E-02	2.93E+00	2.93E-06		7.40E-01	7.40E-07	3.96E+00
Mo-99	NA	3.76E-15	4.38E+02	3.59E+02	9.52E+01	9.27E-01	1.90E-02	7.03E+03	4.39E+01	4.36E-02	9.71E-01	9.71E-07		4.60E+00	4.60E-06	2.11E-01

Table 3-6b. Calculation of Total Skin Dose from Contamination on the Ground (Correcting only for Radioactive Decay) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRCF (mrem/y per µCi/m^2)	First-year EffXP[b] from TF 2.0, 2 - only corrects for decay TF 2.0, 2 (h)	[d]External Skin DCF corrected for GRF, WCF & decay (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron dose-rate factors for skin contam. contamination corrected for WCF and decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
Tc-99m	0.876	1.44E-16	1.68E+01	1.38E+01	9.52E+01	3.55E-02	2.10E-03	7.77E+02	4.86E+00	4.82E-03	4.03E-02	4.03E-08				
Tc-99	1	Note: Tc-99 not considered because its T1/2= 2.13E5 y and is too long to be in equilibrium														
Tc-99	0.124	Note: Tc-99 not considered because its T1/2= 2.13E5 y and is too long to be in equilibrium														
Mo-99/Tc-99m													1.01E-06	NA	NA	NA
Tc-99m, parent		1.44E-16	1.68E+01	1.38E+01	8.69E+00	3.24E-03	2.10E-03	7.77E+02	4.86E+00	4.40E-04	3.68E-03	3.68E-09		7.70E-03	7.70E-09	4.78E-01
Rh-105	NA	1.76E-16	2.05E+01	1.68E+01	5.10E+01	2.33E-02	1.30E-02	4.81E+03	3.01E+01	1.60E-02	3.92E-02	3.92E-08		6.60E-02	6.60E-08	5.94E-01
Ru-103	NA	6.16E-16	7.18E+01	5.89E+01	1.36E+03	2.17E+00	5.80E-03	2.15E+03	1.34E+01	1.90E-01	2.36E+00	2.36E-06		7.80E-01	7.80E-07	3.03E+00
Ru-106	NA	0.00E+00	0.00E+00	0.00E+00	6.34E+03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00		NA	NA	NA
Rh-106	1	1.42E-14	1.66E+03	1.36E+03	6.34E+03	2.33E+02	2.20E-02	8.14E+03	5.09E+01	3.36E+00	2.37E+02	2.37E-04		NA	NA	NA
Ru-106/Rh-106	NA												2.37E-04	8.70E-01	8.70E-07	2.72E+02
Rh-106, parent		1.42E-14	1.66E+03	1.36E+03	1.20E-02	4.41E-04	2.20E-02	8.14E+03	5.09E+01	6.36E-06	4.48E-04	4.48E-10		NA	NA	NA
Sb-127	NA	2.85E-15	3.32E+02	2.72E+02	1.33E+02	9.82E-01	1.80E-02	6.66E+03	4.16E+01	5.77E-02	1.04E+00	1.04E-06		3.40E+00	3.40E-06	3.06E-01
Te-127	0.824	5.40E-16	6.29E+01	5.16E+01	1.33E+02	1.86E-01	1.60E-02	5.92E+03	3.70E+01	5.13E-02	2.37E-01	2.37E-07		NA	NA	NA
Sb-127/Te-127													1.24E-06	NA	NA	NA
Te-127, parent	NA	5.40E-16	6.29E+01	5.16E+01	1.35E+01	1.89E-02	1.60E-02	5.92E+03	3.70E+01	5.21E-03	2.41E-02	2.41E-08		1.00E+00	1.00E-06	2.41E-02
Te-127m	NA	5.20E-17	6.06E+00	4.97E+00	3.40E+03	4.58E-01	4.70E-03	1.74E+03	1.09E+01	3.85E-01	8.43E-01	8.43E-07		9.50E-01	9.50E-07	8.88E-01
Te-127	0.976	5.40E-16	6.29E+01	5.16E+01	3.40E+03	4.76E+00	1.60E-02	5.92E+03	3.70E+01	1.31E+00	6.07E+00	6.07E-06		NA	NA	NA
Te-127m/Te-127													6.77E-06	NA	NA	NA

Table 3-6b. Calculation of Total Skin Dose from Contamination on the Ground (Correcting only for Radioactive Decay) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRCF (mrem/y per µCi/m^2)	First-year EffXP[b] from TF 2.0, 2 - only corrects for decay TF 2.0, 2 (h)	[d]External Skin DCF corrected for GRF, WCF & decay (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron dose-rate factors for skin contam. contamination corrected for WCF and decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
Sb-129	NA	5.10E-15	5.94E+02	4.87E+02	6.23E+00	8.23E-02	1.70E-02	6.29E+03	3.93E+01	2.55E-03	8.49E-02	8.49E-08		NA	NA	NA
Te-129	0.775	5.74E-15	6.69E+02	5.49E+02	6.23E+00	9.26E-02	2.00E-02	7.40E+03	4.63E+01	3.00E-03	9.56E-02	9.56E-08		NA	NA	NA
Sb-129/Te-129													1.59E-07	NA	NA	NA
Te-129m	NA	2.27E-15	2.65E+02	2.17E+02	1.16E+03	6.82E+00	1.30E-02	4.81E+03	3.01E+01	3.64E-01	7.18E+00	7.18E-06		3.60E+01	3.60E-05	2.00E-01
Te-129	0.65	5.74E-15	6.69E+02	5.49E+02	1.16E+03	1.72E+01	2.00E-02	7.40E+03	4.63E+01	5.59E-01	1.78E+01	1.78E-05				
I-129	0.35	Note: I-129 not considered because its T1/2 = 1.57E7 y and is too long to be in equilibrium														
Te-129m/Te-129	NA												1.88E-05	NA	NA	NA
Te-129, parent		5.74E-15	6.69E+02	5.49E+02	1.67E+00	2.48E-02	2.00E-02	7.40E+03	4.63E+01	8.05E-04	2.56E-02	2.56E-08		5.00E-01	5.00E-07	5.13E-02
^{9}Te-131m	NA	2.20E-15	2.56E+02	2.10E+02	4.33E+01	2.47E-01	1.50E-02	5.55E+03	3.47E+01	1.57E-02	2.62E-01	2.62E-07		2.90E-01	2.90E-07	9.05E-01
I-131	0.778	Note: I-131 not considered because its $t_{1/2}$ = 8.04 d and is longer than Te131m $t_{1/2}$ of 1.25 d.														
Xe-131m	1.11E-02	Note: Xe-131m not considered because it is I-131 progeny (and its $t_{1/2}$ = 11.9 d and is longer than ^{9}Te131m $t_{1/2}$ of 1.25 d.														
Te-131	0.222	8.36E-15	9.74E+02	7.99E+02	4.33E+01	9.38E-01	2.30E-02	8.51E+03	5.32E+01	2.40E-02	9.62E-01	9.62E-07				
I-131	1	Note: I-131 not considered because its $t_{1/2}$ = 8.04 d and is longer than Te-131m $t_{1/2}$ of 1.25 d.														
Xe-131m	1.11E-02	Note: Xe131m not considered because it is I-131 progeny (and its $t_{1/2}$ = 11.9 d and is longer than Te-131m $t_{1/2}$ of 1.25 d.														
Te-131m/Te-131													4.76E-07	NA	NA	NA
Te-132	NA	2.99E-16	3.48E+01	2.86E+01	1.13E+02	8.75E-02	7.00E-03	2.59E+03	1.62E+01	1.91E-02	1.07E-01	1.07E-07		5.40E-03	5.40E-09	1.97E+01
I-132	1	7.54E-15	8.79E+02	7.21E+02	1.13E+02	2.21E+00	1.90E-02	7.03E+03	4.39E+01	5.18E-02	2.26E+00	2.26E-06				
Te-132/I-132	NA												2.37E-06	NA	NA	NA
I-132, parent		7.54E-15	8.79E+02	7.21E+02	3.32E+00	6.48E-02	1.90E-02	7.03E+03	4.39E+01	1.52E-03	6.64E-02	6.64E-08		5.00E+01	5.00E-05	1.33E-03
I-131	NA	6.43E-16	7.49E+01	6.15E+01	2.78E+02	4.63E-01	1.50E-02	5.55E+03	3.47E+01	1.01E-01	5.64E-01	5.64E-07		8.70E-01	8.70E-07	6.48E-01

Table 3-6b. Calculation of Total Skin Dose from Contamination on the Ground (Correcting only for Radioactive Decay) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRCF (mrem/y per µCi/m^2)	First-year EffXP[b] from TF 2.0, 2 - only corrects for decay TF 2.0, 2 (h)	[d]External Skin DCF corrected for GRF, WCF & decay (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron dose-rate factors for skin contam. contamination corrected for WCF and decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
I-133	NA	4.55E-15	5.30E+02	4.35E+02	3.00E+01	3.54E-01	1.90E-02	7.03E+03	4.39E+01	1.37E-02	3.67E-01	3.67E-07		NA	NA	NA
[f]Xe-133	NA	6.93E-17	8.08E+00	6.62E+00	1.82E+02	3.27E-02	No data available	0.00E+00	0.00E+00	0.00E+00	3.27E-02	3.27E-08		NA	NA	NA
I-134	NA	9.85E-15	1.15E+03	9.41E+02	1.26E+00	3.22E-02	2.00E-02	7.40E+03	4.63E+01	6.08E-04	3.28E-02	3.28E-08		NA	NA	NA
I-135	NA	4.83E-15	5.63E+02	4.62E+02	9.54E+00	1.19E-01	1.80E-02	6.66E+03	4.16E+01	4.14E-03	1.24E-01	1.24E-07		NA	NA	NA
Cs-134	NA	2.17E-15	2.53E+02	2.07E+02	7.44E+03	4.18E+01	1.20E-02	4.44E+03	2.78E+01	2.15E+00	4.40E+01	4.40E-05		3.30E+01	3.30E-05	1.33E+00
[f]Xe-135	NA	2.09E-15	2.44E+02	2.00E+02	1.31E+01	7.09E-02	No data available	0.00E+00	0.00E+00	0.00E+00	7.09E-02	7.09E-08		NA	NA	NA
Cs-136	NA	2.54E-15	2.96E+02	2.43E+02	4.54E+02	2.99E+00	1.30E-02	4.81E+03	3.01E+01	1.42E-01	3.13E+00	3.13E-06		3.70E-01	3.70E-07	8.46E+00
Cs-137	NA	2.75E-16	3.21E+01	2.63E+01	8.66E+03	6.17E+00	1.40E-02	5.18E+03	3.24E+01	2.92E+00	9.09E+00	9.09E-06		NA	NA	NA
Ba-137m	0.946	1.65E-15	1.92E+02	1.58E+02	8.66E+03	3.70E+01	2.10E-03	7.77E+02	4.86E+00	4.38E-01	3.75E+01	3.75E-05		NA	NA	NA
Cs-137/Ba-137m													4.45E-05	2.90E+01	2.90E-05	1.54E+00
Ba-137m, parent		1.65E-15	1.92E+02	1.58E+02	6.14E-02	2.62E-04	2.10E-03	7.77E+02	4.86E+00	3.11E-06	2.66E-04	2.66E-10		NA	NA	NA
[f]Xe-138	NA	7.65E-15	8.92E+02	7.31E+02	3.41E-01	6.76E-03	No data available	0.00E+00	0.00E+00	0.00E+00	6.76E-03	6.76E-09		NA	NA	NA
Ba-140	NA	1.95E-15	2.27E+02	1.86E+02	4.41E+02	2.23E+00	1.70E-02	6.29E+03	3.93E+01	1.81E-01	2.41E+00	2.41E-06		9.60E+00	9.60E-06	2.51E-01
La-140	1	8.24E-15	9.60E+02	7.88E+02	4.41E+02	9.41E+00	2.00E-02	7.40E+03	4.63E+01	2.13E-01	9.63E+00	9.63E-06		NA	NA	NA
Ba-140/La-140													1.20E-05	NA	NA	NA
La-140	NA	8.24E-15	9.60E+02	7.88E+02	5.81E+01	1.24E+00	2.00E-02	7.40E+03	4.63E+01	2.80E-02	1.27E+00	1.27E-06		1.30E+01	1.30E-05	9.76E-02
Ce-141	NA	1.32E-16	1.54E+01	1.26E+01	1.12E+03	3.83E-01	1.70E-02	6.29E+03	3.93E+01	4.59E-01	8.42E-01	8.42E-07		7.10E-01	7.10E-07	1.19E+00
Ce-143	NA	3.99E-15	4.65E+02	3.81E+02	4.76E+01	4.92E-01	1.90E-02	7.03E+03	4.39E+01	2.18E-02	5.14E-01	5.14E-07		2.30E+00	2.30E-06	2.23E-01

Table 3-6b. Calculation of Total Skin Dose from Contamination on the Ground (Correcting only for Radioactive Decay) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRCF (mrem/y per µCi/m^2)	First-year EffXP[b] from TF 2.0, 2 - only corrects for decay TF 2.0, 2 (h)	[d]External Skin DCF corrected for GRF, WCF & decay (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron dose-rate factors for skin contam. contamination corrected for WCF and decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
Ce-144	NA	2.61E-17	3.04E+00	2.49E+00	5.80E+03	3.92E-01	8.90E-03	3.29E+03	2.06E+01	1.24E+00	1.64E+00	1.64E-06		NA	NA	NA
Pr-144	9.82E-01	1.27E-14	1.48E+03	1.21E+03	5.80E+03	1.91E+02	2.20E-02	8.14E+03	5.09E+01	3.08E+00	1.94E+02	1.94E-04		NA	NA	NA
Pr-144m	1.78E-02	2.67E-17	3.11E+00	2.55E+00	5.80E+03	4.01E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	4.01E-01	4.01E-07		NA	NA	NA
Pr-144	9.99E-01	1.27E-14	1.48E+03	1.21E+03	5.80E+03	1.91E+02	2.20E-02	8.14E+03	5.09E+01	3.08E+00	1.94E+02	1.94E-04		NA	NA	NA
Ce-144/Pr-144/Pr-144m													1.95E-04	1.10E+00	1.10E-06	1.78E+02
Pr-143	NA	2.00E-15	2.33E+02	1.91E+02	4.70E+02	2.44E+00	1.80E-02	6.66E+03	4.16E+01	2.04E-01	2.64E+00	2.64E-06		1.40E+01	1.40E-05	1.89E-01
Pm-147	NA	1.20E-19	1.40E-02	1.15E-02	7.70E+03	2.39E-03	5.40E-03	2.00E+03	1.25E+01	1.00E+00	1.00E+00	1.00E-06		NA	NA	NA
Nd-147	NA	1.10E-15	1.28E+02	1.05E+02	3.80E+02	1.08E+00	1.70E-02	6.29E+03	3.93E+01	1.56E-01	1.24E+00	1.24E-06		4.50E+00	4.50E-06	2.75E-01
Gd-153	NA	1.41E-16	1.64E+01	1.35E+01	5.43E+03	1.98E+00	1.10E-03	4.07E+02	2.54E+00	1.44E-01	2.13E+00	2.13E-06		NA	NA	NA
Yb-169	AN	3.66E-16	4.27E+01	3.50E+01	1.11E+03	1.05E+00	8.80E-03	3.26E+03	2.04E+01	2.35E-01	1.29E+00	1.29E-06		NA	NA	NA
[f]Tm-170	NA	2.12E-15	2.47E+02	2.03E+02	3.83E+03	2.10E+01	No data available	0.00E+00	0.00E+00	0.00E+00	2.10E+01	2.10E-05		NA	NA	NA
Ir-192	NA	1.21E-15	1.41E+02	1.16E+02	2.48E+03	7.77E+00	1.70E-02	6.29E+03	3.93E+01	1.02E+00	8.79E+00	8.79E-06		NA	NA	NA
Ra-226	NA	8.12E-18	9.46E-01	7.76E-01	8.76E+03	1.84E-01	4.20E-04	1.55E+02	9.71E-01	8.87E-02	2.73E-01	2.73E-07		NA	NA	NA
Rn-222	1.00E+00	5.20E-19	6.06E-02	4.97E-02	8.76E+03	1.18E-02	No data available	0.00E+00	0.00E+00	0.00E+00	1.18E-02	1.18E-08		NA	NA	NA
Po-218	1.00E+00	1.17E-20	1.36E-03	1.12E-03	8.76E+03	2.66E-04	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.66E-04	2.66E-10		NA	NA	NA
Pb-214	1.00E+00	9.10E-16	1.06E+02	8.70E+01	8.76E+03	2.07E+01	2.20E-02	8.14E+03	5.09E+01	4.65E+00	2.53E+01	2.53E-05		NA	NA	NA
Bi-214	1.00E+00	8.48E-15	9.88E+02	8.10E+02	8.76E+03	1.93E+02	2.00E-02	7.40E+03	4.63E+01	4.23E+00	1.97E+02	1.97E-04		NA	NA	NA
Po-214	1.00E+00	1.09E-19	1.27E-02	1.04E-02	8.76E+03	2.47E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.47E-03	2.47E-09		NA	NA	NA
At-218	2.00E-04	2.32E-17	2.70E+00	2.22E+00	8.76E+03	5.27E-01	No data available	0.00E+00	0.00E+00	0.00E+00	5.27E-01	5.27E-07		NA	NA	NA
Bi-214	1.00E+00	8.48E-15	9.88E+02	8.10E+02	8.76E+03	1.93E+02	2.00E-02	7.40E+03	4.63E+01	4.23E+00	1.97E+02	1.97E-04		NA	NA	NA
Po-214	1.00E+00	1.09E-19	1.27E-02	1.04E-02	8.76E+03	2.47E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.47E-03	2.47E-09		NA	NA	NA

Table 3-6b. Calculation of Total Skin Dose from Contamination on the Ground (Correcting only for Radioactive Decay) (continued)

		Groundshine Dose to Skin					Skin Contamination Dose				Total Skin Dose			Comparison to EPA Values		
Radio-Nuclide	Branch Fraction	External Skin DCF[a] (Sv/s per Bq/m^2)	External Skin DCF (mrem/y per µCi/m^2)	External Skin DCF corrected for GRCF (mrem/y per µCi/m^2)	First-year EffXP[b] from TF 2.0, 2 - only corrects for decay TF 2.0, 2 (h)	[d]External Skin DCF corrected for GRF, WCF & decay (mrem/y per µCi/m^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin[c] (Sv/y per Bq/cm^2)	Electron dose-rate factor at depth of 7 mg/cm^2 for contamination on skin (mrem/y per µCi/m^2)	Electron dose-rate factor corrected for skin to ground contamination ratio (mrem/y per µCi/m^2)	[e]Electron dose-rate factors for skin contam. contamination corrected for WCF and decay (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per µCi/m^2)	Total Skin Dose Groundshine + skin contamination (mrem/y per pCi/m^2)	Total Skin Dose Groundshine + skin contamination Parent + Progeny (mrem/y per pCi/m^2)	EPA Value (mrem/y per µCi/m^2)	EPA Value (mrem/y per pCi/m^2)	Calculated Value Divided by EPA Value
[f]Ra-226/Rn-222/Po-218/At-218													2.22E-04	NA	NA	NA
Pu-238	NA	9.64E-18	1.12E+00	9.21E-01	8.73E+03	2.18E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.18E-01	2.18E-07		NA	NA	NA
Pu-239	NA	3.67E-18	4.28E-01	3.51E-01	8.76E+03	8.33E-02	0.00E+00	0.00E+00	0.00E+00	0.00E+00	8.33E-02	8.33E-08		NA	NA	NA
Np-239	NA	2.63E-16	3.07E+01	2.51E+01	8.15E+01	5.55E-02	2.30E-02	8.51E+03	5.32E+01	4.52E-02	1.01E-01	1.01E-07		3.40E-02	3.40E-08	2.96E+00
Am-241	NA	8.32E-17	9.70E+00	7.95E+00	8.75E+03	1.89E+00	2.20E-05	8.14E+00	5.09E-02	4.64E-03	1.89E+00	1.89E-06		6.40E-02	6.40E-08	2.95E+01
Cm-244	NA	8.70E-18	1.01E+00	8.31E-01	8.59E+03	1.94E-01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	1.94E-01	1.94E-07		NA	NA	NA
Cf-252	NA	5.75E-18	6.70E-01	5.50E-01	7.70E+03	1.15E-01	4.80E-06	1.78E+00	1.11E-02	8.91E-04	1.16E-01	1.16E-07		NA	NA	NA

[a]From FGR Report #12, Table III.3 (EPA 1993)
[b]From Turbo FRMAC 2.0, RFC 2 (DCFPAK, K. Eckerman).
[c]From Kocher & Eckermann, Health Physics 53(2), p. 135 - 141, 1987)
[d]Assume exposure period to groundshine = 2080 h/y.
[e]Assume exposure period to contamination uniformly distributed on the skin = 800 hours during the year.
[f]Total does not include dose from contamination on the skin if Electron Dose-Rate Factors not available.
[g]Te-131m values in the tables are subject to change pending further development of new parent-daughter rules

3.9 Surface Contamination Control

Areas under the plume can be expected to contain deposited radioactive materials if aerosols or particulate materials were released during the incident. In extreme cases, individuals and equipment may be highly contaminated and screening stations will be required for emergency monitoring and decontamination of individuals and to evaluate the need for medical evaluation. Equipment should be checked at this point and decontaminated as necessary to avoid the spread of contamination to other locations. This screening service would be required for only a few days following plume passage until all such persons have been evacuated or relocated.

After the relocation area is established, based on the PAG for relocation, adults may reenter the relocation area under controlled conditions in accordance with occupational exposure standards. The need for monitoring stations should be evaluated along highways to control surface contamination at exits from the more highly contaminated areas. Because of the possibly high-background radiation levels at control points near exits, significant levels of surface contamination on persons and equipment may be undetectable at these locations. Therefore, additional monitoring and decontamination stations may be needed at nearby low-background locations. Decontamination and other measures should be implemented to maintain low-exposure rates at monitoring stations.

3.9.1 Considerations and Constraints

Surface contamination limits to control routine operations at nuclear facilities and in transporting radioactive material are generally set at levels lower than are practical for situations involving high-level, widespread contamination of the environment. The density of uncontrolled surface contamination cannot be predicted. It is not practical to set contamination limits significantly lower than residual contamination levels in uncontrolled areas.

The principal exposure pathways for loose surface contamination on persons, clothing, and equipment are (a) internal doses from ingestion by direct transfer; (b) internal doses from inhalation of resuspended materials; (c) beta dose to skin from contaminated skin or clothing or from nearby surfaces; and (d) dose to the whole body from external gamma radiation. Because of the difficulties in predicting the density of uncontrolled surface contamination, a contaminated individual or item should not be released to an unrestricted area. On the other hand, a level of contamination comparable to that existing on surfaces immediately outside the relocation area may make materials leaving the relocation area an acceptable action. Persons working in the parts of the relocation areas in which their annual dose from the residual radioactivity would exceed 2 mrem (0.05 mSv) in any hour or 100 mrem (1 mSv) in a year should operate under the controlled conditions normally established for occupational exposure.

The contamination limit should also be influenced by the potential for the contamination to be ingested, inhaled, or transferred to other locations. Therefore, it is reasonable to establish lower limits for surfaces where contamination is loose than for surfaces where the contamination is fixed (except for skin). The expected period of fixed contamination on skin would be longer so a lower limit would be justified.

For routine (non-incident) situations, measurement of gross beta-gamma surface contamination levels is commonly performed with a thin-window geiger counter. Since beta-gamma measurements made with such field instruments cannot be interpreted in terms of dose or exposure rate, the guidance set forth below is related to the background radiation level in the area where the measurement is being made. Supplementary levels are provided for gamma exposure rates measured with the beta shield closed. Guidance levels expressed in this form should be easily detectable and should satisfy the above considerations. Corresponding or lower levels expressed in units related to instrument designations may be adopted for convenience or for ALARA determinations. Smears may also be used to detect loose surface contamination at very low levels. However, they are not considered necessary for emergency response and, therefore, such guidance is not provided.

Certain cases, such as alpha RDD or IND, will require measurements of gross alpha activity per unit area or other special measurements. These alpha deposition measurements are sensitive to details of the measurement technique, such as probe distance, surface texture and, especially surface wetness or cleanliness. Care must be used when using these measurements.

3.9.2 Numerical Relationships

As discussed in Section 3.3.1, a relationship can be established between projected first year doses and instantaneous gamma exposure rates from properly characterized surface contamination. Based on assumed radiological characteristics of releases from fuel melt incidents, gamma exposure rates in areas where the projected dose is equal

to the relocation PAG of 2 rem (20 mSv) in the first year may be in the range of 2 to 5 mR/h during the first few days following the deposition from a type SST-2 accident (See Section F.1.2). (This relationship must be determined for each specific release mixture). Based on relationships in reference (DOE 1988) and a mixture of radionuclides expected to be typical of an SST-2 type accident, surface contamination levels of $2x10^8$ pCi/m^2 would correspond approximately to a gamma exposure rate of 1 mR/h at 1 m (3.28 feet) height.

3.9.3 Population Monitoring and Contamination Limits

Surface contamination must be controlled both before and after relocation protective actions are implemented. Therefore, this section deals with the control of surface contamination on persons and equipment being protected during both the early and intermediate phases of a radiological incident.

For the early and intermediate phase, the following general guidance regarding surface contamination is recommended:

1. Do not delay urgent medical care for decontamination efforts or for time-consuming protection of attendants.

2. In scenarios where it might not be practical or would interfere with other priorities, do not waste effort trying to contain contaminated wash water.

3. Do not allow monitoring and decontamination to delay evacuation from high or potentially high exposure rate areas.

4. (Optional provision, for use only if a major contaminating event occurs, and rapid early screening is needed). After plume passage, it may be necessary to establish emergency contamination screening stations in areas not qualifying as low background areas. Gamma exposure rates in such areas should be less than 5 mR/h. These screening stations should be used only during the early phase and for major releases of particulate materials to the atmosphere to monitor persons emerging from possible high exposure areas, to provide simple (rapid) decontamination if needed, and to make decisions on whether to send them for special care or to a monitoring and decontamination station in a lower background area. Table 3-8 provides guidance on surface contamination levels for use if such centers are needed.

5. Establish monitoring and personnel decontamination (e.g., bathing) facilities at evacuation centers or other locations in low background areas (less than 0.1 mR/h). Encourage evacuated persons, who were exposed in areas where release of particulate materials would have warranted evacuation, to (if possible) change clothes, wash clothes, and wash other exposed surfaces such as cars and trucks and their contents and then report to these centers for monitoring. Table 3-9 provides surface contamination guidance for use at these centers. These screening levels are examples derived primarily on the basis of easily measurable radiation levels using portable instruments.

6. After the relocation area has been established, consider the need to set up monitoring and decontamination stations at exits from the more highly contaminated parts of the area. Because of the probably high background radiation levels at these locations, low levels of contamination may be undetectable. If contamination levels are undetectable, then they probably do not exceed those in some unrestricted areas occupied by the exposed population and no decontamination is required. Nevertheless, these individuals should be advised to bathe and change clothes at their first opportunity and certainly within the next 24 hours. If, after decontamination, persons still exceed the limits for this station, they should be sent for further decontamination or for medical or other special attention. As an alternative to decontamination, contaminated items other than persons or animals may be retained in the relocation area while the radiological contamination decays.

Table 3-8. Recommended Surface Contamination Screening Levels for Emergency Screening of Persons and Other Surfaces at Screening or Monitoring Station in High Background Radiation Areas (0.1 mR/h to 1 mR/h Gamma Exposure Rate)[a]

Condition	Geiger-counter shielded-window reading	Recommended Action
Before Decontamination	<2x existing background	Unconditional release
	>2x existing background	Decontaminate Equipment may be stored or disposed of as appropriate
After Decontamination	<2x existing background	Unconditional release
	>2x existing background	Continue to decontaminate or refer to low background monitoring and decon station. Equipment may be stored for decay or disposed of as appropriate

[a]Monitoring stations in such high exposure rate areas are for use only during the early phase of an incident involving major atmospheric releases of particulates. Otherwise use Table 3-9.

Table 3-9. Recommended Surface Contamination Screening Levels for Persons and Other Surfaces at Monitoring Stations in Low Background Radiation Areas (<0.1 mR/h Gamma Exposure Rate)[a]

Condition	Geiger-counter shielded-window reading	Recommended Action
Before decontamination	<2x existing background	Unconditional release
	>2x existing background	Decontaminate
After simple[b] decontamination effort	<2x existing background	Unconditional release
	>2x existing background	Full decontamination
After full[c] decontamination effort	<2x existing background	Unconditional release
	>2x existing background	Continue to decontaminate persons Release animals and equipment
After additional full decontamination effort	<2x existing background	Unconditional full release
	>2x existing background	Send persons for special evaluation Release animals and equipment

[a]Window thickness of approximately 30 mg/cm^2 is acceptable. Recommended limits for open window readings are expressed as 2x the existing (post-incident) background in the area where measurements are being made. Corresponding levels, expressed in units related to instrument designations, may be adopted for convenience. Levels higher than 2x background (not to exceed the meter reading corresponding to 0.1 mR/h) may be used to speed the monitoring of evacuees in very low background areas.
[b]Flushing with water and wiping is an example of a simple decontamination effort.
[c]Washing or gentle scrubbing with soap or other mild detergent followed by flushing is an example of a full decontamination effort.

3.10 Longer-Term Objectives of the Protective Action Guides for the Intermediate Phase

It is an objective of the PAGs to assure that doses in any single year after the first will not exceed 0.5 rem (5 mSv). For source terms from NPP incidents, the PAG listed in Table 3-1 of 2 rem (20 mSv) projected dose in the first year is expected to meet this longer-term objective through radioactive decay, weathering, and normal part-time occupancy in structures. Decontamination of areas outside the relocation area may be required during the first year to meet these objectives for releases consisting of long-lived radionuclides. For situations where it is impractical to meet these objectives through decontamination, consideration should be given to relocation to a lower projected first-year dose area specified by the relocation PAG.

After the population has been protected in accordance with the PAGs for relocation, return for occupancy of the relocation areas should be governed on the basis of recovery criteria as presented in Chapter 6.

Projected dose considers exposure rate reduction from radioactive decay and, generally, weathering. When one also considers the anticipated effects of shielding from partial occupancy in homes and other structures, persons who are not relocated should receive a dose substantially less than the projected dose. For commonly assumed reactor source-terms, it is estimated that 2 rem (20 mSv) projected dose in the first year will be reduced to about 1.2 rem (12 mSv) by this factor. The application of simple decontamination techniques shortly after the incident can be assumed to provide a further 30% or more reduction so that the maximum first year dose to persons who are not relocated is expected to be less than 1 rem. Taking account of decay rates assumed to be associated with releases from NPP incidents (SNL 1982), and shielding from partial occupancy and weathering, a projected dose of 2 rem (20 mSv) in the first year is likely to amount to an actual dose of 0.5 rem (5 mSv) or less in the second year. The application of simple dose reduction techniques would reduce the dose further. Calculations supporting these projections are summarized in Table F-6 of Appendix F.

1Chapter 4
Protective Action Guidance for Drinking Water

4.1 Introduction

This chapter identifies the projected level of radiation exposure at which protective actions are needed to protect the public from drinking water contaminated with radioactive material. This chapter also provides the concentrations of radionuclides, called derived response levels (DRLs), in drinking water that correspond to EPA's drinking water PAG of 0.5 rem whole-body CEDE (the sum of the products of the weighting factors applicable to each of the body organs that are irradiated and the committed dose equivalent to these organs).

Similar to the PAGs for the early and intermediate phases discussed in Chapters 2 and 3, the drinking water PAG indicates a level of exposure at which protective action should be taken to prevent, reduce, or limit a person's radiation dose during a radiological incident. The drinking water PAG for the intermediate phase is expressed as a numerical dose level.

4.1.1 How the Drinking Water PAG Complements the FDA Food PAG

The drinking water PAGs are based on EPA's DRLs while the Food PAGs are based on FDA's Derived Intervention Levels (DILs). The numerical values of the DIL and DRL for the same contaminant vary due to the inherent differences in their derivations. Although both the DIL and DRL apply to contaminant exposure due to ingestion pathway (of either food or water, respectively), the potential exposure to contamination would depend on the contaminant event. In the event of water intake, which could include beverages (soft drinks, coffee, etc.), simultaneous use of the DRL and DIL is applicable; similarly, food intake of any variety should include some quantity of water intake as a component of the food.

Though drinking water is not regulated by FDA, it was included as a component for the dietary intake values used to derive the DILs as this was consistent with the conservative nature of the other assumptions. Some members of the population ingest little drinking water, consuming only prepared beverages such as soft drinks or coffee. As a result, the water intake of these individuals could be via products within the scope of the DILs. Thus, accounting for the water intake is necessary because though these individuals do ingest substantial water intake from a nutritional perspective, it is from beverages that are within the scope of the DILs. Including water consumption in food intake accounts for the possibility that some radioactive contamination may be ingested via the water component in the food not covered by the DIL. Including water in the food intake value used to derive the DILs results in a bias slightly lower than if it were not included. Thus, both the DIL and DRL can be used either independently or in unison.

Similarly, manufactured or otherwise prepared beverages (excluding drinking water) are not regulated by EPA but the conservative assumption calls for accommodating contamination from drinking water used in production of these beverages. Thus, the drinking water PAGs use a nutritional water intake value in deriving DRLs. The nutritional water intake value was large enough to include water that many people would use to make other beverages. The net result of using a nutritional intake value in this case is that the DRLs are biased slightly lower than if only a strict drinking water value was used. Effectively, this conservative bias accounts for the possibility that drinking water could bring radioactive contamination that would be ingested via the food component that the DRLs do not cover.

4.1.2 Early Phase

Due to the fact that it will take some time after the radioactive airborne plume passes to determine the full extent of any contamination to surface water bodies, actions to protect drinking water supplies would likely be put in place during the intermediate phase of a radiological incident. In fact, it is unlikely that emergency response personnel will have the results from radiation measurements of water supplies until well after the plume has passed.

Water distribution systems all incorporate reserve and storage capacity. During the early phase of the incident it is unlikely that contamination could affect water which is directly available for consumption through distribution systems. It would take some time for radionuclides to be deposited from the plume into the water supply system and then subsequently distributed. The drinking water PAG may also be applied in the event of direct radiological contamination of a water system. During the early phase, recommendations to the public (i.e. about drinking tap water while sheltering-in-place) should reflect these considerations. The public should be advised that the water is safe to drink unless otherwise informed.

The fact that protective action guides for water apply during the intermediate phase should not, however, preclude reasonable precautionary measures (e.g., closing water intakes and using available stored water) during the early phase.

4.1.3 Intermediate Phase

Chapter 3 of this Manual provides a detailed discussion of a variety of protective actions that should be considered during the intermediate phase of a radiological incident. EPA has set the drinking water PAG at a level consistent with the recommendations given in Chapter 3. EPA recommends a drinking water protective action guide of 0.5 rem CEDE. It is assumed that Safe Drinking Water Act (SDWA) levels may be achieved within the first year. If not, a waiver must be obtained.

In recommending this PAG, EPA considered the principles identified as the basis for PAGs in Chapter 1; one of these principles is that "PAGs should not be higher than justified on the basis of optimization of cost and the collective risk of effects on health." EPA also considered consistency with other PAGs already established. The recommended drinking water PAG of 0.5 rem (5 mSv) is consistent with FDA guidance for food ingestion (see Chapter 5). In addition, the International Commission on Radiological Protection Publication 60 (ICRP 1991a) recommends intervention efforts following a radiological incident at a dose level of 0.5 rem (5 mSv). Thus, the drinking water PAG is consistent with both national and international guidance related to radiological incidents.

Estimates of risk associated with the recommended drinking water PAG are in the range of risks which EPA has generally considered to be acceptable or ALARA for protecting the public. Using EPA's estimate of radiogenic cancer risk from low-level, whole-body, low-LET radiation (*Estimating Radiogenic Cancer Risk*, EPA 402-R-93-076, June 1994), a one-year exposure of 0.5 rem (5 mSv) produces an individual lifetime cancer incidence risk of approximately 4 $x10^{-4}$. This is consistent with the upper limits of risk associated with current radionuclide drinking water standards, assuming lifetime exposure.

EPA considers that this dose and risk level will provide an acceptable level of safety to an affected community for the year following a radiological incident. The time period of one year is also consistent with the time period recommended by the FDA PAGs for food.

4.1.4 Late Phase

The need for intervention and an appropriate level of exposure may continue after the first year. Protective actions, such as those explained in Section 4.2, implemented after the first year can significantly reduce exposures. Depending upon individual circumstances, an appropriate dose level and protective action should be chosen for the long-term, with the goal being SDWA levels or below. Chapter 6 provides a framework for choosing appropriate long-term risk levels and protective actions.

4.2 Protective Actions for Water

The drinking water PAG only applies to drinking water. EPA considered the potential radiation dose people could receive from various uses of contaminated water, including showering and bathing. These activities generally represent a smaller risk than drinking contaminated water and are covered by the intermediate phase PAGs of Chapter 3. However, people typically shower and bathe using the same source of water that they use to drink. As a result, protection of a community's drinking water supply will also protect the source of water used for virtually all other needs.

Radiological or nuclear incidents may affect surface water supplies. Some have the potential to contaminate ground water though this occurrence is much less likely. In addition, the time required for ground water to become

contaminated would be sufficient enough to plan for and implement protective actions. The drinking water PAG does apply to ground water; however, it is primarily intended to guide planning and decision making efforts during the early and intermediate phases of a radiological emergency when surface water supplies are particularly vulnerable to contamination from deposition of radioactive materials from the air.

In the case that water supplies may be affected by a radiological incident and that projected radiation doses from drinking contaminated water could exceed 0.5 rem (5 mSv) in the first year, taking actions to protect the water supplies of communities should be considered.

Any of the actions described below may be taken alone or in combination, depending upon the nature of the emergency and the characteristics of the local water system. Consideration of these and other options during emergency planning provides the opportunity to develop state and local emergency plans and implementation procedures that reflect the unique needs of a particular community. Advance planning can provide clarity during the process of making key decisions quickly during a radiological emergency.

To ensure effectiveness of protective actions taken, emergency response plans should include a comprehensive radiological surveillance program. This program will monitor the concentration of radionuclides in the drinking water and will provide an indication of whether any action is necessary or if the actions being taken are effective. The emergency response plan should also include a strategy for keeping the community informed of the actions being taken and ensuring that people understand their role in carrying them out.

It should be noted that use of water tankers and bottled water are the best actions until municipal systems are verified to be unaffected. These actions may be escalated initially in response to a large incident that destroys water treatment and processing. This would virtually eliminate the need to evaluate drinking water supplies in the early stages of the intermediate phase.

Examples of these and other possible protective actions to protect drinking water are discussed in the following sections.

4.2.1. Wait for Flow-By

If radionuclides are deposited from the atmosphere over a river, a section of the water can become contaminated, and can be repeatedly contaminated as rain redeposits radioactive particles. As this section flows down the river, this section of contaminated water is called a plume. This is similar to the atmospheric plume of radiation described in Chapter 1. This action calls for closing down any source intake valves along the path of the plume of contaminated water. While the intake valve is closed, no contaminated water can enter the water supply system, therefore the contaminated water is allowed to flow by the system. During this time, the system's existing storage capacity can be depended upon. Large systems usually possess 12 to 24 hours of water storage capacity. If the stored water supplies could be depleted before the affected valves can be reopened, treatment of the contaminated water while using available stored water supplies should be considered. This assumes that the treatment technology will be in place or readily accessible. In the event of no other option, contaminated water could be temporarily replaced with large quantities of purchased uncontaminated water.

4.2.2 Ration Clean Water Supplies

Rationing uncontaminated water is also a possibility. This is particularly true if water reserves can provide each individual in the community with 1 L (0.264G, about 1 qt) of water per day until the contaminated plume has passed, the contaminated water is treated, or the water supply system has returned to normal operating conditions. If this option is chosen, it is important that contamination is isolated from the system or is in a single area of the system and that efficient methods for rationing are in place. Rationing water might also be required if water treatment capabilities are limited. Consider a scheme to ration water as part of emergency planning efforts.

4.2.3 Treat Contaminated Water

Various treatment options exist for reducing or eliminating the contamination of drinking water by man-made radionuclides. Only a small percentage of all water systems treat specifically for radionuclides. However, typical

technologies used to treat water for other contaminants can reduce the concentration of radioactivity. These technologies include coagulation/filtration, ion exchange, lime softening, and reverse osmosis. Their actual removal efficiency depends on the radionuclide and the type of treatment.

4.2.4 Activate Existing Connections to Neighboring Systems

If the water supply system is part of a larger, regional supply system, activation of existing connections to a neighboring area could be considered. Most large water systems can establish connections with other large systems for emergency purposes. As in many cases, this option may have already been considered during emergency planning. If this option is implemented, steps must be taken to ensure that the "clean" systems do not become contaminated from water backflow.

Smaller water supply systems (i.e., that serve between 10,000 and 75,000 people), or sparsely populated or rural areas, may not be connected to a neighboring system. In this case, regionalization may be a part of emergency planning to explore. This involves connecting smaller systems to larger systems, thus forming a regional water supply system. This is obviously a long-term proposition, but it does have the added advantage of reducing system vulnerability to water shortages or water quality problems other than those resulting from a radiological incident.

4.2.5 Establish Pipeline Connections to Closest Sources/Systems

Running a pipeline from a "clean" water supply system to various distribution centers located throughout the affected community is a routine means of providing clean water. For example, when water mains must be repaired or cleaned of debris, community water needs have been met through the assembly of temporary pipes and hoses. This is a relatively simple procedure, requiring very little construction and technical expertise. For medium- to long-term emergencies, the construction of a temporary pipeline could be cost-effective. PVC pipe, fire hoses, and steel pipe have been used to provide emergency drinking water for periods of up to 2 months when service has been disrupted by earthquakes, drought, or bacterial contamination.

4.2.6 Import Water in Tanker Trucks

If an uncontaminated source of water is close to the affected area, it may be more efficient to arrange to transport water from that source by truck, rail, or barge to distribution centers located throughout the community. The most significant obstacle for the use of this option is the cleanliness and availability of transport vehicles. State and local laws may also affect this option.

4.2.7 Import Bottled Water

Importing bottled water into the affected community is another possible option. The water may come from a nearby water supply system or from a local spring water bottling company. This option may be cost-effective during an emergency if water is needed quickly and if the length of the emergency does not merit long-term action, such as the construction of a temporary pipeline.

4.3 Projecting Radiation Doses Using Derived Response Levels

DRLs are concentrations of radionuclides in water that, using conservative assumptions, correspond to EPA's drinking water PAG of 0.5 rem (5 mSv) in the first year. The use of (DRLs as a basis for protective actions will make it very unlikely that anyone will be exposed at the PAG level of 0.5 rem. A radiological surveillance program will provide information regarding the concentration of various radionuclides in drinking water. This information can be used along with the tables provided in this chapter to determine whether the drinking water PAG is likely to be exceeded.

Under ideal circumstances, event- and site-specific DRLs would be derived based on the unique characteristics of the radiological emergency that has occurred and a specific community's water system. However, such calculations can be difficult and time-consuming, requiring large amounts of data and specialized expertise in hydrology and water resource management. Instead, EPA has developed generic DRLs for use in emergency planning and response activities. These generic DRLs can also be used during an actual emergency until there is sufficient time or a need to develop event- or site-specific DRLs.

Table 4-1 presents generic DRLs for radionuclides likely to contaminate a water supply during a radiological emergency. This is the same list of radionuclides included in Chapters 2 and 3, with the exception of the very short-lived nuclides (half-life less than 1 day) and the noble gases. Short-lived nuclides undergo radioactive decay at a speed that makes their deposition into the water supply unlikely.

EPA derived the DRLs by calculating the radionuclide concentrations in water that will result in a radiation dose of 0.5 rem (5 mSv) TEDE in the first year, assuming the members of the exposed population each consume 2 L (0.52 gal) of water per day for an entire year. DRLs assume that the radionuclide concentrations in the water supply decline by virtue of radioactive decay only.

The DRLs may be calculated using the following formula:

$$DRL = \frac{HL}{hI\ (1 - e^{-LT})}$$

where:

DRL=	the derived response level (pCi/L);
H =	the PAG (mrem)
L =	the effective decay constant (sum of the radioactive decay constant and the removal constant due to natural processes, expressed as the fraction of contamination removed per day)
h =	dose conversion factor (mrem/pCi ingested)
T =	exposure time or interdiction time (d)
I =	the daily water intake (L/d)

This equation simplifies the removal of radiation from a water supply following a contaminating incident. For example, if a radiological incident results in the contamination of the water in a reservoir, then the radionuclide concentration will decline with the effective half-life of the radionuclide in the reservoir. Although factors such as turnover rate of the water supply contribute to the effective half-life, DRLs are calculated on the conservative basis that only radioactive decay contributes to concentration reductions.

The assumed water intake rate, I, is another variable that will influence the DRLs. The water intake rate is assumed to be 2.0 L/d (0.528 gal/d), which represents the water intake of the upper 90th percentile of the U.S. population. The larger the assumed daily water consumption, the lower the DRLs. The DRLs are also affected by the assumed exposure time, T. In this analysis, T is assumed to be 1 year. The longer the assumed exposure time, the lower the DRLs.

The 1-year exposure period assures that protective actions would be in place long enough for water monitoring to reveal a decline in the concentrations of radionuclides in the drinking water supply and/or other variables affecting the potential radiation dose to members of the community. Using a 1-year basis for the DRLs also reduces the uncertainty due to the inability to predict the temporal variations in rain storm events over a shorter period.

4.4 Derived Response Levels

The final step in developing the drinking water PAG was to calculate DRLs. These levels will allow emergency response personnel to determine whether the radionuclide concentrations in water are likely to exceed the drinking water PAG of 0.5 rem (5 mSv).

The list of radionuclides contained in Table 5-3 in ICRP 1984b was reduced by eliminating those radionuclides with a half-life of less than 1 day. Table 5-3 in ICRP 1984b presents the DRLs for the air exposure pathway. As such, even relatively short-lived radionuclides are of concern due to the short time from a release to an exposure via the airborne pathways. For the drinking water pathways, there is substantial delay between the contaminating event and exposure. For example, the typical turnover rate of a reservoir is 1 year. Accordingly, the list of radionuclides for the drinking water PAG was constructed from the list in Table 5-3 in ICRP 1984b by deleting the radionuclides

with half-lives less than 1 day. This option was selected for the drinking water PAG because it includes a comprehensive list of potentially important radionuclides and is consistent with the approach used in developing the implementation guidance for the air exposure PAGs. Table 4-1 provides the actual list of these radionuclides and their associated DRLs.

