Dose Assessment Fundamentals

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1. Introduction

This manual provides an overview of fundamental dose assessment concepts for emergency preparedness personnel responsible for maintaining the station’s dose assessment program, emergency response organization members responsible for performing dose assessments, and emergency response decision makers responsible for determining necessary protective actions. The concepts in this document are not new, but rather a repackaging of existing concepts from various industry guidance documents.

In the early phase, or plume phase, of a radiological emergency, parameters other than projected dose may offer a more proper basis for implementing protective actions. Dose assessors may not have all the real-time inputs needed to estimate projected dose when the plant is operating outside its design basis and a radioactive release into the environment has started, or is imminent. Although initial assessments may have high levels of uncertainty, subsequent assessments should reduce uncertainties as more information on plant conditions and prognosis, effluent radiation monitoring data and environmental data become available. The results of ongoing dose assessments form the basis for refining the initial protective action recommendations (PARs).

The principal protective actions for the plume phase are evacuation and sheltering-in-place. Protective action decisions should weigh the anticipated dose consequence to the affected population against the feasibility of implementing an evacuation safely and in time to avoid unnecessary exposure to radioactive material.

Severe accidents involve conditions that make offsite dose estimates difficult. Plant conditions may exist beyond design basis, unreliable instrument readings may occur, accident progression does not follow an exact timeline, and plant conditions can quickly and unexpectedly change. Early in an accident, available data may limit calculation methods to those that use generic assumptions sacrificing levels of accuracy for improved timeliness. As the accident progresses, more detailed data from sample analysis or field measurements may become available, which the dose assessor can use to refine dose projections.

Dose assessors and decision makers should understand the methods and limitations of the dose assessment model, and be able to interpret dose assessment results in order to make timely and accurate decisions about PARs, emergency exposure controls and issuance of potassium iodide (KI).
2. Definitions

Dose Conversion Factor (DCF)

Coefficient that converts isotopic concentrations in the source term to dose equivalent values for each organ. DCFs are for general use in assessing average person committed doses in any population that is characterized by Reference Man.

Early Phase

The beginning of a radiological incident for which immediate decisions for effective use of protective actions are required, and must therefore be based primarily on the status of the radiological incident and the prognosis for worsening conditions. When available, predictions of radiological conditions in the environment, based on the condition of the source or actual environmental measurements may be used. Protective actions based on the PAGs may be preceded by precautionary actions during the period. This phase may last from hours to days.

Emergency Action Level (EAL)

A pre-determined, site-specific, observable threshold for an Initiating Condition that, when met or exceeded, places the plant in a given emergency classification level.

Emergency Classification Level (ECL)

One of a set of names or titles established by the U.S. Nuclear Regulatory Commission (NRC) for grouping off-normal events or conditions according to (1) potential or actual effects or consequences, and (2) resulting onsite and offsite response actions. The emergency classification levels, in order of severity from lowest to highest, are: Notification of Unusual Event (NOUE), Alert, Site Area Emergency (SAE), and General Emergency (GE).

Evacuation

The urgent removal of people from an area to avoid or lower high-level, short-term exposure from the plume or deposited radioactivity.

Evacuation Time Estimate (ETE)

The estimated time needed to evacuate the public from affected areas of the plume exposure pathway emergency planning zone (EPZ).
Fission Product

The nuclei (fission fragments) formed by the fission of heavy elements, plus the nuclide formed by the fission fragments’ radioactive decay. Fission products are a complex mixture of nuclei belonging to the middle region of the periodic table—barium, strontium, nickel, xenon, iodine—all are radioactive. There are hundreds of possible different fission products produced as nuclear fuel undergoes the fission process in nuclear reactors.

Ingestion Pathway Zone (IPZ)

A geographic area surrounding a commercial nuclear power plant with a radius of about 50 miles from the reactor site. Predetermined plans in place for the IPZ typically include actions to avoid or reduce dose from potential ingestion of radioactive materials. These actions include a ban of contaminated food and water.

Intermediate Phase

The period beginning after the source and releases are brought under control (not necessarily stopped, but no longer growing) and reliable environmental measurements are available for use as a basis for decisions on protective actions and extending until these additional protective actions are no longer needed. 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 lower radiation levels in the environment to acceptable levels are commenced and ending when all recovery actions are complete. This phase may extend from months to years. A protective action guide (PAG) level, or dose to avoid, is not appropriate for long-term cleanup.

Monitor and Prepare

A type of precautionary action intended to tell the public within the EPZ that a serious emergency at the nuclear power plant exists and that the public should monitor the situation and prepare for evacuation, shelter-in-place (SIP), or other protective actions. Further, if an evacuation is underway, officials should ask individuals who are not involved in the evacuation to remain off the roadways to allow those who need to evacuate to do so.
Offsite Response Organizations (OROs)

State, tribal, and local governments with emergency preparedness responsibilities for a commercial nuclear power plant. OROs are responsible for preparing offsite plans and conducting exercises. The Federal Emergency Management Agency (FEMA) evaluates ORO capabilities and preparedness under 44 CFR 350.