EPA derived the DRLs in Table 4-1 using the Federal Guidance Report No. 11 dose coefficients. These dose coefficients were based upon the International Commission on Radiological Protection (ICRP) Publication 30 (ICRP 1979) which established dosimetric models and biokinetic data for adult workers (ICRP 1979). At that time, there were no recommended models and data for members of the general public, which include all age groups. Between 1989 and 1996, the ICRP published new age-dependent dose coefficients for members of the general public (ICRP 1989, ICRP 1993, ICRP 1995). The ICRP used updated age-dependent metabolic models and radionuclide-specific biokinetic data to calculate the new dose coefficients. ICRP summarized these coefficients in ICRP Publication 72 (ICRP 1996).

The ICRP based their age-dependent dose coefficients on a unit intake of radionuclides at six different ages (3 months, 1 year, 5 years, 10 years, 15 years, and adult). The ICRP established age-dependent reference values encompassing both sexes for organ mass and body size, absorption, retention, and clearance. In almost every case, the resulting dose coefficients are the highest for the youngest age group (<1 year). This is due to increased radionuclide intake and smaller organ mass during the first year of life. For many of the radionuclides that are important during the first year of exposure following an accident, the dose coefficients for the youngest age group can be an order of magnitude or more greater than the coefficients for adults. Such differences in dose coefficients would not necessarily result in similar differences in dose, however. DRL calculations include the assumption that water is ingested at the rate of 2 L/d, which is likely to be an overestimate for the youngest age group.

4.5 Applying the DRLs

EPA derived generic DRLs to assist in planning and, in the event of a radiological emergency, quick and effective execution of protective actions. Although site- and event-specific DRLs would reduce some of the uncertainty associated with generic DRLs, it would require real-time analysis that might be difficult to obtain or cause a delay in taking action if time-sequenced data needed to be gathered. However, site-specific DRLs may be developed; the generic values included in this Manual are only intended to act as a guide for decision making.

DRLs can be used to trigger and guide the response to an incident that results in, or that could result in, the contamination of drinking water supplies. For example, action might be taken to protect water supplies as soon as notification of a radiological release is received. Data can then be obtained from monitoring programs, and field measurement programs can be expanded to include drinking water samples. Such programs could include sampling and analysis of water upstream and downstream of a water supply system and in storage within the supply system. Comparison of these data to the DRLs in Table 4-1 can result in informed judgments regarding the need to implement protective actions. Once it is determined that the potential exists for the PAG to be exceeded, actions can be initiated or revised based on the results of comparison of environmental data to Table 4-1. It is important to remember that such actions are likely to be site- and event-specific. However, consideration of various strategies during emergency planning will allow quick and effective decisions to be made in the case of an actual incident.

Developing a reasonably accurate model for predicting the movement of radionuclides from a watershed to their ultimate fate in a reservoir or other water body is a very difficult problem. The uncertainty of precipitation events represents a major obstacle to credibly predicting this movement. Rain may fall in almost any pattern of frequency and intensity. This leads to great uncertainty in predicting the magnitude of dilution in streams and lakes, the lateral movement of a contaminant due to erosion, or the vertical movement of a contaminant due to percolation in a watershed. Therefore, decisions on protective action must be based on measurements rather than calculations alone. A radiological environmental surveillance program in place to collect up-to-date information concerning the concentrations of radionuclides in water supplies is vital.

4.6 DRLs for Multiple Radionuclides

In the case that multiple radionuclides are found in the water supply, divide the actual concentration of each radionuclide in water by its DRL of 0.5 rem (5 mSv). This gives a fraction of the allowed amount in water for each

radionuclide. If the sum of the fractions is 1 or less, the total of the radionuclides is less than the PAG of 0.5 rem. (See the fractions rule, as follows):

$$F = \sum_{i}^{n} \frac{C_i}{DRL_i}$$

where:

F	=	the sum of fractions
C_i	=	the concentration of radionuclide, i, in the water supply (pCi/L)
DRL_i	=	the derived response level for the i^{th} radionuclide (pCi/L)

For example, assume that as a result of a nuclear power plant accident, a water supply is contaminated with 100,000 pCi/L of I-131, 12,000 pCi/L of Cs-137, and 3,500 pCi/L of Sr-90. The DRLs in Table 4-1 for radioactive decay only are 406 504, 13 850, and 4950, respectively. The sum of fractions rule would result in the following:

$$F = 100{,}000/406{,}504 + 12{,}000/13{,}850 + 3{,}500/4{,}950 = 0.25 + 0.87 + 0.71 = 1.83$$

Based on this, the radionuclide contamination levels exceed the DRL for this combination of radionuclides, assuming depletion by radioactive decay only. As such, actions to protect drinking water would be recommended.

If F is less than 1, the contamination is less than the DRL, and there is no need to take action unless the concentrations change. If F is greater than one, the DRL is exceeded. Intervention is recommended until such time that the radionuclide concentrations decline below the DRL and that decision makers are certain that the radionuclide concentrations will not increase above the DRL in the future.

Like all PAGs, the drinking water PAG indicates a level of exposure at which protective action should be taken to prevent, reduce, or limit a person's radiation dose from a release of radioactive material into the environment. Actions to protect the water supply may be implemented at other levels and at any time following a radiological incident. The DRLs are provided as a point of reference to aid emergency response personnel in their decision making. The ICRP suggests that agencies optimize the net benefit of an intervention strategy based on their knowledge of local situations, as well as their assumptions about the monetary value to a particular dose level. EPA recognizes this as an acceptable strategy. Therefore, after a particular situation stabilizes and becomes more clearly defined, local authorities may wish to modify the level of drinking water exposure they consider to be acceptable based upon a desire to implement optimized longer-term dose reduction strategies. In other words, the dose level associated with the drinking water pathway may be controlled to levels below 0.5 rem (5 mSv) depending on the characteristics of an individual community and a site-specific cost-benefit analysis.

Table 4-1 presents generic DRLs for radionuclides likely to contaminate a water supply during a radiological emergency. This is the same list of radionuclides included in Chapters 2 and 3, with the exception of the very short-lived nuclides (half-life less than 1 day) and the noble gases. Short-lived nuclides undergo radioactive decay at a speed that makes their deposition into the water supply unlikely.

EPA derived the DRLs by calculating the radionuclide concentrations in water that will result in a radiation dose of 0.5 rem total effective dose equivalent in the first year, assuming the members of the exposed population each consume 2 L (0.52 gal) of water per day for an entire year. DRLs assume that the radionuclide concentrations in the water supply decline by virtue of radioactive decay only.

Table 4-1. DRLs Associated with a Total Effective Dose Equivalent of 0.5 rem Resulting from 1 Year of Ingestion

			Normalized DRLs mrem per pCi/L (TEDE)		DRLs (pCi/L)	
Radionuclide	Radioactive Decay Constant 1/d	DCF mrem/uCi	Without Radioactive Decay	With Radioactive Decay Only	Without Radioactive Decay	With Radioactive Decay Only
H-3	1.54E-04	1.55E-01	1.13E-04	1.10E-04	4.42E+06	4.54E+06
C-14	3.31E-07	2.15E+00	1.57E-03	1.57E-03	3.19E+05	3.19E+05
Na-22	7.29E-04	1.18E+01	8.61E-03	7.56E-03	5.80E+04	6.61E+04
P-32	4.85E-02	8.88E+00	6.48E-03	3.66E-04	7.71E+04	1.37E+06
P-33	2.73E-02	9.10E-01	6.64E-04	6.67E-05	7.53E+05	7.50E+06
S-35	7.93E-03	2.87E+00	2.10E-03	6.84E-04	2.39E+05	7.31E+05
Cl-36	6.30E-09	3.44E+00	2.51E-03	2.51E-03	1.99E+05	1.99E+05
K-40	1.48E-12	2.28E+01	1.66E-02	1.66E-02	3.00E+04	3.00E+04
Ca-45	4.25E-03	2.63E+00	1.92E-03	9.75E-04	2.60E+05	5.13E+05
Sc-46	8.27E-03	5.48E+00	4.00E-03	1.26E-03	1.25E+05	3.97E+05
Ti-44	3.16E-05	2.15E+01	1.57E-02	1.56E-02	3.19E+04	3.20E+04
V-48	4.27E-02	7.33E+00	5.35E-03	3.43E-04	9.34E+04	1.46E+06
Cr-51	2.50E-02	1.43E-01	1.04E-04	1.14E-05	4.79E+06	4.37E+07
Mn-54	2.22E-03	2.67E+00	1.95E-03	1.34E-03	2.57E+05	3.74E+05
Fe-55	7.03E-04	1.23E+00	8.98E-04	7.92E-04	5.57E+05	6.31E+05
Fe-59	1.56E-02	6.62E+00	4.83E-03	8.46E-04	1.03E+05	5.91E+05
Co-58	9.79E-03	2.77E+00	2.02E-03	5.50E-04	2.47E+05	9.09E+05
Co-60	3.60E-04	1.27E+01	9.27E-03	8.69E-03	5.39E+04	5.76E+04
Ni-63	1.98E-05	5.63E-01	4.11E-04	4.10E-04	1.22E+06	1.22E+06
Zn-65	2.84E-03	1.46E+01	1.07E-02	6.64E-03	4.69E+04	7.54E+04
Ge-68	2.41E-03	4.77E+00	3.48E-03	2.32E-03	1.44E+05	2.16E+05
Se-75	5.79E-03	9.66E+00	7.05E-03	2.93E-03	7.09E+04	1.70E+05
Rb-86	3.71E-02	1.04E+01	7.59E-03	5.61E-04	6.59E+04	8.92E+05
Sr-89	1.37E-02	9.51E+00	6.94E-03	1.38E-03	7.20E+04	3.63E+05
Sr-90	6.52E-05	1.03E+02	7.52E-02	7.43E-02	6.65E+03	6.73E+03
Y-90	2.60E-01	9.96E+00	7.27E-03	7.66E-05	6.88E+04	6.53E+06
Y-91	1.18E-02	8.77E+00	6.40E-03	1.47E-03	7.81E+04	3.41E+05
Zr-93	1.24E-09	4.11E+00	3.00E-03	3.00E-03	1.67E+05	1.67E+05
Zr-95	1.08E-02	3.56E+00	2.60E-03	6.46E-04	1.92E+05	7.73E+05
Nb-94	9.35E-08	6.44E+00	4.70E-03	4.70E-03	1.06E+05	1.06E+05
Nb-95	1.97E-02	2.18E+00	1.59E-03	2.21E-04	3.14E+05	2.26E+06
Mo-99	2.52E-01	2.24E+00	1.64E-03	1.78E-05	3.06E+05	2.81E+07
Tc-99	8.91E-09	2.38E+00	1.74E-03	1.74E-03	2.88E+05	2.88E+05
Ru-103	1.76E-02	2.72E+00	1.99E-03	3.09E-04	2.52E+05	1.62E+06
Ru/Rh-106	1.88E-03	2.59E+01	1.89E-02	1.37E-02	2.64E+04	3.65E+04
Ag-110m	2.77E-03	1.03E+01	7.52E-03	4.73E-03	6.65E+04	1.06E+05
Cd-109	1.49E-03	7.40E+00	5.40E-03	4.17E-03	9.26E+04	1.20E+05
Cd-113m	1.40E-04	8.51E+01	6.21E-02	6.06E-02	8.05E+03	8.26E+03
In-114m	1.40E-02	1.51E+01	1.10E-02	2.14E-03	4.54E+04	2.33E+05
Sn-113	6.02E-03	2.73E+00	1.99E-03	8.06E-04	2.51E+05	6.20E+05
Sn-123	5.36E-03	7.77E+00	5.67E-03	2.49E-03	8.82E+04	2.01E+05
Sn-125	7.19E-02	1.14E+01	8.32E-03	3.17E-04	6.01E+04	1.58E+06
Sn-126	8.25E-09	1.77E+01	1.29E-02	1.29E-02	3.87E+04	3.87E+04
Sb-124	1.15E-02	9.40E+00	6.86E-03	1.61E-03	7.29E+04	3.11E+05
Sb-126	5.59E-02	9.10E+00	6.64E-03	3.26E-04	7.53E+04	1.54E+06

Table 4-1. DRLs Associated with a Total Effective Dose Equivalent of 0.5 rem Resulting from 1 Year of Ingestion

			Normalized DRLs mrem per pCi/L (TEDE)		DRLs (pCi/L)	
Radionuclide	Radioactive Decay Constant 1/d	DCF mrem/uCi	Without Radioactive Decay	With Radioactive Decay Only	Without Radioactive Decay	With Radioactive Decay Only
Sb-127	1.80E-01	6.18E+00	4.51E-03	6.87E-05	1.11E+05	7.28E+06
Te-127	1.78E+00	6.25E-01	4.56E-04	7.02E-07	1.10E+06	7.12E+08
Te-129	1.43E+01	2.33E-01	1.70E-04	3.26E-08	2.94E+06	1.53E+10
Te-129m	2.06E-02	1.10E+01	8.03E-03	1.07E-03	6.23E+04	4.68E+05
Te-131m	5.55E-01	7.22E+00	5.27E-03	2.60E-05	9.49E+04	1.92E+07
Te/I-132	2.13E-01	1.41E+01	1.03E-02	1.32E-04	4.86E+04	3.78E+06
I-125	1.15E-02	5.70E+01	4.16E-02	9.76E-03	1.20E+04	5.12E+04
I-129	1.21E-10	3.92E+02	2.86E-01	2.86E-01	1.75E+03	1.75E+03
I-131	8.62E-02	8.07E+01	5.89E-02	1.87E-03	8.49E+03	2.67E+05
Cs-134	9.20E-04	7.11E+01	5.19E-02	4.41E-02	9.63E+03	1.13E+04
Cs-136	5.29E-02	1.14E+01	8.32E-03	4.31E-04	6.01E+04	1.16E+06
Cs/Ba-137	6.33E-05	5.03E+01	3.67E-02	3.63E-02	1.36E+04	1.38E+04
Ba-133	1.77E-04	5.66E+00	4.13E-03	4.00E-03	1.21E+05	1.25E+05
Ba-140	5.44E-02	9.62E+00	7.02E-03	3.54E-04	7.12E+04	1.41E+06
La-140	4.13E-01	7.48E+00	5.46E-03	3.62E-05	9.16E+04	1.38E+07
Ce-141	2.13E-02	2.63E+00	1.92E-03	2.47E-04	2.60E+05	2.03E+06
Ce-143	5.04E-01	4.15E+00	3.03E-03	1.65E-05	1.65E+05	3.04E+07
Ce/Pr-144	2.44E-03	1.94E+01	1.42E-02	9.38E-03	3.53E+04	5.33E+04
Nd-147	6.31E-02	4.00E+00	2.92E-03	1.27E-04	1.71E+05	3.94E+06
Pm-145	1.07E-04	4.29E-01	3.13E-04	3.07E-04	1.60E+06	1.63E+06
Pm-147	7.23E-04	9.66E-01	7.05E-04	6.20E-04	7.09E+05	8.07E+05
Pm-149	3.13E-01	3.68E+00	2.69E-03	2.35E-05	1.86E+05	2.13E+07
Pm-151	5.86E-01	2.71E+00	1.98E-03	9.25E-06	2.53E+05	5.41E+07
Sm-151	2.11E-05	3.63E-01	2.65E-04	2.64E-04	1.89E+06	1.89E+06
Eu-152	1.42E-04	5.07E+00	3.70E-03	3.61E-03	1.35E+05	1.39E+05
Eu-154	2.16E-04	7.55E+00	5.51E-03	5.30E-03	9.07E+04	9.43E+04
Eu-155	3.83E-04	1.21E+00	8.83E-04	8.24E-04	5.66E+05	6.07E+05
Gd-153	2.86E-03	1.03E+00	7.52E-04	4.67E-04	6.65E+05	1.07E+06
Tb-160	9.59E-03	5.96E+00	4.35E-03	1.21E-03	1.15E+05	4.15E+05
Ho-166m	1.58E-06	7.33E+00	5.35E-03	5.35E-03	9.34E+04	9.35E+04
Tm-170	5.39E-03	4.89E+00	3.57E-03	1.56E-03	1.40E+05	3.20E+05
Yb-169	2.17E-02	2.63E+00	1.92E-03	2.42E-04	2.60E+05	2.06E+06
Hf-181	1.63E-02	4.15E+00	3.03E-03	5.08E-04	1.65E+05	9.84E+05
Ta-182	6.03E-03	5.70E+00	4.16E-03	1.68E-03	1.20E+05	2.97E+05
W-187	6.96E-01	2.33E+00	1.70E-03	6.70E-06	2.94E+05	7.47E+07
Ir-192	9.36E-03	5.07E+00	3.70E-03	1.05E-03	1.35E+05	4.77E+05
Au-198	2.57E-01	3.81E+00	2.78E-03	2.96E-05	1.80E+05	1.69E+07
Hg-203	1.49E-02	7.07E+00	5.16E-03	9.45E-04	9.69E+04	5.29E+05
Tl-204	5.02E-04	4.40E+00	3.21E-03	2.93E-03	1.56E+05	1.70E+05
Pb-210	8.51E-05	2.58E+03	1.88E+00	1.85E+00	2.65E+02	2.70E+02
Bi-207	4.99E-05	4.70E+00	3.43E-03	3.40E-03	1.46E+05	1.47E+05
Bi-210	1.38E-01	4.85E+00	3.54E-03	7.03E-05	1.41E+05	7.11E+06
Po-210	5.01E-03	4.48E+03	3.27E+00	1.50E+00	1.53E+02	3.33E+02
Ra-226	1.19E-06	1.04E+03	7.59E-01	7.59E-01	6.59E+02	6.59E+02
Ac-227	8.72E-05	1.19E+03	8.69E-01	8.55E-01	5.76E+02	5.85E+02
Th-227	3.70E-02	3.34E+01	2.44E-02	1.81E-03	2.05E+04	2.77E+05
U-235	2.70E-12	1.73E+02	1.26E-01	1.26E-01	3.96E+03	3.96E+03

Table 4-1. DRLs Associated with a Total Effective Dose Equivalent of 0.5 rem Resulting from 1 Year of Ingestion

			Normalized DRLs mrem per pCi/L (TEDE)		DRLs (pCi/L)	
Radionuclide	Radioactive Decay Constant 1/d	DCF mrem/uCi	Without Radioactive Decay	With Radioactive Decay Only	Without Radioactive Decay	With Radioactive Decay Only
U-238	4.25E-13	1.65E+02	1.20E-01	1.20E-01	4.15E+03	4.15E+03
Np-237	8.87E-10	3.96E+02	2.89E-01	2.89E-01	1.73E+03	1.73E+03
Np-239	2.94E-01	2.95E+00	2.15E-03	2.01E-05	2.32E+05	2.49E+07
Pu-236	6.66E-04	3.22E+02	2.35E-01	2.09E-01	2.13E+03	2.40E+03
Pu-238	2.16E-05	8.44E+02	6.16E-01	6.14E-01	8.12E+02	8.15E+02
Pu-239	7.89E-08	9.29E+02	6.78E-01	6.78E-01	7.37E+02	7.37E+02
Pu-240	2.90E-07	9.29E+02	6.78E-01	6.78E-01	7.37E+02	7.37E+02
Pu-241	1.32E-04	1.76E+01	1.28E-02	1.25E-02	3.89E+04	3.99E+04
Pu-242	5.04E-09	8.81E+02	6.43E-01	6.43E-01	7.77E+02	7.77E+02
Am-241	4.39E-06	7.55E+02	5.51E-01	5.51E-01	9.07E+02	9.08E+02
Am-242m	1.25E-05	7.07E+02	5.16E-01	5.15E-01	9.69E+02	9.71E+02
Am-243	2.57E-07	7.51E+02	5.48E-01	5.48E-01	9.12E+02	9.12E+02
Cm-242	4.26E-03	4.33E+01	3.16E-02	1.60E-02	1.58E+04	3.12E+04
Cm-243	6.66E-05	5.51E+02	4.02E-01	3.97E-01	1.24E+03	1.26E+03
Cm-244	1.05E-04	4.55E+02	3.32E-01	3.26E-01	1.51E+03	1.53E+03
Cm-245	2.23E-07	7.70E+02	5.62E-01	5.62E-01	8.90E+02	8.90E+02
Cm-246	4.01E-07	7.66E+02	5.59E-01	5.59E-01	8.94E+02	8.94E+02
Cf-252	7.19E-04	3.52E+02	2.57E-01	2.26E-01	1.95E+03	2.21E+03

Chapter 5
Accidental Radioactive Contamination of Human Food and Animal Feeds: Recommendations for State and Local Agencies

5.1 Introduction

This chapter presents, in its entirety, FDA Guidance entitled "Accidental Radioactive Contamination of Human Food and Animal Feeds: Recommendations to State and Local Agencies." FDA announced the availability of this document in the Federal Register of August 13, 1998 (63 FR 43402).

The Guidance provided information regarding radionuclides in food, and recommended new guidance levels for radionuclide activity concentration in food called Derived Intervention Levels (DILs), which is a term that replaced the previous FDA term "Levels of Concern" (LOCs). FDA adopted the DILs as guidance levels for food offered for import and for domestic food in interstate commerce. On November 11, 2005, FDA broadened the scope of coverage to include food that is accidentally or intentionally contaminated with radionuclides. (See www.fda.gov/ora/compliance_ref/cpg/cpgfod/cpg560-750.html)

FDA uses DILs to help determine whether domestic food in interstate commerce or food offered for import into the United States presents a safety concern. FDA determines whether foods contain unsafe levels of radionuclides on a case-by-case basis, considering the totality of the circumstances and the extent to which those circumstances depart from the assumptions that underlie the derivation of DILs. The DILs are not binding on FDA, the regulated industry, or the courts. In any given case, FDA may decide to initiate an enforcement action against food with concentrations below the DILs or decide not to initiate an enforcement action against food with concentrations that meet or exceed the DILs.

Individual Nature of Food and Water PAGs

As noted in Chapter 4, food and water PAGs are necessarily treated and presented separately. Depending on the event, the distribution of contamination between the food and water pathways could vary widely and would not be known quickly. Therefore, separate PAGs are given instead of a total ingestion PAG with specific portioning between the two pathways. As a result, the derivations of the response levels for food and water focus on the individual pathways appropriate to each but provide conservatism that accounts for the fact that both could be contaminated.

As a result of this approach, the derivations provide guidelines that can be used simultaneously or individually since they are neither additive nor restrictive between the two ingestion pathways. The numerical values are necessarily different to account for the different pathways.

The conservatism noted above was accomplished as follows: The food guidelines account for the fact that part of the ingestion PAG could come via drinking water and includes a drinking water component in the intake value of the derivation. Thus, although the food guidelines do not apply to drinking water, it was nevertheless accounted for. Likewise, the water guidelines account for the fact that part of the ingestion PAG could come via food by using a total water intake value (2 L) instead of drinking water only. Thus, although the water guidelines do not apply to food and prepared beverages, the water that is inherently contained in them is accounted for.

ACCIDENTAL RADIOACTIVE CONTAMINATION OF HUMAN FOOD AND ANIMAL FEEDS: RECOMMENDATIONS FOR STATE AND LOCAL AGENCIES

This document is intended to provide guidance. It represents the Agency's current thinking on the above. It does not create or confer any rights for or on any person and does not operate to bind FDA or the public. An alternative approach may be used if such approach satisfies the requirements of the applicable statute, regulations or both.

Radiation Programs Branch
Division of Mammography Quality and Radiation Programs
Office of Health and Industry Programs

Document issued on: August 13, 1998

Comments and suggestions may be submitted at any time for Agency consideration to: Radiation Programs Branch (HFZ-240), Center for Devices and Radiological Health, 1350 Piccard Drive, Rockville, MD 20850. Comments may not be acted upon by the Agency until the document is next revised or updated. For questions regarding the use or interpretation of this guidance document contact Donald Thompson at 301-827-0012 or DLT@cdrh.fda.gov.

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Food and Drug Administration
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ACCIDENTAL RADIOACTIVE CONTAMINATION OF HUMAN FOOD AND ANIMAL FEEDS: RECOMMENDATIONS FOR STATE AND LOCAL AGENCIES

Prepared by: Center for Devices and Radiological Health
Food and Drug Administration

ACCIDENTAL RADIOACTIVE CONTAMINATION OF HUMAN FOOD AND ANIMAL FEEDS: RECOMMENDATIONS FOR STATE AND LOCAL AGENCIES[1]

INTRODUCTION

Recommendations on accidental radioactive contamination of human food and animal feeds were issued in 1982 by the Food and Drug Administration (FDA) (FDA 1982, Shleien et al. 1982). Since then, there have been enough significant advancements related to emergency planning to warrant updating the recommendations. New scientific information and radiation protection philosophy are incorporated, experience gained since 1982 is included, and guidance developed by international organizations is taken into account (Schmidt 1988a, l988b, 1990, Burnett and Rosenstein 1989).

These recommendations provide guidance applicable to accidents at nuclear power plants and many other types of accidents where a significant radiation dose[2] could be received as a result of consumption of contaminated food. These recommendations rescind and replace the 1982 FDA recommendations.

GENERAL PROVISIONS

(a) Applicability.

The recommendations provide guidance to State and local agencies to aid in emergency response planning and execution of protective actions associated with production, processing, distribution, and use of human food and animal feeds accidentally contaminated with radionuclides. The recommendations do not authorize or apply to deliberate releases of radionuclides which are permitted and limited by general controls and/or terms and conditions stipulated by a regulatory agency.

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1 This document is intended to provide guidance. It represents the Agency's current thinking on the above. It does not create or confer any rights for or on any person and does not operate to bind FDA or the public. An alternative approach may be used if such approach satisfies the requirements of the applicable statute, regulations, or both.

2 The term “radiation dose” is used when the intended meaning is general or refers to more than one specific dose quantity.

(b) Scope.

The recommendations advise that health risk to the public be averted by limiting the radiation dose received as a result of consumption of accidentally contaminated food. This will be accomplished by: (1) setting limits, called Derived Intervention Levels (DILs) on the radionuclide activity concentration (concentration) permitted in human food, and (2) taking protective actions to reduce the amount of contamination.

DILs are limits on the concentrations permitted in human food distributed in commerce. They are established to prevent consumption of undesirable amounts of radionuclides and have units of radionuclide activity per kilogram of food, i.e. becquerels per kilogram, Bq/kg (previously used units - picocuries per kilogram, pCi/kg)[3]. Comparable limits were not provided in the 1982 FDA recommendations. DILs apply during the first year after an accident. If there is concern that food will continue to be significantly contaminated beyond the first year, the long-term circumstances need to be evaluated to determine whether the DILs should be continued or if other guidance may be more applicable.

Protective actions would be initiated subject to evaluation of the situation and would continue until, in the absence of the actions, the concentrations remain below the DILs. Protective actions can be taken to:

- avoid or limit, through precautionary measures, the amount of contamination that could become incorporated in human food and animal feeds, or

- delay or limit consumption of human food and animal feeds suspected of being contaminated until the concentration of contamination has been determined, or

- reduce the amount of contamination in human food and animal feeds.

[3] The International System of Units is used throughout this document. Units that were used in previous FDA guidance are shown in parenthesis in the main text of this document as reference points for the reader.

Limits on concentrations permitted in animal feeds are not given in these recommendations. However, protective actions for animal feeds are included as measures to reduce or prevent subsequent contamination of human food.

PROTECTIVE ACTION GUIDES

The 1982 FDA recommendations established two levels of Protective Action Guides (PAGs). PAGs were defined as "projected dose commitment values to individuals in the general population that warrant protective action following a release of radioactive material." The lower level, called the Preventive PAG, was a projected dose commitment of 5 mSv (0.5 rem) to the whole body, active bone marrow, or any other organ except the thyroid, or a projected dose commitment of 15 mSv (1.5 rem) to the thyroid. The Preventive PAG was associated with low-impact protective actions (e.g. placing dairy cows on stored feed). The upper level, called the Emergency PAG, was a projected dose commitment of 50 mSv (5 rem) to the whole body, active bone marrow, or any other organ except the thyroid, or a projected dose commitment of 150 mSv (15 rem) to the thyroid. The Emergency PAG was associated with higher-impact protective actions (e.g., diversion of fresh milk to cheese or milk powder).

The 1982 FDA recommendations were developed from the prevailing scientific understanding of the relative risks associated with radiation as described in the 1960 and 1961 reports of the Federal Radiation Council (FRC 1960, 1961). Since 1982, FDA and the other Federal agencies in the United States have adopted the methodology and terminology for expressing radiation doses provided by the International Commission on Radiological Protection (ICRP) in 1977 (ICRP 1977, ICRP l984a, EPA 1987). The ICRP's dose quantities for radiation protection purposes include effective dose equivalent, committed effective dose equivalent, dose equivalent for a specific tissue, and committed dose equivalent for a specific tissue[4,5].

[4] See Appendix A (Glossary) for explanation of these dose quantities and their use in this document.

[5] The ICRP adopted new recommendations in 1990, which include revisions in its methodology and terminology for expressing radiation doses and the relative risks associated with irradiation of specific organs (ICRP l991a). There is not yet consensus among the Federal agencies on the use of these changes.

These current recommendations replace the Preventive and Emergency PAGs with one set of PAGs for the ingestion pathway. The PAGs are 5 mSv (0.5 rem) for committed effective dose equivalent or 50 mSv (5 rem) committed dose equivalent to an individual tissue or organ, whichever is more limiting. These correspond to the "intervention levels of dose" consensus values set by international organizations (see Appendix B). Intervention levels of dose are radiation doses at which introduction of protective actions should be considered (ICRP 1984b). The FDA guidance retains use of the term Protection Action Guide (PAG) for consistency with U.S. Federal and State needs.

The current nominal estimate for the general population for lifetime total cancer mortality for low-LET (linear energy transfer) ionizing radiation, delivered at low doses and low dose rates, is 4.5×10^{-3} for a reference dose equivalent in the whole body of 100 mSv (10 rem) (CIRRPC 1992). For 5 mSv (0.5 rem) committed effective dose equivalent (the recommended PAG) the associated lifetime total cancer mortality would be $2.25 \times l0^{-4}$ or approximately 1 in 4400.[6] For comparison, the estimate of the normal lifetime total cancer mortality in the United States for the general population, not associated with additional radiation dose from ingestion of contaminated food from an accident, is 0.19 or approximately 1 in 5 (CIRRPC 1992). For example, in a general population of 10,000 individuals, each receiving a committed effective dose equivalent of 5 mSv (0.5 rem), the number of cancer deaths over the lifetimes of the individuals could increase in theory by about 2 cancer deaths, that is from the normal number of 1900 to 1902.

The numerical estimate of cancer deaths presented above for the recommended PAG of 5 mSv (0.5 rem) was obtained by the practice of linear extrapolation from the nominal risk estimate for lifetime total cancer mortality for the general population at 100 mSv (10 rem) dose equivalent in the whole body. Other methods of extrapolation to the low-dose region could yield higher or

[6] The alternate PAG of 50 mSv (5 rem) committed dose equivalent to a specific tissue or organ is always associated with a lifetime cancer mortality for the specific tissue that is as limiting or in some cases more limiting than the lifetime total cancer mortality associated with the PAG of 5 mSv (0.5 rem) for committed effective dose equivalent.

lower numerical estimates of cancer deaths. Studies of human populations exposed at low doses are inadequate to demonstrate the actual magnitude of risk. There is scientific uncertainty about cancer risk in the low-dose region below the range of epidemiological observation, and the possibility of no risk cannot be excluded (CIRRPC 1992).

DERIVED INTERVENTION LEVELS

A DIL corresponds to the concentration in food present throughout the relevant period of time that, in the absence of any intervention, could lead to an individual receiving a radiation dose equal to the PAG, or in international terms, the intervention level of dose. The equation given below is the basic formula for computing DILs.[7]

$$\text{DIL (Bq/kg)} = \frac{\text{PAG (mSv)}}{\text{f x Food Intake (kg) x DC (mSv/Bq)}}$$

Where:

DC = Dose coefficient; the radiation dose received per unit of activity ingested (mSv/Bq).

f = Fraction of the food intake assumed to be contaminated.

Food Intake = Quantity of food consumed in an appropriate period of time (kg).

The FDA DILs provide a large margin of safety for the public because each DIL is set according to a conservatively safe scenario for the most vulnerable group of individuals (see Appendix D). In addition, protective action would be taken if radionuclide concentrations were to reach or exceed a DIL at any point in time, even though such concentrations would need to be sustained throughout the relevant extended period of time for the radiation dose to actually reach the PAG. In practice, when FDA DILs are used, radiation doses to the vast majority of the affected public would be very small fractions of the PAG. As a result, future adjustments in the absolute values

[7] In the previous system of units DIL would be in units of pCi/kg, intervention level of dose in units of mrem and DCs in units of mrem/pCi.

of the PAGs would not necessarily require proportionate modifications in the DILs. Any modification of the DILs would depend on a review of all aspects of the conservatively safe scenario and how the DILs are applied.

Food with concentrations below the DILs is permitted to move in commerce without restriction. Food with concentrations at or above the DILs is not normally permitted into commerce. However, State and local officials have flexibility in whether or not to apply restrictions in special circumstances, such as permitting use of food by a population group with a unique dependency on certain food types.

(a) Use of Derived Intervention Levels for Food Monitoring after the Chernobyl Accident

Developments in the U.S.

Following the Chernobyl accident in 1986, a task group of representatives from FDA and the Food Safety and Inspection Service (FSIS) of the United States Department of Agriculture established DILs for application to imported foods under their respective regulatory control. The FDA DILs were called "Levels of Concern" (LOCs) (FDA l986a, 1986b) and the FSIS DILs were called "Screening Values." Food containing concentrations below the LOCs and Screening Values was allowed to be imported into the U.S.

FDA LOCs were derived from the 1982 Preventive PAGs and used the following assumptions:

- the entire intake of food would be contaminated,
- I-131 could be a major source of radiation dose for only 60 days following the accident
- Cs-134 + Cs-l37 could be a major source of radiation dose for up to one year.

The LOCs provided such a large margin of safety that derivation of LOCs for other radionuclides, judged to be of less health significance, was considered unnecessary.

The FSIS Screening Value for I-131 was the same as the FDA LOC for I-131 in infant foods. The FSIS Screening Value for Cs-l34 + Cs-137 initially differed from the FDA LOC because the

FSIS assumed that only meat and poultry (not 100% of the diet) would be contaminated (USDA 1986a). In November 1986, the FSIS changed the Screening Value for Cs-134 + Cs-137 to be the same as the FDA LOC (USDA 1986b, Engel et al. 1989). The FDA and FSIS DILs for the Chernobyl accident contamination in imported food after November 1986 are given in Table 1.

Table 1

FDA AND FSIS DERIVED INTERVENTION LEVELS FOR IMPORTED FOOD AFTER THE CHERNOBYL ACCIDENT, Bq/kg (pCi/kg)

	FDA LOC		FSIS Screening Value
Radionuclide	Infant Food	Other Food	Meat and Poultry
I-131	55 (1500)	300 (8000)	55 (1500)
Cs-134 + Cs-l37	370 (10,000)	370 (10,000)	370 (10,000)

The food monitoring results from FDA and others following the Chernobyl accident support the conclusion that I-131, Cs-134 and Cs-137 are the principal radionuclides that contribute to radiation dose by ingestion following a nuclear reactor accident, but that Ru-103 and Ru-l06 also should be included (see Appendix C). Also, use of DILs was shown to be a practical way to control the radiation dose from ingestion of food that has been contaminated as a result of a nuclear reactor accident.

International Activities

Efforts by international organizations to develop DILs have been extensive. Derivations have been based on the consensus value for the intervention level of dose, and have been for application within individual countries and in international trade. Each of the various international organizations selected values for the components in the basic formula for computing DILs, and each introduced additional judgments to arrive at its recommended DILs. As a result, the DILs recommended by the various organizations differed. The DILs adopted by the Commission of European Communities (CEC) for use in future accidents and those adopted

by the Codex Alimentarius (CODEX) for use in international trade[8] are presented in Appendix F.

(b) Recommended Derived Intervention Levels

In these recommendations, FDA uses the term Derived Intervention Level (DIL), which is consistent with international usage. DIL is equivalent to, and replaces the previous FDA term Level of Concern (LOC).

The recommended DILs are for radionuclides expected to deliver the major portion of the radiation dose from ingestion during the first year following an accident. The DILs are for accidental releases of radionuclides from large nuclear reactors and for other radiological emergencies where there is a possibility of accidental radioactive contamination of human food. The approach provides the flexibility necessary to respond to special circumstances that may be unique to a particular accident. A summary of the considerations in selecting DILs is given in this section, with a more detailed explanation available in Appendix D.

The types of accidents and the principal radionuclides for which the DILs were developed are:

- nuclear reactors (I-131; Cs-134 + Cs-l37; Ru-l03 + Ru-106),
- nuclear fuel reprocessing plants (Sr-90; Cs-137; Pu-239 + Am-241),
- nuclear waste storage facilities (Sr-90; Cs-137; Pu-239 + Am-241),
- nuclear weapons (i.e., dispersal of nuclear material without nuclear detonation) (Pu-239)
- radioisotope thermoelectric generators (RTGs) and radioisotope heater units (RHUs) used in space vehicles (Pu-238)

The radionuclides listed are expected to be the predominant contributors to radiation dose through ingestion. [9] Several radionuclides could be released by an accident at a nuclear

[8] An application of the CODEX DILs can be found in the International Atomic Energy Agency's (IAEA) interim edition of its basic safety standards for protection against ionizing radiation (IAEA 1994). IAEA based its "generic action levels for foodstuffs," found in Schedule V of IAEA 1994, on CODEX DILs.

[9] A discussion of the principal radionuclides for an accident at a nuclear reactor is given in Appendix C.

reactor, a nuclear fuel processing plant or a nuclear waste storage facility, while only the specific radionuclide used in a nuclear weapon or a space vehicle would be released in that type of accident. When more than one radionuclide is released, the relative contribution that a radionuclide makes to radiation dose from ingestion of subsequently contaminated food depends on the specifics of the accident and the mode of release (NRC 1975, DOE 1989, EPA 1977).

In unique circumstances, such as transportation accidents, other radionuclides may contribute radiation doses through the food ingestion pathway. These situations are not specifically treated in these recommendations. An evaluation of the radiation dose from ingestion of these other radionuclides should be performed, however, to determine if the PAGs would be exceeded. FDA should be notified during such an evaluation.

DILs were calculated for the nine radionuclides noted above. For each radionuclide, DILs were calculated for six age groups using Protective Action Guides, dose coefficients, and dietary intakes relevant to each radionuclide and age group. The age groups included 3 months, 1 year, 5 years, 10 years, 15 years and adult (>17 years). The dose coefficients used were from ICRP Publication 56 (ICRP 1989).

The DILs were based on the entire diet[10] for each age group, not for individual foods or food groups. The calculation presumed that contamination would occur in thirty percent of the dietary intake. The value of thirty percent was based on the expectation that normally less than ten percent of the annual dietary intake of most members of the population would consist of contaminated food. An additional factor of three was applied to account for limited sub-populations that might be more dependent on local food supplies. An exception was made for I-131 in the diets of the 3-month and 1-year age groups, where the entire intake over a sixty-day period was assumed to be contaminated.

[10] The “entire diet” includes tap water used for drinking.

The nine radionuclides comprised five radionuclide groups, each having common characteristics. The five groups are: Sr-90; I-131; Cs-l34 + Cs-137; Ru-103 + Ru-l06; and Pu-238 + Pu-239 + Am-24l. An accident could involve more than one of the five groups.

Protection of the more vulnerable segments of the population and the practicality of implementation were major considerations in the selection of the recommendations. These considerations lead to the single DIL or the single criterion for each radionuclide group that is presented in Table 2, based on the most limiting Protective Action Guide (PAG) and age group for the radionuclide group.[11]

The recommended DILs may be applied immediately following an accident. Early identification of other radionuclides that may be present in food is not required. However, the recommended DILs should be evaluated as soon as possible after an accident to ensure that they are appropriate for the situation. Appendix E presents a discussion on DILs for a number of other radionuclides that could be released from the reactor core of a nuclear power plant.

(c) Imported or Exported Food

The LOCs that applied to radioactive contamination from the Chernobyl accident in imported foods subject to FDA authority were given in an FDA Compliance Policy Guide (FDA 1986b). This guidance remains in effect and would be reviewed and modified as necessary to respond to any future accident resulting in radioactive contamination of imported food.

Food exported from the United States is controlled by standards, regulations and guidance in the importing countries. Two examples of guidance applicable to accidentally contaminated foods exported from the United States are the guidelines issued by the CODEX Alimentarius Commission of the Joint FAO/WHO Food Standards Program and the regulations adopted by the

[11] The PAG of 5 mSv (0.5 rem) for committed effective dose equivalent was most limiting for Cs-l34 + Cs-137 and Ru-l03 + Ru-l06; the PAG of 50 mSv (5 rem) for committed dose equivalent to a single specific tissue or organ was most limiting for Sr-90, I-131 and Pu-238 + Pu 239 + Am-241.

Commission of the European Communities (CEC). The DILs adopted by these two organizations (presented in Appendix F) differ from each other and from the FDA LOCs.

Table 2

Recommended Derived Intervention Level (DIL) or Criterion for Each Radionuclide Group[a],[b]

Radionuclide Group	All Components of the Diet (Bq/kg)	(pCi/kg)
Sr-90	160	4300
I-131	170	4600
Cs-134 + Cs-137	1200	32,000
Pu-238 + Pu-239 + Am-241	2	54
Ru-l03 + Ru-106[c]	$\frac{C_3}{6800} + \frac{C_6}{450} < 1$	$\frac{C_3}{180{,}000} + \frac{C_6}{12{,}000} < 1$

Notes:

(a) The DIL for each radionuclide group is applied independently (see discussion in Appendix D). Each DIL applies to the sum of the concentrations of the radionuclides in the group at the time of measurement.

(b) Applicable to foods as prepared for consumption. For dried or concentrated products such as powdered milk or concentrated juices, adjust by a factor appropriate to reconstitution, and assume the reconstitution water is not contaminated. For spices, which are consumed in very small quantities, use a dilution factor of 10.

(C) Due to the large difference in DILs for Ru-l03 and Ru-106, the individual concentrations of Ru-103 and Ru-106 are divided by their respective DILs and then summed. The sum must be less than one. C3 and C6 are the concentrations, at the time of measurement, for Ru-103 and Ru-106, respectively (see discussion in Appendix D).

PROTECTIVE ACTIONS

Protective actions are steps taken to limit the radiation dose from ingestion by avoiding or reducing the contamination that could occur on the surface of, or be incorporated into, human food and animal feeds. Such actions can be taken prior to and/or after confirmation of contamination. The protective actions for a specific accident are determined by the particulars of

the situation and once initiated they continue at least until the concentrations are expected to remain below the DILs.

For contamination events not effectively managed using DILs, protective actions appropriate to the situation would still be established and applied by the responsible officials. For example, in 1988 FDA developed guidance for use in responding to a contamination event that could have occurred from an uncontrolled reentry of the Russian satellite Cosmos 1900. FDA issued an advisory which specified protective actions against contamination in the form of widely but sparsely distributed discrete radioactive particulates and large pieces of radioactive debris (FDA 1988). The uncontrolled reentry of Cosmos 1900 did not occur.

(a) Protective Actions Prior to Confirmation of Contamination

Protective actions which can be taken within the area likely to be affected and prior to confirmation of contamination consist of:

- simple precautionary actions to avoid or reduce the potential for contamination of food and animal feeds, and
- temporary embargoes to prevent the introduction into commerce of food which is likely to be contaminated.

Protective actions can be taken before the release or arrival of contamination if there is advance knowledge that radionuclides may accidentally contaminate the environment.

For some types of accidents, determination of when and what protective actions would be taken may be facilitated by associating them with the accident classifications designated by the Nuclear Regulatory Commission (NRC) or the Department of Energy (DOE). For accidents involving commercial nuclear power reactors, the NRC has established four emergency classes: Notification of Unusual Event, Alert, Site Area Emergency, and General Emergency. Criteria for declaring these classes were published by the NRC (NRC 1980, 1991).

For accidents at DOE facilities, the DOE has established three emergency classes: Alert, Site Area Emergency, and General Emergency. These classes are comparable to those established by NRC. Incidents considered as Unusual Events by NRC licensees are covered as Unusual Occurrences by DOE (DOE 1992)

Simple precautionary actions include modest adjustment of normal operations prior to arrival of contamination. These will not guarantee contamination in food will be below the DILS but the severity of the forthcoming problem would be significantly reduced. Typical precautionary actions include covering exposed products, moving animals to shelter, corralling livestock and providing protected feed and water.

Precautionary actions should be implemented so as to avoid placing in jeopardy persons implementing the action. For example, in the case of an accident involving a commercial nuclear power plant, if the predictions of the magnitude of future off-site contamination are persuasive, precautionary actions that could be taken and completed before a declaration of Site Area Emergency or General Emergency could be considered. However, precautionary actions that would involve persons either not seeking shelter or leaving the immediate vicinity of shelter should not be taken after declaration of a Site Area Emergency or General Emergency. A temporary embargo on food and agricultural products (including animal feeds) prevents the consumption of food that is likely to be contaminated. Distribution and use of possibly contaminated food and animal feeds is halted until the situation can be evaluated and monitoring and control actions instituted. Temporary embargoes are applied when the concentrations are not yet known. Because there is potential for negative impact on the community, justification for this action must be significant. The embargo should remain in effect at least until results are obtained. For nuclear power plants, a temporary embargo should be issued only upon declaration of a General Emergency and if predictions of the extent and magnitude of the off-site contamination are persuasive. The geographical area under control by the embargo would depend on the accident sequence, the meteorological conditions, and the food affected.

(b) Protective Actions for Foods Confirmed to be Contaminated

Protective actions which should be implemented when the contamination in food equals or exceeds the DILs consist of:

- temporary embargoes to prevent the contaminated food from being introduced into commerce,
- normal food production and processing actions that reduce the amount of contamination in or on food to below the DILs.

A temporary embargo to prevent the introduction into commerce of food from a contaminated area should be considered when the amount of contamination equals or exceeds the DILs or when the presence of contamination is confirmed, but the concentrations are not yet known. The temporary embargo would continue until measurements confirm that concentrations are below the DILs.

Normal food production and processing procedures that could reduce the amount of radioactive contamination in or on the food could be simple, (such as holding to allow for radioactive decay, or removal of surface contamination by brushing, washing, or peeling) or could be complex (Grauby and Luykx 1990, FDA 1982, USDA 1989). The blending of contaminated food with uncontaminated food is not permitted because this is a violation of the Federal Food, Drug and Cosmetic Act (FDA 1991).

Protective actions focus on the specific foods having the greatest sources of radiation dose to the population. Factors that determine which foods are most significant include the agricultural practices in the area of contamination and the stage of the growing or harvest season at the time of the accident. In general, foods consumed fresh, such as milk, leafy vegetables, and fruit, are initially most important. Grains, root crops, other produce, and animal-derived food products are significant later as they come to market.

Specific protective actions to be implemented following an accident are not provided in these recommendations because there is such a wide variety of actions that could be taken. The protective actions would be determined by State and local officials with assistance from the growers, producers, and manufacturers.

(c) Protective Actions for Animal Feeds Confirmed as Contaminated

Protective actions to reduce the impact of contamination in or on animal feeds, including pasture and water, should also be taken on a case-by-case basis. Accurately forecasting the transfer of radioactive contamination through the agricultural pathway, from animal feed to human food, is problematic. The forecast is influenced by many factors, such as: the type of feed (e.g., fresh pasture, grain), other intakes (e.g., other feeds, supplements), the chemical form of the radionuclide, medications being administered, the animal species, and the type of resulting human food (e.g., milk, meat, eggs).

Protective actions that could be taken when animal feeds are contaminated include the substitution of uncontaminated water for contaminated water and the removal of lactating dairy animals and meat animals from contaminated feeds and pasture with substitution of uncontaminated feed. Corralling livestock in an uncontaminated area could also be effective. The protective actions would be determined by State and local officials, with assistance from growers, producers, and manufacturers.

APPENDIX A - GLOSSARY

absorbed dose - the quotient of the mean energy imparted by ionizing radiation, $d\varepsilon$, to matter of mass dm, unit: Gy (ICRU 1993)

averted dose - the radiation dose saved by implementing a protective action. It may be expressed in any of the relevant dose quantities. (ICRP 1991b)

Becquerel (Bq) - the unit of radionuclide activity or expectation value of the number of spontaneous nuclear transitions per unit of time. Bq = 1 transition per second. Unit: 1/s (ICRU 1980) The unit of radionuclide activity used in the previous FDA guidance was the curie (Ci)[12]. 1 Bq = 27 x 10^{-12} Ci = 27 picocuries (pCi).

committed dose equivalent (H_T) - the dose equivalent accruing in an organ or tissue up to a specified number of years after the intake of a radionuclide into the body. In this document, committed dose equivalent is always computed to age 70 years. Unit: Sv (ICRP 1984a)

committed effective dose equivalent (H_E) - committed dose equivalents to individual organs or tissues, multiplied by weighting factors, then summed. In this document, committed effective dose equivalent is always computed to age 70 years. Unit: Sv (ICRP 1984a)

contamination - radionuclides on or in food or animal feed as a result of an accidental release.

concentration - radionuclide activity concentration. Unit: Bq/kg; 1 Bq/kg = 27 pCi/kg.

Derived Intervention Level (DIL) - concentration derived from the intervention level of dose at which introduction of protective measures should be considered. Unit: Bq/kg (IAEA 1985)

[12] The International System of Units is used throughout the document. In this Glossary, the units that were used in previous FDA guidance are given as reference points for the reader in the definitions of the units “Becquerel” and “sievert”.

dose coefficient (DC) - the conversion coefficient for committed dose equivalent or committed effective dose equivalent per unit intake of radionuclide activity. Unit: Sv/Bq (ICRP 1989)

dose equivalent[13] (H_T) - the product of the absorbed dose in an organ or tissue and the quality factor. Unit: Sv (ICRU 1993)

effective dose equivalent (H_E) - sum of weighted dose equivalents for irradiated tissues or organs.

$$H_E = W_T H_T$$

where W_T is a weighting factor representing the proportionate stochastic risk for tissue T, and H_T is the mean dose equivalent received by tissue T. A list of tissues and their weighting factors is given by ICRP (ICRP 1984a). Unit: Sv

gray (Gy) - unit of absorbed dose. 1 Gy = 1 J/kg; 1 milligray (mGy) = 10^{-3} Gy. (ICRU 1993) The unit of absorbed dose in previous FDA publications was the rad. 1 Gy = 100 rad; 1 mGy = 0.1 rad.

intervention level of dose - reference level of dose equivalent to an individual at which introduction of protective actions should be considered. Unit: Sv (ICRP 1977, ICRP 1984b)

Level of Concern (LOC) - concentration in an imported food, set by FDA after the Chernobyl accident, below which unrestricted distribution in U.S. commerce is permitted.

precautionary action - action taken, prior to confirmation of contamination, to avoid or reduce the potential for contamination of food and animal feed.

[13] In this document, dose equivalent and committed dose equivalent are synonymous, and effective dose equivalent and committed effective dose equivalent are synonymous, because they

always refer to the general public, to radionuclides deposited in the body, and to values computed to age 70 years.

protective action - action taken to limit the radiation dose from ingestion by avoiding or reducing the contamination in or on human food and animal feeds.

Protective Action Guide (PAG) - committed effective dose equivalent or committed dose equivalent to an individual organ or tissue that warrants protective action following a release of radionuclides.

quality factor - modifying factor that weights the absorbed dose for the biological effectiveness of the charged particles producing the absorbed dose. (ICRU 1993)

sievert (Sv) - unit of dose equivalent. 1 Sv = 1 J/kg; 1 millisievert (mSv) = 10^{-3}Sv. (ICRU 1993) The unit of dose equivalent used in previous FDA guidance was the rem. 1 Sv = 100 rem; 1 mSv = 0.1 rem.

APPENDIX B - INTERNATIONAL CONSENSUS ON INTERVENTION LEVELS OF DOSE

In 1984, the International Commission on Radiological Protection (ICRP) recommended basic principles for planning intervention in the event of major radiation accidents and provided general guidance on radiation dose levels for the implementation of countermeasures (ICRP 1984b). The term "intervention level of dose" is used by ICRP for these dose levels. The ICRP guidance indicated that for any countermeasure there is a lower level of radiation dose below which the introduction of the countermeasure is unlikely to be warranted, an upper level of radiation dose above which the countermeasure should almost certainly be implemented, and when between these levels, the specifics of the situation determine which actions (if any) would be taken. For the control of food, ICRP indicated lower and upper levels of 5 mSv[14] and 50 mSv, respectively, for committed effective dose equivalent and 50 mSv and 500 mSv, respectively, for committed dose equivalent to an individual organ or tissue (ICRP l984b, ICRP 1977).

Since 1984, a number of international organizations have provided guidance dealing with the ingestion of radionuclides that was consistent with the ICRP guidance. These organizations included the Commission of the European Communities (CEC), the Codex Alimentarius Commission (CODEX), the Food and Agricultural Organization of the United Nations (FAO), the International Atomic Energy Agency (IAEA), the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (NEA), and the World Health Organization (WHO). All have adopted 5 mSv committed effective dose equivalent as the radiation dose level above which intervention was recommended (CODEX 1989, FAO 1987, IAEA 1986, Luykx 1989, NEA 1989, Waight 1988, WHO 1988). All except CODEX also adopted 50 mSv committed dose equivalent to an individual tissue or organ when that value is more limiting.

[14] The International System of Units is used throughout this document. See Appendix A, Glossary, for equivalence to units used in previous FDA guidance.

The ICRP has updated its general concepts on intervention in its Publication 60 (ICRP 1991a). Additional advice for intervention for protection of the public was provided in its Publication 63 (ICRP 1991b). The additional advice included an intervention level of averted dose (10 mSv effective dose[15] in a year) for restriction of a single foodstuff. ICRP considered this level appropriate for almost all cases, excepting when alternative food supplies are not available or population groups might suffer serious disruption of their food supply.

The ICRP approach recommended that in application of this intervention level of averted dose, the net benefit of withdrawing a particular foodstuff be made optimum, based on knowledge of the local situation and other assumptions about the monetary value assigned to the effective dose. The ICRP provided an example of how to evaluate the optimum. Such a procedure requires information that would not be available during the early phases of an accident.

The FDA uses the principles in the general guidance provided by ICRP in 1984 for the immediate response to a major radiation accident, recognizing that at later stages, after the local situation is stabilized and more clearly defined, the longer-term intervention for food can be modified based on more detailed evaluation of local conditions by local authorities. Therefore, the PAGs for the ingestion pathway at the onset of an accident are 5 mSv committed effective dose equivalent or 50 mSv committed dose equivalent to an individual tissue or organ, whichever is more limiting.

[15] Effective dose is the ICRP's revised formulation of effective dose equivalent, as described in its 1990 recommendations (ICRP 1991a)

APPENDIX C - RADIONUCLIDES DETECTED IN FOOD FOLLOWING THE CHERNOBYL NUCLEAR POWER PLANT ACCIDENT OF APRIL 1986

(a) Analyses of Imported Food by the United States and Canada

(1) I-131 and Cs-134 + Cs-l37

Shortly after the accident at Chernobyl on April 26, 1986, the FDA and FSIS of the USDA began sampling imported food for analysis to determine radionuclide activity concentrations. Regulatory actions were based on FDA Levels of Concern (LOCs) and the FSIS Screening Levels which were developed in 1986 and applied to I-131 and Cs-134 + Cs-137.

The regulatory results of FDA and FSIS import monitoring and analyses are summarized in Table C-1[16]. The radionuclide activity concentrations (concentrations) exceeded the FDA LOCs (Cunningham et al. 1992) in 23 out of 2,600 (0.9%) food samples, and exceeded the FSIS Screening Values (equal to the LOCs) (Engel et al. 1989, Randecker 1990) in 107 out of 6,295 (1.7%) meat and poultry samples. In general, Cs-134 and Cs-137 were the principal radionuclides detected by FDA and FSIS in the imported foods analyzed. I-131 was significant for only about two months. Cs-134 and Cs-l37 were also the dominant radionuclides in imported foods analyzed by Canada (NHW 1987). The European countries of the Nuclear Energy Agency (NEA) also found that I-131 and Cs-134 + Cs-137 contributed most of the radiation dose from radionuclides ingested with food contaminated by the Chernobyl accident (NEA 1987, NEA 1989).

(2) Radionuclides Other Than I-131 and Cs-134 + Cs-l37

In addition to the radionuclides used for regulatory actions (I-131, Cs-134 + Cs-137), a number of other radionuclides were detected in imported food entering the U. S. and Canada. Of these,

[16] The International System of Units is used throughout the document. See Appendix A, Glossary, for equivalence to units used in previous FDA guidance.

the most commonly detected radionuclides were Ru-103, Ru-l06, Ba-140, Sr-90, Ce-l44 and Zr-95. The results of FDA and Canadian import sampling for the latter radionuclides are summarized in Table C-2. The data supported the prediction that I-131 and Cs-134 + Cs-l37 were the most significant radionuclides for screening of imported foods, and that the other radionuclides were of significantly less importance.

During 1986, of about 500 imported samples monitored by FDA, Ru-l03 and Ru-106 were above the detection levels for 18 samples and Ba-140 was above the detection levels in 9 samples (Cunningham et al. 1992). These radionuclides were not detected after 1986. Only selected samples were analyzed for Sr-90. Two samples, containing relatively high amounts of Cs-134 + Cs-137 were analyzed for Sr-90 in 1986. In the following years, a total of 40 samples (those having Cs-134 + Cs-137 in excess of 110 Bq/kg) were analyzed for Sr-90. The Sr-90 was above the detection levels in all 42 samples.

For Canadian imported foods, Ru-103 was above detection levels in 46 of 840 samples analyzed during 1986 and 1987, and below detection levels in all samples analyzed later. Ru-l06 was above detection levels in 130 of 936 samples analyzed from 1986 through 1989 (Marshall 1992). Samples were analyzed for Ce-144 and Zr-95 from 1987 through 1989. Out of 486 samples, Ce-144 was above detection levels in 88 samples and Zr-95 was above detection levels in 3 samples.

Concentrations in FDA and Canadian imported samples were generally below 10% of the respective Derived Intervention Levels (DILs) given in Appendices D and E. The main exceptions were for Ru-106 in Canadian samples which ranged up to 42% of the DIL.

The results of analysis for imported samples collected by the U.S. and Canada are representative of collections distant from the accident site. Therefore, not only was the food variety relatively limited, but time delays between accident and sample collection, processing effects, and selective screening that exporters may have applied could have influenced the findings. Consequently, findings from samples collected at countries close to Chernobyl are most useful for U.S. decision-makers responding to a domestic release because these findings are more representative of a local contamination event.

(b) Analyses of Foods Collected Locally at Central and Eastern European Countries

In 1986, FDA received a variety of foods collected locally by United States Embassy staff in Central and Eastern European countries. A total of 48 samples from Bulgaria, Czechoslovakia, Finland, Hungary, Poland, Romania, Russia, and Yugoslavia, were analyzed. Results for Ru-103, Ru-106, and Ba-140 are summarized in Table C-3. The number of samples above detection levels for each radionuclide is given with the ranges of associated percentages relative to the DILs. I-131 and Cs-134 + Cs-l37 (not shown) were also detected in most of the samples. I-131 concentrations exceeded the DIL for 27 samples; while Cs-134 + Cs-137 exceeded the DIL for 2 samples.

Most of the 48 embassy samples were fresh vegetables. The edible portions were leafy for 28 samples and roots, bulbs, shoots, or seedlings for 12 samples. Ru-103 was above detection levels in all vegetables, exceeding its DIL for 6 samples. Ru-106 was above detection levels in all vegetables, exceeding its DIL for 14 samples. Ba-140 was above detection levels in 19, but did not exceed its DIL in any vegetables (maximum, 6.3% of DIL).

Other samples included 3 fresh fruit and 5 processed foods (cheese, yogurt, ice cream, and 2 milk samples). Ru-106 was above detection levels in all fruit (maximum, 14% of DIL) and in 2 processed foods (maximum, 29% of DIL). Ru-103 and Ba-140 were above detection levels but did not exceed 2% of their DILs in the fruit or processed food samples.

In September 1986, 28 samples of spices from Turkey and Greece (not offered for import) were provided by the American Spice Trade Association (ASTA) for testing by FDA. This set of samples represented deposition at a distance comparable to many of the Eastern European embassy samples but were analyzed at a later time after the accident. FDA analyzed spices for gamma-ray emitting radionuclides and Sr-90. Findings are included in Table C-3. Following the advice of CEC (CEC 1989a) and CODEX (CODEX 1989) for minor foods, a dilution factor of ten was applied to the concentrations for herbs, spices and flavorings, because they will be consumed in very small quantities.

Cs-l34 + Cs-l37 (not shown in Table C-3), Ru-103, Ru-106, and Sr-90 were above detection levels in all samples. I-131 and Ba-140 were below detection levels having undergone ten or more half-lives of radioactive decay.

Ru-103, having decayed for over four half-lives, ranged to a maximum of only 4.5% of its DIL while Sr-90, though having decayed very little, reached 10% of the DIL in only 8 samples (maximum, 30% of DIL). Ru-106 exceeded its DIL in 2 samples, was 50% to 100% in 5, and 10% to 50% in another 17.

(c) Conclusions

The results support the expectation that concentrations of I-131 and Cs-134 + Cs-137 would serve as the main indicators of the need for protective actions for imported and local food. However, concentrations of Ru-l06 were consistently in excess or at a significant fraction of the DIL, which suggests that Ru-106 should also serve as an indicator, i.e. be included as a principal radionuclide for nuclear reactor incidents.

Also, for local samples of fresh vegetables harvested during the first week of the incident, half of the samples had Ru-103 concentrations a significant fraction of the DIL and another quarter of the samples had Ru-103 concentrations in excess of the DIL. Consequently, it would be prudent to consider Ru-103 as a principal radionuclide for local deposition, particularly in the early phase of a nuclear reactor incident.

Sr-90 did not exceed 11% of the DIL in imported food (Table C-2). For the series of 28 local (ASTA) spice samples (Table C-3), Sr-90 was less than 30% of its DIL (generally a lower percent of the DIL than found for Ru-106 or Cs-134 + Cs-137). Also, the analytical method for determination of Sr-90 in food is lengthy compared to analysis for the gamma-ray emitting radionuclides, such that protective actions based on the concentration of Sr-90 could not be taken in a timely manner. Therefore, Sr-90 would not be an effective indicator of the need for protective actions in the early phase of a nuclear reactor incident.

During the first year after an accident, concentrations in local or imported food other than for I-131, Cs-134, Cs-137, Ru-l03 and Ru-106 are expected to be significant only when one or more of these principal radionuclides has exceeded its DIL. Therefore, the food would already have been subject to protective action.

Table C-1
SUMMARY OF U.S. REGULATORY FINDINGS FOR IMPORTED FOOD FOLLOWING THE CHERNOBYL ACCIDENT

Agency	Number of Samples Analyzed	Sampling Period	Number of Samples Contaminated Above Regulatory Limits[c]	
			I-131	Cs-134 + Cs-137
FDA[a]	2600	5/86-9/92	2	21
FSIS[b]	6295	5/86-10/88	-	107
Regulatory Limits[c]			300 Bq/kg	370 Bq/kg

(a) Food and Drug Administration
(b) Food Safety and Inspection Service of the U.S. Department of Agriculture
(c) FDA: Levels of Concern FSIS: Screening Levels

Table C-2

Ru-103, Ru-106, Ba-140, Sr-90, Ce-144, and Zr-95

IN IMPORTED FOOD SAMPLES[a] (UNITED STATES AND CANADA)

	Year, Number, and Type of Samples Analyzed[b]				Number of Samples with Measurable Concentration (Maximum Percent of Derived Intervention Level)					
					Ru-103[c]	Ru-106[c]	Ba-140	Sr-90	Ce-144	Zr-95
	U.S. (FDA)	1986	500[d]	Herbs	2 (0.02)	2 (9)				
				Others	16 (1.3)	16 (6)	9 (1.9)	2[e] (8)		
		1987	37[f]	Herbs				24 (3)		
				Others				13 (11)		
		1989	3[f]	Herbs				3 (2)		
	Canada	1986	450[d]	Herbs	26 (0.5)	13 (42)			58 (9)	3 (0.9)
				Others	10 (0.5)	1 (3)				
		1987	390[d]	Herbs	10 (0.05)	75 (22)				
				Others		2 (19)				
		1988	76	Herbs		30 (10)			26 (4)	
		1989	20	Herbs		9 (4)			4 (2)	

(a) For herbs (which include herbs, spices, and flavorings), a dilution factor of ten was applied to the concentrations. No dilution factor was applied for other foods.
(b) Number of samples analyzed for the featured radionuclides. Not equal to number of samples analyzed for principal radionuclides.
(c) The reported Ru-106 concentrations in FDA reports were usually the sum of Ru-103 + Ru-106. Values in this table are the individual Ru-103 and Ru-106 concentrations.
(d) Approximate number.
(e) Number of samples tested for Sr-90, one of which exceeded the 1986 LOC for Cs-134 + Cs-137.
(f) Only samples with Cs-134 + Cs-137 in excess of 0.3 of 1986 LOC were analyzed for Sr-90.

Table C-3

Ru-l03, Ru-106, Ba-140, and Sr-90
IN SAMPLES FROM U.S. EMBASSIES IN CENTRAL AND EASTERN EUROPE
AND FROM THE AMERICAN SPICE TRADE ASSOCIATION (ASTA)

	Type and Number	Number of Samples with Measurable Concentrations in 1986 (Range, as Percent of Derived Intervention Level)			
	of Samples Analyzed	Ru-103[(a)]	Ru-106	Ba-140	Sr-90
EMBASSY SAMPLES	Leafy Vegetables 28	28 (0.1-507)	28 (1-3500)	14 (0.1-6.3)	NA
	Non-leafy Vegetables 12	12 (1-222)	12 (9-1570)	5 (0.2-5.4)	NA
	Fruit 3	3 (0.3-1.4)	3 (4-14)	ND	NA
	Processed Food 5	2 (0.6-2)	2 (4-29)	3 (0.2-1.4)	NA
ASTA SAMPLES	Spices 28	28 (0.2-4.5)	28 (6-1640)	ND	28 (0.9-30)

(a) Embassy samples were received primarily in May and June 1986 and the ASTA samples in September 1986. Due to radioactive decay, the relative concentration of Ru-103 compared to Ru-106 is considerably lower for the ASTA samples than for the embassy samples.

NA Not analyzed.
ND Not detected.

APPENDIX D - DERIVATION OF RECOMMENDED DERIVED INTERVENTION LEVELS

The Derived Intervention Level (DIL) for a specific radionuclide is calculated as follows:

$$\text{DIL (Bq/kg)} = \frac{\text{PAG (mSv)}}{\text{f x Food Intake (kg) x DC (mSv/Bq)}}$$

Where:

DIL	=	Derived Intervention Level
PAG	=	Protective Action Guide
DC	=	Dose coefficient
Food Intake	=	Quantity of food consumed in an appropriate period of time
f	=	Fraction of food intake assumed to be contaminated

The recommended Protective Action Guides (PAGs) are 5 mSv[17] committed effective dose equivalent, or 50 mSv committed dose equivalent to individual tissues and organs, whichever is more limiting. These PAGs are consistent with the consensus of international organizations on the levels of radiation dose below which ingestion pathway interventions are generally not appropriate (see Appendix B).

Dose coefficients (DCs) are given in Table D-l and food intakes are given in Tables D-2 and D-3. The fraction of food intake assumed to be contaminated (f) equals 0.3, except for I-131 in infant diets where f equals 1.0.

(a) Radionuclides

Based upon data on radionuclides in human food following the Chernobyl accident, DILs for I-131, Cs-134, Cs-137, Ru-103 and Ru-106 would facilitate application of food monitoring programs following accidents involving nuclear reactors. For accidents at nuclear fuel

[17] The International System of Units is used throughout the document. See Appendix A, Glossary, for equivalence to units used in previous FDA guidance.

reprocessing facilities and nuclear waste storage facilities, DILs for Sr-90, Cs-137, Pu-239, and Am-241 would be used. For nuclear weapons accidents and accidents involving radioisotope thermal generators (RTGs) and radioisotope heater units (RHUs) used in space vehicles, DILs for Pu-239 and Pu-238, respectively, would be used. The selection of these radionuclides as the major contributors to radiation dose through ingestion is consistent with recommendations on DILs published by NEA, WHO, CODEX, and CEC (NEA 1989, WHO 1988, CODEX 1989, CEC 1989b, IAEA 1994).

(b) Age Groups and Dose Coefficients (DCs)

The general population was divided into six age groups ranging from infants to adults and corresponding to the age groups in ICRP Publication 56 (ICRP 1989) for which ICRP has published DCs. The age groups are 3 months, 1 year, 5 years, 10 years, 15 years, and adult. The radionuclides, age groups and dose coefficients used in the calculations are presented in Table D-l.

(C) Food Intake

Food intake included all dietary components including tap water used for drinking, and is the overall quantity consumed in one year, with exceptions in the period of time for I-131 ($T_{1/2}$ = 8.04 days) and Ru-103 ($T_{1/2}$ = 39.3 days). For these, the quantities consumed were for a 60-day period and a 280-day period, respectively, due to the more rapid decay of these radionuclides. The intake periods for I-131 and Ru-103 are the nearest whole number of days for decay of these radionuclides to less than 1% of the initial activities.

Dietary intakes were derived from a 1984 EPA report which presented average daily food intake by age and gender (EPA 1984a, EPA 1984b). The EPA intakes were based on data from the 1977-1978 Nationwide Food Consumption Survey published by the U. S. Department of Agriculture (USDA 1982, USDA 1983). The age groups and annual dietary intakes for various food classes and the total, calculated from data in the EPA report, are given in Table D-2.

The dietary intakes derived for the ICRP age groups for which DCs are available, using the results in Table D-2, are presented in Table D-3.

(d) Fractions of Food Intake Assumed to be Contaminated (f)

For food consumed by most members of the general public, ten percent of the dietary intakes was assumed to be contaminated. This assumption recognizes the ready availability of uncontaminated food from unaffected areas of the United States or through importation from other countries, and also that many factors could reduce or eliminate contamination of local food by the time it reaches the market[18].

Use of ten percent of the dietary intake as the portion contaminated was consistent with recommendations made by a Group of Experts to the Commission of the European Communities (CEC 1986a) and by the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (NEA 1989). The NEA noted that modification of this value would be appropriate if justified by detailed local findings.

FDA applied an additional factor of three to account for the fact that sub-populations might be more dependent on local food supplies. Therefore, during the immediate period after a nuclear accident, a value of 0.3 (i.e., thirty percent) is the fraction of food intake that FDA recommends should be presumed to be contaminated. If, subsequently, there is convincing local information that the actual fraction of food intake that is contaminated (f) is considerably higher or lower, there will be adequate time for State and local officials to determine whether to adjust the value of f (and therefore adjust the values of the DILs) for the affected area.

For infants, (i.e., the 3-month and 1-year age groups) the diet consists of a high percentage of milk and the entire milk intake of some infants over a short period of time might come from supplies directly impacted by an accident. Therefore, f was set equal to 1.0 (100%) for the infant diet.

[18] In most situations, one would expect less than ten percent of the dietary intakes to be contaminated.

(e) Selection of Recommended Derived Intervention Levels

DILs are presented in Table D-4 for Sr-90, I-131, Cs-134, Cs-l37, Ru-103, Ru-106, Pu-238, Pu-239, and Am-241 for six population age groups and applicable PAGs. To facilitate the execution of food monitoring programs, two criteria were used in selecting FDA's recommended DILs. First, the most limiting DIL for either of the applicable PAGs was selected for each of the nine radionuclides. These DILs are presented in Table D-5 for each of the six age groups. In addition, the average DIL is presented for the radionuclide group Pu + Am, composed of Pu-238, Pu-239, and Am-241, and the radionuclide group Cs, composed of Cs-134 + Cs-137. The three radionuclides in the Pu + Am group deposit on the bone surface and are alpha-particle emitters. The radionuclides in the Cs group are deposited throughout the body and are beta-particle and gamma-ray emitters. The average values are recommended for these groups because the calculated DILs for radionuclides in each group are similar.

The radionuclides Ru-103 and Ru-106 are chemically identical, are deposited throughout the body, and are beta-particle and gamma-ray emitters. However, their widely differing half lives (i.e., 39.3 days and 373 days, respectively) result in markedly differing individual DILs which do not permit simple averaging. Instead, the concentrations of Ru-103 (C_3) and Ru-106 (C_6) are divided by their respective DILs and are then summed[19]. The sum must be less than one.

Therefore,
$$\frac{C_3}{DIL_3} + \frac{C_6}{DIL_6} < 1.0 \qquad \text{(equation D-1)}$$

This assures that the sum of the separate radiation dose contributions from the Ru-l03 and Ru-l06 concentrations will be less than that required by the Protective Action Guide during the first year after an accident.

[19] Laboratories that are not equipped to resolve separately the concentrations for Ru-103 and Ru-106 should contact FDA for alternate procedures.

Second, there are dietary components which are common to all six age groups. A principal example is fresh milk, for which the consumer of particular supplies cannot be identified in advance. Therefore, the most limiting DIL for all age groups in Table D-5, for each radionuclide or radionuclide group, was selected and is applicable to all components of the diet.

These DILs are presented in Table D-6 and were rounded to two significant figures (one significant figure for the Pu + Am group). These are the FDA's recommended DILs.

The DILs in Table D-6 apply independently to each radionuclide or radionuclide group, because they apply to different types of accidents, or in the case of a nuclear reactor accident, to different limiting age groups. However, the DILs for Ru-103 and Ru-106 are used in equation D-l to evaluate that criterion for the radionuclide group Ru-103 + Ru-106.

The FDA recommended DILs in Table D-6 are given in Table 2 in the main text, along with clarifying notes on application of the DILs.

Table D-l

DOSE COEFFICIENTS (mSv/Bq) [(a)]

Radionuclide	Age Group					
	3 month	1 year	5 years	10 years	15 years	Adult
Sr-90 bone srfc	l.0E-03	7.4E-04	3.9E-04	5.5E-04	l.2E-03	3.8E-04
Sr-90	l.3E-04	9.1E-05	4.lE-05	4.3E-05	6.7E-05	3.5E-05
I-131 thyroid	3.7E-03	3.6E-03	2.1E-03	l.1E-03	6.9E-04	4.4E-04
I-131	1.1E-04	1.1E-04	6.3E-05	3.2E-05	2.1E-05	1.3E-05
Cs-134	2.5E-05	1.5E-05	l.3E-05	l.4E-05	2.0E-05	1.9E-05
Cs-137	2.0E-05	1.1E-05	9.0E-06	9.8E-06	1.4E-05	1.3E-05
Ru-103	7.7E-06	5.1E-06	2.7E-06	1.7E-06	1.0E-06	8.1E-07
Ru-106	8.9E-05	5.3E-05	2.7E-05	1.6E-05	9.2E-06	7.5E-06
Pu-238 bone srfc	1.6E-01	1.6E-02	1.5E-02	1.5E-02	1.6E-02	1.7E-02
Pu-238	1.3E-02	1.2E-03	1.0E-03	8.8E-04	8.7E-04	8.8E-04
Pu-239 bone srfc	1.8E-01	1.8E-02	1.8E-02	1.7E-02	1.9E-02	1.8E-02
Pu-239	1.4E-02	1.4E-03	1.1E-03	1.0E-03	9.8E-04	9.7E-04
Am-241 bone srfc	2.0E-01	1.9E-02	1.9E-02	1.9E-02	2.1E-02	2.0E-02
Au-241	1.2E-02	1.2E-03	1.0E-03	9.0E-04	9.1E-04	8.9E-04

(a) Dose coefficients are from ICRP Publication 56 (ICRP 1989). The committed effective dose equivalents or committed dose equivalents are computed to age 70 years.

Table D-2

ANNUAL DIETARY INTAKES (kg/y) [a]

Food Class	AGE GROUP(years) < 1	1-4	5-9	10-14	15-19	20-24	25-29	30-39	40-59	60 & up
Dairy (fresh milk) [b]	208 (99.3)	153 (123)	180 (163)	186 (167)	167 (148)	112 (96.5)	98.2 (79.4)	86.4 (66.8)	80.8 (61.7)	90.6 (70.2)
Egg	1.8	7.2	6.2	7.0	9.1	10.3	10.2	11.0	11.4	10.5
Meat	16.5	33.7	46.9	58.4	69.2	71.2	72.6	73.4	70.7	56.3
Fish	0.3	2.5	4.0	4.9	6.1	6.8	7.6	7.1	8.0	6.3
Produce	56.6	59.9	82.3	96.0	97.1	91.4	99.1	102	115	121
Grain	20.4	57.6	79.0	90.6	89.4	77.3	78.4	73.7	70.2	67.1
Beverage (tap water)[b]	112 (62.3)	271 (159)	314 (190)	374 (226)	453 (243)	542 (240)	559 (226)	599 (232)	632 (268)	565 (278)
Misc	2.0	9.3	13.3	14.8	13.9	10.9	11.9	12.5	13.3	13.0
TOTAL	418	594	726	832	905	922	937	965	1001	930

(a) Computed from daily intake values in grams per day provided in (EPA 1984b). The total annual intakes are rounded to nearest 1 kg/y.
(b) Fresh milk is included in the dairy entry, and tap water used for drinking is included in the beverage entry. The total annual intakes (kg/y) for fresh milk and tap water are also each given separately in parentheses.

Table D-3

DIETARY INTAKES

FOR ICRP AGE GROUPS

ICRP age group	Intake (kg) annual[(a)]	280-day Ru-103	60- day I-131
3 months	418	320	69
1 year	506	387	83
5 years	660	506	109
10 years	779	597	128
15 years	869	666	143
Adult	943	723	155

(a) The annual dietary intakes for the ICRP age groups were obtained by assigning or averaging the appropriate annual dietary intakes given in Table D-2 for the EPA age groups, as follows:

3 months:		<1
1 year:	average	<1 and 1-4
5 years:	average	1-4 and 5-9
10 years:	average	5-9 and 10-14
15 years:	average	10-14 and 15-19
Adult:	average	15-19, 20-24, 25-29, 30-39, 40-59, 60 and up

Table D-4

PAGs AND DERIVED INTERVENTION LEVELS[(a)]

(individual radionuclides, by age groups)

Radionuclide	PAG (mSv)	Derived Intervention Levels(Bq/kg) 3 months	1 year	5 years	10 years	15 years	Adult
Sr-90 bone srfc.	50	400	445	648	389	160	465
Sr-90	5	308	362	616	497	286	505
I-131 thyroid	50	196	167	722	1200	1690	2420
I-131	5	659	548	2410	4110	5540	8180
Cs-l34	5	1600	2190	1940	1530	958	930
Cs-l37	5	2000	2990	2810	2180	1370	1360
Ru-103	5	6770	8410	12200	16400	25000	28400
Ru-106	5	449	621	935	1340	2080	2360
Pu-238 bone srfc.	50	2.5	21	17	14	12	10
Pu-238	5	3.1	27	25	24	22	20
Pu-239 bone srfc.	50	2.2	18	14	13	10	9.8
Pu-239	5	2.9	24	23	21	20	18
Am-241 bone srfc.	50	2.0	17	13	11	9.1	8.8
Am-241	5	3.3	27	25	24	21	20

(a) Derived Intervention Levels were computed using dose coefficients from Table D-1, dietary intakes from Table D-3, and "f" as given below:

0.3 (except for I-131 in infant diets, i.e., the 3-month and 1-year age groups)
1.0 (I-131 in infant diets)

5

(b) The observed trend in Derived Intervention Levels for Sr-90 as a function of age, i.e. minimum values at 15 years, results primarily from the mass of exchangeable strontium in bone as a function of age (Leggett et al. 1982).

Table D-5

DERIVED INTERVENTION LEVELS (Bq/kg)

(individual radionuclides, by age group, most limiting of either PAG)

Radionuclide	3 months	1 year	5 years	10 years	15 years	Adult
Sr-90	308	362	616	389	160	465
I-131	196	167	722	1200	1690	2420
Cs-134	1600	2190	1940	1530	958	930
Cs-l37	2000	2990	2810	2180	1370	1360
Cs group[(a)]	1800	2590	2380	1880	1160	1150
Ru-103	6770	8410	12200	16400	25000	28400
Ru-106	449	621	935	1340	2080	2360
Pu-238	2.5	21	17	14	12	10
Pu-239	2.2	18	14	13	10	9.8
Am-241	2.0	17	13	11	9.1	8.8
Pu+Am group[(b)]	2 .2	19	15	13	9. 6	9. 3

(a) Computed as: (DIL for Cs-134 + DIL for Cs-l37)/2

(b) Computed as: (DIL for Pu-238 + DIL for Pu-239 + DIL for Am-241) /3

Table D-6

DERIVED INTERVENTION LEVELS (Bq/kg)

(radionuclide groups, most limiting of all diets)

Radionuclide Group	Derived Intervention Levels	
Sr-90	160	(15 years)
I-131	170	(1 year)
Cs group	1200	(adult)
Ru-103[(a)]	6800	(3 months)
Ru-106[(a)]	450	(3 months)
Pu + Am group	2	(3 months)

(a) Due to the large differences in DILs for Ru-103 and Ru-106, the individual concentrations of Ru-103 and Ru-106 are divided by their respective DILs and then summed. The sum must be less than one.

APPENDIX E - DERIVED INTERVENTION LEVELS FOR OTHER RADIONUCLIDES IN THE INVENTORY OF THE CORE OF AN OPERATING NUCLEAR REACTOR

After a reactor accident, radionuclides other than the principal radionuclides may also be detected in the food supply, usually at much lower concentrations (See Appendix C). However, in the event other radionuclides are present in significant concentrations, this Appendix presents Derived Intervention Levels (DILs) for a number of other radionuclides commonly found in a reactor core inventory.

The DILs for fifteen other radionuclides were determined by the same procedure used in Appendix D. The Protective Action Guides were also the same, i.e. 5 mSv^{20} committed effective dose equivalent, or 50 mSv committed dose equivalent to individual tissues and organs.

Age groups and their related food intakes for one year were given previously in Table D-3, Appendix D. Dietary intakes for seven of the fifteen other radionuclides that have half-lives much less than one year were computed for the periods of time (i.e. in nearest whole number of days) required for the radionuclides to decay to less than 1% of the initial activities. Table E-l and Table E-2 give the relevant data for these seven radionuclides.

IDose coefficients for seven of the fifteen other radionuclides included in this Appendix are provided in ICRP Publication 56 (ICRP 1989) for all six age groups. For the remaining eight radionuclides, DCs are available in NRPB Publication GS7 (NRPB 1987), but for only three age groups, i.e. 1-year, 10-year and adult. The more limited data in NRPB publication GS7 are supplemented as indicated in the next section.

Fractions of food intake assumed to be contaminated (f) are:

0.3 for all radionuclides except Te-l32, I-133 and Np-239 in infant diets (i.e., the 3-month and 1-year age groups);

1.0 for Te-132, I-133 and Np-239 in infant diets.

[20] The International System of Units is used throughout the document. See Appendix A, Glossary, for equivalence to units used in previous FDA guidance.

SELECTION OF DERIVED INTERVENTION LEVELS

The dose coefficients in ICRP Publication 56 and NRPB Publication GS7 are for individual tissues and the effective dose equivalent, as formulated in ICRP Publication 26. ICRP has also developed dose coefficients for individual tissues and the effective dose, as formulated in ICRP publication 60. These latter dose coefficients were published in ICRP Publication 67 (ICRP 1993) and ICRP 72 Publication (ICRP 1996) for all six age groups. Review of all these DCs demonstrated that the trend for relative values of DCs with age for any given radionuclide or for radionuclides with common biokinetic characteristics and half lives is similar. Therefore, DCs for the missing 3-month, 5-year, and 15-year age groups were derived for the eight radionuclides in NRPB Publication GS7, based on the trends observed in the three sets of ICRP tables. Table E-3 presents the derived DCs for these three age groups and the data from ICRP Publication 67 or 72 used in the derivations. Table E-4 gives the DCs used in computing the DILs for all fifteen radionuclides presented in Table E-5. DILs have been rounded to two significant figures (except one significant figure for Np-237 and Cm-244).

In the same manner as for the principal radionuclides in Appendix D, the most limiting Derived Intervention Level for a radionuclide for either PAG is given in Table E-6 for each age group. Then, the most limiting DIL for a radionuclide for each age group is presented in Table E-7.

During the immediate period after a nuclear reactor accident, decisions on protective actions for food may be required and may need to be based on the general status of the facility or the overall prognosis for worsening conditions. Once food monitoring data is available, the recommended DILs or criterion for the principal radionuclides I-131, Cs-134 + Cs-l37, and Ru-l03 + Ru-106 recommended in Table 2 of the main text should be used.

The more complex radiochemical or gamma-ray spectrometric analyses for the fifteen other radionuclides listed in this Appendix would not be generally available. If other radionuclides are subsequently detected in food, there will be adequate time to review the data on the concentrations of the other radionuclides to evaluate whether their contributions to radiation dose via ingestion are unexpectedly high, and to determine whether additional radionuclides should be controlled by their respective DILs in Table E-7. The evaluation takes place with knowledge of the radiation dose represented by the concentrations of the principal radionuclides, which may already exceed one or more of their DILs.

Table E-1

NEAREST WHOLE NUMBER OF DAYS FOR SHORT-LIVED RADIONUCLIDES TO HAVE DECAYED TO LESS THAN 1% OF INITIAL ACTIVITY (A_o)

Radionuclide	Half-life	Number of Days for Decay to Less Than 1% of A_o
I-133	20.8 h	6
Np-239	2.36 d	16
Te-132	3.26 d	22
Ba-140	12.7 d	85
Ce-141	32.5 d	217
Nb-95[(a)]	35.2 d	236
Sr-89	50.5 d	336

(a) Applies to Nb-95 existing in core inventory of an operating reactor at the time of release. Nb-95 produced as a result of decay of released parent Zr-95 is accounted for in the treatment of Zr-95.

TABLE E-2

DIETARY INTAKES

ICRP Age Group (annual intake, kg) [(a)]	Radionuclide and days[(b)] for decay to 1%						
	Sr-89	Nb-95	Ce-141	Ba-140	Te-132	Np-239	I-133
	336	236	217	85	22	16	6
	Intake (kg)						
3 months (418)	385	270	249	97	25	18	6.9

1 year	(506)	466	327	301	118	31	22	8.3
5 years	(660)	608	427	392	154	40	29	11
10 years	(779)	717	503	463	181	47	34	13
15 years	(869)	799	562	517	202	52	38	14
Adult	(943)	868	610	561	220	57	41	16

(a) The annual intakes (from Table D-3) are for radionuclides which do not decay to less than 1% of initial activity within a year.
(b) Time periods for intakes are for specified radionuclides (from Table E-1) which decay to less than 1% of the initial activity within a year.

Table E-3

DOSE COEFFICIENTS (mSv/Bq) DERIVED FOR THE 3-MONTH, 5-YEAR AND 15-YEAR AGE GROUPS[a] NOT AVAILABLE IN NRPB PUBLICATION GS7, USING DATA IN ICRP PUBLICATIONS[b]

Radionuclide [c]		References Used	Dose Coefficients by Age Group					
			3 months	1 year	5 years	10 years	15 years	Adult
Sr-89	HE	NRPB GS7	**3.0E-05**	l.5E-05	**7.7E-06**	5.2E-06	**3.5E-06**	2.2E-06
Sr-89	E	ICRP 72	3.6E-05	1.8E-05	8.9E-06	5.8E-06	4.0E-06	2.6E-06
Y-91	LLI	NRPB GS7	**3.3E-04**	2.1E-04	**1.1E-04**	7.1E-05	**3.8E-05**	3.0E-05
Y-91	E	ICRP 72	2.8E-05	1.8E-05	8.8E-06	5.2E-06	2.9E-06	2.4E-06
Te-132	THY	NRPB GS7	**4.6E-04**	2.2E-04	**1.3E-04**	6.0E-05	**3.5E-05**	1.9E-05
Te-132	THY	ICRP 67	6.2E-04	3.0E-04	l.6E-04	7.1E-05	4.6E-05	2.9E-05
I-133	THY	NRPB GS7	**9.6E-04**	8.6E-04	**5.0E-04**	2.3E-04	**1.5E-04**	8.3E-05
I-133	E	ICRP 72	4.9E-05	4.4E-05	2.3E-05	1.0E-05	6.8E-06	4.3E-06
Ba-140	LLI	NRPB GS7	**2.1E-04**	1.8E-04	**9.7E-05**	6.0E-05	**3.1E-05**	2.6E-05
Ba-140	LLI	ICRP 67	2.2E-04	1.9E-04	9.9E-05	5.7E-05	3.lE-05	2.9E-05
Ce-141	LLI	NRPB G57	**9.3E-05**	6.0E-05	**3.3E-05**	2.0E-05	**1.2E-05**	8.7E-06
Ce-l41	LLI	ICRP 67	9.8E-05	6.3E-05	3.2E-05	l.9E-05	1.1E-05	8.7E-06
Cm-242	BS	NRPB GS7	**2.1E-02**	2.6E-03	**1.4E-03**	8.9E-04	**5.6E-04**	4.5E-04

Cm-242	E	ICRP 72	5.9E-04	7.5E-05	3.9E-05	2.4E-05	1.5E-05	l.2E-05
Cm-244	ES	NRPB GS7	**2.5E-01**	2.5E-02	**1.6E-02**	l.2E-02	**9.9E-03**	9.8E-03
Cm-244	E	ICRP 72	2.9E-03	2.9E-04	1.9E-04	l.4E-04	1.2E-04	1.2E-04

(a) The dose coefficients (DCs) derived for age groups not available in NRPB Publication GS7 are indicated in bold font.

5 (b) The derived DCs were obtained by multiplying the DC for the NRPB age group contiguous to the missing NRPB age group by the following: the ratio of the DC for the desired age group to the DC of the contiguous age group, from the supporting ICRP data. When there were two contiguous age groups (i.e. for the 5-year and 15-year age groups), the two resulting DCs for the missing NRPB age groups were averaged.

(c) The dose quantity used is noted for each radionuclide. LLI is lower large intestine, THY is thyroid, BS is bone surface, H_E is effective dose equivalent, and E is effective dose.

Table E-4 DOSE COEFFICIENTS (mSv/Bq)[(a)]

Radionuclides	AGE GROUP					
	3 months	1 year	5 years	10 years	15 years	Adult
Sr-89 lower large intestine	2.8E-05	1.4E-04	7.1E-05	4.8E-05	2.3E-05	2.1E-05
Sr-89	3.0E-05	1.5E-05	7.7E-06	5.2E-06	3.5E-06	2.2E-06
Y-91 lower large intestine	3.3E-04	2.1E-04	1.1E-04	7.1E-05	3.8E-05	3.0E-05
Y-91	2.8E-05	1.7E-05	8.8E-06	5.7E-06	3.1E-06	2.4E-06
Zr-95	1.0E-05	6.6E-06	3.6E-06	2.2E-06	1.4E-06	1.1E-06
Nb-95	5.2E-06	3.7E-06	2.1E-06	1.3E-06	8.6E-07	6.8E-07
Te-132 thyroid	4.6E-04	2.2E-04	1.3E-04	6.0E-05	3.5E-05	1.9E-05
Te-132	3.0E-05	1.9E-05	1.1E-05	6.4E-06	3.4E-06	2.0E-06
I-129 thyroid	3.7E-03	4.3E-03	3.5E-03	3.8E-03	2.8E-03	2.1E-03
I-129	1.1E-04	1.3E-04	1.0E-04	1.1E-04	8.4E-05	6.4E-05
I-133 thyroid	9.6E-04	8.6E-04	5.0E-04	2.3E-04	1.5E-04	8.3E-05
I-133	2.9E-05	2.6E-05	1.8E-05	7.0E-06	4.3E-06	2.5E-06
Ba-140 lower large intestine	2.1E-04	1.8E-04	9.7E-05	6.0E-05	3.1E-05	2.6E-05
Ba-140	2.5E-05	1.4E-05	7.6E-06	5.1E-06	3.7E-06	2.3E-06
Ce-141 lower large intestine	9.3E-05	6.0E-05	3.3E-05	2.0E-05	1.1E-05	8.7E-06
Ce-141	7.8E-06	4.9E-06	2.5E-06	1.6E-06	9.0E-07	7.0E-07
Ce-144 lower large intestine	7.6E-04	4.9E-04	2.4E-04	1.5E-04	8.2E-05	6.6E-05
Ce-144	8.0E-05	4.3E-05	2.1E-05	1.3E-05	7.2E-06	5.8E-06
Np-237 bone surface	1.0E-01	8.9E-03	9.3E-03	9.9E-03	1.2E-02	1.2E-02
Np-237	5.5E-03	4.9E-04	4.3E-04	4.0E-04	4.7E-04	4.5E-04
Np-239 lower large intestine	9.8E-05	6.4E-05	3.2E-05	1.9E-05	1.1E-05	8.8E-06
Np-239	9.6E-06	6.3E-06	3.2E-06	1.9E-06	1.1E-06	8.7E-07
Pu-241 bone surface	3.3E-03	3.4E-04	3.5E-04	3.9E-04	3.9E-04	3.7E-04
Pu-241	2.2E-04	2.2E-05	2.1E-05	2.0E-05	2.0E-05	1.9E-05
Cm-242 bone surface	2.1E-02	2.6E-03	1.4E-03	8.9E-04	5.6E-04	4.5E-04
Cm-242	1.4E-03	1.8E-04	9.8E-05	6.4E-05	3.8E-05	3.0E-05
Cm-244 bone surface	2.5E-01	2.5E-02	1.6E-02	1.2E-02	9.9E-03	9.8E-03
Cm-244	1.4E-02	1.4E-03	9.2E-04	6.7E-04	5.9E-04	5.4E-04

(a) When dose coefficients were available from ICRP Publication 56 (ICRP 1989), they were given for all six age groups. When dose coefficients were available only from NRPB GS7 (NRPB 1987), they were given for only 3 age groups (i.e. 1 year, 10 years, and adult),and derived for the other 3 age groups (see Table E-3). The committed effective dose equivalents or committed dose equivalents are computed to age 70 years.