Plume Exposure Emergency Planning Zone (EPZ)

A geographic area surrounding a commercial nuclear power plant for which emergency planning is necessary to make sure that offsite response organizations (OROs) can take prompt and effective actions to protect public health and safety during a radiological accident. The plume pathway EPZ is about 10 miles in radius, although the actual size may be larger or smaller depending on site specific licensing basis requirements.

Potassium Iodide (KI)

Provides thyroid protection by partially blocking the uptake of radioiodine by the thyroid. KI is effective only against uptake of radioiodine, and is best taken before or just after exposure. The protective effect of a single dose of KI lasts about 24 hours. ORO health officials direct KI administration until the risk of significant exposure to radioiodine (either by inhalation or ingestion) no longer exists (i.e., once the plume has passed). KI is a supplemental action, secondary to evacuation or sheltering, and is not a substitute for evacuation or sheltering-in-place.

Process Reduction Factors

The various fission product removal mechanisms a radioactive plume encounters as it migrates from the reactor core to the environment. Examples include: filtration systems, spray systems, plate-out, and natural reduction factors.

Protective Action Decision (PAD)

A decision made by OROs on what protective action is necessary to protect public health and safety. Once the decision is made, the applicable ORO relays protective action decisions to the public in a timely manner. OROs take recommendations from the plant emergency response decision makers into consideration along with other information such as impediments and resource availability to make sure they select the most effective protective action.
Protective Action Guide (PAG)

Numerical guides for the principal protective actions available to public officials during a radiological incident. A PAG is the projected dose to an individual from a release of radioactive material at which a specific protective action to lower or avoid that dose is recommended. PAGs are guides to help official’s select protective actions under emergency conditions during which exposures would occur for relatively short time periods. They are not meant as strict numeric criteria, but rather as guidelines in the context of incident specific factors.

Protective Action Recommendation (PAR)

A recommendation made by responsible nuclear power plant staff to the applicable ORO decision maker on how to protect the public. PARs may be based on evaluation of plant conditions and dose assessment results.

Reference Man

A person assumed to have the anatomical and physiological characteristics of an average person. These assumed characteristics are used in calculations assessing internal dose.

Relocation

The removal or continued exclusion of people (households) from contaminated areas to avoid chronic radiation exposure.

Shelter-In-Place (SIP)

A type of protective action in which instructions are given to members of the public to stay indoors, turn off heating or air conditioning (per the region and season), close windows, monitor communication channels, and prepare to evacuate. Those people who are not at home (shopping, dining, or working) should stay in their current location. SIP is typically used when it is safer to shelter than evacuate or to make sure that the public remains off roadways to allow other areas that are under an evacuation order to evacuate unimpeded. The intent is that members of the public should remain where they are or should seek shelter close by, but they should not return home to shelter.
3. Dose Assessment Overview

Emergency response personnel use dose assessment to estimate the size and impact of the release of radioactive materials to the public and emergency workers. Dose assessment includes the projected size and location of doses, allowing decision makers to take actions to protect the health and safety of the public by preventing unnecessary exposure.

Emergency response personnel use dose assessment results to classify emergency conditions based on offsite radiological impacts. This is to evaluate protective measures for onsite emergency workers, and to formulate PARs for consideration by responsible Offsite Response Organizations (OROs).

As a condition of operating, licensees maintain the ability to perform dose assessment using effluent release information and real-time meteorology. This is done through use of computer models or other calculation methods such as hand calculations or spreadsheets. On-shift dose assessment may use simplified methods requiring minimal operator input or easy-to-use graphs, charts, etc. in order to lower the burden for on-shift personnel. These simplified dose assessment methods still need to take into account effluent release data and real-time meteorological conditions such as wind speed, wind direction, and atmospheric stability (NRC EPPOS No. 3).

The augmented Emergency Response Organization (ERO) should perform more advanced dose assessments. By the time the augmented ERO arrives, more information on plant conditions and meteorology may be available to refine the initial dose assessment. The augmented ERO dose assessments may improve the level of precision of projected dose to the public, as well as begin the process of calculating total integrated dose to affected areas, and provide a basis for refining PARs.

The dose assessor uses various inputs such as meteorological data, the type of accident occurring, and release rate to estimate the dose impacts to the public from a radioactive release. Most stations use computerized dose assessment models that employ complex algorithms as a way of expressing the actual mixing of released radioactive material in the atmosphere based on various assumptions and inputs. Understanding the underlying assumptions and inputs, and how they influence the output of the dose assessment model is critically important.
## 4. Dose Assessment Example

The following is a simplified look at the dose assessment process. Each of the following sections of this manual discusses the underlying concepts involved in each of these steps.

<table>
<thead>
<tr>
<th>Step</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Abnormal conditions exist indicating potential emergency plan entry</td>
<td>An accident occurs involving the actual or potential release of radioactive material to the environment</td>
</tr>
<tr>
<td>On-shift staff or ERO identify release to the environment</td>
<td>Emergency response personnel determine if a radiological release is in progress by observing plant indications and use of field team reports</td>
</tr>
<tr>
<td>Determine release path</td>
<td>Once personnel confirm a release is in progress, dose assessors use radiation monitors and plant indications to determine the release path</td>
</tr>
</tbody>
</table>
| Perform dose assessment | The dose assessor performs an assessment of the release in progress using a computerized dose assessment program or manual calculations, with inputs for:  
• Source term  
• Release Pathway  
• Process Reduction Factors  
• Effluent Flow Rate  
• Radiological Data  
• Meteorological Data  
• Release Duration |
| Compare results against EALs and PAGs | The dose assessor compares dose assessment results against emergency actions levels (EALs) and protective action guides (PAGs) |
| Decision makers decide if classification or PAR is required | Emergency response decision makers use dose assessment results to evaluate the need to classify emergencies and determine if PARs are necessary for public health and safety |
| Dose assessment continues for duration of release | Dose assessment continues for the duration of the release with all inputs to the dose assessment model continuously monitored and updated when conditions change or new information becomes available |
| Field team data used to validate model | The dose assessor uses field team data to validate the dose assessment model results or as an input during an unmonitored release |
5. Identifying a Release in Progress

Identifying a release in progress is important to both the dose assessment team and those responsible for communicating accurate information to the public. Radioactive release status is information required as part of the initial and follow-up emergency notifications to OROs as part of the Nuclear Regulatory Commission (NRC) risk significant planning standards.

An incorrect determination of radioactive release status jeopardizes credibility, and may result in ineffective protective measures for the public. As an example, during an actual emergency, a utility spokesperson communicated there was no release in progress during a steam generator tube rupture (SGTR), and the NRC commission later communicated that there was a release. This resulted in public trust concerns with the utility and OROs (OIG Case No. 00-03S). An important lesson learned is to make sure the site-specific definition of a release accounts for scenarios with an elevated release that is within “normal” permissible limits.

The NRC does not define what a release in progress is, specific to emergency planning requirements. Nuclear Energy Institute (NEI) 99-02, Regulatory Assessment Performance Indicator Guideline (rev. 7), defines a release as “a radiological release attributable to the emergency event.” It is important to note that the release thresholds selected for the purposes of communicating release status to OROs during a declared emergency may differ from thresholds used in other licensee programs, such as the Offsite Dose Calculation Manual (ODCM) and technical specifications.

The dose assessor should understand the site-specific release in progress definition and have a thorough understanding of indications available to evaluate release status, including:

- Specific effluent radiation monitors for monitored release paths
- Direct field measurements (portable survey instrument or air sample)
- Indications of an unmonitored radioactive release
- How to decide if the information is valid (such as not attributable to shine)

Dose assessment is only required if there is a valid pathway between a source of radioactive material and the environment. Some sites have dose assessors do “what if” assessments based on potential release pathways. It is important that these “what if” assessments are clearly marked as such to avoid misunderstandings and to prevent actions based on hypothetical information.
6. Fission Product Barriers and Release Pathways

During normal reactor operations, the principle source of radiation exposure is from irradiated corrosion and wear product deposition commonly known as crud. Although other exposure sources are present, engineering, design and administrative controls minimize on-line exposure to plant personnel. Survey techniques are typically limited to gamma radiation surveys of components such as piping, valves and tanks. Plant staff typically only encounter beta radiation when they open plant components. Beta radiation is also typically found in areas with high levels of fixed or loose surface contamination. Particulate airborne radioactive contamination can occur requiring respiratory protection if concentrations are high or during some aggressive maintenance activities.

Plant design effectively contains the millions of curies of fission products contained within the nuclear fuel by the use of three fission product barriers: fuel cladding, reactor coolant system, and the containment building. During maintenance periods when entry into containment structures are performed, the biological reactor vessel shielding is effective at reducing the radiation levels from fission products contained within the fuel cladding to a minimal source of radiation exposure. The intact cladding contains the radioactive material and the deep fuel pools provide shielding to limit radiation exposure from spent fuel stored within spent fuel pools.

During a reactor accident, rapidly changing conditions may compromise the fission product barriers resulting in the normally contained fission products migrating outside the designed structures. This shifts the source of exposure away from corrosion and wear products to the migrating fission products. The order of magnitude of the shift is a function of the degree of fuel damage as well as the extent of the fission barriers compromised.

In order to understand how fission products migrate, and ultimately the impact on plant worker and general public exposure during a reactor or spent fuel accident, we need to understand the composition of the fission products. The chemical properties of the fission products dictate the migration pathways. As we look at typical nuclear fuel composition, we know that stacks of ceramic fuel pellets with fissionable material (3-5% enriched U-235) are sealed inside thin metal tubes (cladding) made of zirconium alloys. Nuclear fission occurs when thermal neutron radiation bombards fissionable material such as U-235. As the fuel absorbs thermal neutrons, the fission process occurs within the ceramic pellets. The uranium fuel atoms fission or split into fission products (atoms), gamma radiation, neutron radiation and heat. Typically, the majority of the fission products remain locked inside the ceramic pellet.