TABLE E-5 PAG AND DERIVED INTERVENTION LEVELS[(a)]

Radionuclide	PAG (mSv)	Derived Intervention Levels (Bq/kg) 3 months	1 year	5 years	10 years	15 years	Adult
Sr-89 lower large intestine	50	1600	2600	3900	4800	9100	9100
Sr-89	5	1400	2400	3600	4500	5800	8700
Y-91 lower large intestine	50	1200	1600	2300	3000	5300	5900
Y-9l	5	1500	1900	2900	3800	6200	7400
Zr-95	5	4000	5000	7000	9700	14000	16000
Nb-95	5	12000	14000	19000	26000	35000	40000
Te-132 thyroid	50	4400	7300	35000	59000	89000	150000
Te-132	5	6700	8500	38000	55000	94000	150000
I-129 thyroid	50	110	76	72	56	69	84
I-129	5	360	250	250	200	230	280
I-133 thyroid	50	7600	7000	30000	56000	79000	130000
I-133	5	25000	23000	84000	180000	280000	420000
Ba-140 lower large intestine	50	8200	7900	11000	15000	27000	29000
Ba-140	5	6900	10000	14000	18000	22000	33000
Ce-141 lower large intestine	50	7200	9200	13000	18000	27000	34000
Ce-141	5	8600	11000	17000	23000	36000	43000
Ce-144 lower large intestine	50	530	670	1100	1400	2300	2700
Ce-144	5	500	770	1200	1700	2700	3100
Np-237 bone surface	50	4	37	27	22	16	15
Np-237	5	7	67	59	54	41	39
Np-239 lower large intestine	50	28000	36000	180000	260000	400000	460000
Np-239	5	29000	36000	180000	260000	400000	470000
Pu-241 bone surface	50	120	970	720	550	490	480
Pu-241	5	180	1500	1200	1100	960	930
Cm-242 bone surface	50	19	130	180	240	340	390
Cm-242	5	29	180	260	330	510	590
Cm-244 bone surface	50	2	13	16	18	19	18
Cm-244	5	3	24	27	32	33	33

(a) Derived Intervention Levels derived using dose coefficients from Table E-4, dietary intakes from Table E-2 and "f" as given below:
0.3 (except for I-133, Te-132 and Np-239 in infant diets, i.e., the 3-month and 1-year age groups)
1.0 for I-133, Te-132 and Np-239 in infant diets.

TABLE E-6

DERIVED INTERVENTION LEVELS (Bq/kg)

Most limiting of Derived Intervention Levels for 5 mSv H_E or 50 mSv H_T
(individual radionuclides, by age group)

Radionuclide	3 months	1 year	5 years	10 years	15 years	Adult
Sr-89	1400	2400	3600	4500	5800	8700
Y-91	1200	1600	2300	3000	5300	5900
Zr-95	4000	5000	7000	9700	14000	16000
Nb-95	12000	14000	19000	26000	35000	40000
Te-132	4400	7300	35000	55000	89000	150000
I-129	110	76	72	56	68	84
I-133	7600	7000	30000	56000	79000	130000
Ba-140	6900	7900	11000	15000	27000	29000
Ce-141	7200	9200	12000	18000	29000	34000
Ce-144	500	670	1100	1400	2300	2700
Np-237	4	37	27	22	16	15
Np-239	28000	36000	180000	260000	400000	460000
Pu-241	120	970	720	550	490	480
Cm-242	19	130	180	240	340	390
Cm-244	2	13	16	18	19	18

TABLE E-7

DERIVED INTERVENTION LEVELS (Bq/kg)
(radionuclide groups, most limiting of all diets)

Radionuclide Group	Derived Intervention Level	
Sr-89	1400	(3 months)
Y-91	1200	(3 months)
Zr-95	4000	(3 months)
Nb-95	12000	(3 months)

Te-132	4400	(3 months)
I-129	56	(10 years)
I-133	7000	(1 year)
Ba-140	6900	(3 months)
Ce-141	7200	(3 months)
Ce-144	500	(3 months)
Np-237	4	(3 months)
Np-239	28000	(3 months)
Pu-241	120	(3 months)
Cm-242	19	(3 months)
Cm-244	2	(3 months)

APPENDIX F - DERIVED INTERVENTION LEVELS ADOPTED BY THE COMMISSION OF THE EUROPEAN COMMUNITIES AND THE CODEX ALIMENTARIUS COMMISSION FOR INTERNATIONAL TRADE

Foods exported from the U.S. are subject to the criteria used by the importing country, such as the recommendations of the CODEX Alimentarius Commission (CODEX) or the regulations of the Commission of the European Communities (CEC). CODEX is operated by the Joint Food Standards Programme of the Food and Agriculture Organization of the United Nations (FAO) and World Health Organization (WHO). CODEX develops and recommends standards and other guidance which are widely used in international trade. CEC regulations govern trade within the European Economic Community (EEC) and between the EEC and other countries. U.S. food exporters need to be familiar with the guidance from these organizations.

A discussion of CEC and CODEX Derived Intervention Levels (DILs)[21] is given below to provide insight into their differences.

(a) Commission of The European Communities: DILs for Future Accidents

The CEC adopted regulations in 1987 and 1989, establishing DILs for human food and animal feeds following a nuclear accident or any other case of radiological emergency (CEC 1987, 1989a, 1989b). These were established for use following any future accident and do not apply to residual contamination from the accident at Chernobyl. DILs addressing radioactive contamination from the Chernobyl accident were adopted by the CEC in 1986 (CEC 1986b).

The DILs for foods contaminated by future accidents are presented in Table F-l. DILs were given for four radionuclide groups and four food categories. The radionuclide groups include: isotopes of strontium, notably Sr-90; isotopes of iodine, notably I-131; alpha-emitting isotopes of _____

[21] The International System of Units is used throughout the document. See Appendix A, Glossary, for equivalence to units used in previous FDA guidance.

plutonium and transpiutonium elements, notably Pu-239 and Am-241; and all other radionuclides of half-life greater than 10 days, notably Cs-134 and Cs-137. For each group, CEC specified DILs for four food categories: baby foods, dairy produce, other food except minor food, and liquid foods.

Baby foods were defined as "foodstuffs intended for the feeding of infants during the first four to six months of life, ... and are put up for sale in packages which are clearly identified and labeled food preparation for infants". Dairy produce, liquid food, and minor foods were defined by reference to specific CEC regulations and nomenclature. Liquid foods included tap water and the CEC stated the "same values should be applied to drinking water supplies at the discretion of competent authorities of member states". Dried products referred to the products as prepared for consumption. Dilution factors were not specified and the CEC permitted member States to specify the dilution conditions.

DILs for minor foods such as spices were established, in a separate regulation, at ten times the DILs specified for "other foods" (CEC 1989a). Each DIL is to be applied independently. However, for each radionuclide group, the concentrations within the group are to be added when more than one radionuclide is present. The DILs are to be reviewed within three months following an accident to determine if they should be continued.

(b) CODEX Alimentarius Commission: DILs for Use in International Trade

CODEX adopted guidance in 1989 establishing DILs for food contaminated with radionuclides. The CODEX DILs were issued as guideline levels following an accidental nuclear contamination event (CODEX 1989). The guidance was developed from earlier publications of FAO (FAO 1987, Lupien and Randall 1988) and WHO (Waight 1988, WHO 1988). The DILs are presented in Table F-2. They were given for several radionuclide groups categorized by the magnitude of their dose coefficients and two food groups.

The food groups are milk and infant foods and foods destined for general consumption. CODEX defined infant food as a food prepared specifically for consumption by infants in the first year of life and stated that such foods are packaged and identified as being for this purpose (CODEX 1989). The radionuclides were grouped according to the magnitude of their dose coefficients (DCs). The specific groupings differed for the two food groups. CODEX listed representative radionuclides for each DC group. CODEX guidelines were not restricted to these radionuclides; any radionuclide can be placed into the appropriate DC group.

CODEX DILs apply for one year following a nuclear accident. They are intended to be applied to food prepared for consumption. Each DIL is to be applied independently. However, for each, the concentrations within the group are to be added. No guidance is provided for foods which are consumed in small quantities, although CODEX stated that application of the DILs to products of this type may be unnecessarily restrictive (CODEX 1989).

Table F-1

DILs ADOPTED BY CEC FOR FUTURE ACCIDENTS[a] (CEC 1989b)

Radionuclide Group	Derived Intervention Levels(Bq/kg) Baby Foods	Dairy Produce	Other except minor foods	Liquids
Isotopes of strontium, notably Sr-90	75	125	750	125
Isotopes of iodine, notably I-131	150	500	2000	500
Alpha-emitting isotopes of Pu and transplutonium elements, notably Pu-239, Am-241	1	20	80	20
All other radionuclides of half-life greater than 10 days, notably Cs-134, Cs-137	400	1000	1250	1000

(a) Do not apply to residual contamination from the accident at Chernobyl.

Table F-2

DIL VALUES RECOMMENDED BY CODEX (CODEX 1989)

FOODS DESTINED FOR GENERAL CONSUMPTION		
Approximate Dose Coefficient (Sv/Bq)	Representative Radionuclides	DIL (Bq/kg)
10^{-6}	Am-241, Pu-239	10

10^{-7}	Sr-90	100
10^{-8}	I-131, Cs-134, Cs-137	1000

MILK AND INFANT FOODS

Approximate Dose Coefficient (Sv/Bq)	Representative Radionuclides	DIL (Bq/kg)
10^{-5}	Am-241, Pu-239	1
10^{-7}	I-131, Sr-90	100
10^{-8}	Cs-134, Cs-137	1000

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Chapter 6
The Late Phase

6.1 Introduction

During the early and intermediate phases of a radiological incident, evacuation, relocation, and other actions may be necessary to protect the general public from the incident's immediate effects until cleanup and decontamination efforts can be completed. As defined in Section 1.2.3, the late or recovery phase of a radiological incident is the period initiated when action is taken to reduce radioactive contamination from the event to a level that allows the public to expect that further response actions will not occur.

The actions taken during this phase are referred to as recovery actions. The recovery phase itself ends upon their completion, a span of time that will be dependent upon the scale of the affected area. In cases in which the affected area is small, the cleanup and recovery effort may be completed very rapidly. However, if the affected area is large, the late phase may continue for years or even decades. This chapter provides guidance for decision making for the cleanup of the affected area, identification of areas in which specific cleanup techniques should be applied, and planning for subsequent site recovery and restoration.

The information presented in this chapter is intended for radiological incidents and facility accidents in which the affected area may extend far beyond the boundaries of a facility's site. It is not intended to impact site cleanups occurring under other statutory authorities such as EPA's Superfund program, NRC's decommissioning program, or State-administered cleanup programs.

Radiation levels should diminish with time during the course of the late phase, though the rapidity at which this occurs will be dependent upon the environmental and physical half-lives of the radionuclides contaminating the environment. Protective actions taken during the intermediate phase of the incident should still be in force. This circumstance will allow more time to make decisions on the need for long-term protective actions and what actions might be appropriate. By this phase a large amount of data about the location, type, and amounts of contaminants should be available. Information will also be available to facilitate evaluation of the consequences of implementing different long-term protective actions. It should be noted that EPA does not propose a specific numerical PAG for the late phase because it concluded that no single numerical value would be appropriate for all situations. Instead, EPA recommends utilization of an optimization method that allows for balancing of the benefits against the detriments of various possible cleanup actions in order to develop incident location-specific cleanup recovery plans. For the late phase guidance, there are different approaches depending upon whether the incident is an RDD/IND or a non-RDD/IND incident. All approaches involve evaluating cleanup and recovery options using an optimization process on an incident- and locality-specific basis.

The principles of optimization are applicable to all types of radiological incidents. Optimization is a general principle that exposure to radiation should be controlled so as to achieve the lowest risks or doses reasonably attainable given consideration of social, economic and public welfare factors. As an element of radiation protection guidance it is the logical consequence of the assumption of a linear relationship between exposure to radiation and risk at the radiation levels corresponding to environmental exposures. All EPA radiation standards and guidance incorporate this concept. Because of the case-specific application of optimization principles, numerical radiation criteria depend on the specific circumstance. The application of optimization is therefore the essential common element in maintaining a consistent approach to radiation protection across the wide variety of exposure situations.

6.2 The Optimization Process for Recovery

Optimization, in a broad definition, is finding the best and most preferred of many possible solutions to a problem. The process involves balancing the many factors that influence the decision. The simplest example is one where two competing factors (e.g., cost and benefit) influence a decision, such as buying a car. If additional benefit also increases the cost, the question is how much additional cost an extra amount of benefit is worth. Generally, all the possible answers can work. The objective of optimization is to determine which balance of complex factors works "best."

Optimization is a concept that is common to many State and Federal risk management programs that address radionuclides and chemicals, although it is not always identified as such. In the case of radiation, optimization generally takes the form of evaluating cleanup actions to determine which action reduces radiation exposures as much as is practical. The optimized exposure level is the level that results from taking actions that reduce doses and risk from radiation exposure as much as is practically possible, considering the circumstances, the costs including other risk, and the benefits (financial, social, and other). It is the level that is considered acceptable for the situation. In the case of radiation, optimization generally takes the form of reducing radiation exposure or dose.

Optimization activities are quantitative and qualitative assessments applied at each stage of site restoration decision making – from evaluation of remedial options to implementation of the chosen alternative. Evaluation of options for late phase recovery after a radiological incident should balance all of the relevant factors. Some issues that should be considered include the following (all may not be relevant in all circumstances) (DHS 2006):

- Areas impacted (e.g., size, location relative to population)
- Types of contamination (chemical, biological, in addition to radiological)
- Other hazards present
- Human health
- Public welfare
- Ecological risks
- Actions already taken during the early and intermediate phases
- Projected land use
- Preservation or destruction of places of historical, national, or regional significance
- Technical feasibility
- Wastes generated
- Waste disposal options and costs
- Costs and available resources to implement and maintain remedial options
- Potential adverse impacts (i.e., to human health, the environment, and the economy) of remedial options
- Long-term effectiveness
- Timeliness
- Public acceptability, including local cultural sensitivities
- Economic effects (e.g., tourism, business, and industry)

Determination of the optimized actions for recovery may require consideration of the net health benefits of reducing radiation risks or doses and the burden that the dose or risk reduction may place upon society. It is important that members of the affected population and other stakeholders be involved in this process. To accomplish this, EPA recommends the formation of a work group whose sole purpose is to focus on recovery and site restoration issues. Smaller incidents may not require the establishment of a formal workgroup, but appropriate stakeholders should be involved in the optimization process. The workgroup should draw upon the expertise of various technical disciplines, members of the affected areas, government agencies, and public interest groups. At a minimum, individuals with the following areas of expertise should be involved:

- Health physics and worker, public, and environmental radiation protection
- Environmental fate and transport sciences
- Decontamination technologies
- Radiation measurements
- Site-specific demographics, land uses, and local public works
- Local community needs, wants and wishes
- Government
- Waste disposal alternatives

The work group should be organizationally and functionally flexible, as the needs of the specific situation dictate. State and local public officials also need to be included in the process. In some cases it may be feasible to have a single group that includes all the relevant skills and knowledge. Under other circumstances, it may be desirable to have separate groups of technical experts, local community members, other stakeholders and government officials.

6.3 Recommendations for Cleanup and Recovery Exposure Levels

For RDD/IND incidents, EPA adopts and defers to the guidance for cleanup and recovery contained in the document titled "Application of Protective Action Guides for Radiological Dispersal Device and Improvised Nuclear Device Incidents" (RDD/IND PAG Application) (DHS 2006) released by DHS for public comment in 2006. The RDD/IND PAG Application provides an optimization process for evaluating cleanup and recovery actions. The portions of the RDD/IND PAG Application relevant to cleanup and recovery are found in Appendix G. As described in the document, optimized cleanup and recovery actions should be established for an affected area on an incident-specific basis.

For non-RDD/IND incident recovery, there are a variety of methods available for evaluating cleanup options: (1) the same method recommended above for RDD/IND; (2) a streamlined version of the process delineated in the RDD/IND PAG Application; (3) methods used by EPA in EPA-led cleanup activities (both removals and remedial actions); (4) methods used by the NRC in their decommissioning program; (5) methods used by DOE in the cleanup of their sites; and (6) the optimization processes that are inherent in some of the benchmarks provided in Table 6-1 and Section 6.5. Benchmarks are points of reference or comparison and provide decision makers, stakeholders, and technical assessors with a spectrum of values that have been deemed acceptable for other cleanup scenarios. A preliminary range of cleanup levels may be selected and used as a starting point for optimization with incident-specific considerations. Any of these will allow selection of optimized cleanup options that best suit the characteristics of the radiological incident and the particular needs of the affected area.

Table 6-1. Examples of United States Benchmarks of Potential Use in Evaluating Long-Term Cleanup Options during the Late Phase[a,b]

Example Organizations or Cleanup Programs	Summary of selected program-specific human health protection goals or concepts as applied to the cleanup of radiological contamination.
States NRC Agreement State Decommissioning Programs	Varies across states. Usually, decommissioning programs seek to achieve: • 25 mrem/y primary dose constraint; • 100 mrem/y allowable exemption • Lower levels based on the ALARA concept. • Some States have more stringent dose limits (e.g., 19, 15, or 10 mrem/y)
Environmental Department Contaminated Site Cleanup Programs	Varies across states. Usually, programs seek to achieve risk-based goals or a range of acceptable risk outcomes. Goals typically: • fall within a risk range of 10^{-4} to 10^{-6} excess lifetime cancer risk; and • include meeting existing applicable or relevant environmental regulations/standards. • Some States have single risk-based standards or goals (e.g., 10^{-4}, 10^{-5}, or 10^{-6}).
Federal NRC and DOE decommissioning and site remediation programs	• remediate or decommission to levels that are as low as reasonably achievable not to exceed 25 mrem/y given anticipated use either without restriction or with restriction • if restricted use is anticipated, dose estimates, given assumed failure of restriction (institutional controls), should be less the 100 mrem/y or 500 mrem/y if durable institutional controls are employed. • exemptions to the 25 mrem/y constraint for anticipated use may be approved if conditions are met. For more details see 10 CFR Part 20 Subpart E and DOE 5400.5 and associated guidance.
EPA Superfund remedial site cleanup program	Generally, remedial actions achieve human exposures that meet: • 10^{-4} to 10^{-6} excess cancer risk; • Hazard Index of one for non-cancer toxicity or less; and, • All Applicable or Relevant and Appropriate Requirements (ARARs). These may be waived under specific circumstances.

	• Cleanup levels should be developed considering the reasonably anticipated land use (For further information see: 40 CFR 300.430)
EPA Uranium and Thorium Mill tailings Standards	• 5pCi/g surface and 15 pCi/g subsurface for Ra-226 and Ra-228 in soil above background • 0.02 Working Levels (WL) not to exceed 0.03 WL for radon in habitable structures • 20 mR/h above background in habitable structures • options available for supplemental limits to be used under certain conditions (e.g. actions to meet the limits would pose a clear and present risk of injury to the workers or public notwithstanding reasonable measures to avoid the risk). For further details see 40 CFR Part 192 Subpart B: Uranium Mill Tailings Radiation Control Act (UMTRCA).

[a]Table presents examples only. Final cleanup goals and/or actual cleanup outcomes for a particular incident may vary depending on the circumstances of the incident. No single cleanup target is recommended for all possible incidents.
[b]Although many response programs often articulate target cleanup goals or limits in planning guidance, whether these levels are met or exceeded on a response-specific basis generally depends on the program context and the site-specific circumstances. Levels and concepts in this table are presented for illustration only and should not be applied to a specific incident cleanup without a thorough understanding of their derivation and application in the originating programs.

6.4 Optimization Plan for Radiological Cleanup Actions

A primary objective of the workgroup responsible for recovery should be the development of a plan for optimization. The plan serves to describe the conditions, document the basis for decisions, and provide a framework for future actions. The optimization plan can be very simple or as complex as dictated by the event. The plan should be a living document that adjusts to new information and changing conditions, and does not need to be completed before response actions are initiated. Its primary purpose is to provide a framework that addresses the elements of the optimization process. The following is an example outline of an optimization plan.

1. Introduction
   - Event summary
   - Responses
   - Protective Action Recommendations

2. Data
   - Background levels
   - Contamination levels and distribution
   - Demographic data
   - Data gaps

3. Radiation Exposures
   - Pathways
   - Modeled/estimated risks or doses
   - Measured doses
   - Dose projections for future
   - Sensitive/special populations

4. Other risks
   - Other risks and impacts from the event
   - Impacts related to protective actions
   - Other impacts

5. Completed Actions
   - Risks or doses averted
   - Costs
   - Implementation

6. Alternatives for future actions
    - Time required
    - Resource and materials costs
    - Opportunity costs
    - Benefits
    - Social impacts

7. Goals and objectives
    - Qualitative
    - Quantitative
    - Context of risks and objectives

8. Public participation
    - Stakeholders
    - Public information
    - Feedback to planning

9. Actions
    - Methods
    - Criteria
    - Schedule

9. Follow-up to action
    - Review results
    - Public participation
    - Review/revision to optimization plan

6.5 Recommendations for Recovery after Radiological Incidents Other than RDD and IND

The information needed to develop recovery cleanup options depends on the unique characteristics of a specific incident. There may be issues beyond the radiological contamination situation that need to be considered. For example, the presence, nature, type, and size of non-radiological hazards may have an impact that must be considered. Under some circumstances, the hazards of non-radiological problems can surpass those of radiation exposure. This information will vary from incident to incident. In the following, EPA presents alternative approaches for recovery from a non-RDD/IND incident.

Due to the extremely broad range of potential impacts that may occur as a result of radiological incidents (i.e., ranging from light contamination of one building to destruction and disruption of a metropolitan area), rather than a pre-established numeric guideline, a process should be available that will permit the community to determine the societal objectives for expected land uses and the options and approaches available, to select the most acceptable cleanup options. For an event that leaves a large, complex, or difficult problem, EPA recommends a process like that for an RDD or IND incident. When large areas are affected, it is likely that areas with different land uses will have different cleanup options. Thus, a cleanup action that is appropriate for one area might be quite different than the options selected for other areas with different land uses. For incidents of a lesser scale, such as transportation accidents of limited size, it might reasonably be expected that a complete return to pre-incident conditions can be achieved within a short period of time. For such circumstances, use of a more streamlined process is recommended. In the streamlined optimization process, only those items relevant to the situation's specifics need be considered, and then only to the degree that the situation warrants. A streamlined approach is not inconsistent with the process in the DHS RDD/IND PAG Application, which recognizes that under certain circumstances (e.g., small area affected) following the entire process is not necessary. Any of the approaches are designed to allow a community and other stakeholders to determine incident- and site-specific cleanup actions. In some of these cases, cleanups may be more conducive to little or no optimization analysis and may instead utilize a readily available concentration based on a risk, dose, field measurement, or regulatory level. EPA believes that the best course of action for devising a long-

term response plan is the selection of optimized cleanup actions in conjunction with technical experts and representatives of the affected population.

6.6 Development of Specific Concentration Levels for Cleanup

Long-term cleanup levels should be based on the reasonably anticipated use of the facility or affected area. In some situations, an area may reasonably be expected to support a wide range of uses. Thus, cleanup goals may be different for different parts or uses of the affected area. In cases in which benchmarks are considered, various levels have been used by EPA, NRC, and DOE. However, it is important to consider the contexts for which these benchmarks were developed.

6.7 Implementing Site Cleanup and Restoration

Planning for site cleanup and restoration is most effective when conducted as soon as possible after an incident. Once the initial protective measures are in place and stabilization and initial characterization of the affected areas are complete, recovery and site restoration will become the dominant task. Activities undertaken as part of the initial and intermediate response will be vital to the recovery of the contaminated area and will have an impact on the eventual cleanup. Therefore, it is important to have some initial plan for site cleanup and restoration available early in the process. Generally, a more detailed recovery strategy is then developed during the intermediate phase of the radiological incident response.

The working group or groups formed to decide upon an optimal exposure level, or cleanup level, could provide additional assistance to develop and oversee implementation of the site cleanup and restoration plan.

6.8 Establishing Criteria for Recovery

Some potential starting points or benchmarks are provided in Table 6-1. As described in the optimization process, the unique characteristics of the radiological incident and its location will drive the final optimized cleanup actions. EPA recommends the use of the processes and assumptions associated with the benchmark being used. If the benchmark is not associated with any optimization process, EPA suggests determining with stakeholder input which of the existing optimization processes to use rather than creating a new process.

Comparison to existing regulatory criteria may be useful in selecting predetermined residual levels that are presumptively acceptable. One approach to cleanup that may work well for small or uncomplicated events is cleanup using a presumptive or default cleanup level. These are levels that can be determined in advance to be acceptable to EPA and the public. In selecting such levels, comparisons with existing regulatory standards may be useful. Regulatory criteria represent levels that have been determined to be appropriate for specific contexts and sources, and those considerations need to be taken into account when applying them to different circumstances, such as an RDD. As examples, EPA criteria for cancer risk at Superfund sites is generally 10^{-4} to 10^{-6}, and NRC decommissioning criteria for radioactive material licensees include a dose limit of 25 mrem/y (0.25 mSv/y).

6.9 Making Decisions about Area Cleanup and Restoration

The first step in ensuring that reasonable decisions are made during the late phase is commitment to involve stakeholders in the decision-making process. It is important that the affected population has an opportunity to voice concerns and suggestions. Stakeholder input should be considered throughout the recovery process, as appropriate. Stakeholders should be included in the analysis of risks and doses, evaluation of these options, selection of an approach, and evaluation of the effectiveness of actions taken. Inclusion of stakeholders early and often in the decision-making process will not only aid in the provision of a reasonable basis for the actions to be taken, but will also help to ensure that those actions are accepted by the affected community. The working group can serve as a focal point for soliciting and processing stakeholder input. The type and extent of stakeholder input should be tailored to the needs of the specific incident.

6.10 Completion of Cleanup

The development of the cleanup plan, as described above, must contain the process and criteria for determining that the cleanup is complete. Cleanup plans should identify the constituents that will make the final decision that the cleanup is complete, as well as the criteria upon which this decision is reached. Cleanup can be complete even though ongoing monitoring, sampling, and other actions may continue for an extended period of time.

Documents such as the Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM) (MARSSIM 2000) provide one set of technical criteria for doing surveys and monitoring. The MARSSIM process has been agreed upon by the four agencies of the MARSSIM working group – DoD, DOE, EPA, and NRC. The MARSSIM process is technically defensible over a broad range of situations. Its performance-based approach generally involves more planning and less re-work than other methods.

Other approaches are discussed in EPA's "Methods for Evaluating the Attainment of Cleanup Standards – Volume 3: Reference-Based Standards for Soil and Solid Media" (EPA 1989a), "Soil Screening Guidance for Radionuclides: User's Guide" (EPA 2000) and guidance that apply to various aspects of site study, data collection, and decision making strategies, quality assurance project plans, sampling designs, and modeling found at the following web address: (http://www.epa.gov/quality/qa_docs.html).

Another example is the guidance document entitled: "Improving Sampling, Analysis, and Data Management for Site Investigation and Cleanup" (EPA 2004). This describes the three-pronged "triad approach" that forms the basis of EPA's national strategy for site characterization and site assessment. This streamlined approach to site assessment focuses on the conduct of systematic planning to ensure the effective use of resources, the preparation of a dynamic work plan to support decision making in the field, the use of on-site, real-time analytical tools rapid sampling platforms and on-site, real-time data interpretation. More information about the triad approach can be found at the following web address: (http://www.triadcentral.org/over/index.cfm).

However, the optimization process is more likely to yield information on what methods should be applied than to give concentration limits. The question to be answered is whether or not the remediation is performed properly, not if the area meets a limit. As this approach to cleanup is not as common in current site cleanup and survey programs, new paradigms and processes may have to be developed, or modified from existing ones, to meet demands brought about by the cleanup.

Generally, the particular method used to evaluate attainment of cleanup goals should be consistent in scope and complexity with the decisions to be made at the affected areas (usually defined in the quality assurance project plan). In other words, smaller incidents may require very simple judgments about whether the contamination has been addressed. Larger and/or more complex contamination problems may require significantly greater (and more resource-intensive) site study and post-cleanup evaluation. Other less intensive means for evaluating attainment of cleanup levels include the use of a "2x background" criterion (see section 3.4), exposure-area averaging, and/or "not-to-exceed" decision rules.

6.11 Other Recovery Issues

Essential services should be restored as quickly as possible. These include any services that are necessary to the general health and well-being of the population, such as:

- Law Enforcement and Public Safety
- State, County, and Local Government
- Public Services and Utilities
- Transportation
- Human Services
- Social and Psychological Assistance

There may be other services that are essential to specific communities. The overall goal of the late phase is to reduce potential risks or doses to the extent practical. This goal should be kept in mind during determination of which services should be restored first.

Members of the public returning to an area affected by a radiological incident, whether for work or to reside, will require reassurance regarding their safety. It is important that they be provided with accurate and reliable information, and it is critical to the success of the late phase that they perceive that this information is, indeed, true. In addition, they should be provided with updated information frequently, such as residual radioactivity levels. The public will depend upon a variety of sources for this information, including governmental, educational, scientific, and medical organizations, as well as the various forms of public media. These potential information sources should be included in the distribution of these materials.

6.12 Long-Term Environmental Monitoring

It may be desirable or necessary to establish a regular, long-term environmental radiation monitoring program in areas affected by the incident. EPA maintained environmental monitoring in and around Middletown, Pennsylvania (the nearest town to the Three Mile Island nuclear power plant) for more than 10 years after occurrence of the Three Mile Island accident in 1979. This monitoring should be established at a level that is appropriate for the nature and size of the radiological incident. Another working group, or the same group(s) that provided input during the cleanup planning process, could be used to inform decision makers about what the community would desire in a post-cleanup monitoring program.

When a long-term monitoring program is established, it is important to be sure that it has designated review times, criteria and reviewers, and that there is a clear path to end the monitoring. Potential criteria for ending the program could include a set number of years after certain measurement conditions are achieved or when other predetermined criteria are met.

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Appendix B: Glossary and List of Acronyms

Acute health effects: Prompt radiation effects (those that would be observable within a short period of time) for which the severity of the effect varies with the dose and for which a practical threshold exists.

Alpha Particle: A positively charged particle made up of two neutrons and two protons emitted by certain radioactive nuclei. Alpha particles can be stopped by thin layers of light materials, such as a sheet of paper, and pose no direct external radiation threat; however, they can pose a serious health threat if inhaled or ingested.

As Low As Reasonably Achievable (ALARA): An approach to control or manage radiation exposures (both individual and collective to the workforce and the public) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. ALARA is not a dose limit; it is a practice that has as its objective the attainment of dose levels as far below applicable limits as possible.

Beta Particle: An electron or positron emitted by certain radioactive nuclei. Beta particles can be stopped by aluminum. They can pose a serious direct or external radiation threat. They also pose a serious internal radiation threat if inhaled or ingested.

Buffer Zone: An area surrounding the preliminary relocation area designation selected by additional relocation until the stability of radioactivity levels in the area is confirmed.

Committed Dose Equivalent: The dose equivalent to an organ that will be received from an intake of radioactive material by an individual during a 50-year period following the intake.

Committed Effective Dose Equivalent: The sum of the products of the weighting factors applicable to each of the body organs that are irradiated and the committed dose equivalent to these organs.

Delayed health effects: Radiation effects which are manifested long after the relevant exposure. The vast majority of these effects are stochastic, that is, the severity is independent of dose and the probability is assumed to be proportional to the dose, without a threshold.

Derived Intervention Level (DIL): A protection action guideline issued in draft form only by the Food and Drug Administration pertaining to contamination of human foodstuffs and based upon a committed effective dose equivalent of 5 mSv, or a committed dose equivalent to individual tissues and organs of 50 mSv, whichever is more limiting.

Derived Response Level (DRL): A level of radioactivity in an environmental medium that would be expected to produce a dose equal to its corresponding Protective Action Guide.

Dose Conversion Factor (DCF): Any factor that is used to change an environmental measurement to dose in the units of concern.

Dose equivalent: The product of the absorbed dose in *rad*, a quality factor related to the biological effectiveness of the radiation involved and any other modifying factors.

Early phase: The period at the beginning of a nuclear incident when immediate decisions for effective use of protective actions are required and must be based primarily on predictions of radiological conditions in the environment. This phase may last from hours to days. For the purpose of dose projection, it is assumed to last for 4 days. This phase is also known as the emergency phase.

Effective dose equivalent: The sum of the products of the dose equivalent to the organ and the weighting factors applicable to each of the body organs that are irradiated.

Evacuation: The urgent removal of people from an area to avoid or reduce high-level, short-term exposure, usually from the plume or from deposited activity. Evacuation may be a preemptive action taken in response to a facility

condition rather than an actual release.

Exposed population: Those individuals affected by the radioactive emissions of a nuclear incident.

Exposure pathway: The way in which an individual is exposed to radiation.

External exposure: Radiation exposure due to radioactive material that is outside a person's body, such as radioactive particles deposited on the ground.

Gamma exposure: Exposure to gamma radiation emanating from radioactive materials deposited on the ground.

Genetic effect: An effect in a descendant resulting from the modification of genetic material in a parent.

Groundshine: Gamma radiation emitted from radioactive materials deposited on the ground.

Half-life: The time in which one-half of the atoms of a radioactive isotope disintegrate into another form. Half-lives vary from billionths of a billionth of a second to billions of years. Also called physical or radiological half-life.

Incident Phase: Any one of the three defined phases of a radiological or nuclear incident: (1) early phase; (2) intermediate phase; and (3) late phase.

Intermediate phase: The period beginning after the incident source and releases have been brought under control and reliable environmental measurements are available for use as a basis for decisions on additional protective actions and extending until these protective actions are terminated. This phase may overlap the early and the late phases and may last from weeks to many months. For the purpose of dose projection, it is assumed to last for 1 year.

Internal exposure: Radiation exposure resulting from breathing or eating food or drinking water that is radioactively-contaminated.

Isopleth: The line or area represented by an isoconcentration.

Isoconcentration: More than one sample point exhibiting the same isolate concentration.

Late phase: The period beginning when recovery action designed to reduce radiation levels in the environment to permanently acceptable levels are commenced, and ending when all recovery actions have been completed. This period may extend from months to years. It may also be referred to as the recovery phase.

Low LET Radiation: Linear Energy Transfer (LET) describes the rate at which gamma radiation or a charged particle transfers its energy to an absorber per unit distance traveled in the absorber. Low-LET radiation means energy is deposited at a low rate but along a long path. Low-LET radiation is relatively penetrating

Radiological Incident: An event or series of events, either deliberate or accidental, leading to the release, or potential release, into the environment, of radioactive materials in sufficient quantity to warrant consideration of protective actions.

Nuclear Incident: An event or series of events that results in a yield-producing, nuclear explosion.

Optimization: A flexible, multi-attribute decision-making process that seeks to consider and balance many factors for cleanup in the late phase.

Plume: A column or band of smoke or exhaust gases which moves through the atmosphere. In the case of a nuclear incident, the plume is the airborne band of radioactive release from the facility.

Potassium Iodide: Potassium Iodide has been approved by the FDA as a nonprescription drug for use as a "blocking agent" to prevent the human thyroid gland from absorbing radioactive iodine.

Projected dose: Future dose calculated for a specific time period.

Protective Action: An activity conducted in response to an incident or potential incident to avoid or reduce radiation dose to members of the public.

Protective Action Guidance: Protective Action Guides and their corresponding protective action recommendations.

Protective Action Guide (PAG): The projected dose to reference man, or other defined individual, resulting from a radiological incident at which a specific protective action to reduce or avoid that dose is warranted.

Recovery: The process of reducing radiation exposure rates and concentrations of radioactive material in the environment to levels acceptable for unconditional occupancy or use.

Reentry: Temporary entry into a relocation area under controlled conditions.

Relocation: The removal or continued exclusion of people (households) from contaminated areas to avoid chronic radiation exposure.

Relocation Area: An area from which residents should be relocated because radiation doses are expected to exceed the intermediate phase PAGs.

Return: The reoccupation of areas previously designated as relocation areas because potential radiation doses are below the intermediate phase PAGs.

Secular Equilibrium: A state of parent-daughter equilibrium that is achieved when the half-life of the parent is much longer than the half-life of the daughter. In this case, if the two are not separated, the daughter will eventually decay at the same rate at which it is being produced. At this point, both parent and daughter will decay at the same rate until the parent is essentially exhausted.

Sheltering-in-place: The use of a structure for radiation protection from an airborne plume and/or deposited radioactive materials.

Short-lived daughters: Radioactive progeny of radioactive isotopes that have half-lives on the order of a few hours or less.

Total effective dose equivalent (TEDE): the projected sum of the effective dose equivalent from external radiation exposure (i.e.,groundshine) and the committed effective dose equivalent from inhaled radioactive material.

Weathering factor: The fraction of radioactivity remaining after being affected by average weather conditions for a specified period of time.

Acronyms Used in this Document

ALARA	As Low as Reasonably Achievable
AMS	Aerial Measurement System
ARAR	All Applicable and Relevant Requirement
AvCRP	Average Combined Removal Parameter
AvEffXP	Average Effective Exposure Period
BEIR	Biological Effects of Ionizing Radiation
BR	Breathing Rate
CDF	Committed Dose Conversion Factor
CEDE	Committed Effective Dose Equivalent
CERCLA	Comprehensive Environmental Response, Compensation, and Liability Act
CF	Conversion Factor
CFR	Code of Federal Regulations
CRP	Combined Removal Parameter
DCF	Dose Conversion Factor
DCP	Dose Conversion Parameter
DHS	Department of Homeland Security
DOD	Department of Defense
DOE	Department of Energy
DOL	Department of Labor
DRL	Derived Response Level
DRL_Dp	Deposition Derived Response Level
DRL_XR	Derived Response Level for Exposure Rate
DRP	Derived Response Parameter
DXCF	Dose to Exposure Conversion Factor
EAL	Emergency Action Level
EDP	Effective Dose Parameter
EDP_Dp	External Dose Parameter for Deposition
EffXP	Effective Exposure Period
EPA	Environmental Protection Agency
EPZ	Emergency Planning Zone
ExDC	External Dose Coefficient
ExDF	External Dose Factor
ExXC	External Exposure Coefficient
ExXF	External Exposure Factor (effective)
ExXR	External Exposure Rate
FBI	Federal Bureau of Investigation
FDA	Food and Drug Administration
FEMA	Federal Emergency Management Agency
FGR	Federal Guidance Report
FRC	Federal Radiation Council
FRMAC	Federal Radiological Monitoring and Assessment Center
FRP	Federal Response Plan
FRPCC	Federal Radiological Preparedness Coordination Committee
GRF	Ground Roughness Factor
HAZWOPER	Hazardous Waste Operations and Emergency Response
HHS	Department of Health and Human Services
HI	Hazard Index
IAEA	International Atomic Energy Agency
IC/UC	Incident Command/Unified Command
ICRP	International Commission on Radiological Protection
IDCF	Inhalation Dose Conversion Factor

IIMG	Interagency Incident Management Group
IND	Improvised Nuclear Device
K	Resuspension Factor
KI	Potassium Iodide
KP	Resuspension Parameter
LD_{50}	Median Lethal Dose
LET	Linear Energy Transfer
MARSSIM	Multi-Agency Radiation Survey and Site Investigation Manual
NAS	National Academy of Sciences
NASA	National Aeronautics and Space Administration
NCRP	National Council on Radiation Protection & Measurements
NHSC	National Highway Safety Council
NIH	National Institutes of Health
NNPP	Naval Nuclear Propulsion Program
NPL	National Priorities List
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRP	National Response Plan
OSHA	Occupational Safety and Health Administration
PAG	Protective Action Guide
RDD	Radiological Dispersal Device
SDWA	Safe Drinking Water Act
TDE	Total Dose, Effective
TDP_XR	Total Dose Parameter for Exposure Rate
TED	Total Effective Dose
TEDE	Total Effective Dose Equivalent
TSH	Thyroid Stimulating Hormone
UMTRCA	Uranium Mill Tailings Radiation Control Act
UN	United Nations
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USDA	United States Department of Agriculture
WF	Weathering Factor
WHO	World Health Organization

Appendix C: Protective Action Guides for the Early Phase: Supporting Information

C.1 Introduction

This appendix discusses the rationale used by EPA in selecting the PAGs for the early phase of a radiological incident. Response to a radiological emergency will normally be carried out in three phases, as discussed in Chapter 1. The early phase, which may also be called the emergency phase, is the period at the beginning of the event when the source is out of control. This phase may last from hours to days. Decisions made during the early phase will often be based on predicted or potential radiological conditions in the environment rather than on actual measurements. The first, and primary, protective action taken during this phase is evacuation and/or sheltering-in-place. This action will prevent members of the community from being exposed to direct radiation. Supplementary protective actions, such as washing and changing clothes to reduce exposure of the skin and the use of stable iodine (see Section C.2.3) to reduce the uptake of radioiodine in the thyroid, may also be taken.

In particular, this appendix examines the potential magnitude and consequences of predicted radiation exposures during the early phase of a radiological incident. These outcomes are then compared in detail to the benefits and possible negative impacts of evacuation and sheltering-in-place. For the purposes of this analysis, EPA evaluated several nuclear reactor accident scenarios. Due to the number of nuclear reactor facilities, the size of the source, and the amount of energy available to drive a release, EPA believes that the evaluation of accident scenarios involving nuclear reactors will yield an upper bound on the magnitude of potential releases and exposures.

In an analysis separate to the 1992 PAG Manual entitled "Implementing Protective Actions for Radiological Incidents at Other Than Nuclear Power Reactors," EPA evaluated the applicability of the early phase PAG to radiological incidents other than a power plant accident, such as a terrorist event (EPA 1992, DHS 2006). EPA concluded that the early phase PAG can apply to these types of incidents. The analysis included the likely sources of radiation in an RDD, also known as a "dirty bomb," and indicated that the impact distances and areas associated with the early phase following the detonation were similar to or less than the reactor release scenarios used by EPA to develop the early phase protective action guidance. This was also true for the transportation incidents that EPA reviewed as part of the analysis.

Atmospheric releases from other types of radiological incidents are likely to be of smaller consequence, affecting fewer individuals than a release from a nuclear reactor. The costs and benefits of protective actions in these situations are expected to be proportional to the magnitude of the incident. Therefore, the basic conclusions developed in this appendix with regard to nuclear reactor incidents will remain valid for other types of radiological incidents.

C.1.1 Existing Federal Guidance

In the 1960s, the FRC defined the concept of the PAG and established guidance for limiting the ingestion of Sr-89, Sr-90, Cs-137, and I-131 (FRC 1964; FRC 1965). That guidance applied to restricting the use of food products that had become contaminated as the result of the release of radioactivity to the stratosphere from weapons testing. The PAGs presented in Chapter 2 of this Manual do not supersede the FRC guidance, but rather address a different concern. During the period immediately following a radiological incident, the critical source of radiation exposure is likely to be an atmospheric plume. Thus, the principal exposure pathways will be direct exposure and inhalation and the primary protective actions will be evacuation and sheltering-in-place. Therefore, the early phase PAGs simply expand the application of protective action guides to exposure pathways and situations not considered by the original FRC guidance.

C.1.2 Principal Exposure Pathways

As radioactive materials are released from a source, such as a nuclear reactor, people in the surrounding area can be exposed to radiation via several different pathways. Most immediately is direct radiation exposure resulting from the cloud of radioactive material carried by prevailing winds. The airborne plume can contain radioactive noble gases, iodines, and/or particulate materials, depending on the source involved and conditions of the incident. These materials emit gamma rays, which are not significantly absorbed by air, and will expose the entire bodies of nearby individuals.

As the plume moves through the atmosphere, people are submerged in the cloud of radioactive materials. In this case, radioactive materials are inhaled, and the skin and clothes may be contaminated. Inhaled radioactive materials, depending on their solubility in body fluids, may either remain in the lungs or move via the blood to other organs. Many radionuclides that enter the bloodstream tend to be predominantly concentrated in a single organ. For example, if radioiodines are inhaled, a significant fraction will tend to move rapidly from the lungs through the bloodstream to the thyroid gland where much of the iodine will be deposited and most of the dose[20] will be delivered. Although dose to skin from materials deposited on the skin and clothing could be significant, it will contribute less to the overall risk of fatal cancer than the dose from inhalation, particularly if early protective actions include washing of exposed skin and changing of contaminated clothes.

Radioactive materials may also settle onto the ground and other surfaces. People remaining in the area will continue to be exposed through ingestion and external radiation, and through inhalation of resuspended materials. The total dose from such deposited materials may be more significant than that due to direct exposure to the plume because the length of exposure can be much longer. However, longer term exposures are not included in the predicted dose considered in the PAGs for the early phase. As discussed in Chapter 2, a 4-day exposure period is assumed for the purposes of projecting dose during the early phase of a radiological incident. Radiation doses associated with longer term exposures are addressed by the PAGs for relocation and for contaminated food and water (see Chapters 4 and 5). Thus, the PAG levels for the early phase of a radiological incident are expressed in terms of the estimated doses from exposure due to external radiation, inhalation, and contamination of the skin occurring in the first 4 days following the projected (or actual) start of a release.

C.2 Practicality of Implementation

Chapter 1 presented three principles that EPA used in establishing the PAGs contained in this Manual. These principles are:

1. Prevent acute effects.

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20 In this and all subsequent references, the word "dose" refers to the committed dose equivalent to the specified organ, or, if no organ is specified, the sum of the committed effective dose equivalent from intake of radionuclides and the effective dose equivalent from external sources of radiation.