Although the physical properties of fission products for each nuclear fission are random, overall there is a predictable distribution of fission products based on the total number of nucleons (number of neutrons and protons within the nucleus of an atom). The following figure shows a standard fission yield curve with two peaks indicating the most probable total number of nucleons. We see...
these peaks are at about 90 and 138 total nucleons. This corresponds to various isotopes including the radioactive gases such as Kr-88 and Xe-138, radioactive particulates such as Rb-88 and Cs-137, as well as radioactive iodine isotopes such as I-131, I-135 and I-136. Of course, the exact chemical element is dependent on the number of protons, and the number of neutrons determines the isotope.

A. Fission Product Barriers

Most of the particulate fission products including radioactive iodine remain locked inside the ceramic fuel pellet. Some gaseous fission products created during fission escape through thermally induced cracks in the fuel pellets. The rate of release increases with temperature. The fission product gases that escapes the fuel pellet ends up in the gap formed by the fuel cladding or the fuel pin plenum. Some of the fission products will undergo radioactive decay where decay chains include radioactive iodine. In simple terms, the ceramic fuel pellets lock a high percentage of particulate fission products and radioactive iodine inside while the gap space inside the cladding contains the noble gases and a smaller concentration of radioactive iodine.

Fuel pellet damage due to overheating results in the fission products typically locked inside the ceramic pellet being released. This is referred to as fuel melt. Fuel melt results in the release of the fission products that had been locked inside the ceramic pellets. The isotopes associated with fuel
melt typically include a higher concentration of radioactive iodine and more of the particulate fission products. The particulate fission product mix from fuel melt includes nuclides not typically seen in standard waste stream analysis such as Promethium, Samarium or Cerium. Identification of these rare nuclides can help determine the magnitude of fuel damage during a reactor accident.

One thing to keep in mind is that there will always be clad damage when there is fuel pellet damage. Even if there was a mechanism by which the ceramic pellet is damaged resulting in the release of fission products, if the cladding is intact then the fission products are locked inside. By design, the ceramic pellet has a much higher melting point than the cladding. If the fuel pellet heats to the point of damage, it would also damage the cladding. When the fuel pellet is damaged to the point of releasing fission products, there are more particulate and iodine nuclides than noble gases.
During accident conditions, radionuclides (halogens, and noble gases) may escape from a fuel pellet and eventually find their way to the environment. For the fission products generated within the core to reach the environment, they must pass through multiple fission product barriers. This concept relies on multiple physical barriers, any one of which, if maintained intact, precludes the release of significant amounts of radioactive fission products to the environment. The primary fission product barriers are: Fuel cladding, the Reactor Coolant System (RCS), and the containment building.

**Fuel Cladding**

Fuel cladding refers to the zirconium alloy (stainless steel) tubes that house the fuel pellets. The zirconium cladding retains most of the fission products. The cladding helps contain the fuel and fission products within the fuel assembly. Gaseous fission products created during power production escape through thermal induced cracks in the fuel pellets. The rate of release increases with temperature. The fission product gas that escapes the fuel pellet ends up in the gap formed by the fuel cladding or the fuel pin plenum. The fission products collected in the voids and plenum is often referred to as gas gap activity or simply gap activity.

Although fuel fabrication rules are stringent and use high quality assurance procedures to reduce manufacturing defects, fuel failures can take place during reactor operations. In addition, limited fuel failures can take place due to other operational phenomena such as pellet cladding interaction and crud deposition.

Design criteria for fuel rods consider fuel integrity during normal operations and during off-normal events. Fuel design limits are prescribed to ensure cladding integrity under the severe reactor operating conditions and during off-normal conditions. There are design limits set by the NRC for cladding temperatures and heat fluxes, in addition to limits on cladding oxidation and hydrogen generation from chemical reaction between water/steam with cladding. The cladding temperature limit is 2200°F (1204°C) for zircalloy cladding of Light Water Reactor (LWR) fuel. Scenarios when such a high temperature limit is exceeded include a loss of coolant accident (LOCA) with possibility of evaporation of the water cooling the fuel rods, as the water temperature rise while heat generation in the fuel continues (even if the reactor is shut down, heat generation from decay heat, at the rate of a few percent of the fission heat generation, can lead to overheating of the cladding).

**Reactor Coolant System**

The RCS includes the reactor vessel and attached piping that contains the coolant, usually water, which flows through the core. The coolant transfers the core’s fuel heat to the feedwater to produce steam for the turbine. If fission products escape through the fuel rod cladding, they are generally contained within the RCS. The RCS provides a physical barrier between the radionuclides contained within the reactor coolant and the atmosphere. Atmosphere as it relates to the RCS barrier includes containment atmosphere as well as any atmosphere around RCS components outside of
containment. As long as the RCS barrier is intact, there is no potential for offsite dose impacts because the fission products have no direct pathway to the environment.

Fission products that escape the fuel pellet and accumulate in the “gap” region of the fuel cladding, release into the RCS through manufacturing defects or small pinhole leaks. Under normal plant operations, cleanup systems remove these fission products and other radionuclides to keep the reactor coolant activity within technical specification limits. Under accident conditions, the normal cleanup systems cannot keep up with the volume of activity released into the reactor coolant.