2. Reduce risk of chronic effects.

3. Require optimization to balance protection with other important factors and ensure that actions taken cause more benefit than harm.

This section examines issues related to the latter principle. Protective actions come at a resource cost to implement and may place people at some risk. In developing the early phase PAG, EPA analyzed both the costs and risks associated with the actions most likely to be taken and weighed them against the benefits achieved by these actions in terms of radiation dose avoided.

The primary protective actions likely to be taken during the early phase are evacuation and/or sheltering-in-place. In some cases, washing and changing of clothing, or the use of stable iodine, may also be appropriate actions. EPA found that the costs, risks, and degrees of protection associated with evacuation are generally higher than those for sheltering-in-place. In addition, although some costs and risks may exist with the use of other protective actions, they are likely to be small and not readily quantifiable. Therefore, only the costs and risks associated with evacuation are discussed below. By undertaking an analysis of these factors, EPA can ensure that the early phase PAG is set at a level which provides an appropriate balance between the costs and risks associated with evacuation and the benefits of averting radiation dose.

C.2.1 Cost of Evacuation

Costs incurred to reduce the radiation risk from radiological incidents fall into several major categories. The first category, applicable to incidents at nuclear facilities, includes the design, construction, and operation of nuclear facilities in such a manner as to minimize the probability and consequences of radiological incidents. It is recognized that the probability and consequences of such incidents usually cannot be reduced to zero. Therefore, a second category is necessary for nuclear facility incidents and is also applicable for non-facility incidents: the development of emergency response plans to invoke actions which would reduce exposure of potentially exposed populations, and consequently their risks, if a major radiological incident should occur.

For nuclear facility incidents, both of the above categories of costs are properly attributed to the cost of design and operation of a nuclear facility. A third category of costs is the actual expenses incurred by taking protective actions as the result of an incident. In general, the choice of levels for PAGs will affect only this third category of costs. That is, all costs in the first two categories are assumed to be unaffected by decisions on the levels of PAGs. (This will be the case unless the PAGs were to be set so high as to never require protective action, in which case response plans would be unnecessary). Therefore, the costs associated with implementing the protective actions based on the PAGs are evaluated only in terms of the actual cost of response. Similarly, the risk incurred by protective actions is compared only to the risk associated with the radiation dose that would be avoided by the action, and is unaffected by any other measures taken to reduce risks that fall in the first two categories of cost identified above.

C.2.1.1 Cost Assumptions

The analyses in this section are based on evaluating the costs of evacuation and the doses that would be received in the absence of protective actions for nuclear reactor incidents. These were calculated as a function of offsite location, meteorological condition, and incident type. Dose and cost data are based on the following assumptions:

- Airborne releases are those associated with fuel melt incidents at nuclear reactor facilities followed by containment failure.
- Meteorological conditions range from stable to unstable, and windspeeds are those typical of the stability class.
- Plume dispersion follows a Gaussian distribution, with a 0.01 m/s dry deposition velocity for iodine and particulate materials.
- Doses are those incurred from whole body gamma radiation from the plume, inhalation of radioactive material in the plume, and from 4 days of exposure to deposited radioactive material.

- Population distributions are the average values observed around 111 nuclear power reactor plants, based on 1970 data.

- The cost of evacuation is $185 per person for a 4-day evacuation involving a 100-mile round trip, with an average of three persons per household. These evacuation costs include wages and salaries of personnel directing the evacuation, transportation costs of evacuees to and from the staging location, food and shelter for the evacuees during the evacuation period, loss of personal and corporate income during the evacuation period, and the costs of any special supplies.

The estimated costs and doses avoided are based on the following idealized evacuation area model (see Figure C.1):

- All people within a 2-mile radius of the incident are evacuated for all scenarios.

- People are also evacuated from a downwind area bounded by equivalent rays on either side of the center line of the plume, which define the angular spread (70, 90, or 180 degrees) of the area evacuated by an arc at the distance beyond which the evacuation dose would not be exceeded on the plume centerline.

Figure C-1 shows the relationship between the area in which the evacuation dose would be exceeded and the larger area that might be evacuated. The figure shows the plume centered in idealized evacuation area.

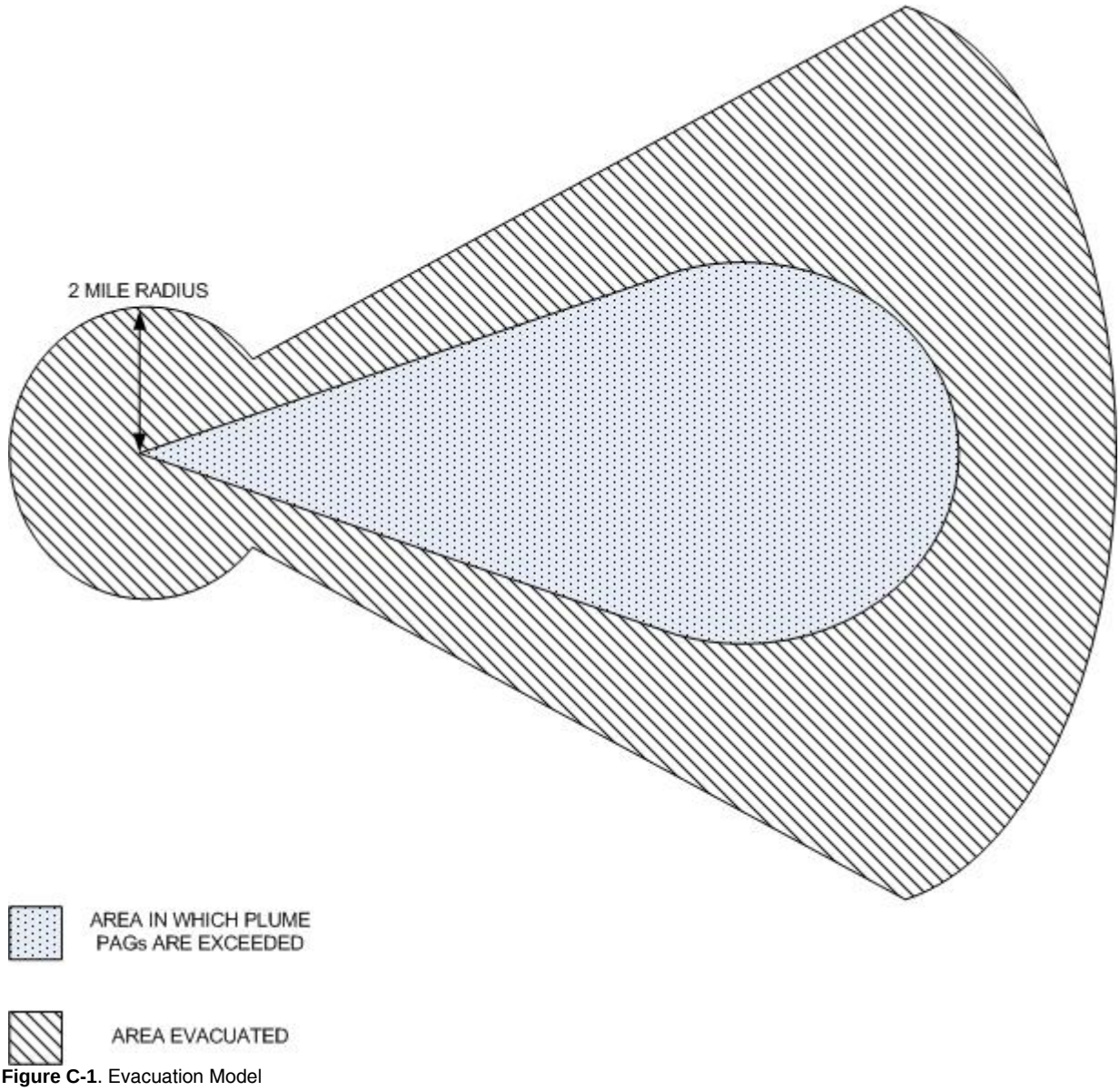

Figure C-1. Evacuation Model

C.2.1.2 Analysis

Evaluation of costs for evacuation and doses to populations as a function of the area evacuated depends on a variety of assumptions. Three fuel-melt accident categories, six meteorological stability classes, and the three evacuation area models discussed in the previous section were examined. Detailed assumptions and data are reported elsewhere (EPA 1987). Selected data are present in Tables C-1, C-2, and C-3. Each table includes the cost per unit of collective dose (person-rem) avoided for the population identified in Figure C-1. Data in these three tables are also based on the median accident category examined (SST-2). (SST accident categories are described in Section F.1.2). Each table presents data associated with a different stability class, specifically Stability Class A, Stability Class C, and Stability Class F.

The data are presented for both the total area and the incremental area evacuated for each change in dose level examined. When evaluating the cost per person-rem avoided for a specific set of circumstances, it is appropriate to assess the ratio of the total cost to the total dose avoided to calculate the average cost per person-rem avoided. However, when one is comparing the cost versus dose avoided to make a judgment between a variety of different limiting dose values, it is appropriate to compare the dose savings and costs at the margin since the cost of

evacuating the additional area is incurred to avoid the incremental collective dose. Therefore, the appropriate quantities of cost and dose savings are the cost and risk for the additional area evacuated. Results of analyses on both a total and incremental basis are presented in Tables C-1, C-2, and C-3 for accident category SST-2. This is the smallest category of fuel-melt accident that could yield effective dose equivalents during the first 4 days of exposure greater than 0.5 rem outside the assumed 2-mile evacuation circle for all stability classes. Data on costs versus collective dose avoided for all three accident categories are summarized in Table C-4.

Changes in population density would not affect the above results since both cost and collective dose are proportional to the size of the population affected. Factors that could affect these results are evacuation costs, accident scenarios, and evacuation area models. The results will be directly proportional to different assumptions for the cost of evacuation. Some data on the variation among different accident scenarios are presented in the next section. In situations in which different widths of evacuation area are assumed, the change in cost per unit dose avoided will be approximately proportional to the change in width in degrees. This approximation is more accurate for the higher stability classes (E and F). Evacuation within a 2-mile radius and a 90-degree sector in the downwind direction is generally considered to be adequate for release durations not exceeding a few hours and for which reliable wind direction forecasts are available.

Table C-1. Costs for Implementing Various PAGs for an SST-2 Type Accident (Stability Class A)[a]

Evacuation Angle (Degrees)	PAG Value (rem)	Total Area			Marginal Area		
		Cost (Dollars)	Dose Avoided (Person-rem)	Dollars/ Person-rem Avoided	Δ Cost (Dollars)	Δ Dose Avoided (Person-rem)	Δ Dollars/ (Person-rem Avoided)
70	0.5	2.83E+07	8.97E+04	315	2.16E+07	4.91E+04	440
	1	6.68E+06	4.06E+04	164	5.19E+06	2.33E+04	223
	2	1.49E+06	1.73E+04	88	1.19E+06	1.21E+04	98
	5	2.99E+05	5.22E+03	57	9.70E+04	2.44E+03	40
	10	(a)	(a)	(a)			
90	0.5	3.63E+07	9.29E+04	391	2.78E+07	5.05E+04	550
	1	8.54E+06	4.24E+04	201	6.68E+06	2.42E+04	276
	2	1.86E+06	1.82E+04	102	1.54E+06	1.28E+04	120
	5	3.26E+05	5.41E+03	60	1.25E+05	2.63E+03	47
	10	(a)	(a)	(a)			
180	0.5	7.16E+07	9.33E+04	767	5.49E+07	5.06E+04	1080
	1	1.67E+07	4.27E+04	391	1.32E+07	2.43E+04	543
	2	3.48E+06	1.84E+04	190	3.04E+06	1.29E+04	235
	5	4.48E+05	5.46E+03	82	2.47E+05	2.68E+03	92
	10	(a)	(a)	(a)			

[a]The 4-day dose does not exceed the PAG outside the 2-mile radius of the accident site. The total cost of evacuating everyone within the 2-mile radius is 2.02E5 dollars; the total dose avoided is 2.78E3 person-rem; and the total cost per person-rem avoided is $73.

Table C-2. Costs for Implementing Various PAGs for an SST-2 Type Accident (Stability Class C)[a]

Evacuation Angle (Degrees)	PAG Value (rem)	Total Area			Marginal Area		
		Cost (Dollars)	Dose Avoided (Person-rem)	Dollars/ Person-rem Avoided	Cost (Dollars)	Dose Avoided (Person-rem)	Dollars/ ☎Person-rem Avoided)
70	0.5	4.95E+07	1.13E+05	439	3.71E+07	4.95E+04	750
	1	1.23E+07	6.31E+04	195	9.87E+06	2.58E+04	382
	2	2.46E+06	3.73E+04	66	1.68E+06	1.02E+04	165
	5	7.82E+05	2.71E+04	29	3.89E+05	6.15E+03	63
	10	3.93E+05	2.10E+04	19	1.32E+05	4.75E+03	28
	20	2.60E+05	1.62E+04	16	3.40E+04	2.50E+03	10
	50	(a)	(a)	(a)			
90	0.5	6.35E+07	1.13E+05	564	4.77E+07	4.95E+04	964
	1	1.58E+07	6.32E+04	250	1.27E+07	2.58E+04	491
	2	3.11E+06	3.74E+04	83	2.16E+06	1.02E+04	212
	5	9.48E+05	2.72E+04	35	5.00E+05	6.16E+03	81
	10	4.47E+05	2.10E+04	21	1.70E+05	4.76E+03	36
	20	2.77E+05	1.63E+04	17	3.40E+04	2.50E+03	14
	50	(a)	(a)	(a)			
180	0.5	1.25E+08	1.13E+05	1110	9.44E+07	4.95E+04	1910
	1	3.10E+07	6.32E+04	491	2.51E+07	2.58E+04	971
	2	5.95E+06	3.74E+04	159	4.28E+06	1.02E+04	419
	5	1.68E+06	2.72E+04	62	9.90E+05	6.16E+03	161
	10	6.87E+05	2.10E+04	33	3.36E+05	4.77E+03	70
	20	3.51E+05	1.63E+04	22	6.70E+04	2.50E+03	27
	50	(a)	(a)	(a)			

[a]The 4-day dose does not exceed the PAG outside the 2-mile radius of the accident site. The total cost of evacuating everyone within the 2-mile radius is 2.02E5 dollars; the total dose avoided is 2.78E3 person-rem; and the total cost per person-rem avoided is $73.

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Table C-3. Costs for Implementing Various PAGs for an SST-2 Type Accident (Stability Class F)

Evacuation Angle (Degrees)	PAG Value (rem)	Total Area			Marginal Area		
		Cost (Dollars)	Dose Avoided (Person-rem)	Dollars/ Person-rem Avoided	Δ Cost (Dollars)	Δ Dose Avoided (Person-rem)	Δ Dollars/ Δ Person-rem Avoided
70	0.5	8.95E+07	4.61E+05	194	4.01E+07	1.98E+04	2020
	1	4.95E+07	4.41E+05	112	2.12E+07	2.17E+04	977
	2	2.83E+07	4.19E+05	67	1.59E+07	3.66E+04	436
	5	1.23E+07	3.83E+05	32	5.65E+06	2.93E+04	193
	10	6.68E+06	3.53E+05	19	3.03E+06	3.18E+04	95
	20	3.65E+06	3.22E+05	11	9.70E+05	3.10E+04	32
	50	1.49E+06	2.68E+05	5.6			
90	0.5	1.15E+08	4.61E+05	250	5.15E+07	1.98E+04	2600
	1	6.35E+07	4.41E+05	144	2.72E+07	2.17E+04	1260
	2	3.63E+07	4.19E+05	87	2.05E+07	3.66E+04	560
	5	1.58E+07	3.83E+05	41	7.26E+06	2.93E+04	248
	10	8.54E+06	3.53E+05	24	3.90E+06	3.18E+04	123
	20	4.64E+06	3.22E+05	14	1.30E+06	3.10E+04	41
	50	1.86E+06	2.68E+05	6.9			
180	0.5	2.27E+08	4.61E+05	493	1.02E+08	1.99E+04	5120
	1	1.25E+08	4.41E+05	285	5.39E+07	2.17E+04	2480
	2	7.16E+07	4.19E+05	171	4.05E+07	3.66E+04	1110
	5	3.10E+07	3.83E+05	81	1.44E+07	2.92E+04	492
	10	1.67E+07	3.53E+05	47	7.71E+06	3.18E+04	242
	20	8.98E+06	3.22E+05	28	2.40E+06	3.10E+04	80
	50	3.51E+06	2.68E+05	13			

C.2.1.3 Conclusions

As shown in Tables C-1, C-2, and C-3 for an SST-2 accident, the cost per unit dose avoided is greatest for wide angle evacuation and for the most stable conditions, Stability Class F. Although a few emergency plans call for evacuation over wider angles (up to 360 degrees), the model shown in Figure C-1 with a 90 degree angle is most common.

To estimate an upper bound on a dose level sufficient to trigger dose for evacuation based on cost, EPA first considered common values placed on avoiding risk. As one input into its risk management decisions, EPA has used a range of $400,000 to $7,000,000 as an acceptable range of costs for avoiding a statistical death from pollutants other than radiation. For a risk of 3×10^{-4} cancer deaths per person-rem, these dollar values are equivalent to a range of about $120 to $2,000 per person-rem avoided. These values can be compared to the marginal cost-effectiveness (dollars per person-rem) of evacuation over an angle of 90 degrees. The resulting ranges of upper bounds on dose are shown in Table C-4 for SST-1, SST-2, and SST-3 accident scenarios. The maximum upper bounds (based on minimum costs for avoiding risk) range from 1 to 10 rem, with most values being approximately 5 rem. The minimum upper bounds (based on maximum costs for avoiding risk) range from 0.15 to 0.8 rem (1.5 to 8 mSv), with 0.5 rem (5 mSv) being representative of most situations. From these data EPA concludes that, based on the cost of evacuation, a PAG outside the range of 0.5 to 5 rem (5 to 50 mSv) would be incompatible with EPA's commitment, in its third guiding principle, to consider both the costs and benefits of evacuation.

C.2.2 Risk of Evacuation

EPA's fourth guiding principle requires that the risk of the protective action not exceed the risk associated with the radiation dose that will be avoided by the action. Risk from evacuation can come from several sources, including: (1) transportation incidents for both pedestrians and vehicle passengers; (2) exposure to severe weather conditions or a competing disaster; and (3) in the case of immobile persons, anxiety, unusual activity, and separation from medical

care or services. The first source, transportation incidents, is the only category for which the risk has been quantified. An EPA report (Hans et al. 1975) evaluated the risk of transportation fatalities associated with emergency evacuations that have actually occurred and concluded that the risk of death per mile traveled is about the same as that for routine automobile travel. Using this as a basis, the risk of death from travel is about $9x10^{-8}$ deaths per person-mile, or $9x10^{-6}$ deaths per person for the 100-mile round trip assumed for evacuation. Assuming a risk of fatal cancer from radiation of approximately $3x10^{-4}$ per person-rem, such an evacuation risk is equivalent to a dose of about 0.03 rem (0.3 mSv).

Table C-4. Upper Bounds on Dose for Evacuation, Based on the Cost of Avoiding Fatalities[a]

Accident Category	Atmospheric Stability Class	Dose Upper Bounds[b,c]	
		Maximum (rem)	Minimum (rem)
SST-1	A	5	0.4
	C	5	0.4
	F	10	0.8
SST-2	A	1	0.15
	C	3.5	0.25
	F	10	0.7
SST-3	A	(d)	(d)
	C	(d)	(d)
	F	5	0.45

[a]Based on data from EPA 1987.
[b]Windspeeds typical of each stability class were chosen.
[c]Based on an assumed range of $400,000 to $7,000,000 per life saved.
[d]For stability classes A and C, the dose from an SST-3 accident is not predicted to exceed 0.5 rem outside a 2-mile radius. It is assumed that evacuation inside this radius would be carried out based on the emergency condition on the site. No differential evacuation costs were calculated within this area.

In comparing this risk (or, more exactly, its equivalent in dose) to the risk avoided by evacuation, it is important to note that protective actions must be implemented over a larger population than will actually be exposed at the level of the PAG. Because of uncertainty or unpredictable changes in wind direction, the exact location of the plume will not be precisely known. Dose projections are made for the maximum exposed individuals—those at the assumed location of the plume centerline. To assure that these individuals will be protected, it is necessary that others on either side take protective action at exposures that are less than at the plume centerline, and, in some cases, are zero. Thus, the entire evacuated population could incur, on the average, a risk from the protective action which exceeds the risk of the radiation dose avoided.

Although it is impossible to assure that all individuals incur risks from evacuation less than their radiation risks, it can be assured that this does not occur, on the average, at the outer margin of the evacuation area. For this reason, EPA also examined the average dose avoided for the incrementally evacuated population for various choices of evacuation levels. Table C-5 presents the results, which are derived from the data in Tables C-1, C-2, and C-3. For the levels analyzed, the average dose avoided is always significantly greater than 0.03 rem (0.3 mSv). In other words, at a centerline dose at or above 0.5 rem (5 mSv), the risk of evacuation is less than the resulting radiation dose for persons in the general population whose risk from evacuation is primarily the normal risk of transportation.

Table C-5. Average Dose Avoided per Evacuated Individual for Incremental Dose Levels for Evacuation

Centerline Dose (rem)	Average Dose Avoided (rem per Individual) by Stability Class		
	A	C	F
0.5 to 1	0.34	0.19	0.07
1 to 2	0.67	0.38	0.15
2 to 5		0.87	0.33
5 to 10			0.75

As previously discussed, hazardous environmental conditions (e.g., severe weather or a competing disaster) could create transportation risks from evacuation that would be higher than normal. It is therefore appropriate to make an

exception to allow higher projected doses for evacuation decisions under these circumstances. In the absence of any definitive information on such higher risks from evacuation, EPA has assumed that it would be appropriate to increase the recommended projected dose for evacuation of the general population under hazardous environmental conditions up to a factor of five higher than that used under normal environmental conditions.

It is also recognized that those persons who are not readily mobile are at a higher risk from evacuation than are the average members of the population. It would be appropriate to adopt higher PAG levels for evacuation of individuals who would be at greater risk from evacuation itself than for the typically healthy members of the population who are at low risk from evacuation. In the absence of definitive information on the higher risk associated with the evacuation of this group, EPA has assumed that it is appropriate to adopt a PAG a factor of 5 higher for evacuation of high risk groups under normal environmental conditions. If both conditions exist, (high-risk groups and hazardous environmental conditions) projected doses up to 10 times (10x) higher than the PAGs for evacuation of the general population under normal conditions may be justified.

C.2.3 Thyroid Blocking

The ingestion of stable potassium iodide (KI) to block the uptake of radioiodine by the thyroid has been identified as an effective supplementary protective action. The FDA updated its guidance on the use of KI in radiation emergencies (FDA 2001). In this revised guidance, the FDA lowered the exposure situations for which KI could be used from a projected adult thyroid dose level of 25 rem (250 mSv) to a projected child thyroid dose level of 5 rem (50 mSv). The FDA based its new dose level recommendations on a review of the thyroid cancer data from the Chernobyl reactor accident of April 1986. The recommended KI dosages for children and adults were also revised based on information from Poland where millions of people received KI after the Chernobyl release.

It should be stated that exposure to radioiodines is predominantly through the ingestion pathway, and the Chernobyl data support this conclusion (FDA 2001). Beginning within a week of the Chernobyl accident, the government made direct measurements of thyroid exposures in hundreds of thousands of individuals across three republics of the former Soviet Union. Milk consumption data were also used to complement some dosimetric analyses. A report published in 2005 by the International Atomic Energy Agency's (IAEA's) Chernobyl Forum on the health effects of the Chernobyl accident included a finding that ingestion of contaminated milk products was the primary cause of the thyroid cancers found in children living in the surrounding regions. Consequently, interdiction of contaminated milk and use of stored feed would have prevented most of the thyroid cancers found in these children. The Chernobyl reactor accident resulted in massive releases of I-131 and other radioiodines. Approximately 4 years after the accident, scientists began to observe a sharp increase in the incidence of thyroid cancer among children and adolescents in Belarus and Ukraine, the area covered by the plume. In some regions, the number of observed thyroid cancer cases exceed the expected number by 30 to 60 times (30x-60x). During the ensuing years, in the most heavily affected areas, the incidence of thyroid cancer was as much as 100 times greater than the pre-accident rate of this type of cancer. Most cases occurred in children who apparently received less than 30 rem (300 mSv) to the thyroid. A few cases occurred in children exposed to doses of <1 rem, but uncertainty exists as to the causal role of radioiodine at this dose level.

The FDA concluded that the best dose-response data from Chernobyl show a marked increase in the risk of childhood thyroid cancer with thyroid exposures of 5 rem (50 mSv) or greater (FDA 2001). In addition, the incidence of thyroid cancer among children born more than 6 months after the accident in areas traversed by the radioactive plume did not exceed pre-accident rates. These data are consistent with the short half-life of I-131.

The FDA also made changes to the recommended dosage of KI. New recommendations are based on a large body of evidence concerning the side effects of stable iodine and the experience with KI use in Poland after the Chernobyl accident (FDA 2001). The side effects of stable iodine include thyrotoxicosis, which is more common in older people, iodine goiter, and hypothyroidism. Iodine goiter and hypothyroidism usually result from chronic high doses of stable iodine and are therefore not considered to be likely side effects from the administration of KI in an emergency situation.

Approximately 10.5 million Polish children under the age of 16 and 7 million adults received at least one dose of KI after the Chernobyl accident. The FDA reported that some newborns who received single doses of 15 mg KI (0.37%) showed transient increases in TSH (thyroid stimulating hormone) and decreases in free thyroxin. Other side

effects among adults and children were generally mild and not clinically significant and included gastrointestinal distress, rash and allergic reactions.

The FDA recommendations for KI intervention encompass different threshold thyroid radioactive exposures for different groups within the population. Several factors were involved in this approach. During the Chernobyl accident, younger people exposed to radioactive iodine (especially children 0 to 4 years old) were most sensitive to its carcinogenic effects. In the years following the accident, most children who subsequently developed thyroid cancer apparently received internal thyroid radioactive exposures of less than 30 rad (30 cGy), and the best dose-response information supports increased risk in children receiving 5 rad (5 cGy) or more.

As age increases, the risk of thyroidal side effects following excess (i.e., nonradioactive) iodine ingestion increases. This is due to the increasing prevalence of underlying thyroidal illness with older age (e.g., Graves' disease, thyroid nodules, Hashimoto's thyroiditis).

- Thyroid irradiation in older adults (i.e., over 40 years of age) is associated with an extremely low incidence of cancer. Therefore, KI is only recommended if a very large internal radioactive dose to the thyroid is projected. In such a situation, KI would be ingested to prevent destruction of the thyroid gland, which, if it occurred, would lead to lifelong dependence on thyroid hormone replacement therapy.

- After careful review of the data from Chernobyl relating radiation dose and cancer risk in exposed children, the FDA recommends administration of KI to children 0-18 years and pregnant or lactating women in the event of a projected radiation dose to the thyroid of 5 rem (50 mSv) or greater. For adults up to 40 years of age, the FDA recommends administration when projected thyroid doses exceed 10 rem (100 mSv). They also recommend that adults over 40 need only take KI when projected thyroid doses exceed 500 rem (5 Sv) to prevent hypothyroidism. The FDA emphasized that KI should only be used as an adjunct to evacuation, sheltering-in-place, and food control. The FDA recommendations are summarized in Table C-6 below.

Table C-6. Threshold Thyroid Radioactive Exposures and Recommended Doses of KI for Different Risk Groups

Age Group	Thyroid Exposure Level	KI Dose (mg)	# of 130 mg Tablets	# of 65 mg Tablets	Milliliters (mL) of Oral Solution, 65 mg/mL
Adults over 40 y	≥500 rem	130	1	2	2
Adults over 18-40 y	≥10 rem	130	1	2	2
Pregnant or lactating women	≥5 rem	130	1	2	2
Adolesc. over 12-18 y*	≥5 rem	65	1/2	1	1
Children over 3-12 y	≥5 rem	65	1/2	1	1
Over 1 month - 3 y	≥5 rem	32	1/4	1/2	0.5
Birth-1 month	≥5 rem	16	1/8	1/4	0.25

*Adolescents approaching adult size (~70 kg) should receive the full adult dose (130 mg)

For optimal protection against inhaled radioiodines, the FDA recommends that KI should be given before or immediately coincident with the passage of the radioactive cloud. However, KI may still have a substantial protective effect even if taken 3 or 4 hours after exposure. If the release of radioactive iodine into the atmosphere is protracted, then even a delayed administration of KI may still reduce or eliminate the total radiation dose to the thyroid.

In its guidance, the FDA states that the protective effect of KI lasts approximately 24 hours. For optimal prophylaxis, KI should be given daily until the risk of significant exposure to radioiodines, by either inhalation or ingestion, no longer exists. Individuals intolerant of KI at protective doses and pregnant and lactating women (in whom repeat doses of KI may raise particular safety issues) should be given priority with regard to other protective measures like sheltering-in-place, evacuation, and food supply control.

The FDA also recommends that, once the plume has passed and radiation protection measures are in place, the uptake of ingested radioiodines should be reduced by using food control measures and not by repeated administration of KI. Because of radioactive decay, grain products and canned milk and vegetables from sources affected by radioactive fallout should pose no radiation risks from radioactive iodine if they are stored for weeks to

months after production and before consumption. Thus, KI prophylaxis at the time these products are consumed is not required.

In a supplement to its present guidance (FDA 2002), FDA added that a KI administration program that sets different projected thyroid radioactive exposure thresholds for treatment of different population groups may be logistically impractical to implement during a radiological emergency. If emergency planners reach this conclusion, FDA recommends that KI be administered to both children and adults at the lowest intervention threshold (i.e., > 5 rem (50 mSv) projected internal thyroid exposure in children).

As a rule, however, individuals with known allergy to KI or with pre-existing thyroid disease (e.g., Graves' disease, thyroid nodules, Hashimoto's thyroiditis) that might predispose them to adverse reactions should avoid KI. Most likely these will be adults who have little or no risk of developing thyroid cancer from radioactive exposure to the thyroid and who may, in these cases, incur substantial risks from taking KI.

An oral solution of potassium iodide (containing 65 mg of KI in each mL) is available to facilitate dosing of children. KI tablets can be dissolved in liquids and the appropriate volume administered. For example, if a 130 mg tablet were dissolved in 8 ounces of liquid, 1 ounce would contain about 16 mg of KI. FDA has conducted studies of the palatability, solubility, and stability of KI dissolved in a number of different liquids, including juice and formula. Emergency planners and others should understand that absolute precision in dosing is generally not critical to safety or efficacy.

FDA's guidance on dosing KI in radiation emergencies adheres to principles of minimum effective dose and therefore recommends graded dosing according to age (and thus in effect, body size). There is ample evidence that the recommended doses, as well as higher doses (i.e., up to 130 mg), will effectively block thyroidal uptake of radioactive iodine if taken in advance of exposure. Furthermore, particularly among school-age children, higher milligram doses are extremely safe (FDA 2002).

FDA also realizes that a scheme of graded dosing may be difficult to implement during a radiological emergency involving large numbers of people. However, they continue to emphasize attention to KI dosing in infants. Excess iodine intake can lead to transient iodine-induced hypothyroidism. As FDA has said in its guidance, individuals who are intolerant of KI at protective doses, as well as neonates, pregnant, and lactating women, should be given priority with regard to other protective measures (i.e., sheltering-in-place, evacuation, and control of the food supply).

In summary, if local emergency planners conclude that graded dosing is logistically impractical, FDA believes that for populations at risk for radioiodine exposure, the overall benefits of taking up to 130 mg of KI instead of the lower doses recommended for certain age groups far exceed the small risks of overdosing (FDA 2002). However, where feasible, adherence to FDA guidance should be attempted when dosing infants.

FDA recommendations differ from the 1999 World Health Organization (WHO) (WHO 1999) guidelines for KI prophylaxis in two areas. WHO recommends a 130 mg dose of KI for adults and adolescents over 12. For the sake of logistical simplicity in dispensing and administering KI to children, FDA recommends a 65 mg dose as the standard for all school-age children, while allowing an adult dose (i.e., a single 130 mg tablet or 2 x 65 mg tablets) in adolescents approaching adult size. A second difference in the recommendations of these 2 organizations relates to the threshold exposure level (i.e., the level at which KI prophylaxis should be used) to those up to 18 years of age and to pregnant or lactating women. WHO recommends a 1 rem threshold for these groups. The FDA concluded from the Chernobyl data that the most reliable evidence suggests a significant increase in the risk of childhood thyroid cancer at exposures of 5 rem or greater. Thus, FDA recommends a 5 rem threshold for these groups.

FEMA has published a federal policy developed by the Federal Radiological Preparedness Coordinating Committee regarding the use of KI as a protective action (FEMA 2002). In summary, the policy recommends that KI be stockpiled in sufficient quantities to allow its distribution during emergencies to emergency workers and institutionalized persons. However, FEMA does not recommend that enough KI be stockpiled to allow its distribution to the general public during an emergency. The policy recognizes, however, that options on the distribution and use of KI rest with the states. Hence, FEMA's policy permits state and local governments, within the limits of their authority, to take measures beyond those recommended or required nationally.

C.2.4 Sheltering-in-Place

Sheltering-in-place means staying inside a structure with the doors and windows closed and, generally, with any exterior ventilation systems shut off. Sheltering-in-place (i.e., at or near the location of an individual when a radiological incident occurs) is a low-cost, low-risk protective action that can provide protection with an efficiency ranging from almost 100% to zero, depending on the circumstances. It can also be particularly useful to assure that a population is positioned so that, if the need arises, communication can be carried out expeditiously. The degree of protection provided by a structure is governed by the attenuation of radiation levels by structural components (e.g., the mass of walls, ceilings, etc.) and by the outside-inside air exchange rate of the building. These two protective characteristics are considered separately.

The protection factor for either of these two protective characteristics may be described by a dose reduction factor (DRF) defined as:

$$DRF = \frac{\text{dose with protective action}}{\text{dose without protective action}}$$

The shielding characteristics of most structures for gamma radiation can be categorized based on whether they are "small" or "large." Small structures are primarily single-family dwellings, and large structures include office, industrial, and commercial buildings. The typical attenuation factors given in Table C-7 show the importance of the type of structure in terms of its ability to provide protection from external gamma radiation (EPA 1978a). If the structure is a wood frame house without a basement, then, according to Table C-7, its DRF = 0.9, which means that only 10% of the dose would be avoided. The DRFs shown in Table C-7 are initial values prior to infiltration of contaminated air, and therefore apply only to short duration plumes. The values will increase (and the effectiveness will decrease) with increasing time of exposure to a plume due to a structure's outside-inside air exchange rate. However, this reduction in efficiency is not dramatic for source terms involving primarily gamma radiation because most of the dose arises from outside a structure and not from the small volume of contaminated air inside a shelter. Therefore, most shelters will retain their efficiency as shields against gamma radiation even if the concentration of radiation inside equals the concentration of radiation outside.

Table C-7. Representative Dose Reduction Factors for External Radiation

Structure	DRF	Effectiveness (%)
Wood frame house (first floor)	0.9	10
Wood frame house (basement)	0.6	40
Masonry house	0.6	40
Large office or industrial building	0.2 or less	80 or better

The second factor is the outside-inside air exchange rate. This factor primarily affects a shelter's ability to protect against exposure by inhalation of airborne radionuclides with half lives long compared to the air exchange rate. The factor is expressed as the number of air exchanges per hour, L (h^{-1}), or the volume of fresh air flowing into and out of the structure per hour divided by the volume of the structure. Virtually any structure that can be used for sheltering-in-place has some degree of outside-inside air exchange due to natural ventilation, forced ventilation, or uncontrollable outside forces, primarily wind.

Assuming constant atmospheric and source conditions and no effects from filtration, deposition, or radioactive decay, the following model can be used to estimate the buildup of radioactivity indoors, for a given outdoor concentration, as a function of time, after the appearance of the plume, and of ventilation rate:

$$C_i = C_o(1 - e^{-Lt}),$$

where:

C_i = concentration inside,
C_o = concentration outside,
L = ventilation rate (h^{-1}), and
t = elapsed time (h)

Typical values for ventilation rates range from one-fifth to several air exchanges per hour. In the absence of measurements, an air exchange rate of 1.0/h may be assumed for structures with no special preparation except for closing the doors and windows. An air exchange rate of 0.3/h is appropriate for relatively air-tight structures, such as well-sealed residences, interior rooms with doors chinked and no windows, or large structures with outside ventilation shut off. Using the above model to calculate indoor concentration relative to outdoor concentration after one, two, and four complete air exchanges, the indoor concentration would be about 64%, 87%, and 98% of the outside concentration, respectively. It is apparent from these numbers, that sheltering-in-place for long periods of time (i.e., beyond that required for one or two complete air exchanges) is not very effective in reducing inhalation exposure.

The inhalation DRF is equal to the ratio of the average inside to outside air concentration over the period of sheltering-in-place. Studies have been conducted of typical ventilation rates for dwellings (EPA 1978a) and for large commercial structures (GR 86). In each case, the rate varies according to the air tightness of the structure, windspeed, and the indoor-to-outdoor temperature difference. For the purpose of deriving PAGs, average ventilation rates were chosen for the two types of structures that are of greatest interest. Table C-8 shows calculated dose reduction factors for inhalation exposure as a function of plume duration (for beta-gamma source terms), assuming average ventilation rates for these structures.

Table C-8. Dose Reduction Factors for Sheltering -in-Place from Inhalation of Beta-Gamma Emitters

Ventilation Rate (Air Changes/h)	Duration of Plume Exposure (h)	DRF
0.3[a]	0.5	0.07
	1	0.14
	2	0.25
	4	0.41
	6	0.54
1.0[b]	0.5	0.21
	1	0.36
	2	0.56
	4	0.75
	6	0.83

[a]Applicable to relatively "airtight" structures such as well-sealed residences, interior rooms with tight-fitting doors and no windows, or large structures with outside ventilation shut off.
[b]Applicable to structures with no special preparation except for closing of doors and windows.

A potential problem associated with sheltering-in-place is that persons may not leave the shelter as soon as the plume passes; as a result, they will receive exposure from radioactive gases trapped inside. The values for DRFs tabulated in Table C-8 ignore this potential additional contribution to dose. This effect is generally minor for gamma dose (generally less than a 10% increase in the dose received during plume passage (EPA 1978b)), but can be greater for inhalation dose.

Doses from inhalation during sheltering-in-place can be reduced in several ways, including reducing air exchange rates by sealing cracks and openings with cloth or weather stripping, tape, etc., and filtering the inhaled air with commonly available items like wet towels and handkerchiefs. Analyses for some hypothesized incidents, such as situations involving short-term transuranic releases, show that sheltering-in-place in residences and other buildings

can be more effective for these types of radionuclide releases than for those involving beta-gamma emitters. In these situations, sheltering-in-place may provide adequate protection and may be more effective than evacuation when evacuation cannot be completed before plume arrival (DOE 1990). However, effectiveness of sheltering-in-place for the inhalation exposure pathway can be reduced drastically by open windows and doors or by forced air ventilation.

Therefore, reliance on an assumed level of protection provided by sheltering-in-place should be accompanied by a cautious examination of possible failure mechanisms. Further, except in very unusual circumstances, sheltering-in-place should not be relied upon at projected doses greater than 10 rem (100 mSv). An evaluation of the use of sheltering-in-place as a protective action should be based on realistic or "best estimate" dose models and should consider any unavoidable dose associated with the use of evacuation as an alternative means of protection.

C.3 Recommended PAG for Exposure to a Plume during the Early Phase

The risks of health effects from radiation (Principles 1 and 2) are presented in Appendix E and an analysis of the costs and risks associated with evacuation (Principle 3) are presented in this appendix. These results are summarized in Table C-9 and may be applied to the early phase of a radiological incident.

The following describes how these results led to the selection of the PAGs outlined in Chapter 2. Conformance to Principle 1 (avoidance of acute health effects) is assured by the low risk required to satisfy Principle 2, and thus requires no additional consideration. Principle 2 requires that EPA weigh the risk of delayed health effects that may be considered adequately protective of public health under emergency conditions. The ICRP recommends intervention efforts at a dose level of 0.5 rem following a radiological emergency. In other words, protective actions should be taken to avoid further radiation dose at levels of avoidable dose above 0.5 rem during the early phase of a radiological incident..

Lifetime risk levels that are higher than 0.5 rem may be justified on the basis of the nature of the situation. Nuclear incidents represent events that are not predictable or controllable to the same degree as normal facility operations. In addition, protective actions for nuclear incidents have a wide range of costs, implementation issues, and potentially adverse consequences. Given these considerations, and in the context of the entire radiological incident, EPA considers that a combined risk level, from all phases of a nuclear incident, of up to an order of magnitude higher (5 rem) may be justifiable as a basis for protective action decisions. This forms the basis for the early and intermediate phase PAGs and is still much lower than needed to avoid acute health effects. EPA acknowledges that this is a generic assessment that covers a wide range of accident types and considers the possible contributions to individual risk from exposure during all phases of the incident.

Principle 4 (risk from the protective action must be less than that from the radiation risk avoided) supplies a lower bound of 0.03 rem (0.3 mSv) on the dose at which evacuation of most members of the public is justified (see Section C.2.2). Finally, under Principle 3 (cost/risk considerations), evacuation is justified only at risk levels equal to or greater than 0.5 rem (5 mSv). However, lower values may be required for purely health-based reasons under Principle 2. However, our analysis demonstrates that this will not be the case. EPA did consider a purely health-based risk level of 0.1 rem (1 mSv) in developing the early phase PAG level, but did not adopt it because, although it is a perfectly valid health-based criterion for chronic exposure, it is an unrealistic criterion for emergency situations.

In summary, EPA has selected the value of 0.5 rem (5 mSv) as the basis for the early phase PAG because: (1) it limits the risk of delayed effects on health to levels adequately protective of public health under emergency conditions; (2) the cost of implementation of a lower value is not justified; and (3) it satisfies EPA's guiding principles to avoid acute radiation effects and to avoid increasing risk through the protective action itself. EPA notes that this choice also satisfies the criterion for acceptable risk to the fetus of occupationally exposed mothers as well as falling well below the dose values at which abortion is recommended.

In Chapter 2, EPA strongly recommends that evacuation and/or sheltering-in-place occur when projected dose levels to the public are at 1 rem TEDE or higher. This considers the two pathways (inhalation and ingestion) that could possibly result in exposure to the public as the plume passes over. At a dose level of 1 rem (10 mSv), the radiation risk avoided is usually much greater than the risk of evacuation. In addition, EPA also assumes that by evacuating the affected community, you will be avoiding at least one-half of their projected dose. Thus, by taking

protective action at 1 rem, the resultant exposures to the population in general should be on the order of 0.5 rem (5 mSv).

Table C-9. Summary of Considerations for Selecting the Evacuation PAGs

Dose (rem)	Consideration	Principle	Section
50	Assumed threshold for acute health effects in adults.	1	E.2.1.4
10	Assumed threshold for acute health effects in the fetus.	1	E.2.1.4
5	Maximum acceptable dose for normal occupational exposure of adults.	2	C.5
5	Maximum dose justified to average members of the population, based on the cost of evacuation.	3	C.2.1.3
0.5	Maximum acceptable dose to the general population from all sources from nonrecurring, non-accidental exposure.	2	E.4.4
0.5	Minimum dose justified to average members of the population, based on the cost of evacuation.	3	C.2.1.3
0.5	Maximum acceptable dose[a] to the fetus from occupational exposure of the mother.	2	C.5
0.1	Maximum acceptable dose to the general population from all sources from routine (chronic), nonaccidental exposure.	2	E.4.4
0.03	Dose that carries a risk assumed to be equal to or less than that from evacuation.	1	C.2.2

[a]This is also the dose to the 8- to 15-week-old fetus at which the risk of mental retardation is assumed to be equal to the risk of fatal cancer to adults from a dose of 5 rem.

The above considerations apply to evacuation of typical members of the population under normal circumstances and apply to effective doses (i.e., the weighted sum of doses to all organs). As discussed in previous sections, it may be appropriate to adjust the value for special groups of the population at unusually high risk from evacuation, and to provide for situations in which the general population may be at a higher than normal risk from evacuation. These are addressed below.

Special risk groups include fetuses and persons who are not readily mobile. As noted in Sections E.4.1.3 and E.3, EPA assumes that the risk of radiation-induced cancer is about five to 10 times (10x) higher for fetuses than for adults. The risk of mental retardation in fetuses exposed during the eighth to 15th weeks of gestation is about 10 times (10x) higher than the risk of fatal cancer in equivalently exposed adults. However, due to the difficulty of rapidly evacuating only pregnant women in a population, and the assumed higher-than-average risk associated with their evacuation, it is not considered appropriate to establish separate PAGs for pregnant women. EPA notes that the PAG is chosen sufficiently low to satisfy federal guidance for limiting exposure of the fetus in pregnant workers.

Higher PAG levels for situations involving higher risks from evacuation were discussed in Section C.2.2. Under normal, low-risk, environmental conditions, PAGs for evacuation of groups who present higher than average risks from evacuation (e.g., persons who are not readily mobile) are recommended at projected doses of up to 5 rem. Evacuation of the general population under high-risk environmental conditions is also recommended at projected doses of up to 5 rem (50 mSv). If evacuation of high-risk groups under hazardous environmental conditions is being considered, projected doses of up to 10 rem (100 mSv) may be justified.

Short-term sheltering-in-place is recognized as a low-cost, low-risk protective action primarily suited for protection from exposure to an airborne plume. Sheltering-in-place will usually be clearly justified to avoid projected doses above 0.5 rem (5 mSv), on the basis of avoidance of health risks. However, data are not available to establish a lower level at which sheltering-in-place is no longer justified because of its cost or the risk associated with its implementation. Sheltering-in-place will usually have other benefits related to emergency communication with

members of the public. It is expected that protective action planners and decision-making authorities will take into account the added benefits of sheltering-in-place (e.g., communication and established planning areas) for decisions on sheltering-in-place at levels below 0.5 rem (5 mSv).

Bathing and changing of clothing are effective for reducing beta dose to the skin of persons exposed to an airborne plume of radioactive materials. Since these are also low-cost, low-risk actions, no PAG is recommended for initiating their implementation. It is expected that any persons exposed in areas where evacuation is justified based on projected dose from inhalation will be routinely advised by emergency response officials to take these actions within 12 hours after exposure.

If procedures are included in the applicable emergency response plan, use of stable iodine should be considered for any such situation in which evacuation or sheltering-in-place will not be effective in preventing thyroid doses of 5 rem (50 mSv) (see also Section C.2.3).

C.4 Comparison to Previous PAGs

This section compares the level of protection provided by the previously published PAGs for evacuation (1 rem external gamma dose from the plume and 5 rem committed dose to the thyroid from inhalation, under normal evacuation circumstances) with the PAGS presented in this Manual. In order to do so, airborne releases were calculated for radionuclide mixes postulated for three nuclear power plant accident scenarios. The doses were then normalized for each accident so that they represent a location in the environment where the controlling dose would be equal to the revised PAG. These results are shown in Table C-10.

Table C-10. Comparison of Projected Doses for Various Reactor Accident Scenarios[a]

Accident Category[b]	Effective Dose Equivalent[c] (rem)	Skin Dose[d] (rem)	Thyroid Dose[e] (rem)	External Dose[f] (rem)
SST-1	0.7	6	5	0.2
SST-2	1	5	5	0.4
SST-3	0.4	6	5	0.1

[a]Doses are normalized to the limiting PAG.
[b]See Table F-1 for a description of these accident scenarios.
[c]The dose is the sum of the doses from 4-day exposure to external gamma radiation from deposited materials, external exposure to the plume, and the committed effective dose equivalent from inhalation of the plume.
[d]The dose equivalent from external beta radiation from the plume and from 12 hours of exposure to materials deposited on the skin and clothing.
[e]Committed dose equivalent to the thyroid from inhalation.
[f]External gamma dose equivalent from the plume.

C.5 Dose Limits for Workers Performing Emergency Services

Dose limits for workers during emergencies are based on avoiding acute health effects and limiting the risk of delayed health effects in the context of the need to assure protection of the population and valuable properties. It is assumed that most emergency workers are accustomed to accepting an element of risk as a condition of their employment. Examples of occupations that may be affected include law enforcement, firefighting, radiation protection, civil defense, traffic control, health services, environmental monitoring, animal care, and transportation services. In addition, selected utility, industrial, and farm and other agribusiness workers may be required to protect others or to protect valuable property during an emergency. The above are examples – not designations – of workers that may be exposed to radiation during emergencies.

Radiation exposure of workers during an emergency should normally be governed by the Federal Radiation Protection Guidance for Occupational Exposure (EPA 1987). This guidance specifies an upper bound of 5 rem (50 mSv) committed effective dose equivalent per year for most workers. (Declared pregnant women, who, under this guidance, should not normally engage in work situations that involve more than approximately 50 mrem (0.5 mSv) per month, would normally be evacuated as part of the general population). The guidance also specifies that: (1) doses to workers should be maintained as low as reasonably achievable; (2) doses should be monitored; and (3) workers should be informed of the risks involved and of basic principles for radiation protection.

There are some emergency situations, however, for which higher doses may be justified. These include lifesaving operations and the protection of valuable property. International guidance (ICRP 1977) recognizes two additional dose levels for workers under specially justified circumstances: two times (2x) the annual limit (10 rem (100 mSv)) for any single event, and five times (5x) the annual limit (25 rem (250 mSv)) in a lifetime. The dose limits recommended in this Manual include 10 rem for operations limited to the protection of valuable property. A dose limit of 25 rem (250 mSv) may be permitted for situations involving lifesaving operations or activities that are essential to preventing substantial risks to populations. In this context, "substantial risks" means collective doses that are significantly larger than those incurred through the protective activities engaged in by the workers. Workers should not operate under dose limits higher than 5 rem (50 mSv) unless the following conditions are satisfied:

- Lower doses through minimizing time, maximizing distance, employing shielding, and other commonly used dose reduction methods are not possible.

- Instrumentation is available to measure their exposure. In addition to the limitation on effective dose equivalent, the dose equivalent received in any year by workers under normal occupational conditions is limited to 15 rem (150 mSv) to the lens of the eye and 50 rem (500 mSv) to any other organ, tissue (including skin), or extremity of the body. [Extremity is defined as the forearms and hands or the lower legs and feet (EPA 1987).] By analogy to these dose limits for organs and extremities, the limits for workers performing the various categories of emergency services are established at numerical values that are five times (5x) the limits for effective dose to the lens of the eye and 10 times (10x) the limits for effective dose to any other organ, tissue (including skin), or extremity of the body.

Situations may occur in which a dose in excess of 25 rem (250 mSv) would be required for lifesaving operations. It is not possible to prejudge the risk that one person should be allowed to take to save the life of another. However, persons undertaking an emergency mission in which the dose would exceed 25 rem (250 mSv) to the whole body should do so only on a voluntary basis and with full awareness of the risks involved, including the numerical levels of dose at which acute effects of radiation will be incurred and numerical estimates of the risk of delayed effects.

The risk of acute health effects is discussed in E.2. Table C-11 presents estimated cancer mortality rates for a dose of 25 rem (250 mSv) as a function of age at the time of exposure. The risk of cancer from moderately higher doses will increase proportionately. These values were calculated using risk estimates from BEIR-3 (NA 80) as discussed in Section E.4, and life table analyses that assume the period of cancer risk lasts for the worker's lifetime (Bunger et al. 1981). The risk was calculated for the midpoint of each age range. Roughly equivalent risks of nonfatal cancer and serious genetic effects (if gonadal tissue is exposed) will also be incurred.

The dose limits of 75 rem (750 mSv) to the whole body previously recommended by EPA and 100 rem (10 Sv) recommended by NCRP (GL-57) for lifesaving action represents a very high level of risk for both acute and delayed health effects. A dose of 100 rem (10 Sv) is expected to result in an approximately 15% risk of temporary incapacity from nonlethal acute effects and an indeterminate, but less than 5%, chance of death within 60 days. This is in addition to a risk of about one in 30 of incurring fatal cancer. Such high risk levels can only be accepted by a recipient who has been made fully aware of the risks involved. Therefore, no absolute dose limit for lifesaving activities is offered.

Table C-11. Cancer Risk to Emergency Workers Receiving 25 Rem Whole Body Dose

Age of the Emergency Worker at the Time of Exposure (Years)	Approximate Risk of Premature Death (Deaths per 1,000 Persons Exposed)	Average Years of Life Lost if Premature Death Occurs (Years)
20 to 30	9.1	24
30 to 40	7.2	19
40 to 50	5.3	15

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Appendix D: Basis for Calculations

D.1 Combined Removal Parameter Calculation for Table 2-9

The CRP adjusts the groundshine dose for radioactive decay and weathering over the time period (i.e., early phase) under consideration. Although weathering has little effect on the groundshine dose over the first 96 hours, it is included to be complete and consistent with the methods used to calculate radiation doses over the other time periods (e.g., first year, second year).

$$CRP_{TP,i} = \int_{T_1}^{T_2} \left(EffXP_{TP,i} * WF_{TP,i} \right) dT \quad , \quad h = h * unitless$$

Where:

$CRP_{TP,\,i}$ = Combined Removal Parameter, value that adjusts the external (groundshine) dose from radionuclide *i* for radioactive decay and weathering effects which decrease the groundshine dose over the time phase under consideration, h,

$WF_{TP,\,i}$ = Weathering Factor, value that adjusts the groundshine dose from radionuclide *i* for weathering forces that reduce the radionuclide activity near the surface over time, and, thereby decreases the external dose rate, unitless, and

$EffXP_{TP,\,i}$ = Effective Exposure Period, value that adjusts the groundshine dose from radionuclide *i* for radioactive decay that occurs over the time phase under consideration, h.

D.1.1 Weathering Factor Calculation for Table 2-9

The Weathering Factor (WF) adjusts the external exposure rate for the decrease that occurs over time as the deposited material migrates deeper into the soil column. The WF model was developed using data from the Chernobyl nuclear power plant accident (HPS 2002). It should be noted that This WF model may not be appropriate for the environmental conditions existing in the area under investigation. An alternate WF model may be substituted, with EPA approval, if the alternate model can be shown to more accurately model the weathering in the area under investigation.

$$WF = 0.4e^{(-1.46E^{-}8 * t)} + 0.6e^{(-4.44E^{-}10 * t)}$$

$$unitless = \left(e^{\left(-\frac{1}{s} * s \right)} \right) + unitless \left(e^{\left(-\frac{1}{s} * s \right)} \right)$$

Where:

0.4 = fraction of material that undergoes rapid weathering, unitless,

0.6 = fraction of material that undergoes slow weathering, unitless,

1.46E-8 = rate constant representing the removal rate for the fraction of material that is rapidly ($t_{1/2}$ = 1.5 y) weathered, s^{-1},

4.44E-10 = rate constant representing the removal rate for the fraction of material that is slowly ($t_{1/2}$ = 50 y) weathered, s^{-1}, and

t = seconds.

D.1.2 Effective Exposure Period Calculation for Table 2-9

The EffXP adjusts the external (groundshine) dose for radioactive decay over the time phase under consideration and is calculated using the equation below.

$$EffXP_{TP,i} = CF * \int_{T_1}^{T_2} e^{(-T*\lambda_i)} dT \,, \qquad h = \frac{h}{s} * \int_{s}^{s} e^{\left(-s*\frac{1}{s}\right)}$$

Or

$$EffXP_{TP,i} = CF * \left(\frac{e^{(-T_2*\lambda_i)} - e^{(-T_1*\lambda_i)}}{-\lambda_i} \right), \quad h = \frac{h}{s} * \left[\frac{\left(e^{\left(-s*\frac{1}{s}\right)} - e^{\left(-s*\frac{1}{s}\right)} \right)}{-\frac{1}{s}} \right]$$

Where:

CF = Unit conversion factor, 2.77E-04 h/s,
λ_i = Decay constant for radionuclide *i*, s^{-1},
T_1 = Time of the beginning of the integration period, 0 y, and
T_2 = Time of the end of the integration period, 4 days.

D.1.3 Combining the WF and EffXP to Calculate the CRP for Table 2-9

Ignoring the unit conversion factor of 2.77E-04 h/s and multiplying the WF: ($WF_{TP,i} = 0.4 * e^{(-T*1.46E^{-}8)} + 0.6 * e^{(-T*4.44E^{-}10)}$) by the EffXP ($EffXP_{TP,i} = e^{(-\lambda_i * T)}$) yields:

$$CRP = e^{(-\lambda_i * T)} * * (0.4 * e^{(-T*1.46E^{-8})} + 0.6 * e^{(-T*4.44E-10)})$$

Which simplifies to:

$$CRP = (0.4 * e^{(-T*(\lambda_i + 1.46E-8))} + 0.6 * e^{(-T*(\lambda_i + 4.44E-10))})$$

$$unitless = unitless * \mathrm{e}^{\left(-\left(\frac{1}{s}+\frac{1}{s}\right)*s\right)} + unitless * \mathrm{e}^{\left(-\left(\frac{1}{s}+\frac{1}{s}\right)*s\right)}$$

Integrating over a time period of interest yields the following CRP:

$$CRP = \frac{0.4 * (e^{(-T_2*(\lambda_i + 1.46E-8))} - e^{(-T_1*(\lambda_i + 1.46E-8}))}{-(\lambda_i + 1.46E-8)} + \frac{0.6 * (e^{(-T_2*(\lambda_i + 4.44E-8))} - e^{(-T_1*(\lambda_i + 4.44E-8))})}{-(\lambda + 4.44E-10)}$$

$$s = \frac{unitless * \left(e^{\left(s*\left(\frac{1}{s}+\frac{1}{s}\right)\right) - \left(s*\left(\frac{1}{s}+\frac{1}{s}\right)\right)} \right)}{-\left(\frac{1}{s}+\frac{1}{s}\right)} + \frac{unitless * \left(e^{\left(s*\left(\frac{1}{s}+\frac{1}{s}\right)\right) - \left(s*\left(\frac{1}{s}+\frac{1}{s}\right)\right)} \right)}{-\left(\frac{1}{s}+\frac{1}{s}\right)}$$

NOTE: If desired (as in Tables 7.2 and 7.5 of the 1992 PAG Manual), the WF can be ignored (turned off) by setting the WF exponent terms to zero. This reduces the previous formula to:

$$CRP = \frac{0.4 * (e^{(-T_2*(\lambda_i + 0))} - e^{(-T_1*(\lambda_i + 0))})}{-(\lambda_i + 0)} + \frac{0.6 * (e^{(-T_2*(\lambda_i + 0))} - e^{(-T_1*(\lambda_i + 0))})}{-(\lambda_i + 0)}$$

Which further reduces to:

$$CRP = \frac{0.4 * (e^{(-T_2 * \lambda_i)} - e^{(-T_1 * \lambda_i)})}{-\lambda_i} + \frac{0.6 * (e^{(-T_2 * \lambda_i)} - e^{(-T_1 * \lambda_i)})}{-\lambda_i}$$

Because 0.4 + 0.6 =1, the above equation reduces to the following equation which is equivalent to the effective exposure period and only adjusts for radioactive decay:

$$CRP = \frac{(e^{(-T_2 * \lambda_i)} - e^{(-T_1 * \lambda_i)})}{-\lambda_i} = EffXP \quad , \quad s = \frac{\left(s * \frac{1}{s}\right) - \left(s * \frac{1}{s}\right)}{-\left(\frac{1}{s}\right)}$$

D.2 Resuspension Parameter (KP) Calculation for Table 2-9

The Resuspension Parameter (KP) adjusts the inhalation dose for radioactive decay and the time-dependent resuspension factor (K) that occurs over the time period under consideration. The KP integral below does not have an exact solution when "K" is in a time-dependent form. Therefore, the integral cannot be solved analytically and must be solved using a software program that capable of numerical integration. It should be noted that the K model described below may not be appropriate for the environmental conditions existing in the area under investigation. An alternate K model may be substituted, with EPA approval, if the alternate model can be shown to more accurately model the resuspension in the area under investigation.

$$KP_i = CF * \int_{T_1}^{T_2} K * e^{(-\lambda_i * T)} dT \quad , \qquad \frac{h}{cm} = \frac{h/cm}{s/m} * \int_s^s \frac{1}{m} * e^{\left(\frac{-\frac{1}{s} * s}{\frac{1}{s}}\right)}$$

Where:

KP_i = Resuspension Parameter, adjusts the inhalation dose from radionuclide *i* for radioactive decay and the time-dependent resuspension factor (K), h/cm,

CF = Unit conversion factor, 2.78E-06 h/cm per s/m,

$$\frac{2.78E-6\, h/cm}{s/m} = \frac{s}{m} * \frac{m}{100\, cm} * \frac{h}{3600\, s}$$

T_1 = Time at the start of the time phase (integration period) under consideration, s,

T_2 = Time at the end of the time phase (integration period) under consideration, s,

K = Resuspension Factor, based on the time-varying formula from NCRP Report No. 129, *Recommended Screening Limits for Contaminated Surface Soil and Review of Factors Relevant to Site-Specific Studies*, (NCRP 1999), m^{-1},

- K = 1.00E-06 m^{-1} for t <1 d or
- K = 1.00E-06 m^{-1}/t for t >1 and ≤1,000 d or,
- K = 1.00E-09 m^{-1} for t >1000 d.

λ_i = Decay constant for radionuclide *i*, s^{-1}.

D.3 Combined Removal Parameter (CRP) Calculation for Tables 3-3 and 3-4

The Combined Removal Parameter (CRP) adjusts the groundshine dose for radioactive decay and weathering over the time period under consideration.

$$CRP_{i,TP} = \int_{T_1}^{T_2} \left(EffXP_{i,TP} * WF_{i,TP} \right) dT \;,\; s = s * unitless$$

Where:

$CRP_{i, TP}$ = Combined Removal Parameter, value that adjusts the external (groundshine) dose from radionuclide *i* for radioactive decay and weathering that decrease the radioactivity over the time phase under consideration, s,

$WF_{i, TP}$ = Weathering Factor, value that adjusts the groundshine dose from radionuclide *i* for weathering forces that reduce the radionuclide activity near the surface over time, and, thereby decreases the external dose rate, unitless, and

$EffXP_{i, TP}$ = Effective Exposure Period, value that adjusts the groundshine dose from radionuclide *i* for radioactive decay that occurs over the time phase under consideration, s.

D.3.1 Weathering Factor Calculation

The Weathering Factor (WF) adjusts the external exposure rate for the decrease that occurs over time as the deposited material migrates deeper into the soil column. The WF model was developed using data from the Chernobyl nuclear power plant accident (HPS 2002). It should be noted that this WF model may not be appropriate for the environmental conditions existing in the area under investigation. An alternate WF model may be substituted, with EPA approval, if the alternate model can be shown to more accurately model the weathering in the area under investigation.

$$WF = 0.4e^{(-1.46E^{-}8 * t)} + 0.6e^{(-4.44E^{-}10 * t)}$$

$$unitless = \left(e^{\left(-\frac{1}{s} * s \right)} \right) + unitless \left(e^{\left(-\frac{1}{s} * s \right)} \right)$$

Where:

0.4 = fraction of material that undergoes rapid weathering, unitless,

0.6 = fraction of material that undergoes slow weathering, unitless,

1.46E-08 = rate constant representing the removal rate for the fraction of material that is rapidly ($t_{1/2}$ = 1.5 y) weathered, s^{-1},

4.44E-10 = rate constant representing the removal rate for the fraction of material that is slowly ($t_{1/2}$ = 50 y) weathered, s^{-1}, and

t = seconds.

D.3.2 Effective Exposure Period Calculation

The EffXP adjusts the groundshine dose for radioactive decay over the time phase under consideration and is calculated using the following equation.

$$EffXP_{TP,i} = \int_{T_1}^{T_2} e^{(-T * \lambda_i)} dT \,, \qquad s = \int_{s}^{s} e^{\left(-s * \frac{1}{s}\right)}$$

Or

$$EffXP_{TP,i} = \frac{e^{(-T_2 * \lambda_i)} - e^{(-T_1 * \lambda_i)}}{-\lambda_i}, \quad s = \frac{\left(e^{\left(-s * \frac{1}{s}\right)} - e^{\left(-s * \frac{1}{s}\right)}\right)}{-\frac{1}{s}}$$

Where:

λ_i = decay constant for radionuclide *i*, s^{-1},

T_1 = the start of the time phase (integration period) under consideration, s, and

T_2 = the end of the time phase (integration period) under consideration, s.

D.3.3 Combining the WF and EffXP to Calculate the CRP

Multiplying the WF ($WF_{TP,i} = 0.4 * e^{(-T * 1.46E^{-}8)} + 0.6 * e^{(-T * 4.44E^{-}10)}$) by the EffXP ($EffXP_{TP,i} = e^{(-\lambda_i * T)}$) yields:

$$CRP = e^{(-\lambda_i * T)} * \left(0.4 * e^{(-T * 1.46E^{-8})} + 0.6 * e^{(-T * 4.44E^{-}10)}\right)$$

Which simplifies to:

$$CRP = (0.4 * e^{(-T*(\lambda_i + 1.46E-8))} + 0.6 * e^{(-T*(\lambda_i + 4.44E-10))}$$

$$unitless = unitless * \mathrm{e}^{\left(-\left(\frac{1}{s}+\frac{1}{s}\right)*s\right)} + unitless * \mathrm{e}^{\left(-\left(\frac{1}{s}+\frac{1}{s}\right)*s\right)}$$

Integrating over a time period of interest yields the following CRP:

$$CRP = \frac{0.4 * \left(\mathrm{e}^{\left(-T_2 * \left(\lambda_i + 1.46E^{-}8\right)\right)} - \mathrm{e}^{\left(-T_1 * \left(\lambda_i + 1.46E^{-}8\right)\right)}\right)}{-(\lambda_i + 1.46E^{-}8)} + \frac{0.6 * \left(\mathrm{e}^{\left(-T_2 * \left(\lambda_i + 4.44E^{-}8\right)\right)} - \mathrm{e}^{\left(-T_1 * \left(\lambda_i + 4.44E^{-}8\right)\right)}\right)}{-(\lambda_i + 4.44E^{-}10)}$$

$$s = \frac{unitless * \left(e^{\left(s * \left(\frac{1}{s}+\frac{1}{s}\right)\right) - \left(s * \left(\frac{1}{s}+\frac{1}{s}\right)\right)}\right)}{-\left(\frac{1}{s}+\frac{1}{s}\right)} + \frac{unitless * \left(e^{\left(s * \left(\frac{1}{s}+\frac{1}{s}\right)\right) - \left(s * \left(\frac{1}{s}+\frac{1}{s}\right)\right)}\right)}{-\left(\frac{1}{s}+\frac{1}{s}\right)}$$

D.3.4 Adjusting the Groundshine Dose only for Radioactive Decay

If desired (as in Tables 7.2 and 7.5 of the 1992 PAG Manual) the Weathering Factor (WF) can be ignored when calculating the external dose from groundshine. To ignore the WF and to adjust the groundshine dose for only radioactive decay, substitute the Effective Exposure Period (EffXP).

D.4 External Dose Factor Calculation

The External Dose Factor (ExDF) gives the effective dose from groundshine per unit activity deposited on the ground and adjusted for the ground roughness factor (GRF).

$$ExDF_{ground,E,i} = GRF * \mathrm{ExDC}_{ground,E,\mathrm{i}} * CF$$

$$\frac{mrem \cdot m^2}{s \cdot \mu Ci} = unitless * \frac{Sv \cdot m^2}{s \cdot Bq} * \frac{mrem \cdot Bq}{\mu Ci \cdot Sv}$$

Where:

$ExDF_{ground, E, i}$ = External Dose Factor for Deposition, the effective dose rate from the external exposure to radionuclide *i* per unit activity deposited on the ground and adjusted for the ground roughness factor, mrem·m^2/ s·μCi,

GRF= Ground Roughness Factor, a unitless constant (0.82) that compensates for the fact that the external exposure is not coming from an infinite flat plane (HPS, 2002),

$ExDC_{ground,E,i}$ = External Dose Coefficient, the effective dose rate from the external exposure to radionuclide *i* per unit activity deposited on the ground, Sv·m^2/s·Bq, (values from ICRP 60+ dosimetry models, DCFPAK, 2006), and

CF = Unit Conversion Factor to convert Sv/Bq to mrem/μCi, 3.7E+09 mrem/μCi per Sv/Bq,

$$\frac{3.7E9\, mrem \,/\, \mu Ci}{Sv \,/\, Bq} = \frac{Sv}{Bq} * \frac{10^5\, mrem}{Sv} * \frac{Bq}{dps} * \frac{3.7E4\, dps}{Bq}$$

D.5 Total Dose Parameter for Surface Deposition Calculation

The Total Dose Parameter for Surface Deposition (90_S) is the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground.

$$\mathrm{TDP\_Dp}_{E,i,\mathrm{TP}} = \mathrm{EDP}_{\mathrm{inh},E,TP,\mathrm{i}} + \mathrm{ExDP\_Dp}_{ground,E,i,TP},$$

$$\frac{mrem \cdot m^2}{\mu Ci} = \frac{mrem \cdot m^2}{\mu Ci} + \frac{mrem \cdot m^2}{\mu Ci}$$

Where:

$TDP\_Dp_{E, i, TP}$ = Total Dose Parameter for Surface Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground, mrem·m^2/μCi,

$EDP_{inh, E, i, TP}$ = Effective Dose Parameter, the committed effective dose received from the inhalation of the resuspended radionuclide *i* over the time phase under consideration and per unit of radioactivity of radionuclide *i* deposited on the ground, mrem·m^2/μCi, and

$ExDP\_Dp_{ground, E, i, TP}$ = External Dose Parameter for Deposition, the groundshine dose received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground and adjusted for the ground roughness factor, mrem·m^2/μCi.

D.5.1 Deposition Total Dose Parameter Calculation

The Deposition Total Dose Parameter for Deposition (Dp_TDP_Dp) is the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, from the deposited radioactivity level of all parent radionuclide(s) and any short-lived daughter radionuclides.

$$\mathrm{Dp\_TDP\_Dp}_{E,i,TP} = \sum_{i}^{P+D} \left(Dp_i * \mathrm{TDP\_Dp}_{E,i,TP} \right)$$

Where:

$\sum_{i}^{P+D}$ = Represents the summation of values from all parent (P) and short-lived daughter (D) radionuclide(s),

$Dp\_TDP\_Dp_{E,\,i,\,TP}$ = Deposition Total Dose Parameter for Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, from the deposited radioactivity level of all parent radionuclide(s) and any short-lived daughters, mrem,

Dp_i = Deposition, the radioactivity level of radionuclide *i* per unit area of ground, $\mu Ci/m^2$, and

$TDP\_Dp_{E,\,i,\,TP}$ = Total Dose Parameter for Surface Deposition, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground, $mrem \cdot m^2/\mu Ci$.

D.6 Calculating Total Dose Parameter for Air Concentration (TDP-χ) Values

The total dose parameter for air concentration (TDP-χ) values for radionuclides, are provided in units of mrem per $\mu Ci/m^3$. TDP-χ values include the effective dose from groundshine and the committed effective dose from the inhalation of resuspended material. TDP-χ values are provided for the EPA time phases (i.e., early, first year, second year). As described below, TDP_χ values can be used to estimate the TDE that an adult receptor would receive over each of the three time phases, based on the air concentration measurements made after the plume has passed and the material has been deposited on the surface.

The methods described below are based on the air sample(s) being taken at 4 hours after deposition and are designed to estimate TDE values. This method will provide reasonable TDE estimates when the air sample is taken approximately at 4 hours (e.g., 1-10 hours) post deposition and the radionuclide has a reasonably long half-life (e.g., >1 day). If more accurate estimates are desired, then the method below should be modified to consider the actual time the air sample was collected.

D.6.1 Method Used to Calculate TDP_χ Values

The following method is used to calculate the TDP_χ.

Step 1: Estimate the Ground Deposition Based on the Air Sample

The ground concentration is estimated using the resuspension factor (K) which is described in Section 3.5.2.3. The method below is used to estimate K at 4 hours.

$$K_T = \frac{1E-06\,m^{-1}}{T_d} = \frac{1E-06\,m^{-1}}{4\,h \,/\, 24\,h/d} = 6.0E-06\,m^{-1}$$

Where:

K_T = Resuspension Factor at 0.17 d post deposition and

T_d = Time post deposition, 0.17 d

Step 2: Estimate the Ground Concentration

The method below is used to estimate the ground concentration that would produce an air concentration (χ) of 1 pCi/m^3.

$K_T = \frac{\chi_i}{Dp_i}$ Rearranging the terms to solve for Dp_i,

$$Dp_i = \frac{\chi_i}{K_t} = \frac{1 pCi / m^3}{6.0E-06 / m} = \frac{1.67E+05 pCi}{m^2}$$

Where:

K_T = Resuspension Factor at 0.17 d post deposition,

Dp_i = Deposition concentration of radionuclide *i*, pCi/m^2, and

χ_i = Air concentration of radionuclide *i*, pCi/m^3.

Therefore based on K, it is assumed that at 4 hours post deposition a deposition concentration of 1.67E+05 pCi/m^2 will produce a unit air concentration of 1 pCi/m^3.

Step 3: Calculate the TDP_χ

The following method is used to calculate the TDP_χ.

$$TDP\_\chi_{E,i,TP} = \frac{Dp_i}{\chi_i} * TDP\_Dp_{E,i,TP} \quad , \quad \frac{mrem}{pCi/m^3} = \frac{\frac{pCi}{m^2}}{pCi/m^3} * \frac{mrem}{pCi/m^2}$$

Where:

$TDP\_\chi_{E, i, TP}$ = Total Dose Parameter for Air Concentration, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i*, and any short-lived daughters, in the air, mrem per pCi/m^3,

Dp_i = Deposition concentration of radionuclide *i*, 1.67E+05 pCi/m^2,

χ_i = Air concentration of radionuclide *i*, pCi/m^3, and

$TDP\_Dp_{i, E, TP}$ = Total Dose Parameter for Surface Deposition, the sum of the effective dose from groundshine and the committed effective dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i* deposited on the ground (See Section 3.5.4 for calculation details), values from Table 3-3 or 3-4 depending on whether or not adjustment for weathering effects is desired, mrem·m^2/μCi.

D.6.2 Example Calculation of Dose Projection using TDP_χ

The following equation can be used to estimate the TDE that an adult receptor would receive over each of the three time phases, based on the air concentration that is measured 4 hours after the material has been deposited.

$$TDE_{i,TP} = \chi_i * TDP\_\chi_{E,i,TP} \quad , \quad mrem = \frac{pCi}{m^3} * \frac{mrem}{pCi/m^3}$$

Where:

$TDE_{i, TP}$ = Total Dose, Effective, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended radionuclide *i*, and any short-lived daughter radionuclides, over the time phase of interest, mrem,

χ_i = Air concentration of radionuclide *i*, pCi/m^3, and

TDP_$\chi_{E, i, TP}$ = Total Dose Parameter for Air Concentration, the sum of the external dose from groundshine and the internal (committed effective) dose from inhalation of resuspended material received, over the time phase under consideration, per unit of radioactivity of radionuclide *i*, and any short-lived daughters, in the air, mrem per pCi/m^3.

$$TDE_{Cs-137,1st-y} = \frac{1.4E+03\,pCi}{m^3} * \frac{7.96mrem}{pCi/m^3} = 11\,144mrem$$

D.6.2.1 Calculating Total Dose Parameters for Air Concentration (TDP_χ) and Derived Response Levels for Air Concentration (DRL_χ)for Mixtures

Complete the following steps if, it is desired to calculate total dose parameters for air concentration (TDP_χ) and derived response levels for air concentration (DRL_χ) for a given mixture.

1. Develop a table for the desired mixture which is similar to Table 3-5a or 3-5b (depending on whether or not adjustment for weathering effects is desired), but include TDP_XR values
2. Calculate the TDE_{mix} values following the method in Section 3.7.4 and replacing the TDP_XR values with the TDP_χ values for the mixture.
3. Calculate the DRL_{mix} values following the method in Section 3.7.5 and replacing the TDP_XR values for the TDP_χ values for the mixture.

Appendix E: Risks to Health from Radiation Doses that May Result From Radiological or Nuclear Incidents

E.1 Introduction

This appendix reviews the risks from radiation that form the basis for the choice of PAGs for the response to a radiological or nuclear incident, as well as the choice of limits for occupational exposure during a radiological or nuclear incident.

E.1.1 Units of Dose

The objective of protective action is to reduce the risk to health from exposure to radiation. Ideally, one would like to assure the same level of protection for each member of the population. However, protective actions cannot take into account individual variations in radiosensitivity since these are not known. Therefore, these PAGs are based on assumed average values of risk. EPA further assumes that these risks are proportional to the dose for any level of dose below the threshold for acute effects (see Section E.2).

The dose from exposure to radioactive materials may be delivered during the period of environmental exposure only (i.e., external gamma radiation), or over a longer period (i.e., inhaled radionuclides which deposit in body organs). In the latter case, dose is delivered not only at the time of intake from the environment but continues until all of the radioactive material has decayed or is eliminated from the body. Because of the variable time over which such doses may be delivered, the PAGs are expressed in terms of a quantity called the "committed dose." Conceptually, committed dose is the dose delivered over an individual's remaining lifetime following an intake of radioactive material. However, due to differences in physiology and remaining years of life, the committed dose to a child from internal radioactivity may differ from that to an adult. It should be noted that absorbed dose rather than committed dose is used to estimate acute health effects.

Another important consideration is that different parts of the body are at different risk from the same dose. Since the objective of protective actions is the reduction of health risk, it is appropriate to use a quantity called "effective dose." Effective dose is the sum of the products of the dose to each organ or tissue of the body and a weighting factor representing the relative risk. These weighting factors (ICRP 1991) are chosen as the ratio of mortality (from delayed health effects) from irradiation of particular organs or tissues to the total risk of such mortality when the whole body is irradiated uniformly at the same dose.

Finally, doses from different types of radiation (i.e. alpha, beta, gamma, and neutron radiation) have different biological effectiveness. In radiation protection, these differences are customarily accounted for by multiplicative modifying factors. A dose modified by these factors is designated the “dose equivalent.” The PAG levels are therefore expressed in terms of committed effective dose equivalent. The PAGs are augmented by limits for a few specific organs (skin and thyroid) which exhibit special sensitivity. These are expressed in terms of committed dose equivalent (rem). In the process of developing PAG values, it is necessary to evaluate the threshold dose levels for acute health effects. These levels are generally expressed in terms of absorbed dose (rad) to the whole body from short term (1 month or less) exposure. Other units (Roentgens, rem, and rets) are also used in information cited from various references. They are all approximately numerically equivalent to rads in terms of the risk of acute health effects from beta and gamma radiation.

PAGs are intended to apply to all individuals in a population other than workers performing emergency services. However, there may be identifiable groups that have different average sensitivity to radiation or, because of their living situation, will receive higher or lower doses. In addition, some groups may be at greater risk from taking a given protective action. These factors are separately considered, when it is appropriate, in establishing values for the PAGs.

E.1.2 Principles for Establishing Protective Action Guides

The following three principles provide the basis for establishing levels for PAGs:

- Prevent acute effects.

- Reduce risk of chronic effects.

- Require optimization to balance protection with other important factors and ensure that actions taken cause more benefit than harm.

These principles are similar to those set forth by ICRP Publication 40 (ICRP 1984b) and ICRP Publication 80 (ICRP 2000) as the basis for establishing intervention levels for radiological incidents. EPA examines, below, the basis for estimating effects on health for use in applying the first two of these principles.

E.2 Acute Effects

This section provides information relevant to the first principle: avoidance of acute effects on health from radiation.

Acute radiation health effects are those clinically observable effects on health which are manifested within 2 or 3 months after exposure. Their severity depends on the amount of radiation dose that is received. Acute effects do not occur unless the dose is relatively large, and there is generally a level of dose (i.e., threshold) below which an effect is not expected to occur. Acute effects may be classified as severe or nonsevere clinical pathophysiological effects. Severe pathophysiological effects are those which have clinically observable symptoms and may lead to serious disease and death. Other pathophysiological effects, such as hematologic deficiencies, temporary infertility, and chromosome changes, are not considered to be severe, but may be detrimental in varying degrees. Some pathophysiological effects, such as erythema, nonmalignant skin damage, loss of appetite, nausea, fatigue, and diarrhea, when associated with whole body gamma or neutron exposure, are prodromal (forewarning of more serious pathophysiological effects, including death).

E.2.1 Review of Acute Effects

This section summarizes the results of a literature survey of reports of acute effects from short-term (taken as received in 1 month or less) radiation exposure in some detail. Many reports of observed effects at lower doses differ, and some are contradictory; however, most have been included for the sake of completeness. The results of the detailed review described in this section are summarized in Section E.2.2.

The biological response to the rapid delivery of large radiation doses to man has been studied since the end of World War II. Dose-response relationships for prodromal (forewarning) symptoms and for death within 60 days have been developed from data on the Japanese A-bomb survivors, Marshall Island natives exposed to fallout, and patients undergoing radiotherapy. This work has been supplemented by a number of animal studies under controlled conditions.

The animal studies, usually using lethality as the end point, show that many factors can influence the degree of response. The rate at which the dose is delivered can affect the median lethal dose (LD_{50}) in many species, particularly at dose rates less than 5 R/min (Page 1968, Bateman 1968). However, in primates there is less than a 50% increase in the LD_{50} as dose rates are decreased from 50 R/min to about 0.01 R/min (Page 1968). There is good evidence of species specificity (Page 1968, Bond 1969). The LD_{50} ranges from about 100 rad for burros to over 1,000 rad for lagomorphs (i.e., rabbits). Response is modulated by: age (Casarett 1968), extent of shielding (partial body irradiation) (Bond et al. 1965), radiation quality (Page 1968, Bond 1969), diet, and state of health (Casarett 1968).

While animal studies provide support and supplemental information, they cannot be used to infer the response for man. This lack of comparability of man and animals had already been noted by a review committee for the National Academy of Sciences as early as 1956, in considering the length of time over which acute effects might be expressed (NAS 1956): "Thus, an LD_{50}, 30-day consideration is inadequate to characterize the acute lethal dose response of man, and an LD_{50}, 60 days would be preferable."[21]

[21] The committee (known as the BEAR Committee) also noted "the reservation must be made here that the exposed Japanese population was heterogeneous with respect to age, sex, physical condition and degree of added trauma from burns or blast. The extent to which these factors affected the survival time has not been determined. In studies on laboratory animals the converse is true - homogeneous populations are studied" (NA-56, p.I-6).

Several estimates of the levels at which acute effects of radiation occur in man have been published. For example, an early estimate of the dose-response curves for prodromal (forewarning) symptoms and for lethality was made in the first edition of "The Effects of the Nuclear Weapons" (1957) (Glasstone 1957), and a more recent and well documented estimate is given in a NASA publication, "Radiobiological Factors in Manned Space Flight" (Langham 1967).

E.2.1.1 The Median Dose for Lethality

The radiation dose that would cause 50% mortality in 60 days was estimated as 450 R in early reports (NAS 1956; Glasstone 1957, RD 1951). The National Commission on Radiation Protection and Measurements (NCRP) calculated that this would correspond to a midline absorbed dose of 315 rad (NCRP 1974). The ratio of 315 rad to 450 R is 0.70, which is about the estimated ratio of the active marrow dose, in rads, to the tissue kerma in air, in rads (Kerr 1980). The BEAR Committee noted that the customary reference to LD_{50} in animal studies, as if it were a specific property, independent of age, was not justifiable (NAS 1956): "...it is evident, now, that the susceptibility of a whole population is not describable by a single LD_{50}. The published values are usually obtained for young adults and are therefore maximal or nearly maximal for the strain. In attempts to estimate LD_{50} in man, this age dependence should be taken into consideration" (NAS 1956, pp.4-5). They observed that the LD_{50} approximately doubled as rats went from neonates to young adults and then decreased as the animals aged further. Finally, the BEAR Committee concluded: "the situation is complex, and it became evident that it is not possible to extrapolate with confidence from one condition of radiation exposure to another, or from animal data to man" (NAS 1956, p.I-8). Nevertheless, results from animal studies can aid in interpreting the human data that are available.

The NCRP suggested the $LD_{50/60}$ might be 10% to 20% lower for the old, very young, or sick, and somewhat greater for healthy adults of intermediate age (RD 1951). Other estimates of adult $LD_{50/60}$ have ranged from about 300 rad to 243 ± 22 rad. These lower estimates are apparently based on a ratio of air to tissue dose similar to those calculated for midline organs in the body— 0.54 to 0.66 (Kerr 1980, O'Brien and Sanner 1976, Kocher 1981).

A NASA panel examined all patient and accident studies, tried to remove confounding factors, and concluded: "On this basis, it may be assumed that the LD_{50} value of 286 rad obtained by a normal fit to the patient data is the preferred value for healthy man" (Langham 1967). The $LD_{50/60}$ of 286 ± 25 rad (standard deviation) midline absorbed dose and an absorbed dose/air dose ratio of 0.66, suggested by the National Academy of Science (Langham 1967), is probably a reasonable value for healthy males.

2

The most detailed estimates of $LD_{50/60}$ are probably those of Fujita, et al.(Fujita, S., H. Kato and W.J. Schull, *The LD_{50} Associated with Exposure to the Atomic Bombing of Hiroshima and Nagasaki: A Review and Reassessment*, RERF TR 17-87, Radiation Effects Research Foundation, Japan, 1990) who reviewed previous studies and data on atomic bomb early effects. An $LD_{50/60}$ was estimated from data on 7,586 individuals who were within 1,600 m of the hypocenter at the time of the atomic explosion. The analysis used DS65 dosimetry to estimate free-in air [FIA] kerma to tissue and/or bone marrow doses, adjusting for shielding. $LD_{50/60}$ and $LD_{95/60}$ were calculated using complementary loglog, logit, probit, and weighted linear models. Estimates of bone marrow $LD_{50/60}$ ranged from 2.2 to 2.6 Gy (220 to 260 rad). If these estimates were adjusted by 17.5% to account for severely injured who survived the first day but later succumbed to their injuries, the range of bone marrow $LD_{50/60}$s would be 2.7 to 3.1 Gy (270 to 310 rad). The analysis found that estimates of $LD_{50/60}$ were insensitive to the type of model fit but the estimates of $LD_{95/60}$ were "unstable."

In the absence of more complete information, EPA assumes that a value of 300 rad ⌖ 30 rad is a reasonable reflection of current uncertainties for average members of the population.

E.2.1.2 Variation of Response for Lethality

Uncertainty in the dose-response function for acute effects has been expressed in various ways. The slope of the estimated dose-response function has most commonly been estimated on the basis of the percent difference in the LD_{50} and the $LD_{15.9}$ or $LD_{84.1}$ (one standard deviation from the LD_{50}), as was done by NASA (Glasstone 1957). These and other parameters derived in a similar manner describe the uncertainty in the central risk estimate for the dose-response function.

Another means is to use an estimate of upper and lower bounds for the central risk estimate, i.e., the 95% fiducial limits. At any given response point on the dose-response function, for example, the LD_{10}, the dose causing that

response has a 95% probability of lying between the lower and upper bounds of the 95% fiducial limit for that point. To estimate this value, probability analyses were run for each species using data in published reports (Kocher 1981, Taylor et al. 1971). This provided estimates for each species for comparability analyses. The 95% fiducial limits at the LD_{50} response for $LD_{50/30}$ studies averaged ±9% (range -9 to +26%) and for $LD_{50/60}$ studies ±17% (range −20 to +45%). At the LD_{15} response, values were ±16% (range −12 to +50%) for $LD_{15/30}$ data and +26% (range −20 to +65%) for $LD_{15/60}$ data. For the LD_{85} response, values were ±17% (range −36 to +36 %) for the $LD_{85/30}$ data and ±24% (range −46 to +31%) for $LD_{85/60}$ data.

The differences in the magnitude of the fiducial limits are a function of the differences in age, sex, radiation quality, degree of homogeneity of the experimental animals, husbandry, and other factors. The estimates show that the fiducial limits, expressed as a percent of the dose at any response, get greater the farther from the LD_{50} the estimate is made. For the purpose of estimating fiducial limits for humans, the 95% fiducial limits will be considered to be LD_{l5} +15%, LD_{50} +10%, and LD_{85} +15%. Beyond these response levels, the fiducial limits are too uncertain and should not be used.

If the median lethal dose, $LD_{50/60}$, is taken as 300 +30 rad midline absorbed dose, the response to higher and lower doses depends on the degree of biological variation in the exposed population. The NASA panel decided the wide variation in the sensitivity of patients was a reflection of the heterogeneity of the sample and that the variation in sensitivity, the slope of the central estimate of the response function, would be stated in the form of one standard deviation calculated as 58% of the LD_{50}. They further decided the deviation in the patients (58%) was too great, and the standard deviation for "normal" man should be closer to that of dogs and monkeys (18%) (Langham 1967). (The rationale for selecting these species was not given).

Jones attempted to evaluate the hematologic syndrome from mammalian lethality studies using the ratio of dose to LD_{50} dose as an indicator of the steepness of the slope of the dose-response function (Jones 1981). However, he evaluated LD_{50} studies only of species having a rather steep slope, i.e., dogs, monkeys, mice, and swine. He also looked at several different statistical models for dose-response functions and pointed out the problems caused by different models and assumptions, particularly in evaluating the tails of the dose-response function (less than LD_{10} and greater than LD_{50}). Jones recommended using a log-log model, which he felt provided a better fit at low doses (Jones 1981).

Scott and Hahn also evaluated acute effects from mammalian lethality, but suggested using a Weibull model (Scott and Hahn 1980). One of the advantages of the Weibull model is that in addition to developing the dose-response function, it can also be used to develop hazard functions. These hazard functions, if developed using the same model, can be summed to find the joint hazard of several different types of exposure (Scott 1983). This would allow estimation of the total hazard from multiple organ exposures to different types of radiation.

As mentioned earlier, the human median lethal dose is commonly reported in terms of the $LD_{50/60}$. Most laboratory animal median lethal doses are reported in terms of the $LD_{50/30}$. For those cases in which estimates of both $LD_{50/30}$ and $LD_{50/60}$ are available, i.e. the burro (Still et al. 1969), the variation (that is, the slope of the dose-response curve) is greater in the $LD_{50/60}$ study than in the $LD_{50/30}$ study. Both the dog and the monkey data are for $LD_{50/30}$, and so are not appropriate for direct comparison to man.

If an estimate of the deviation is made for data from other studies and species, those where most of the fatalities occur within 30 days (like dogs and monkeys) have standard deviations of from around 20% [swine (X-ray) (Still et al. 1969), dogs (Nachtwey et al. 1966), hamsters (Ainsworth 1965), primates (Macaca) (Dalrymple et al. 1965)] to 30% [swine (Co-60) (Holloway 1968)]. Those in which most deaths occur in 60 d, like man, have deviations from around 20% [sheep (Chambers 1964)] to 40% [goats (Page et al. 1968), burros (Taylor et al. 1971)]. Nachtwey, et al. (Nachtwey et al. 1966) suggested the steepness of the slope of the exposure response curve depends on the inherent variability of the subjects exposed and any variation induced by uncontrolled factors, i.e., temperature, diurnal rhythm, and state of stimulation or arousal. So, while the slope of the response curve for the patients studied by the NASA panel may be unrealistically shallow for normal human populations, there is no reason to think it should be as steep as those for dogs and monkeys.

The average deviation for those species (burros, sheep, and goats) for which the standard deviation of the $LD_{50/60}$ is available has been used as an estimator for man. The mean value is 34 ± 13%. This is only slightly greater than the average value for all physically large animals (swine, burros, sheep, and goats), 32 ±12%.

E.2.1.3 Estimated Lethality vs. Dose for Man

As noted in Section E.2.1.1, dose-response estimates vary for a number of reasons. Some factors affecting estimates for humans are:

- Age: Studies on rats indicate the LD_{50} is minimal for perinatal exposure, rises to maximum around puberty, and then decreases again with increasing age (Casarett 1968). The perinatal LD_{50} is about one-third of that for the healthy young adult rats; that for the geriatric rat is about one-half of that for the young adult rat.
- Sex: Females are slightly more sensitive than males in most species (Casarett 1968).
- Health: Animals in poor health are usually more sensitive than healthy animals (Casarett 1968), unless elevated hematopoietic activity is occurring in healthy animals (Sugahara et al. 1969).

While these and other factors will affect the $LD_{50/60}$ and the response curve for man, there are no numerical data available.

The variation in response at a given dose level increases as the population at risk becomes more heterogeneous and as the length of time over which mortality is expressed increases. In general, larger species show greater variance and longer periods of expression than do small mammals, i.e., rodents. It is likely that the human population would show at least the same amount of variation as do the larger animals, i.e., a coefficient of variation of about one-third.

The degree of variation exhibited in animal studies follows a Gaussian distribution as well as or better than a log normal distribution over that range of mortality where there are reasonable statistics. EPA has assumed here that the functional form of human response is Gaussian. Generally, sample sizes for extreme values (the upper and lower tails of the distribution) are too small to give meaningful results. Therefore, EPA has not projected risks for doses more than two standard deviations from the $LD_{50/60}$. EPA recognizes that estimates of acute effects may not be reliable even beyond one standard deviation for a population containing persons of all ages and states of health. However, in spite of these uncertainties, previous estimates have been made of the acute effects caused by total body exposure to ionizing radiation as a function of the magnitude of the exposure (NCRP 1971; Lushbaugh et al. 1968; Fabrikant 1973; NATO 1973).