**Containment Building**

In the event that the other barriers are lost, the containment barrier provides the last line of defense. The design features of containment buildings, which include the ability to withstand postulated accident conditions such as the major pressure increases associated with a large break LOCA, prevent the escape of fission products to the environment. Loss of containment provides a release pathway for the fission products to the environment and ultimately to members of public. Some emergency operating procedures and severe accident guidelines direct operations to intentionally vent containment to control pressure and maintain containment integrity. Intentional venting may result in a temporary loss of containment and a release to the environment that requires a dose assessment to quantify the public impacts.

**B. Determination of Release Pathways**

The various release pathways fall into three generic categories: (1) containment leakage, (2) containment bypass, and (3) steam generator tube ruptures. There are some differences within these categories between pressurized water reactors (PWRs) and boiling water reactors (BWRs). Refer to Section 7, Source Term – D. Fission Products Removed on the way to the Environment, for more details.

Accurately determining the release pathway is essential to understanding the potential offsite consequences of a radiological release. Certain release pathways contain removal mechanisms or process reduction factors that affect the output of the dose assessment. Examples of process reduction factors include filtration systems, spray systems, and natural processes such as plate out. Selecting an incorrect release pathway has the potential to result in misapplication of these reduction factors, resulting in an inaccurate dose projection and potentially unwarranted or unrecognized PAR.

**Containment Leakage**

Dissolved fission product gases from the RCS pass through the containment atmosphere first, before entering the environment. For example, a LOCA results in the release of coolant from the RCS into the containment atmosphere. Containment building leakage then occurs resulting in the transport of the gaseous radioactive material from inside containment to outside containment. This release may
go directly to the environment or into an enclosed structure, which vents to the environment such as an auxiliary building. Examples of containment leakage pathways include:

- Stuck open valves such as sampling valves, purge exhaust valves, isolation valves including main steam isolation valves in BWRs
- Failed penetrations
- Failed airlock or equipment hatch seals
- Catastrophic containment failure due to hydrogen explosion or over-pressurization
- Loss of secondary containment (BWR)

**Containment Bypass**

Dissolved fission product gases from the RCS bypass the containment atmosphere. A number of low-pressure systems connect to the RCS. In some cases, low-pressure lines and components like pumps are physically outside containment and separated from the RCS by check valves and isolation valves. If these valves fail, the high-pressure coolant from the RCS could rupture the low-pressure system outside containment. This would result in a coolant release from the RCS to an auxiliary building or directly into the environment without first passing through the containment atmosphere.

- As an example, an in service emergency core cooling system (ECCS) pump develops a leak resulting in the transport of radioactive material directly from the RCS into the auxiliary building. Then, ventilation systems in the auxiliary building transport the radioactive material through the plant vent into the environment.

**Steam Generator Tube Ruptures (SGTR)**

Dissolved fission product gases from the RCS pass into the secondary side through the secondary relief valves, turbine exhaust, condenser steam-jet air-ejector exhaust, or directly through cracks in a pipe or a fitting. The ruptured steam generator may release steam directly into the environment or into an enclosed structure, which vents to the environment.

- As an example, a SGTR occurs. The contaminated main steam goes to the condenser, before main-steam isolation, where the steam-jet air-ejectors remove the radioactive gases and discharge the remaining material into the environment either directly or through the plant vent stack.

**Other Potential Release Pathways**

Not all release pathways at a nuclear power plant originate from an operating reactor core. Many plant designs include waste gas decay tanks or liquid holdup tanks that are potential sources of a radioactive release if a tank ruptures. Spent fuel pool accidents are another possible release pathway. In some cases, ventilation systems may direct radioactive releases from these events through the plant vent stack or through other designed safety system ventilation components.
Sites with an Independent Spent Fuel Storage Installation (ISFSI) should have the ability to assess the consequences of a release originating from a damaged cask or storage container. The offsite impacts from this accident type are lower than the other types of accidents.

**Multi-Unit & Multi-Source Term Release Pathways**

The accident at Fukushima Dai-ichi presented challenges with respect to dose assessment because of the multi-unit nature of the release. While monitoring the accident, dose assessors at the NRC had to use makeshift, ad hoc methods to consider the source term from these multiple concurrent releases and overlay the release points to arrive at a site-wide dose projection. The difficulty of conducting this dose assessment highlighted a potential gap in ability for some U.S. plants to do multi-unit dose assessment.

At the time, most dose assessment software tools in use such as computer code Radiological Assessment System for Consequence AnaLysis (RASCAL) were not designed to model multi-unit accidents. The presence of releases from multiple units and spent fuel pools at Fukushima highlighted the need for the ability to project doses from releases at multiple units. Without a software solution, licensees had to develop manual hand calculation guidance that outlined how to conduct a multi-unit dose assessment, including spent fuel pools as release points.

In the NRC’s Near-Term Task Force (NTTF) Report, “Recommendations for Enhancing Reactor Safety in the 21st Century,” issued July 12, 2011 (Agency wide Documents Access and Management System (ADAMS) Accession No. ML 111861807), the NTTF recommended that an order be issued requiring licensees to develop and add guidance to their emergency plans about how to do multi-unit dose assessments.