Given the large uncertainties in the available data, a median lethal dose value of about 300 rad at the midline, with a standard deviation of 100 rad, may be assumed for planning purposes. Such risk estimates should be assumed to apply only in the interval from 5% to 95% fatality, as shown in Figure E-1. (See also Section E.2.1.4).

Figure E-1 is based on the following values:

Dose (rad)	Percent Fatalities (%)
<140	none[a]
140	5
200	15
300	50
400	85
460	95

[a]The risk of fatality below 140 rad is not necessarily zero; rather, it is indeterminate and likely to remain so. This also applies to prodromal effects below 50 rad.

The estimates of low dose response made above are similar to estimates made by others. For example, estimates that an $LD_{15/60}$ would be around 200 rad and an $LD_{05/60}$ around 140 rad (*Health Effects Models for Nuclear Power Plant Accident Consequence Analysis. Part I: Introduction, Integration, and Summary*, NUREG/CR-4214, Rev. 2, Part I / ITRI-171, USNRC, Washington, 1993). The response at still lower doses, as was mentioned, is indeterminate: there may be some risk, or there may be a threshold.

For moderately severe prodromal (forewarning) effects, EPA believes the dose at which the same percentage of exposed would show effects would be approximately half of that causing fatality. This yields the following results (see also Figure E-1):

Dose (rad)	Percent Affected (%)
50	<2
100	15
150	50
200	85
250	98

Although some incidence of prodromal effects has been observed at doses in the range of 15- 20 rads in patients (C. C. Lushbaugh et al., Clinical Evidence of Dose-Rate Effects in Total-Body Irradiation in Man, pp. 17.1-17.24 in *Dose-Rate in Mammalian Radiation Biology,* CONF-68041, D. G. Brown, R. G. Cragle, and T. R. Noonan, USAEC, Washington, 1968) or even lower, in the upper end of the 0-10 rads range of dose, in the Japanese A-bomb survivors (D.L. Summers and W. J. Slosarik, *Biological Effects of Initial Nuclear Radiation Based on the Japanese Data*, DNA 5427F, Defense Nuclear Agency, Washington, 1980). In such a case, the number of persons affected by such low doses would be quite small (Lushbaugh et al. 1968, Summers 1980, Gilbert 1984) and there is a great uncertainty in interpreting the data. Patients may be abnormally sensitive, so that the dose-response function in patients may represent the lower bound of doses that would show a response in a healthy population (Lushbaugh et al. 1967). The response of Japanese survivors in the low dose ranges is complicated by the blast and thermal exposure that occurred at the same time (Summers and Slosarik 1980). For these reasons, care should be taken in applying estimates of prodromal effects. The prodromal dose-response function listed above is more likely to overestimate the proportion of persons affected than to underestimate it.

n

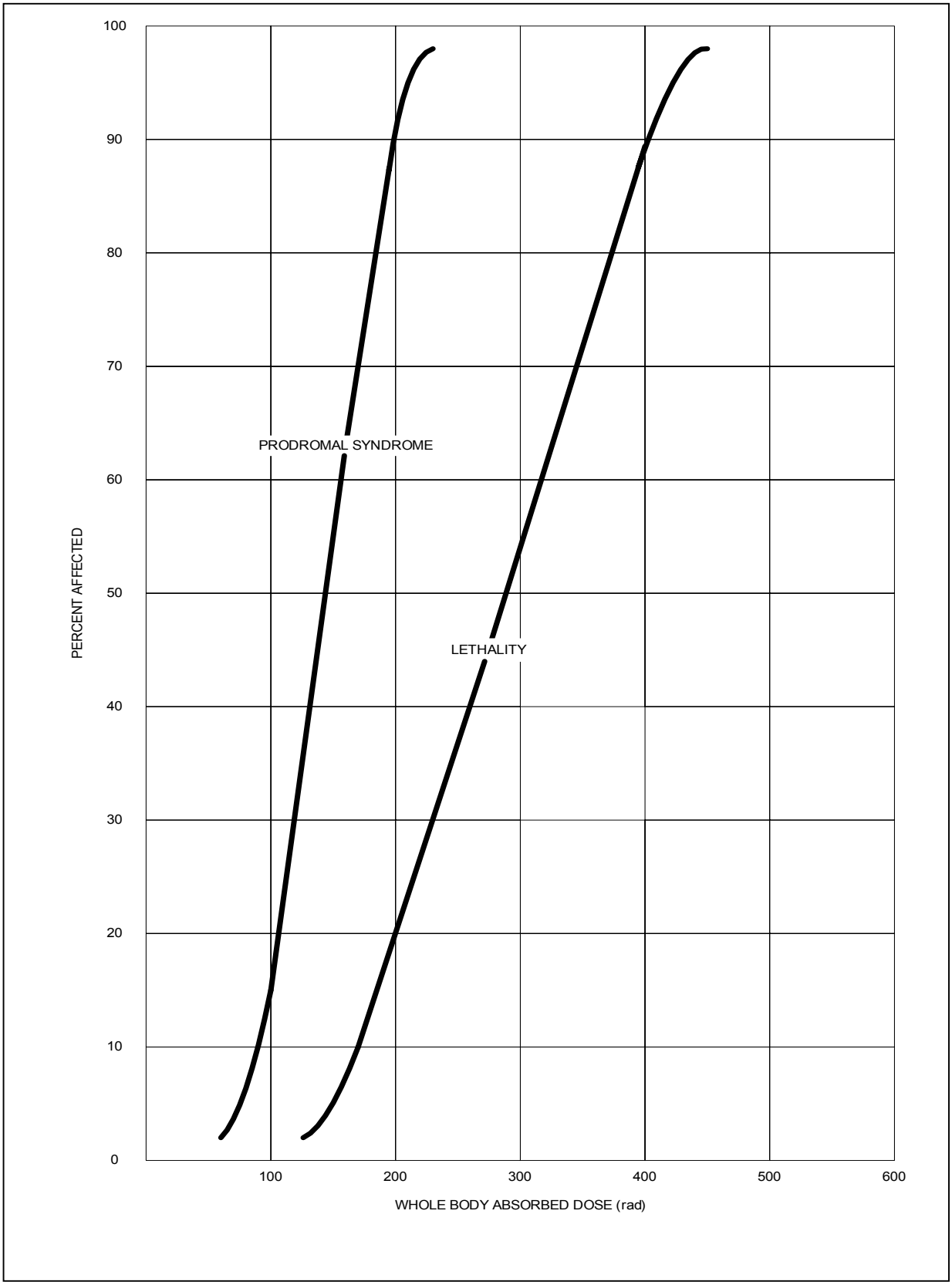

Figure E-1. Acute Health Effects as a Function of Whole Body Dose

These estimated ranges and effects are in agreement with estimates made for manned space flights (Langham 1967, Lushbaugh et al. 1967), which included consideration of the effect of abnormal physiology or sickness in the

patients to which the data apply. Uncertainty in estimates of the biological effects of radiation exposure is great. It is probably due in part to variation in the health of individuals in exposed populations. These estimates assume a healthy young adult population and may not be a conservative estimate of risk for other population groups, such as children or the elderly. Lushbaugh, et al. (Lushbaugh et al. 1968) found that prodromal effects probably occur in both healthy and ill persons in about the same dose range. However, Lushbaugh et al. (Lushbaugh et al. 1968) and NATO (NATO 1973) suggest that acute mortality in a population which is ill, injured, or in other ways debilitated will occur in 50% of that population at doses of 200-250 rad in about 60 d ($LD_{50/60}$), in contrast to an $LD_{50/60}$ from doses of 230-310 rad for a healthy young adult population. Thus, the ill or injured are assumed to have an increased risk of acute mortality at high doses.

The above estimates for $LD_{50/60}$ are also based on the assumption of minimal medical care following exposure. UNSCEAR (UN 1988) estimates that the threshold for mortality would be about 50% higher in the presence of more intense medical care.

E.2.1.4 Threshold Dose Levels for Acute Effects

This section summarizes information available in the literature regarding the thresholds for health effects. It also reviews actions that have been taken as a result of radiation exposure to provide insight on dose levels at which actions to avoid dose may be appropriate.

Some acute effects, such as cellular changes, may occur at low doses with no dose threshold. Most such effects have a minimum threshold of detectability; for example, 5 rad is about the lower limit of whole body dose which causes a cellular effect detectable by chromosome or other special analyses (NCRP 1971; Fabrikant 1973). This value is recommended by UNSCEAR as the starting point for biological dosimetry (UN 1969). Purrott, et al. have reported a lower limit of detection of chromosome aberrations of 4 rad for X-rays and 10 rad for gamma rays (Purrott et al. 1975).

More recent advanced chromosome banding techniques permit detection of increased incidence of chromosome abnormalities from continuous exposure to systematically deposited radioisotopes or radioisotopes deposited in the lung at very low levels, i.e., body burdens of 100 to 1,200 pCi of Pu-239 (Brandom 1977). While the exact dose associated with such burdens is not known, it is probably on the order of 10 to 100 mrem per year. Lymphocytes exposed to 5 rem in-vitro show severe metabolic dysfunction and interphase cell death (Stefani 1964). The extent to which similar effects occur after in vivo exposure is unknown. While chromosome abnormalities in circulating lymphocytes are reported to persist for long periods (UN 1969), the significance of such abnormalities is not known (Brandom 1977).

Transient, dose dependent abnormalities in encephalographic patterns have been noted immediately after brain doses as low as 1 rad; but they are considered of little clinical importance (Medical Effects of Ionizing Radiation, 2nd edition, F.A. Mettler, Jr. and A. C. Upton, W.B. Saunders Company, Philadelphia, 1995). The cause of these effects is unknown.

Hug has suggested 5 rem as the lower limit of exposure which might produce acute effects (WHO 1965). 5 rad is also in the low dose, short-term exposure range defined by Cronkite and Haley, and is below the 10 rad which they thought would cause only a slight detectable physiological effect of unknown clinical significance (Cronkite and Haley 1971).

Although the ICRP has suggested that annual doses of 15 rad would not impair the fertility of normal fertile men (ICRP 1969), an acute dose of 15 rad causes “moderate” oligospermia (approximately 70% reduction in sperm count) that lasts for some months (Langham 1967). Popsecu and Lancranjan reported alterations of spermatogenesis and impaired fertility in men exposed to from 500 mrad to 3 rad per year for periods of varying from 2-22 years (Pompescu and Lancranjan 1975). The shortest exposure period in which abnormal spermatogenesis was reported was 31 to 41 months (Pompescu and Lancranjan 1975); at the highest dose rate reported (3 rad/a), this is a cumulative dose of 8 to 10 rem. While more study is required, these results suggest the need to restrict acute doses to below 10 rem to avoid this effect, because a given acute dose is anticipated to be more effective than the same cumulative dose given over a longer period of time (NAS 1956; UN 1958). Doses of 150-650 rad to the ovary can cause temporary sterility or reduced fertility. (Medical Effects of Ionizing Radiation, 2nd edition, F.A. Mettler, Jr. and A. C. Upton, W.B. Saunders Company, Philadelphia, 1995).

Many observations have indicated that doses of 10 rem or more to the pregnant woman are hazardous to the fetus. Mental retardation due to exposure of the fetus is discussed in Section E.3; this discussion is restricted to acute effects. The World Health Organization (WHO) indicates that there is no evidence of teratogenic effects from short term exposure of the fetus to a dose less than 10 rad during the early gestation, the period when the fetus is most sensitive to these effects (WHO 1984).

A number of authorities have recommended that exposures of 10 R or higher be considered as an indication for carrying out induced abortion (Hammer 1959, Devick 1970, Brent and Gorson 1972). Brent and Gorson also suggest that 10 rad is a "practical" threshold for induction of fetal abnormalities (Brent and Gorson 1972). The Swedish Government Committee on Urban Siting of Nuclear Power Stations stated the situation as follows: "What we have called unconditional indication of abortion involves the exposure of pregnant women where radiation dose to the fetus is higher than 10 rad. When such doses are received in connection with medical treatment, it has hitherto been assumed that the probability of damage to the fetus is so high that an abortion is recommended. The probability for such injury is still moderate compared with the normal frequency of similar fetal injuries, and the probability is particularly reduced when the dose is received late in the pregnancy" (SOU 1974).

E.2.1.5 Acute Effects in the Thyroid

Acute effects are produced in the thyroid by doses from radioiodine on the order of 3,000 to 100,000 rad. Ablation of the thyroid requires doses of 100,000 rad (Beirwalter and Wagner 1968). The thyroid can be rendered hypothyroid by doses of about 3,000 to 10,000 rad (ICRP 1971). A thyroid dose from radioiodines of 1000 rad in adults and 400 rad in children implies an associated whole body dose of about 1 rad due to radioiodines circulating in the blood. Following inhalation of I-131, the committed thyroid dose is about 1 rad/pCi intake of I-131 in adults. In the developing fetus, the thyroid dose ranges from 1 to 6 rad per pCi of I-131 entering the mother's body (Il'in et al. 1974).

Although acute clinical effects are only observed at high doses, subclinical acute thyroid radiation effects may occur at lower doses (Doniach 1972). Impaired thyroid capability may occur above a threshold of about 200 rad (Doniach 1972). A dose dependent excess of thyroid disease [nontoxic nodular goiter, diffuse goiter, thyrotoxicosis, chronic lymphocytic thyroiditis and hypothyroidism] was reported in atomic bomb survivors who were under age 30 at the time of exposure [relative risk 1.24 at 100 rad] and doubling of autoimmune thyroid disease was observed in women following repeated chest fluoroscopic examinations [thyroid dose between 11 and 112 rad] (Mettler and Upton, 1995).

Effects of radiation exposure of the thyroid have been shown in animal experiments. Walinder and Sjoden found that doses in excess of 3,000 rad from I-131 caused noticeable depression of fetal and juvenile mouse thyroid development (Walinder and Sjoden 1969). Moore and Calvin, working with the Chinese hamster, showed that an exposure as low as 10 R (X-rays) would give rise to 3% aberrant cells when the thyroid was cultured (Moore and Calvin 1968). While the direct relationship of these results to human effects is not certain, mammalian thyroid cells can be injured at exposures as low as 10 R.

E.2.1.6 Acute Effects in the Skin

The first stage of skin reaction to radiation exposure is erythema (reddening) and epilation (loss of hair) with a reported threshold of from 500 to 800 rad; however, these effects have been observed to occur at doses as low as 200 to 300 rad (AAPM 2001) Acute exudative radiodermatitis results from doses of 1200 to 2000 rad (WHO 1984) and dermal necrosis after doses above 1,500 rad (AAPM 2001, Mettler and Upton 1995)

E.2.1.7 Clinical Pathophysiological Effects

A large amount of anecdotal information is available on the injury of organ tissues by high doses of radiation. Acute injury to tissue includes swelling and vacuolation of the cells which make up the blood vessels, increased permeability of vessels to fluids so that exudates form, formation of fibrin clots and thrombifribrinoid thickening in the walls of blood vessels, and the swelling and vacuolization of parenchymal cells. In summary, there is an initial exudative reaction followed in time by fibrosis and sclerosis (White 1976, Casarett 1976, Mettler and Upton 1995).

Estimates of radiation doses necessary to cause severe tissue response in various organs are given in Table E-1. These tissue dose estimates are based on response to radiotherapy treatment, which is normally given on a fractionated dose basis, but also may be given as a continuous exposure. Therefore, these estimates must be adjusted

to the equivalent single radiation dose for use in the present analysis. The formalism of Kirk, et al. (Kirk et al. 1971) is used to estimate the equivalent dose for a single acute exposure in rad-equivalent therapy units (rets: the dose calculated from the fractionated exposure which is equivalent to a single acute exposure for a specific biological endpoint). Table E-2 lists acute exposure equivalents in rets for various organs.

With the exception of bone marrow, the exposures required to cause 5% injury within 5 years (TD 5/5) in internal organs are in the range of 1,000 to 5,000 rad. Since, with this type of injury, the dose response is nonlinear and has a threshold (i.e., is nonstochastic), there is an exposure below which injury is not expected. If the shape of the injury dose-response curve is the same for all internal organs as it is for lungs, plotting the two acute exposure equivalents (TD 50/5 and 5/5) for each organ on log probability paper allows a crude estimation of the number of clinical pathophysiological effects per 1000 persons exposed as a function of dose level. If one acute effect per 1,000 persons within 5 years (TD 0.1/5) is taken as the threshold for the initiation of clinical pathophysiological effects in organs other than the thyroid, the equivalent dose level for most organs is 550 rets or more; testes 440±150 rets, ovary 170±70 rets, and bone marrow 165 rets.

Table E-1. Radiation Doses Causing Acute Injury to Organs (Rubin and Casarett 1972, Rubin and Casarett 1973, Mettler and Upton 1995)

Organ	Volume or Area of Exposure[a]	Risk of Injury in 5 Years		Type of Injury
		5% (rad)	50% (rad)	
Bone marrow	whole	250	450	aplasia and pancytopenia
	localized	3000	4000	
Liver	segment	3000	4000	acute and chronic hepatitis
	whole	2500	4000	
Stomach	100cm^2	4500	5500	ulcer, perforation, hemorrhage
Intestine	400cm^2	4500	5500	ulcer, perforation, hemorrhage
	100cm^2	5000	6500	
Lung	whole	1500	2500	acute and chronic puemonitis
	100cm^2	3000	3500	
	lobe	4000	6000	
Kidney	whole	2000	2500	acute and chronic nephrosclerosis
Brain	whole	5000	>6000	infarction, necrosis
Spinal cord	10 cm	4500	>5500	infarction, necrosis
Heart	60%	4500	5500	pericarditis and pancarditis
Skin	100cm^2	5500	7,000	ulcers, fibrosis
Fetus	whole	200	400	death
Lens of eye	whole	500	1200	cataracts
Ovary	whole	200-300	625-1200	permanent sterilization
Testes	whole	500-1500	2000	permanent sterilization

[a]Dose delivered in 200-rad fractions, 5 fractions/week.

The radiation exposure to organs in rad units that will cause clinical pathophysiological effects within 5 years to 0.1% of the exposed population as a function of the duration of a continuous level of exposure can then be estimated using Goitein's modification of the Kirk methodology (Gortein 1976). This relationship is shown in Table E-3.

Bone marrow is an organ of particular concern because radionuclides known to concentrate in this organ system occur in radiological and nuclear incidents. The acute lethality due to hematologic syndrome (Langham 1967) is estimated to occur in the range of 200 to 1,000 rad, so that the difference is small between exposure levels that might cause acute lethality and exposure levels that might cause only "severe clinical pathophysiology," as derived from radiotherapy data.

Table E-2. Acute Radiation Exposure as a Function of Rad Equivalent Therapy Units (rets)

Organ	Volume or Area of Exposure	Risk of Injury in 5 Years	
		5%(rets)	50% (rets)
Bone marrow	whole	230	340
	segment	1135	1360
Liver	whole	1000	1360
Stomach	$100cm^2$	1465	1665
Intestine	$400cm^2$	1465	1665
	$100cm^2$	1570	1855
Lung	whole	720	1000
	$100cm^2$	1135	1245
	75%	770[b]	---
Kidney	whole	875	1000
Brain	whole	1770	1950
Spinal cord	10cm	1465	1665
Heart	60%	1465	1665
Skin	---	1665	1950
Fetus	whole	200	315
Lens of eye	whole	355	620
Ovary	whole	200-430[a]	410-875[a]
Testes	whole (sterilization)	340-720[a]	410-875[a]

[a]For a 200-rad/treatment, 5 treatments/week schedule (Lushbaugh and Casarett 1976).
[b]Reference WA 1973.
---Unspecified.

Table E-3. Radiation Exposure to Organs Estimated to Cause Clinical Pathophysiological Effects Within 5 years to 0.1% of the Exposed Population (G)-76)

Duration of Exposure (days)	Ovary	Bone marrow (rad)	Testes (rad)	Other organs (rad)
(acute)	(170 rets)[a]	(165 rets)	(440 rets)	(550 rets)
1	315	300	810	1020
2	390	380	1010	1260
4	470	450	1210	1510
7	550	540	1430	1790
30	840	820	2190	2740
365[b]	1740	1690	4510	5640

a The dose in rets is numerically equal to the dose in rads.
b Assuming tissue recovery can continue at the same rates as observed during 30-60-day therapeutic exposure courses.

In summary, organ systems are not expected to show symptoms of severe clinical pathophysiology for projected short-term exposure doses less than a few hundred rad. Projected doses to bone marrow at this high level are relatively more serious and more likely to result in injury than doses to other organ systems.

Even if severe clinical pathophysiological effects can be avoided, there is still a possibility of clinical pathophysiological effects of a less severe or transitory nature. The 1982 UNSCEAR report (UN 1982) reviewed much of the data on animals and man. In the animal studies, there were reports of changes in stomach acid secretion and stomach emptying at 50 to 130 rad; stunting in growing animals at the rate of 3- 5% per 100 rad; degeneration

of some cells or functions in the brain at 100 rad, particularly in growing animals; temporary changes in weight of hematopoietic tissues at 40 rad; and more damage in ovaries and testes caused by fractionated doses rather than acute doses. Some of the effects are transitory, others are long-lasting, but with only minor reductions in functional capacity.

Human data are limited and are reported primarily in the radiotherapy literature. The data suggest most tissues in man are more radiation resistant than those in animals. However, the human hematpoietic system shows a transient response, reflected by decreased circulating white cells and platelets, at about 50 rad. Temporary sterility has been observed after doses of 150 rad to the ovaries and 10 rad to the testes when given as fractionated doses.

There is not sufficient data to determine dose-response functions or to describe the duration and severity of dysfunction expected.

E.2.2 Summary and Conclusions Regarding Acute Effects

Based on the foregoing review of acute health effects and other biological effects from large doses delivered over short periods of time, the following whole body doses from acute exposure provide useful reference levels for decision making for PAGs:

- 50 rad - Less than 2% of the exposed population would be expected to exhibit prodromal (forewarning) symptoms.
- 25 rad - Below the dose where prodromal symptoms have been observed.
- 10 rad - The dose level below which a fetus would not be expected to suffer teratogenesis (but see Section E.3, Mental Retardation).
- 5 rad - The approximate minimum level of detectability for acute cellular effects using the most sensitive methods. Although these are not severe pathophysiological effects they may be detrimental.

Based on the first principle to be satisfied by PAGs (paragraph E.1.6), which calls for avoiding acute health effects, values of 50 rem for adults and 10 rem for fetuses appear to represent upper bounds.

E.3 Mental Retardation

Brain damage to the unborn is a class of injury reported in atomic bomb survivors which does not fall into either an acute or delayed effect category, but exhibits elements of both. What has been observed is a significant, dose-related increase in the incidence and severity of mental retardation, microencephaly (small head size), and microcephaly (small brain size) in Japanese exposed to radiation in-utero during the eighth to 15th week after conception (Blot and Miller 1973, Miller and Mulvihill 1976). While the actual injury may be acute, it is not identified until some time after birth.

In an early study, Mole (Mole 1982) suggested that, although radiation may not be the sole cause of these conditions, it is prudent to treat the phenomenon as radiation-related. More recently, Otake and Schull (Otake and Schull 1983) have concluded: (1) there is no risk of mental retardation to live-born due to doses delivered up to 8 weeks after conception; (2) most damage occurs at the time when rapid proliferation of neuronal elements occurs, i.e., 8 to 15 weeks of gestational age; (3) the dose-response function for incidence during this period appears to fit a linear model; (4) the risk of occurrence is about five times (5x) greater during the period 8-15 weeks of gestation than in subsequent weeks; and (5) in later stages of gestation, i.e., after the 15^{th} week, a threshold for damage may exist.

In their published reports, Otake and Schull (Otake and Schull 1983) evaluated the incidence of severe mental retardation using the T-65 dosimetry and the dosimetry estimates developed in the ongoing dose reassessment program for the atomic bomb survivors, and using 2 tissue dose models. Their estimated ranges of risk were:

- 8 to 15 weeks after gestation: $3\text{-}4 \times 10^{-3}$ cases/rad

- 16 or more weeks after gestation: 5–7x10^{-4} cases/rad

The higher values are based on the T-65 dosimetry and the Oak Ridge National Laboratory estimate of tissue dose. The lower values are based on Oak Ridge National Laboratory dosimetry and the Japanese National Institute of Radiological Sciences estimates of tissue dose. Later estimates based on the dose reassessment completed in 1986 are consistent with these published results (Schull 1987).

More recent reviews of the data have not found any compelling reason to change the estimates (BEIR V 1990, USCEAR 1993, ICRP 2003). However, there are analyses reporting the dose response as linear with a threshold in the range of 31-61 rem for the 8-15 week group and 28-107 rem for the 16-25 week group of gestational exposures (ICRP 2003). The 1993 UNSCEAR Report pointed out the extensive sources of uncertainties in the estimates; only 18 cases with exposures above 100 rem, absence of individual in-utero dose estimates, uncertainty in gestational age estimates for time of exposure, etc. Even with these limitations, an increased incidence of 0.22 per 100 rem in unprovoked seizures could be estimated using a linear dose response model with a threshold whose lower limit of uncertainty included zero (UNSCEAR 1993). If similar mechanisms induce both mental retardation and unprovoked seizures, then mental retardation may also be non-threshold.

Federal Radiation Protection Guidance, adopted in 1987, recommends that dose to occupationally exposed pregnant women be controlled to keep the fetal dose below 0.5 rem over the entire term of pregnancy and that no dose be delivered at more than the uniform monthly rate that would satisfy this limit (i.e., approximately 50-60 mrem/month) (EPA 1987). The NCRP has, for many years, recommended a limit of 0.5 rem (NCRP 1971). ICRP recommends controlling exposure of the fetus to less than 0.5 rem in the first two months to provide appropriate protection during the essential period of organogenesis (ICRP 1977).

E.4 Delayed Health Effects

This section addresses information relevant to the second principle (paragraph E.1.5) for establishing PAGs, the risk of delayed health effects in exposed individuals. The following subsections summarize the estimated risks of cancer and genetic effects, the two types of delayed effects caused by exposure to radiation.

E.4.1 Cancer

Because the effects of radiation on human health have been more extensively studied than the effects of many other environmental pollutants, it is possible to make numerical estimates of the risk as a result of a particular dose of radiation. Such estimates, may, however, give an unwarranted aura of certainty to estimated radiation risks. Compared to the baseline incidence of cancer and genetic defects, radiogenic cancer and genetic defects do not occur very frequently. Even in heavily irradiated populations, the number of cancers and genetic defects resulting from radiation is known with only limited accuracy. In addition, all members of existing exposed populations have not been followed for their full lifetimes, so data on the ultimate numbers of effects are not yet available. Moreover, when considered in light of information gained from experiments with animals and from various theories of carcinogenesis and mutagenesis, the observed data on the effects of human exposure are subject to a number of interpretations. This, in turn, leads to differing estimates of radiation risks by individual scientists and expert groups. In summary, the estimation of radiation risks is not a fully mature science and the evaluation of radiation hazards will continue to change as additional information becomes available.

Most of the observations of radiation-induced carcinogenesis in humans are on groups exposed to low-LET radiations. These groups include the Japanese A-bomb survivors and medical patients treated with X-rays for ankylosing spondylitis in England from 1935 to 1954 (Smith and Doll 1978). The National Academy of Science Committee on the Biological Effects of Ionizing Radiations (BEIR) (NAS 1980) and UNSCEAR (CBEIR 1990, CBEIR 2005, UN 1994, UN 2000, UN 2001, UN 1977) have provided knowledgeable and exhaustive reviews of these and other data on the carcinogenic effects of human exposures. The most recent of the BEIR studies was published in 2005 and is here designated BEIR-7 to distinguish it from previous reports of the BEIR committee.

The most important epidemiological data on radiogenic cancer is that from the A-bomb survivors. The Japanese A-bomb survivors have been studied for more than 40 years, and most of them have been followed in a major, carefully planned and monitored epidemiological survey, the Life Span Study Sample, since 1950 (Kato and Schull 1982, Wakabayashi 1983, Shimizu et al. 1989, Shimizu Kato and Schull 1990). The survivors were exposed to a

wide range of doses and are the largest group that has been studied. They are virtually the only group providing extensive information on the response pattern at various levels of exposure to low-LET radiation.

However, the numerical risk estimates used in this document are derived from the EPA Report on cancer risk models (EPA 1994) with some changes made in Federal Guidance Report No. 13 (EPA 1999). Lifetime risks for intakes of radionuclides calculated using these models have been tabulated in Federal Guidance Report No. 13 (EPA 1994) and its CD Supplement (EPA 2002).

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Age- and gender-specific risk model coefficients used in the EPA Report are summarized for cancers other than leukemia and for leukemia. Risk model coefficients for esophagus, stomach, colon, lung, ovary, bladder, leukemia, and "residual" are based on updated information on the Japanese atomic bomb survivors and are derived using a slightly modified version of a model of Land and Sinclair (Land and Sinclair 1991) The risk model coefficients for these sites are obtained by taking the geometric mean of model coefficients derived from the two equally plausible methods used by Land and Sinclair for transporting risk from one population to another. Both methods assume a constant excess relative risk coefficient beginning 10 years after an exposure and continuing throughout the rest of life for each cancer site, excluding leukemia. One method (multiplicative) assumes that the relative risk estimator is the same across populations. The other (NIH) assumes that the relative risk model coefficients for the target population should yield the same risks as those calculated with the additive risk model coefficients from the original population over the period of epidemiological follow-up, excluding the minimal latency period. These excess relative risk model coefficients are then used to project the risk over the remaining years of life. The data considered in deriving risk model coefficients consisted of cancers observed 10-40 years after exposure for solid tumors and 5-40 years after exposure for leukemia. (Forthcoming risk estimates will include 60 years of followup).

Some modifications in the method of calculation of the NIH model coefficients have been made to remove inconsistencies in the derived coefficients. Some but not all of these changes were made in the EPA report on radiation risk models (EPA 1994) and additional ones in Federal Guidance Report 13 (EPA 1999).

The estimated cancer risk from low-LET, whole body, lifetime exposure presented here is based on a life table analysis using a linear response model. EPA used the geometric mean of relative and absolute risk projections (EPA 1999) for solid cancers and an absolute risk projection for thyroid, skin and bone cancer. For whole body dose, this yields an estimated 575 (with a possible range from a factor of two or three three higher to a factor of two or three lower) fatalities per million person-rem for a population cohort representative of the 1990 U.S. population. In like manner, EPA estimates 846 total cancers (fatal and non-fatal) per million person-rem in the cohort. EPA assumes this estimate also applies to high-LET radiation (e.g., alpha emitters); no reduction has been applied for dose rate. Preliminary calculations using BEIR-7 models suggest cancer mortality risk will increase slightly to 583 fatalities per million person-rem and morbidity much more to 1,160 cases per million person-rem. While the numerical estimate of fatal cancers appears similar to the current value, risks for individual organs may be appreciably different in the two models. The extent of such differences has not yet been explored.

Whole body dose means a uniform dose to every organ in the body. In practice, such exposure situations seldom occur, particularly for ingested or inhaled radioactivity. Inhaled radioactive particulate materials may be either soluble or insoluble. Soluble particulate materials deposited in the lung will be rapidly absorbed – 17 to 18% of type M particles and 46 to 47% of type F particles – and the radionuclides associated with them distributed throughout the body by the bloodstream. As these radionuclides are transported in the blood, they irradiate the entire body. Usually, they then redeposit in 1 or more organs, causing increased irradiation of that organ. Insoluble particulate materials, type S, on the other hand, are only partially absorbed into body fluids. (This fraction is typically assumed to be about 2.5%). This absorption occurs over a period of years, with a portion entering the bloodstream and another retained in the pulmonary lymph nodes. The balance (97.5%) of inhaled insoluble particulate materials are removed from the lung within a few days by passing up the air passages to the pharynx where they are swallowed. Inhaled insoluble particulate materials thus irradiate both the lung and the gastrointestinal tract, with a small fraction being eventually absorbed into the bloodstream (Legget and Eckerman 2003). These nonuniform distributions of dose (and therefore risk) are taken into account through use of the weighting factors for calculating effective dose.

There is a latent period associated with the onset of radiation-induced cancers, so the increased risk is not immediately apparent. The increased risk is assumed to commence 2 to 10 years after the time of exposure and continue the remainder of the exposed individual's lifespan (NAS 1980).

For uniform exposure of the whole body, about 66% of radiation-induced cancers in women and about 71% in men are fatal (NAS 1980). Therefore, 1 rem of low-LET radiation would be expected to cause a total of about 850 cancer cases if delivered to a population of 1 million. (In the case of thyroid and skin, the ratio of non-fatal to fatal cancers is much higher. These are addressed separately below). This corresponds to an average annual individual probability of developing cancer of about 11.3×10^{-6} per year. For perspective, the average annual risk of dying of cancer from all causes in the United States for 2001-2003 was about 2.9×10^{-3} and the risk of developing cancer of about 5.5×10^{-3} (based on SEER Results 2003 online).

E.4.1.1 Cancer Risk Due to Radiation Exposure of the Thyroid

Exposure of the thyroid to extremely high levels of radiation may cause it to degenerate. At moderate levels of exposure some loss of thyroid function will occur. At lower levels of exposure, there may be delayed health effects, which take the form of both thyroid nodules and thyroid malignancies (NAS 1972; NAS 1980). Doses as low as 14 rad to the thyroid have been associated with thyroid malignancy in the Marshall Islanders (Conrad et al. 1970). The increased risk of radiation-induced cancer is assumed to commence about 10 years after initial exposure and to continue for the remaining lifespan of an exposed individual.

The true nature of thyroid nodules cannot be established until they are surgically removed and examined histologically, and those that are malignant can lead to death if not surgically removed (Samosan et al. 1968; DeGroot and Paloyan 1973). Although thyroid malignancies are not necessarily fatal, effects requiring surgical removal of the thyroid cannot be considered benign. In this analysis, all thyroid cancers, both fatal and nonfatal, are counted for the purpose of estimating the severity of thyroid exposures.

Based on findings in BEIR-3, EPA estimates that 1 rem of thyroid exposure carries a risk of producing a thyroid cancer of 3.2×10^{-5}, of which a small fraction (on the order of 1 in 10) will be fatal (EPA 1999). Since the calculation of effective dose equivalent does not include consideration of nonfatal thyroid cancers and the severity of the medical procedures for their cure, it is appropriate to limit the dose to the thyroid by an additional factor beyond that provided by the PAG expressed in terms of effective dose equivalent. Protective action to limit dose to thyroid is therefore recommended at a thyroid dose five times (5x) the numerical value of the PAG for effective dose.

E.4.1.2 Cancer Risk Due to Radiation Exposure of the Skin

The risk of fatal skin cancer is estimated to be on the order of 1% of the total risk of fatal cancer for uniform irradiation of the entire body (ICRP 1978), and perhaps as low as 0.1% (EPA 1999). However, since the weighting scheme for calculating effective dose equivalent does not include skin, the PAG expressed in terms of effective dose does not provide protection against radionuclides which primarily expose skin. As in the case of the thyroid, the ratio of nonfatal to fatal cancers from irradiation of the skin is high (on the order of 100 to 1). It would not be appropriate to ignore this high incidence of nonfatal skin cancers by allowing 100 times (100x) as much dose to the skin as to the whole body. For this reason, protective actions are recommended at a skin dose 50 times (50x) the numerical value of the PAG for effective dose.

E.4.1.3 Cancer Risk Due to Radiation Exposure of the Fetus

The fetus is estimated to be five times (5x) to ten times (10x) as sensitive to radiogenic cancer as an adult (Fabrikant 1973; WHO 1965). Stewart reports increased relative incidence of childhood cancers following prenatal X-ray doses as low as 0.20 to 0.25 rem and doubling of childhood cancers between 1-4 rem (Stewart 1973). She concluded that the fetus is about equally sensitive to cancer induction in each trimester. Her findings are supported by similar results reported by MacMahon and Hutchinson (MacMahon and Hutchinson 1964), Kaplan (Kaplan 1958), Polhemus and Kock (Polhemus and Kock 1959), MacMahon (MacMahon 1963), Ford, et al. (Ford et al. 1959), Stewart and Kneale (Stewart and Kneale 1970), and an AEC report (AEC 1961). MacMahon reported that although there were both positive and negative findings, the combination of weighted data indicates a 40% increase in childhood cancer mortality after in vivo exposure to diagnostic X-rays (1.0 to 5.0 rad): about one cancer per 2,000 exposed children in the first 10 years after birth (MacMahon 1963). He concluded that although the range of dose within which these effects are observed is wide, effects will be fewer at 1 rad than at 5 rad.

Graham, et al., investigating diagnostic X-ray exposure, found a significantly increased relative risk of leukemia in children – by a factor of 1.6 following preconception irradiation of mothers or in utero exposure of the fetus; by a

factor of 2 following postnatal irradiation of the children; and by a factor of 2 following preconception irradiation of the mother and in utero exposure of the child (Graham and other 1966).

E.4.1.4 Age Dependence of Doses

Almost all dose models are based on ICRP "Reference Individuals," which adopts the characteristics of male and female adults of working age [25 years of age for bone seeking isotopes; 20 years of age for other isotopes], a 15 year old, a 10-year-old, a 5-year-old, a 1-year-old or an infant [3 months old]. ICRP 30 dosimetric models, which use "Reference Man" as a basis, are appropriate for only adult workers and do not take into account differences in dose resulting from the differences in physiological parameters between children and adults, e.g., intake rates, metabolism, and organ size. The new ICRP 72 dosimetric models used in this document do take these factors into account and can provide age- and sex-specific dose and risk estimates. (4ICRP 1996). Lifetime risk estimates for radionuclides are based on models where the age specific doses are changing throughout the life span reflecting the changing intakes, organ masses, biokinetic models, etc. They are considered reasonable estimates based on data available now.

E.4.2 Genetic Risk

An average parental dose of 1 rem before conception has been estimated to produce five to 75 significant genetically-related disorders per million liveborn offspring (NAS 1980). The BEIR-5 report (NRC 1990) estimates for the same conditions were 6 to 45 significant genetically-related disorders per million liveborn offspring. For this analysis EPA uses the geometric mean of this range, i.e., 1.8×10^{-5}. This estimate applies to effects in the first generation only, as a result of dose to parents of liveborn offspring. The sum of effects over all generations is estimated to be approximately 12 times (12x) greater – that is, 2.2×10^{-4}. In addition, since any radiation dose delivered after a parent's last conception has no genetic effect, and not all members of the population become parents, less than half of the entire dose in an average population is of genetic significance. Taking the above factors into account, EPA estimates that the risk of genetically related disorders in all generations is 1×10^{-4} per person-rem to a typical population.

The BEIR-7 Report (BEIR 2005) treats genetic risk in a different way. First generation risks are estimated to be 0.004 to 0.006% of the baseline frequency of genetic diseases per rem per million progeny and second generation risks, under conditions of continuous radiation exposure, from 0.0053 to 0.0091% of the baseline frequency. The risk estimates for 2005 are 30 to 47 per rem per million progeny, geometric mean 3.7×10^{-5}, about twice the earlier estimates.

Although the overall severity of the genetic effects included as "significant" in the above estimates is not well known, rough judgments can be made. The 1980 BEIR report referred to "...disorders and traits that cause a serious handicap at some time during lifetime" (NAS 1980). From the types of defects reported by Stevenson (Stevenson 1959), it can be estimated that, of all radiation-induced genetic effects, 50% lead to minor to moderate medical problems (i.e., hair or ear anomalies, polydactyl, strabismus, etc.), 25% lead to severe medical problems (e.g., congenital cataracts, diabetes insipidus, deaf mutism, etc.), 23% would require extended hospitalization (e.g., mongolism, pernicious anemia, manic-depressive psychoses, etc.), and 2% would die before age 20 (e.g., anencephalus, hydrocephalus, pancreatic fibrocytic disease, etc.).

E.4.3 Summary of Risks of Delayed Effects

Table E-4 summarizes average lifetime risks of delayed health effects based on results from the above discussion. Because of the nature of the dose-effect relationships assumed for delayed health effects from radiation (linear, nonthreshold), risks are known to decrease to zero or similar value below which no risk can be assumed to exist.

Table E-4. Average Risk of Delayed Health Effects in a Population[a]

	Effects Per Person-rem		
	Whole Body	Thyroid[c]	Skin
Fatal cancers	5.75E-04	1.6 to 3.2 E-06	6E-6 to 6E-04
Nonfatal cancers	2.71E-04[b]	3.20E-05	3.00E-05
Genetic disorders (all generations)	1.00E-04		

[a]EPA assumes a population with the same age distribution as that of the U.S. population in 1990.
[b]Risk to the fetus may be 5 to 10 times (5-100x) higher.
[c]Risk to young children is estimated to be about two to three times (2-3x) as high.

E.4.4 Risks Associated with Other Radiation Standards

A review of radiation standards for protection of members of the general population from radiation shows a range of values spanning several orders of magnitude. This occurs because of the variety of bases (risk, cost, practicability of implementation, and the situations to which they apply) that influenced the choice of these standards. Some source-specific standards are relatively protective. Similarly, regulations under the Clean Air Act limit the dose due to emissions of radionuclides to air alone from all DOE and NRC facilities to 0.01 rem per year, which corresponds to a cancer risk of 2×10^{-4} for lifetime exposure (risk estimate using 2006 risk coefficients and vital statistics -4.22×10^{-4}).

Federal Radiation Protection Guidance for nonemergency situations recommends that the dose from all sources combined (except from exposure to medical and natural background radiation) to individuals in the population not exceed 0.5 rem in a single year (FRC 1960) and that the dose to the fetus of occupationally exposed mothers not exceed 0.5 rem during the 9-month gestation period (EPA 1987). This dose corresponds to an annual incremental risk of fatal cancer to members of the general population of about 2.88×10^{-4}. If exposure of the fetus is limited to 1/9 of 0.5 rem per month over a 9-month gestation period, as recommended, the risk of severe mental retardation in liveborn is limited to about 7×10^{-4}.

The ICRP recommends that the dose to members of the public not exceed 0.5 rem per year due to nonrecurring exposure to all sources of radiation combined, other than natural sources or beneficial medical uses of radiation (ICRP 1977). They also recommend a limiting dose to members of the public of 0.1 rem per year from all such sources combined for chronic (i.e., planned) exposure (ICRP 1984a). These upper bounds may be taken as representative of acceptable values for the situations to which they apply. That is, these are upper bounds of individual risk that are acceptable for the sum of all sources and exposure pathways under international recommendations, for circumstances that are justified on the basis of public benefit, and when actual doses from individual sources are ALARA within these upper bounds.

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Appendix F: Intermediate Phase Protective Action Guides Background Information

F.1 Introduction

This Appendix provides background information for the choice of PAGs for relocation and other protective actions for reducing exposure to deposited radioactive materials during the intermediate phase of the response to a radiological incident. The resulting PAGs, corresponding protective actions, and associated implementing guidance are provided in Chapters 2 through 5.

This analysis is based on the assumption that an airborne plume of radioactive material has already passed over an area and left a deposit of radioactive material behind, or that such material exists from some other source, and that the public has already been either sheltered-in-place or evacuated, as necessary, on the basis of PAGs for the early phase of a radiological incident. PAGs for subsequent relocation of the public and other protective actions, as well as dose limits for persons reentering the area from which the public is relocated, are addressed in this appendix.

EPA first sets forth the assumptions used to derive information pertinent to choosing the dose level at which relocation of the public is appropriate. This is followed by an examination of information relevant to this decision, and selection of the PAG for relocation. This Appendix concludes with a brief discussion of the basis for dose limits for persons temporarily reentering areas from which the public has been relocated.

F.1.1 Response Duration

In deciding whether to initiate relocation of the public from specific areas, it is necessary to predict the dose that would be avoided. One factor in this prediction is the duration of the exposure to be avoided. Relocation can begin as soon as patterns of exposure from deposited radioactivity permit relocation areas to be identified. For the purpose of this analysis, relocation of persons who have not already been evacuated from the relocation area is assumed to take place beginning on the fourth day after the incident. Return of evacuated persons to their residences outside the relocation area and transition to relocation status of persons already evacuated is assumed to occur over a period of 1 week or more.

The period of exposure avoided by relocation ends when the relocated person either returns to his property or is permanently resettled in a new location. At the time of relocation decisions, it will usually not be possible to predict when either of these actions will occur. Therefore, for convenience of dose projection, it is assumed that the period of exposure avoided is 1 year and that any extension beyond this period will be determined on the basis of recovery criteria. This assumption corresponds to emergency response planning guidance by ICRP (ICRP 1984) and IAEA (IAEA 1985).

Table F-1. Brief Descriptions Characterizing Various Nuclear Power Plant Accident Types (SNL 1982)

Type	Description
SST-1	Severe core damage. Essentially involves loss of all installed safety features. Severe direct breach containment.
SST-2	Severe core damage. Containment fails to isolate. Fission product release mitigating systems (i.e., sprays, suppression pool, fan coolers) operate to reduce release.
SST-3	Severe core damage. Containment fails by base-mat melt-through. All other release mitigation systems function as designed.
SST-4	Modest core damage. Containment systems operate in a degraded mode.
SST-5	Limited core damage. No failures of engineered safety features beyond those postulated by the various design basis incidents. Containment is assumed to function for even the most severe incidents in this group.

F.1.2 Source Term

The "source term" for this analysis is comprised of the quantities and types of the particulate radioactive material

found in the environment following a radiological incident. Radiological incidents can be postulated with a wide range of release characteristics. The characteristics of the source terms assumed for the development of these PAGs are those postulated for releases from various types of fuel-melt incidents at nuclear power plants (SNL 1982). Table F-1 provides brief descriptions of these accident types. Radionuclide releases have been estimated for the 3 most severe accident types (SST-1, SST-2, SST-3) based on postulated core inventories and release fractions (Table F-2). The other types (SST-4 and SST-5) would generally not produce offsite doses from exposure to deposited material sufficient to warrant consideration of relocation.

Table F-2. Release Quantities for Postulated Nuclear Reactor Incidents

Principal Radionuclides Contributing to Dose from Deposited Materials	Half-Life (d)	Estimated Quantity Released[a] (Ci)		
		SST-1	SST-2	SST-3
Zr-95	6.52E+01	1.4E+06	4.5E+04	1.5E+02
Nb-95	3.50E+01	1.3E+06	4.2E+04	1.4E+02
Ru-103	3.95E+01	6.0E+06	2.4E+05	2.4E+02
Ru-106	3.66E+02	1.5E+06	5.8E+04	5.8E+01
Te-132	3.25	8.3E+07	3.9E+06	2.6E+03
I-131	8.05	3.9E+07	2.6E+05	1.7E+04
Cs-134	7.50E+02	8.7E+06	1.2E+05	1.3E+02
Cs-137	1.10E+04	4.4E+06	5.9E+04	6.5E+01
Ba-140	1.28E+01	1.2E+07	1.7E+05	1.7E+02
La-140	1.67	1.5E+06	5.1E+04	1.7E+02

[a]Based on the product of reactor inventories of radionuclides and estimated fractions released for 3 accident categories (SNL 1982).

For other types of source terms, additional analysis may be necessary to assure adequate protection. For example, if the release includes a large proportion of long-lived radionuclides, doses will continue to be delivered over a long period of time, and, if no remedial actions are taken, the dose delivered in the first year may represent only a small portion of the total dose delivered over a lifetime. On the other hand, if the release consists primarily of short-lived radionuclides, almost the entire dose may be delivered within the first year.

From the data in Table F-2, it is apparent that, for the groups of incidents listed, both long- and short-lived radionuclides would be released. Consequently, doses due to deposited materials from such incidents would be relatively high during the first year followed by long term exposures at lower rates.

F.1.3 Exposure Pathways

The principal exposure pathway to members of the public occupying land contaminated by deposits of radioactive materials from reactor incidents is expected to be exposure of the whole body to external gamma radiation. Although it is normally expected to be of only minor importance, the inhalation pathway would contribute additional doses to internal organs. The health risks from other pathways, such as beta dose to the skin and direct ingestion of dirt, are also expected to be minor in comparison to the risks due to external gamma radiation (Aaberg 1989). Skin and inhalation dose would, however, be important exposure pathways for source terms with significant fractions of pure beta emitters, and inhalation dose would be important for source terms with significant fractions of alpha emitters.

F.1.4 Response Scenario

This section defines the response zones, population groups, and the activities assumed for implementation of protective actions during the intermediate phase.

After passage of the radioactive plume, the results of environmental monitoring will become available for use in making decisions to protect the public. Sheltering-in-place, evacuation, and other actions taken to protect the public from the plume will have already been implemented. The tasks immediately ahead will be to: (1) define the extent and characteristics of deposited radioactive material and identify a relocation area in accordance with the PAG for relocation; (2) relocate persons from and control access to the relocation area; (3) allow persons to return to areas outside the relocation area; (4) control the spread of and exposure to surface contamination; and (5) apply simple decontamination and other low-cost, low-risk techniques to reduce the dose to persons who are not relocated.

Because of the various source term characteristics and the different protective actions involved (evacuation, sheltering-in-place, relocation, decontamination, and other actions to reduce doses to ALARA levels), the response areas for different protective actions may be complex and may vary in size with respect to each other. Figure F-1 shows a generic example of some of the principal areas involved. The area covered by the plume is assumed to be represented by Area 1. In reality, variations in meteorological conditions would almost certainly produce a more complicated shape.

Based on plant conditions or other considerations prior to or after the release, members of the public are assumed to have already been evacuated from Area 2 and sheltered in Area 3. Persons who were evacuated or sheltered as a precautionary action for protection from the plume but whose homes are outside the plume deposition area (Area 1) are assumed to return to their homes or discontinue sheltering-in-place when environmental monitoring verifies the outer boundary of Area 1.

Area 4 is the relocation area and is defined as the area where projected doses are equal to or greater than the relocation PAG. The portion of area 1 outside of Area 4 is designated as a study zone and is assumed to be occupied by the public. However, contamination levels may exist here that would be of concern for continued monitoring and decontamination to maintain radiation doses ALARA.

The relative positions of the boundaries shown in Figure F-1 are dependent on areas evacuated and sheltered. For example, Area 4 could fall entirely inside Area 2 (the area evacuated) so that relocation of persons from additional areas would not be required. In this case, the relocation PAG would be used only to determine areas to which evacuees could return.

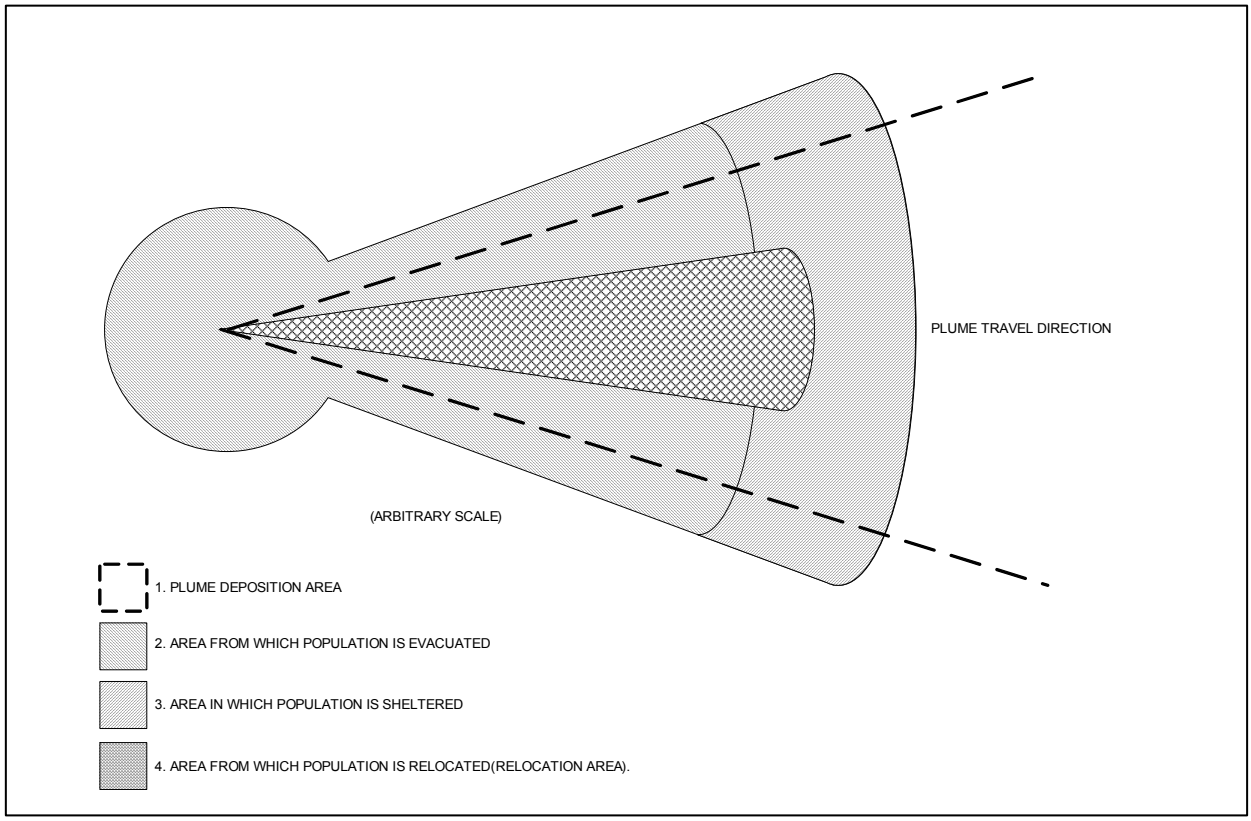

Figure F-1. Generalized Response Areas

Figure F-2 provides, for perspective, a schematic representation of the response activities expected to be in progress in association with implementation of the PAGs during the intermediate phase of the response to a radiological incident.

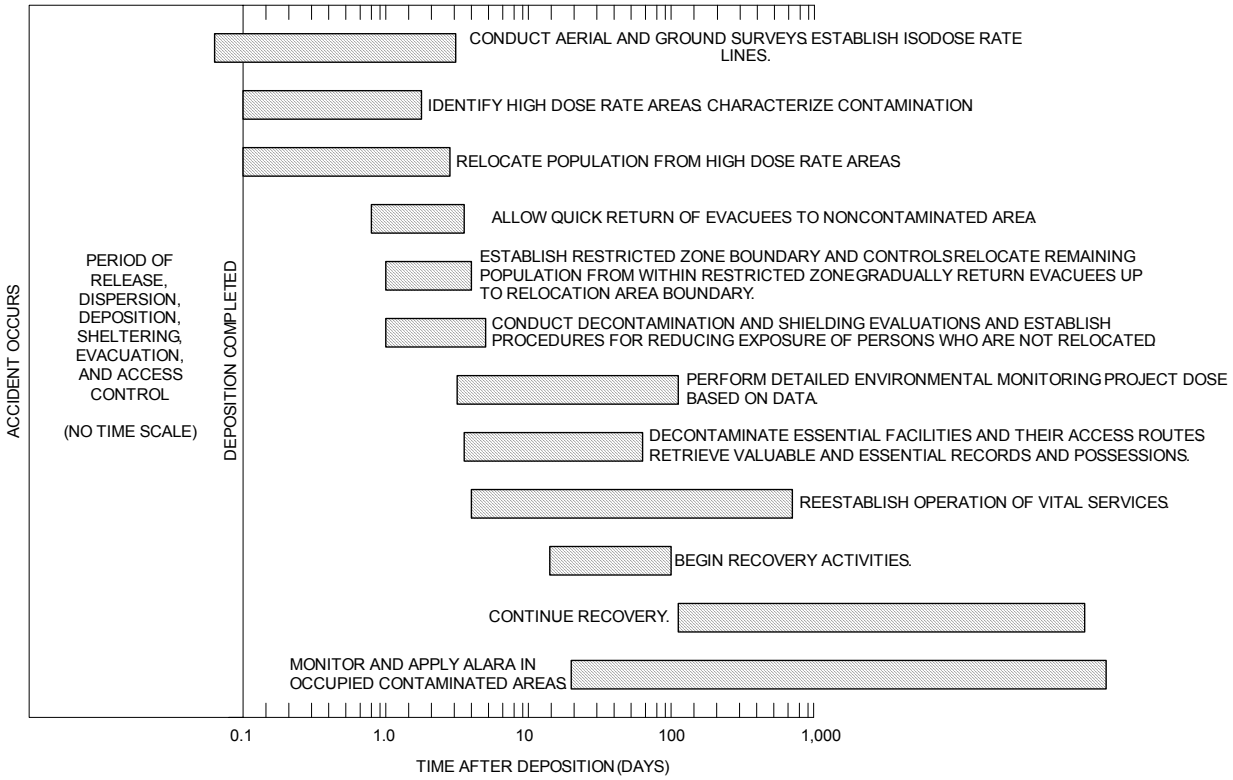

Figure F-2. Potential Time Frame of Response to a Radiological Incident

F.2 Considerations for Establishing PAGs for the Intermediate Phase

The major considerations in selecting values for these PAGs for relocation and other actions during the intermediate phase are the 3 principles that form the basis for selecting all PAGs. Those are discussed in Section F.2.3. Other federal radiation protection guidance considerations are discussed in sections F.2.4 .

F.2.1 Derivation of Intermediate Phase PAGs for Nuclear Power Plant Incidents

A planning group consisting of state, federal, and industry officials provided recommendations in 1982 which EPA considered in the development of the format, nature, and applicability of PAGs for relocation. Abbreviated versions of these recommendations are as follows:

a. The PAGs should apply to commercial, light-water power reactors.

b. The PAGs should be based primarily on health effects.

c. Consideration should be given to establishing a range of PAG values.

d. The PAGs should be established as high as justifiable because at the time of the response, it would be possible to lower them, if justified, but it probably would not be possible to increase them.

e. Only 2 zones (restricted and unrestricted) should be established to simplify implementation of the PAGs.

f. The PAGs should not include past exposures.

g. Separate PAGs should be used for ingestion pathways.

h. PAGs should apply only to exposure during the first year after an incident.

Although these PAGs apply to any radiological incident, primary consideration was given to the case of commercial U.S. reactors. In general, EPA has found it possible to accommodate most of the above recommendations.

F.2.2 Additional Considerations for Intermediate Phase PAGs for Terrorist Events

In a symposium entitled Implementing Protective Actions for Radiological Incidents at Other Than Nuclear Power Reactors (EPA 1992), EPA evaluated the applicability PAGs to a non-reactor incident. In the May, 2003 Top Officials (TOPOFF) 2 interagency exercise, DHS evaluated the applicability of the early and intermediate phase PAGs to a radiological incident involving an RDD and an IND (DHS 2006). These analyses showed that the PAG would be applicable; however, because of the characteristics of the radionuclides involved, doses may not decrease as quickly as they might if they had resulted from a power plant accident. Consequently, it is very important that particular attention be paid to the second and succeeding year(s) doses from these types of incidents.

F.2.3 Principles

In selecting values for these PAGs, EPA has been guided by the three principles that were set forth in Chapter 1. They are repeated here for convenience:

- Prevent acute effects.
- Reduce risk of chronic effects.
- Require optimization to balance protection with other important factors and ensure that actions taken cause more benefit than harm.

Appendix E analyzed the risks of health effects as a function of dose (Principles 1 and 2). Considerations for selection of PAGs for the intermediate phase of a radiological incident differ from those for selection of PAGs for the early phase primarily with regard to implementation factors (i.e., Principle 3). Although sheltering-in-place is not generally a suitable alternative to relocation, other alternatives (e.g., decontamination and shielding) are suitable. These considerations are reviewed in the sections that follow.

F.2.3.1 Protection of Special Groups

Contrary to the situation for evacuation during the early phase of an incident, it is generally not practical to leave a few persons behind when most members of the general population have been relocated from a specified area for extended periods of time. Further, no data are available on differing risks of relocation for different population groups. In the absence of such data, EPA has assumed that these risks will be similar to those from evacuation. Those risks were taken as equivalent to the health risk from doses of 30 mrem for members of the general population and of 150 mrem for persons at high risk from evacuation.

Fetuses are a special group at greater risk of health effects from radiation dose than is the general population, but not at significantly greater risk from relocation itself. The risk of mental retardation from fetal exposure (see Appendix E) is significant. It is affected by the stage of pregnancy relative to the assumed 1-year exposure because the 8^{th} to 15^{th} week critical period during which the risk is greatest (4×10^{-3} cases/rem) must be considered in relation to the rapidly changing dose rate. Taking these factors into account, it can be postulated that the risk of mental retardation due to exposure of the fetus during the intermediate phase will range from 1 times (1x) 5 times (5x) the cancer risk of an average member of the public (8.46×10^{-4} cases/rem), depending upon when conception occurs relative to the time of the incident. The elevated risk of radiation-induced cancer from exposure of fetuses is less significant, as discussed in Appendix E.

It will usually be practicable to reduce these risks by establishing a high priority for efforts other than relocation to reduce the dose in cases where pregnant women reside near the boundary of the relocation area. However, women who are less than 7 months pregnant may wish to relocate for the balance of their pregnancy if the projected dose during pregnancy cannot be reduced below 0.5 rem.

F.2.4 Federal Radiation Protection Guides

The choice of a PAG at which relocation should be implemented does not mean that persons outside the boundary of the relocation area should not be the subject of other protective actions to reduce dose. Such actions are justified on the basis of existing federal radiation protection guidance (FRC 1965) for protecting the public, including implementation of the principle of maintaining doses ALARA.

The intended actions to protect the public from radiation doses on the basis of PAGs are those related to source control. Although it is reasonable for members of the public to receive higher exposure rates prior to the source term being brought under control, the establishment of acceptable values for relocation PAGs must include consideration of the total dose over the average remaining lifetime of exposed individuals.

The nationally and internationally recommended upper bound for dose in a single year from man-made sources, excluding medical radiation, is 500 mrem per year to the whole body of individuals in the general population (ICRP 1977, FRC 1965). These recommendations were not developed for radiological incidents. They are also not appropriate for chronic exposure. The ICRP recommends an upper bound of 100 mrem per year, from all sources combined, for chronic exposure (ICRP 1977). EPA has chosen to limit (a) the projected first year dose to individuals from an incident to the Relocation PAG; and (b) the projected second year dose to 500 mrem. Due to the extended duration of exposures and the short half-life of important radioiodines, no special limits for thyroid dose are needed.

F.3 Dose from Reactor Incidents

Doses from an environmental source will be reduced through the natural processes of weathering and radioactive decay and from the shielding associated with part time occupancy in homes and other structures. Based on studies reported in WASH-1400 (NRC 1975), the most conservative dose reduction factor for structures (frame structures) is about 0.4 (dose inside divided by the dose outside) and the average fraction of time spent in a home is about 0.7. Combining these factors yields a net dose reduction factor of about 0.6.

F.4 Alternatives to Relocation

Persons who are not relocated, in addition to dose reduction provided by partial occupancy in homes and other structures, can reduce their doses by the application of various techniques. Dose reduction efforts can range from the simple processes of scrubbing and/or flushing surfaces, removal and disposal of small spots of soil found to be highly contaminated (e.g., from settlement of water), and spending more time than usual in lower exposure rate areas (e.g., indoors), to the difficult and time consuming processes of removal, disposal, and replacement of contaminated surfaces. It is anticipated that simple processes would be most appropriate to reduce exposure rates for persons living in contaminated areas outside the relocation area. Many of these can be carried out by the residents with support from officials for monitoring, guidance on appropriate actions, and disposal. The more difficult processes will usually be appropriate for recovery of areas from which the population is relocated.

Decontamination experiments involving radioactive fallout from nuclear weapons tests have shown reduction factors for simple decontamination methods in the vicinity of 0.1 (i.e., exposure rate reduced to 10% of original values). However, experiments at the Riso National Laboratory in Denmark (Warming 1982, Warming 1984), using fire hoses to flush asphalt and concrete surfaces contaminated with radioactive material of the type that might be deposited from reactor incidents, show decontamination factors for radionuclides chemically similar to cesium that are in the range of 0.5 to 0.95, depending on the delay time after deposition before flushing is applied. The factor for ruthenium on asphalt was about 0.7 and was independent of the delay of flushing. The results of these experiments indicate that decontamination of the important reactor fission products from asphalt or concrete surfaces may be much more difficult than decontamination of nuclear weapons fallout. Other simple dose reduction methods listed above would be effective to varying degrees. The average dose reduction factor for gamma radiation from combinations of simple decontamination methods is estimated to be at least 0.7.

Table F-3. Summary of Considerations for Selecting PAGs for Relocation

Dose (rem)	Consideration	Principle
50	Assumed threshold for acute health effects in adults.	1
10	Assumed threshold for acute health effects in the fetus.	1
6	Maximum projected dose in first year to meet 0.5 rem in the second year[a].	2
5	Maximum acceptable annual dose for normal occupational exposure of adults.	2
5	Minimum dose that must be avoided by 1 year relocation based on cost.	3
3	Minimum projected first-year dose corresponding to 0.5 rem in the second year[a].	2
0.5	Maximum acceptable single-y dose to the general population from all sources from non-recurring, non-incident exposure.	2
0.5	Maximum acceptable dose to the fetus from occupational exposure of the mother.	2
0.1	Maximum acceptable annual dose to the general population from all sources due to routine (chronic), non-incident, exposure.	2

[a]Assumes the source term is from a reactor incident and that simple dose reduction methods are applied during the first month after the incident to reduce the dose to person not relocated from contaminated areas.

Analyses based on Principle 3 (cost/risk) indicate that considering cost alone would not drive the PAG to values less than 5 rem.

Based on the above, 2 rem projected committed effective dose equivalent from exposure in the first year is selected as the PAG for relocation. Implementation of relocation at this value will provide reasonable assurance that, for a reactor accident, a person relocated from the outer margin of the relocation zone will, by such action, avoid an exposure rate which, if continued over a period of 1 year, would result in a dose of about 1.2 rem. This assumes that 0.5 rem would be avoided without relocation through normal partial occupancy of homes and other structures. This PAG will provide reasonable assurance that persons outside the relocation zone, following a reactor accident, will not exceed 1.2 rem in the first year, and 0.5 rem in the second year. The implementation of simple dose reduction techniques, as discussed in Section F.4, will further reduce dose to persons who are not relocated from contaminated areas. Table F-4 summarizes the estimated maximum dose that would be received by these persons for various reactor accident categories with and without the application of simple dose reduction techniques. In the case of non-reactor incidents these doses will, in general, differ, and it may be necessary to apply more restrictive PAGs to the first year in order to assure conformance to the second year and lifetime objectives noted above.

Since effective dose does not include dose to the skin (and for other reasons discussed in Appendix E), protective action to limit dose to skin is recommended at a skin dose 50 times (50x) the numerical value of the PAG for effective dose. This includes consideration of the risk of both curable and fatal cancers.

Table F-4. Estimated Maximum Doses to Nonrelocated Persons from Areas Where the Projected Dose is 2 rem[a]

Accident Category	Dose (rem)			
	No Additional Dose Reduction		Early Simple Dose Reduction[b]	
	Year One	Year Two	Year One	Year Two
SST-1	1.2	0.5	0.9	0.35
SST-2	1.2	0.34	0.9	0.24
SST-3	1.2	0.2	0.9	0.14

[a]Based on relocation at a projected dose of 2 rem in the first year and 40% dose reduction to nonrelocated persons from normal, partial occupancy in structures. No dose reduction is assumed from decontamination, shielding, or special limitations on time spent in high exposure rate areas.
[b]The projected dose is assumed to be reduced 30% by the application of simple dose reduction techniques during the first month. If these techniques are completed later in the first year, the first-year dose will be greater.

F.5 Criteria for Reentry into the Relocation Area

Persons may need to reenter the relocation area for a variety of reasons, including radiation monitoring, recovery work, animal care, property maintenance, and factory or utility operation. Some persons outside the relocation area, by nature of their employment or habits, may also receive higher than average radiation doses. Tasks that could cause such exposures include (1) changing of filters on air handling equipment (including vehicles); (2) handling and disposal of contaminated vegetation (e.g., grass and leaves); and (3) operation of control points for the relocation area.

Individuals who reenter parts of the relocation area where the dose would exceed 2 mrem in an hour or who perform tasks involving exposure rates that would cause their radiation dose to exceed that permitted by the PAGs, should do so in accordance with existing federal radiation protection guidance for occupationally exposed workers (EPA 1987). The basis for that guidance has been provided elsewhere (EPA 1987).

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5Appendix G: Federal Implementation for Cleanup Following an Improvised Nuclear Device or Radiological Dispersal Device

G.1 Introduction

This appendix provides a summary of the DHS RDD/IND Application, and how federal departments and agencies may interact with state and local government counterparts and the public. The plan does not attempt to provide detailed descriptions of state and local roles and expertise. It is assumed those details would be provided in state-, area-, and local-level planning documents that address radiological incidents.

This site cleanup implementation plan is intended to function under the National Response Plan (NRP) (DHS 2004) with federal agencies performing work consistent with their established roles, responsibilities and capabilities. Agencies should be tasked to perform work under the appropriate Emergency Support Function (ESF), as a primary or support agency, as described in the NRP.

This plan is designed to be compatible with the Incident Command/Unified Command (IC/UC) structure embodied in the National Incident Management System (NIMS). The functional descriptions and processes in this plan are provided to address the specific needs and wide range of potential impacts of an RDD or IND incident. During the intermediate phase, site restoration planners should begin the process described below, in coordination with the on-site IC/UC. Federal activities may organize along IC/UC functional lines coordinating with the on-site organization to avoid redundancy. After early and intermediate phase activities have come to conclusion, and only long-term cleanup and site restoration activities are ongoing, the IC/UC structure may continue to support planning and decision making for the long-term cleanup. The IC/UC may make personnel changes and structural adaptations to suit the needs of a lengthy, multifaceted and highly visible remediation process. For example, a less formal and structured command, more focused on technical analysis and stakeholder involvement, may be preferable for site restoration than what is required under emergency circumstances. Some of the Teams described below, such as the Decision Team or the Recovery Management Team, may be coordinated from, or coincident with, functional portions of the IC/UC at the site. Although the makeup of the Teams may vary, the functions should remain the same.

Radiological and nuclear terrorism incidents cover a broad range of potential scenarios and impacts. For the sake of this appendix, it is assumed that the incident is of sufficient size to trigger a state request for federal assistance and that the federal government is the primary funding agent for site restoration. The process described for the late phase assumes an incident of larger size. For smaller incidents, all of the elements in this appendix may not be warranted. The process should be tailored to the circumstances of the particular incident. It should be recognized that for some radiological incidents, states will take the primary leadership role and contribute significant resources toward restoration of the site. This appendix does not address such a circumstance.

As described earlier in the document, radiological emergency responses are often divided roughly into three phases: (1) the early phase, when the plume is active and field data are lacking or not reliable; (2) the intermediate phase, when the plume has passed and field data are available for assessment and analysis; and (3) the late phase, when long-term issues are addressed, such as restoration of the site. For purposes of this appendix, the response to a radiological or nuclear terrorism incident is divided into two separate, but interrelated and overlapping, processes. The first is comprised of the early and intermediate phases of response, which consist of the immediate on-scene actions of state and local emergency responders under IC/UC as well as those of federal teams and officials, to perform incident stabilization, lifesaving activities, access control and security, emergency decontamination of persons and property, “hot spot” removal actions, dose reduction actions for members of the public and emergency responders, and resumption of basic infrastructure functions.

The second process pertains to environmental restoration which is initiated soon after the incident (during the intermediate phase) and continues through the late phase. The process starts with the convening of stakeholders and technical subject matter experts to begin identifying and evaluating options for the restoration of the site. The environmental restoration process overlaps the intermediate phase activities described above and should be coordinated with those activities.

This implementation plan does not address law enforcement coordination during terrorism incident response,

including how the Federal Bureau of Investigation (FBI) and DHS will manage on-scene actions immediately following an act of terror. Also, victim triage and other medical response aspects are not addressed. The plan presented in this appendix is not intended for use at site cleanups occurring under other statutory authorities such as EPA's Superfund program, the NRC's decommissioning program, or state-administered cleanup programs.

G.2 Late Phase – Recovery and Site Restoration Activities Process Overview

As noted earlier, the long-term recovery process should be initiated during the intermediate phase. This process is interrelated with the ongoing intermediate phase activities and the intermediate phase protective actions continue to apply through the late phase until cleanup is complete. However, long-term recovery during the late phase is likely to involve separate individuals who can focus on long-term restoration issues while others continue working on intermediate phase activities.

Cleanup planning and discussions should begin as soon as practicable after an incident to allow for selection of key stakeholders and subject matter experts, planning, analyses, contractual arrangements, and cleanup activities. These activities should proceed in parallel with ongoing intermediate phase activities and coordination between these sets of activities should be maintained. Preliminary remediation activities carried out during the intermediate phase – such as emergency removals, decontamination, resumption of basic infrastructure function, and some return to normalcy in accordance with intermediate phase guidelines – should not be delayed for the final site remediation decision.

Presented below is a process for addressing environmental contamination using an optimization process. Optimization (described more fully in Chapter 6) is a flexible process in which numerous factors are considered to achieve an end result that balances local needs and desires, health risks, costs, technical feasibility, and other factors. The general process outlined below provides decision makers with input from both technical experts and stakeholder representatives, as well as providing an opportunity for public comment. The extent and complexity of the process for an actual incident should be tailored to the needs of the specific incident; for smaller incidents, the teams discussed below may not be necessary.

The goals of the process described below are: (1) transparency – the basis for cleanup decisions should be available to stakeholder representatives, and ultimately to the public at large; (2) inclusiveness – representative stakeholders should be involved in decision-making activities; (3) effectiveness – technical subject matter experts should analyze remediation options, consider dose and risk benchmarks, and assess various technologies to assist in identifying a final solution that is optimal for the incident; and (4) shared accountability – the final decision to proceed will be made jointly by DHS, state, and local officials.

DHS/FEMA would issue mission assignments to the involved federal agencies to participate in the overall recovery and restoration process. Additional funding may be provided to state/local governments to perform response/restoration activities through other mechanisms. The components of the process are as follows:

Teams

1. Decision Team

Makeup: The Decision Team consists of the secretary of DHS, the governor of the state, the mayor or equivalent, and the head of the federal lead technical agency (or their respective designated representatives with authority to commit resources on behalf of affected persons).

Function: The function of the Decision Team is to make the final decision on recommendations received from the Recovery Management Team, commit resources, and commence cleanup activities. The Decision Team will raise unresolved national level policy issues to the Interagency Incident Management Group (IIMG) and/or to the Assistant to the President for Homeland Security, as appropriate.

2. Recovery Management Team

State and DHS officials should select a Recovery Management Team as soon as possible after the incident. The size and makeup of the team will be dependent on the incident but would be expected to consist of senior-level officials. The Recovery Management Team will normally be located at the JFO to enhance information flow

and response coordination.

Makeup: The Recovery Management Team should include DHS, affected state and/or local representatives, and the federal lead technical agency. The Recovery Management Team should be co-chaired by a DHS and state official. The makeup is flexible and may accommodate other individuals, as necessary.

Functions: The functions of the Recovery Management Team are to select participants for the Stakeholder and Technical Working Groups; provide oversight and guidance during the cleanup analyses and decision making process; oversee working group interactions; maintain communications between working groups; receive and review options and recommendations; ensure the development and implementation of community involvement and public information strategy; and prioritize recommendations when they are forwarded to the Decision Team for action.

3. Stakeholder Working Group

The Stakeholder Working Group should be convened as soon as practicable, normally within weeks of the incident.

Makeup: The Stakeholder Working Group should include selected federal, state, and local representatives; local non-governmental representatives; and local business interests. The exact selection and balance of stakeholders is incident specific. The Stakeholder Working Group should be co-chaired by DHS and state and/or local representatives.

Function: The function of the Stakeholder Working Group is to provide input to the Technical Working Group and the Recovery Management Team concerning local needs and desires for site restoration, proposed cleanup options, and recommendations for recovery.

4. Technical Working Group

The Technical Working Group should be convened as soon as practicable, normally within weeks of the incident.

Makeup: The Technical Working Group should include selected federal, state, local, and private sector subject matter experts in such fields as environmental fate and transport modeling, risk analysis, technical remediation options analysis, cost risk and benefit analysis, health physics/radiation protection, construction remediation practices, and relevant regulatory requirements. The exact selection and balance of subject matter experts is incident specific. The Technical Working Group should be chaired by the federal lead technical agency assigned responsibility for performing cleanup operations and co-chaired by the state/local technical agency.

Function: The Technical Working Group provides expert input on technical issues, analysis of relevant regulatory requirements and guidelines, risk analyses, and evaluation of options as directed by the Recovery Management Team. The actual technical analyses will be the responsibility of the federal lead technical agency for cleanup. The Technical Working Group should also receive input from the Stakeholder Working Group. Technical Working Group written products are provided to the Recovery Management Team.

Activities

1. Optimization and Recommendation (lasts weeks to months)

The Recovery Management Team, in consultation with the Stakeholder Working Group and Technical Working Group, will develop a process for the three teams to work together to provide the opportunity for local concerns to inform the work of the Technical Working Group. The Technical Working Group and Recovery Management Team should assist in answering questions the Stakeholder Working Group may have regarding technical issues and provide information regarding cleanup options.

The Stakeholder Working Group should present local goals, needs, and desires for the use of the site and

prioritize current and future potential land uses and functions, such as utilities and infrastructure, light industrial, downtown business, and residential land uses. The lead technical agency will oversee technical optimization analyses for site cleanup in collaboration with the Recovery Management Team, Technical Working Group, and Stakeholder Working Group. The Technical Working Group will analyze assumptions, review risk analyses for various proposed remediation options, assess technical feasibility and cost of the options, and identify the estimated time to complete restoration options and their potential impacts on the local community.

The Stakeholder Working Group will provide input to the Technical Working Group but may also provide options and recommendations directly to the Recovery Management Team. The Technical Working Group will consider input from the Stakeholder Working Group in its analyses and provide input to the Recovery Management Team on remediation options and recommended approaches and rationale. It is important that the Technical Working Group and the Stakeholder Working Group maintain confidentiality concerning all aspects of the analyses. All outside contacts, such as press interviews, concerning the ongoing work and deliberations should be coordinated through the Recovery Management Team.

As the Technical Working Group completes its analyses and formulates its recommendations, it will present this information to the Recovery Management Team for final review. The Recovery Management Team will present the Decision Team with options, recommendations for final action, and supporting documentation.

2. Public Review of Decision

The Decision Team should publish a summary of the process, the options analyzed, and the recommendation for public comment. Public meetings may also be convened as appropriate. Public comment should be considered and incorporated as appropriate. A reconvening of the Recovery Management Team, Stakeholder Working Group, and Technical Working Group may be useful for resolving some issues.

3. Execute Cleanup

DHS/FEMA should issue mission assignments to the federal departments and agencies that have the capability to perform the required cleanup or remediation activities. For significant de-contamination efforts, decision makers may choose to employ a technical peer review advisory committee to conduct a review of the effectiveness of the cleanup.

Appendix H: Cleanup Principles and Potential Benchmarks

H.1 Some Benchmarks for Consideration during Cleanup

H.1.1 EPA Guidance

EPA has issued a number of CERCLA guidance documents for addressing contaminants, including radiological contaminants. EPA CERCLA guidance documents for addressing cleanup in general may be found on the Internet at: http://www.epa.gov/superfund/action/guidance/remedy/index.htm. EPA CERCLA guidance documents for addressing radionuclides in particular may be found at: http://www.epa.gov/superfund/resources/radiation/index.htm.

Under CERCLA, applicable or relevant and appropriate requirements (ARARs), which are federal, and more stringent state environmental standards, are often the determining factors in setting cleanup levels for long-term remedial actions pursuant to CERCLA. (Relevant and appropriate requirements are those standards and regulations which address circumstances considered to be sufficiently similar to the circumstances being addressed at the particular site). In cases where standards do not exist or may not be sufficiently similar to the actual situation, or may not be applicable or relevant and appropriate, or the ARAR is not sufficiently protective or has been waived, site-specific cleanup levels are generally set for:

- Carcinogens at a level such that a highly-exposed individual may have a 1 in 10,000 to a 1 in 1 million increased chance of developing cancer because of an exposure to a site-related carcinogen (10^{-4} to 10^{-6} cancer risk range); and

- Non-carcinogens such that the cumulative risks from exposure will not result in adverse human health effects. To assess the potential for cumulative non-carcinogenic effects posed by multiple contaminants, EPA has developed a hazard index that is derived by adding the non-cancer risks for site contaminants. Generally, a hazard index (HI) of less than 1 is considered protective.

The specific cleanup levels account for exposures from all potential pathways and through all environmental media (soil, ground water, surface water, sediment, air, animals or plants). Risk-based cleanup levels are developed using the reasonably anticipated land use. If meeting protective levels using the reasonably anticipated land use is not both practical and cost-effective, EPA looks to more restrictive land uses through institutional and engineering controls to achieve further reduction in potential for human exposure. EPA also has developed dose assessment methodology for demonstrating compliance with dose based ARARs that is consistent with its methodology for radiological and chemical risk assessment.

However, the economic or other impacts of institutional or engineering controls on a radiological incident affected area may be so significant that it would become impracticable or too costly to meet EPA's CERCLA standards. These cases would be identified through an evaluation of remedial alternatives considering various target risk levels (possibly with and without institutional and engineering controls). For example, cleanup to industrial/commercial at 1×10^{-6}, 1×10^{-5}, 1×10^{-4}, 1×10^{-3}, and 1×10^{-2} cancer risk levels. If this is true on a site-specific basis, EPA would expect that evaluating the options that are not normally considered protective, with those options that do meet EPA CERCLA standards, pose the best chance of providing stakeholders with a clear rationale for why a cleanup level was selected at a specific radiological incident site that would not normally be considered at an EPA site.

Site decision makers may choose to use standards/approaches other than EPA's to establish cleanup levels after a radiological incident. However, EPA is often asked whether a cleanup conducted by another Agency would be considered protective by EPA. EPA should evaluate the cleanup based on the proposed, planned, or actual post remedy site-specific cleanup level concentrations and not on the dose or risk levels on which they are based. EPA should not evaluate the cleanup using all of the procedures EPA would use if it were conducted under an EPA CERCLA response action. Instead, EPA should evaluate whether the cleanup will: (1) achieve the risk rangeand Hazard Index of less than 1 for noncarcinogens; and (2) achieve ARAR for federal and state environmental requirements. In making this determination, risk assessments should be performed on a site-specific basis using EPA guidance to the extent possible. Compliance with standards that EPA would likely consider potential federal or state requirements to an EPA cleanup site should be used in evaluating the attainment of EPA cleanup levels to the extent

those potential requirements can be readily identified.

H.1.2 NRC's Decommissiong Process (Non-Emergency)

The NRC has established (1997) 25 mrem (250 mSv) Total Effective Dose Equivalent (TEDE) per year from all pathways as the dose constraint below NRC's public dose limit that is appropriate for the decommissioning of licensed nuclear facilities. NRC's decommissioning process applies to licensed sites after normal operations have ceased. NRC's decommissioning process and the cleanup levels required under this process are only applicable to non-emergency situations and would not be appropriate for the types of immediate actions necessary to protect life and property in the event of an attack using an RDD/IND. The NRC decommissioning process is not designed as an emergency response process and achieving the cleanup levels required under this process would in all likelihood be cost prohibitive.

For further clarification, the following text is taken directly from chapters 2 and 3 of IAEA Safety Series DS-162, "Remediation of Areas Contaminated by Past Activities and Accidents." "Protection of the Public in Situations of Prolonged Radiation Exposure" (ICRP 1982) contains similar text and the same generic reference levels for remediation.

> 2. OBJECTIVES IN THE REMEDIATION OF CONTAMINATED AREAS
>
> Remedial measures shall do more good than harm and shall provide optimized protection arrangements to maximize the net benefit to society.
>
> The preferred goal of remediation activities is the release of the area from regulatory control with no restrictions. However, there are situations in which the release of the area from control cannot reasonably be achieved. In such cases, at least the unacceptable risks to human health and the environment shall be removed. In these cases, any restrictions on access to the area, use of the area or any other restrictions shall be established on the basis of an optimization process to maximize the net benefit to society. In the choice of the optimized remediation option, a wide variety of factors shall be considered, and health, safety and environmental impacts shall be considered together with technical, social and financial factors. Non-radiological hazards shall be considered in conjunction with the radiological hazards. Remediation shall be aimed at reducing existing exposures and averting potential prolonged exposures. Remediation shall:
> (a) reduce doses to individuals or groups of individuals being exposed;
>
> (b) avert doses to individuals or groups of individuals likely to arise in the future;
>
> (c) avoid or reduce environmental impacts from the radionuclides present in the contaminated area.
>
> 2.3. Reductions in exposures of individuals and reduced environmental impacts shall be achieved by interventions to remove the existing sources of contamination, to modify the pathways of exposure or to reduce the numbers of individuals or other receptors exposed to radiation from the source.
>
> 3. RADIATION PROTECTION IN REMEDIATION SITUATIONS
>
> 3.1. For contamination resulting from past activities and accidents, the required level of remediation shall be established on a site-specific basis and in accordance with the radiation protection principles that apply to intervention situations. Consequently, remedial measures and protection measures to be implemented thereafter shall be justified and optimized.
>
> The justification of remediation shall be assessed by means of a decision aiding process requiring a positive balance of all relevant attributes relating to the contamination. In addition to the avertable annual doses, both individual and collective, other relevant attributes shall be assessed. These attributes shall include at least: the health detriments; the expected reduction in the anxiety caused by the situation; and the benefit, social cost, disruption and environmental effects that may

be caused by the implementation of remedial measures.

The optimization of remediation actions shall be performed following the general approach to optimization of protection in the context of practices. The optimum nature, scale and duration of the remedial measures shall be selected from a set of justified options of remediation. For some remediation situations the restricted use of human habitats can be the outcome of the optimization process.

The results of such a decision aiding process of justification/optimization shall be used as an input into a decision making process which may encompass other considerations (such as residual doses) and involve relevant interested parties. The objective is that those concerned with the contamination situation shall be involved and be given the opportunity to have input into the decision making process.

3.2. A generic reference level for aiding decisions on remediation is an individual existing annual effective dose of 10 mSv (1 rem) from all sources including natural background. This will normally be assessed as the mean dose in an appropriately defined critical group. Remedial measures would often be justified below the generic reference level and national authorities may define a lower level for identifying sites that might need remediation. The use of generic reference levels shall not encourage a 'trade-off' of remedial measures among the various components of the existing annual dose.

3.3. If remediation is justified below the generic reference level to reduce a dominant component of the existing annual dose, a reference level specific to particular components can be established on the basis of appropriate fractions of the generic reference level. National authorities shall approve such specific reference levels (such as intervention levels and action levels) for particular prolonged contamination situations amenable to intervention on the basis of the optimization process specified in para. 3.1. Specific reference levels can be expressed in terms of the avertable annual dose, or a subsidiary quantity such as activity concentration (Bq/g) or surface contamination density (Bq/cm2).

3.4. Situations in which the annual equivalent dose thresholds for deterministic effects in relevant organs could be exceeded shall in all cases require the implementation of remedial measures or restrictions on access. An existing annual equivalent dose of 100 mSv (10 rem) (inclusive of all existing contributions, including doses due to the natural background) to any organ shall justify intervention under almost any conceivable circumstances, unless national authorities specifically determine that such measures are not justified.

3.5. In the implementation of remedial measures, the exposure of workers shall be controlled under the system of radiation protection for practices. No worker undertaking remedial measures shall be exposed in excess of the annual dose limit for occupational exposure. If the implementation of remedial measures could result in the temporary exposure of members of the public beyond the dose limit for practices, such additional exposure shall be justified on the basis of the resulting net benefit, including the final reduction of the existing annual dose.

H.1.3 DOE Cleanup Programs and Property Control and Release Directives

DOE is responsible for ensuring health and safety of workers and the public from radioactivity and radioactive materials derived from its research, development and production activities. This includes releases from normal operations and off-normal operations including accidents and other potential disasters. Depending on circumstances, DOE operations may be subject to DOE internal requirements, EPA requirements, and in some cases, NRC requirements. DOE conducts cleanup operations under the EPA CERCLA process using EPA standards and releases property for DOE radiological control under the Department's radiation protection directives. As a result, DOE employs both lifetime-risk-based criteria and dose-based criteria and has developed tools that assess dose-based and risk-based impacts from residual radioactive material in the environment.

Under its Atomic Energy Act (AEA) authorities, DOE has established requirements for the control and release of property containing or potentially containing residual radioactive material. When DOE activities are subject to CERCLA, DOE employs procedures and processes described in the EPA CERCLA cleanup discussion addressed in a separate attachment. In such situations, DOE typically conducts analyses to ensure both EPA and DOE public protection requirements are met.

When DOE is releasing property under its AEA authorities, its uses requirements contained in its own directives. These requirements are dose-based and employ the ALARA concept. The requirements are similar to those established by the NRC for decommissioning. DOE requires that all public doses be ALARA and below a 100 mrem/year all pathways/all sources dose limit. To ensure releases of property do not cause the all sources dose to exceed the primary dose limit, DOE uses a 25 mrem/year dose constraint for real property. Hence, analyses must demonstrate that doses associated with a release of property will be as far below 25 mrem/year as is reasonable. The requirements allow for the consideration of cost/benefit or multi-attribute analyses through the ALARA process that would be relevant for late-phase RDD/IND activity; however, some requirements such as the dose constraints were not developed considering factors unique to RDD/IND events. Due to the potential for multiple exposures to personal property, generally DOE releases of such property use a 1 mrem/year dose constraint for establishing criteria. Criteria for personal property employing a constraint of greater than 1 mrem/year must have DOE headquarters written approval.

Table H-1 gives some examples of benchmarks that have been employed by various U.S. agencies for cleanup of radiologically contaminated sites under a variety of circumstances and includes some international benchmarks. Several of these benchmarks include their own optimization procedures, for example the 7 balancing and modifying criteria for CERCLA remedial actions or the ALARA process for NRC decommissioning actions. EPA recommends using the optimization process associated with the applicable benchmark. If the benchmark is not associated with any optimization process, EPA suggests determining with stakeholder input which of the existing optimization processes to use rather than creating a new process.

Table H-1. Examples of U.S. Benchmarks of Potential Use in Evaluating Long-Term Cleanup Options during the Late Phase[a,b]

Example Organizations or Cleanup Programs	Summary of selected program-specific human health protection goals or concepts as applied to the cleanup of radiological contamination.
States NRC Agreement State Decommissioning Programs	Varies across states. Usually, decommissioning programs seek to achieve: • 25 mrem/y primary dose constraint; • 100 mrem/y allowable exemption • Lower levels based on the ALARA concept. • Some States have more stringent dose limits (e.g., 19, 15, or 10 mrem/y)
Environmental Department Contaminated Site Cleanup Programs	Varies across states. Usually, programs seek to achieve risk-based goals or a range of acceptable risk outcomes. Goals typically: • fall within a risk range of 10^{-4} to 10^{-6} excess lifetime cancer risk; and • include meeting existing applicable or relevant environmental regulations/standards. • Some States have single risk-based standards or goals (e.g., 10^{-4}, 10^{-5}, or 10^{-6}).
Federal NRC and DOE decommissioning and site remediation programs	• remediate or decommission to levels that are as low as reasonably achievable not to exceed 25 mrem/y given anticipated use either without restriction or with restriction • if restricted use is anticipated, dose estimates, given assumed failure of restriction (institutional controls), should be less the 100 mrem/y or 500 mrem/y if durable institutional controls are employed. • exemptions to the 25 mrem/y constraint for anticipated use may be approved if conditions are met. For more details see 10 CFR Part 20 Subpart E and DOE 5400.5 and associated guidance.
EPA Superfund remedial site cleanup program	Generally, remedial actions achieve human exposures that meet: • 10^{-4} to 10^{-6} excess cancer risk; • Hazard Index of one for non-cancer toxicity or less; and, • All Applicable or Relevant and Appropriate Requirements (ARARs). These may be waived under specific circumstances. • Cleanup levels should be developed considering the reasonably anticipated land use (For further information see: 40 CFR 300.430)
EPA Uranium and Thorium Mill Tailings Standards	• 5pCi/g surface and 15 pCi/g subsurface for Ra-226 and Ra-228 in soil above background • 0.02 Working Levels (WL) not to exceed 0.03 WL for radon in habitable structures • 20 mR/h above background in habitable structures • options available for supplemental limits to be used under certain conditions (e.g. actions to meet the limits would pose a clear and present risk of injury to the workers or public notwithstanding reasonable measures to avoid the risk). For further details see 40 CFR Part 192 Subpart B: Uranium Mill Tailings Radiation Control Act (UMTRCA).

[a]Table presents examples only. Final cleanup goals and/or actual cleanup outcomes for a particular incident may vary depending on the circumstances of the incident. No single cleanup target is recommended for all possible incidents.
[b]Although many response programs often articulate target cleanup goals or limits in planning guidance, whether these levels are met or exceeded on a response-specific basis generally depends on the program context and the site-specific circumstances. Levels and concepts in this table are presented for illustration only and should not be applied to a specific incident cleanup without a thorough understanding of their derivation and application in the originating programs.

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