All operating licensees submitted letters to the NRC on multi-unit and/or multi-source dose assessment capabilities. These multi-unit/multi-source dose assessment capabilities are typically automated rather than relying on manual calculation. The automated method should demonstrate the ability to model offsite dose assessment conducive to developing PARs when presented with a multi-unit/multi-source event.
7. Source Term

A. Defining Source Term

In its simplest form, source term is the types and amounts of radioactive or hazardous material released into the environment following an accident. Taking this concept one-step further, source term combines multiple estimates of:

- The total inventory of fission products in the core
- The fission product inventory released from the core for a given accident type
- The fission product inventory released from the core that does not make it into the environment – based on the release pathway and associated removal mechanisms that will act to lower the release potential
- The release rate

Source term is dynamic as the combination of nuclides available for release changes during the movement of fission products from the core into the environment.

B. Total Inventory of Fission Products in the Core (Core Inventory)

To understand source term in reactor accidents, it is necessary to understand how the various radionuclides originate and why they are hazardous to human health. The first consideration in determining the impact of a particular fission product to the overall source term is how much of the radionuclide is available in the core at the time of the accident. A number of factors such as burnup, power density, power level, and reactor type influence fission product inventory making this is a difficult estimate.

The next consideration in determining the impact of a particular fission product to the overall source term is the level of impact to offsite dose consequences. Source terms do not include all possible radionuclides because many nuclides contribute little or nothing to offsite dose consequences. The major groups of fission product isotopes included in most source terms are:

- Noble gases (Krypton and Xenon)
- Halogens (Iodine and Bromine)
- Alkali metals (Cesium and Rubidium)
- Tellurium group (Tellurium, Antimony, Selenium)
- Barium and Strontium
- Noble metals (Ruthenium, Rhodium, Palladium, Molybdenum, Technetium, Cobalt)
- Cerium group (Cerium, Plutonium, Neptunium)
• Lanthanides (Lanthanum, Zirconium, Neodymium, Europium, Niobium, Promethium, Samarium, Yttrium, Curium, Americium)

A number of studies in reactor safety suggests that most of the noble gases (xenon, krypton) make a small contribution to long-term radiological health effects. However, noble gases are the most likely group of fission products released to the environment following a severe accident because they are chemically inert, are available in large quantities, and would not be affected by the various reduction mechanisms that would remove other fission products before they reach the environment. For this reason, noble gases are a primary contributor to short-term dose impacts during the plume phase. Depending on the release pathway and process reduction mechanisms applicable to that release pathway, iodine may also be a major contributor to offsite dose consequence.

Generic factors typically accounted for when stations develop core inventories for source term calculations include, curies per megawatts thermal, megawatt days per metric ton of uranium (MWd/MTU), enrichment level of the fuel, radionuclide half-lives, etc. The dose assessor does not directly develop the core inventory, however, dose assessors should know and understand the basis for the core inventory estimates used in their site-specific dose assessment model. Common methods used to develop core inventory estimates include:

• Accident analysis contained in the Final Safety Analysis Report (FSAR) – Licensees analyze various design basis accidents (DBAs), some of which result in offsite releases. Part of the licensing process requires each applicant for a construction permit or operating license to provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the goal of assessing the risk to public health and safety resulting from operation of the facility. Section 15 of the FSAR lists these postulated accidents with a corresponding inventory and relative abundance of nuclides. The DBAs were not intended to represent real event sequences, but rather, to give a framework to enable deterministic evaluation of the response of a station’s engineered safety features. These accident analyses are intentionally conservative in order to compensate for known uncertainties in accident progression, fission product transport, and atmospheric dispersion. The result is that for the FSAR accident conditions analyzed, the projected offsite doses are likely much greater than projections based on actual plant conditions at the time of an event.

• Severe accident consequence studies – This method uses available data from reactor safety studies and employs complex calculations to develop a list of postulated radionuclides and relative abundances. These calculations result in data that is generically representative of a typical core inventory. For example, RASCAL uses normalized radionuclide core inventories (curies per megawatts thermal) based on calculations made by NRC staff in December 2003 using the SAS2H control module of SCALE (Standardized Computer Analyses for Licensing Evaluation). The inventories do not include radionuclides with half-lives of 10 minutes or less.
However, where it makes sense, the inventories include short-lived daughters with their longer-lived parents for dose calculations.

- Site-specific core inventory – Some sites take an individualized approach to develop estimated core inventories using a combination of methods described above, or by developing their own method based on plant operating experience. For example, some sites have experience with operational transients that resulted in levels of fuel degradation that they were able to sample and analyze. The site used the results of these analyses to refine available core inventory and relative abundances postulated in the FSAR. Some sites use their routine RCS sample results to keep a near real-time nuclide inventory list for their normal RCS source term.

C. Fission Product Inventory Released from the Core

The fission product inventory released from the core for a given accident type correlates to the level of fuel damage. Overheating the core and mechanical damage are the main mechanisms of fuel failure. Engineering typically performs core damage assessment using available indications such as reactor shutdown time to account for radiological decay and in-growth, containment radiation readings, area radiation monitors in ECCS rooms, thermocouple temperatures, and RCS sample results.

Methods such as containment hydrogen concentrations and the amount of time that a reactor core is uncovered aid engineers in determining the type and degree of core damage. Dose assessors should understand the indications, methods, and tools engineering uses to give core damage estimates, because the dose assessment model will use a different set of nuclides and relative abundances based upon the fuel damage estimate.

Many dose assessment models characterize the fraction of fission products released from the core as one of the following.

**Normal Coolant**

Conditions are within normal operating temperatures (≤600°F) and there are no abnormal indications on radiation monitors.

**Spiked Coolant**

Under normal operating conditions, equilibrium in pressure between the RCS and the fuel cladding gap region keeps the fission product gases within the fuel cladding. A reactor transient, or response to a reactor transient (changes in reactor power levels), may cause RCS pressure to rapidly change, causing a temporary increase, or “spike,” in the escape rate of fission products from the gap region of the fuel cladding into the RCS. Spiking factors are only applied when using normal coolant as a
source term. The radioactivity order of magnitude when using clad damage or core melt source
terms makes spiking factors insignificant.

Coolant water may also enter fuel rods through cladding defects. If the RCS pressure suddenly
decreases, this water could leach off iodine and cesium salts deposited on the inner cladding
surfaces, thus increasing the iodine and cesium available for escape during the transient. This
phenomenon typically only occurs during accident scenarios with fuel damage within tech spec
limits, because scenarios with any real fuel damage make the increase in activity due to spiking
insignificant.

**Cladding Failure**

Coolant temperatures are in excess of 1200°F up to 2100°F, which correspond to fuel cladding
temperatures high enough to result in the failure of the fuel pins/walls. If the fission product gas
pressure within the fuel pins is considerably less than the primary system pressure, the cladding may
buckle or collapse at about 1300°F. The cladding may balloon and oxidation may become extremely
rapid between 1400°F and 2000°F, leading to rapid fuel pin failure. After a fuel pin cladding failure
occurs, most of the fission gas in the fission gas plenum releases into the RCS. Fuel temperature
throughout the core could vary considerably, which is why use of core temperature by itself may not
provide the best indication. A key indicator of a gap release during a LOCA into containment are the
post-accident high range containment radiation monitors.

**Core Melt**

Coolant temperatures are in excess of 3000°F, or engineering’s core damage assessment indicates
more than 100 percent gap release or localized fuel melt. Localized fuel melt is possible without 100
percent fuel clad failure. Engineering and dose assessors may see indications of this degree of core
damage on core exit thermocouple readings (PWR), containment hydrogen monitors or post-
accident high range containment radiation monitors. Once the cladding fails the release rate of
various fission products increases rapidly with temperature. Fuel melt could occur in as little as an
hour once the fuel is uncovered.

**Spent Fuel**

Spent fuel accidents typically have their own defined source term based on the type of spent fuel
storage accident. The most common spent fuel storage accidents include loss of spent fuel pool
inventory and mechanical fuel damage underwater (such as dropped bundle or fuel handling
accident). Fuel handling accidents may only involve longer lived noble gases such as Krypton-85,
which is a beta emitter and likely have minimal or no risk to the public. The important aspect of this
type of event is knowledge of how long the affected fuel assembly(s) have been out of the core.
• Cold gap – fuel damage occurs without excess heating such as a dropped bundle underwater or mechanical damage to the cladding within an ISFSI container.

• Hot gap – damage to spent fuel occurs and cooling is lost for longer than the specified thermal limit, but before the temperature reaches the threshold for oxidation of the zirconium cladding.

• Clad burning (zirc fire) – Spent fuel in a spent fuel pool must stay covered with water or otherwise cooled to remove decay heat, or the zirconium cladding may heat up and undergo rapid oxidation or “burning” that will eventually spread to adjacent assemblies in the pool.

D. Fission Products Removed on the way to the Environment

The release pathway through which the radioactive fission products migrate from the reactor core to the environment influences concentrations of the various fission products. As the fission products migrate from the core towards the environment, multiple process reduction factors change the nuclide mix including properties and quantity of nuclides. For example, as a pipe leaks into a building, the volume of radioactivity changes through atmospheric dilution and the time the radioactive material resides inside the building (hold-up time), allows for the decay of short-lived fission products.

To estimate the amount of fission products released from the core that actually reach the environment, the dose assessor should account for the effectiveness of the various fission product process reduction factors applicable to the selected release pathway. The effectiveness of these reduction factors could remove as much as 99 percent of the nuclides reaching the environment.

Plate Out in the RCS

Depending on the conditions in the system through which the fission products pass, large quantities of the aerosols and particulates could condense and plate out (chemically adhere) on system surfaces like piping. There are many factors influencing this affect including:

• surface area and temperature of the interfacing system components
• coatings
• flow rates
• particle size
• chemical properties of nuclides

Gravitational Deposition and Hold-Up Time

Natural processes such as gravitational deposition lower the airborne concentrations of heavier aerosols and particulates depending on how long the fission products stay within plant structures before release to the environment.
Hold-up time is the time the radioactive material resides inside a building (such as containment or an auxiliary building), before release to the environment. Hold-up time allows for the decay of short-lived fission products and is the primary way to lower offsite dose consequences attributable to noble gases. Hold-up time relies on volume and ventilation flow rate. The larger the volume of a specific building or containment structure, the longer the radioisotopes remain in the structure. The lower the volumetric flow rate of the ventilation systems, the longer the radioisotopes remain inside the structure.

Normal and emergency ventilation systems within these buildings contribute to hold-up time. Dose assessors should understand how normal and emergency ventilation systems respond during accident conditions for applicable release pathways, including during electrical power transients. For example, a fan is assumed running, providing a flow path for a release path to a filtered stack. However, when offsite power is lost, the booster fan stops leaving no force to move the gaseous release to the filtered vent. The dose assessor may need to adjust the hold-up time in this scenario to account for changes in plant conditions.

**Spray Removal Reduction Factors**

Reactor containment buildings have spray systems designed to entrap airborne fission products, and to condense steam to prevent over-pressurization following a LOCA. Other large structures, such as the auxiliary building, may have spray systems that could remove airborne fission products as well. The effectiveness of spray reduction factors typically increase with time based on the changing characteristics of core inventory release fractions.

**Filtration Systems**

Aerosol and particulate fission products released from the core may meet a number of filtration systems before release to the environment. These systems are likely effective during design basis accidents. However, filters will lose their effectiveness as they saturate with aerosols and particulates during large releases with long release durations. The level of moisture in the release may also challenge the effectiveness of the filters, as the filter material gets wet.

- **BWR Standby Gas Treatment System (SBGTS)** - The SBGTS collects and filters releases from the BWR primary containment to the plant’s stack. SBGTS use HEPA and charcoal systems to filter particulate and iodine as well as provide some degree of noble gas hold-up to allow some amount of noble gas decay prior to release.

- **Containment Re-circulating Filter System** - Many plants have containment air re-circulating systems to trap fission product iodine following an accident. These systems have moisture separators to remove water droplets; however, they do not remove large quantities of aerosols expected in the containment atmosphere during a severe core damage accident.
• Auxillary Building Filter Systems - These filters treat exhaust air from equipment areas and locations outside containment. As with other filter systems, these systems use high-efficiency particulate air (HEPA) and activated charcoal absorbers and are very effective in removing particulates and aerosols. These filters are susceptible to clogging depending on the accident conditions.

BWR Suppression Pool

Some BWRs have pressure suppression pools as part of their containment’s, designed to condense steam following a LOCA. The suppression pool is a large mass of water stored inside a torus structure or water pool depending on the model of the BWR. Steam relief valves (safety or electrically controlled relief valves) discharge into the water volume to condense the steam to limit containment pressure transients. Although the suppression pool is not a designed fission product removal system, a benefit of the suppression pool process is the scrubbing of fission products as the steam passes through the pool. Additionally, many post Fukushima installed hardened vent systems use torus suppression pool iodine scrubbing as the primary filtration mechanism required by regulation. The system's ability to remove fission products will depend on the energy of the steam and the ambient pool temperature.

PWR Ice Condenser

Ice condenser containment’s are an alternative to the large-volume containment traditionally used for PWRs. Like the large, dry containment, the ice condenser containment can accommodate the large steam quantities associated with a LOCA. However, the ice condenser design permits a much smaller containment (50 percent less volume). In addition to suppressing the rise in containment pressure following a LOCA, the ice condenser also reduces the fission products in the containment air by entrapment and dissolution as the steam melts the ice as it passes by.

Steam Generator Partitioning

When the steam generator (SG) tubes fail, a pathway directly to the atmosphere that bypasses containment may exist. The fission products from the higher-pressure primary coolant system pass into the secondary side and possibly into the atmosphere. The steam generator’s ability to partition out fission products will depend on where the tube failure is.

• If the failure is below the secondary side water level in the SG, there is a large partition factor, such as 50. In other words, non-noble gas concentration in the steam is one-fiftieth (0.02) of the concentration in the SG water.

• If the break is above the secondary side water level in the SG, the partition factor is much lower, such as two. In other words, non-noble gas concentration in the steam is one-half (0.5) of the concentration in the SG water.
The partition factors account for the slowing of the release of radionuclides from the SG, but SG partitioning does not prevent the release of radioactive material. Additionally, if the ruptured generator overfills, the partitioning factor will be negligible as all fission products in the water carry over to the release point where the water flashes to steam sending the fission products directly to the atmosphere.

**Liquid Releases**

Liquid releases such as RCS or other contaminated water may transport fission products to the atmosphere as well. The levels of contamination could be very high. For example, a severely damaged core could be re-flooded during extended cool down operations (cold leg re-circulation) resulting in vastly increased fission product concentrations in the reactor coolant.

Another source of contaminated water could be the extra water in containment following spray operations to scrub fission products from the containment atmosphere. In these cases, contaminants released to the atmosphere is a function of the rate the fission products escape from the liquid through evaporation. Just like gaseous releases, liquid releases require a pathway to the environment before they pose a risk to offsite dose consequence.
8. Release Start Time, Release Rate, and Release Duration

A. Release Start Time

Release pathways discussed earlier focused on the various ways in which radioactive material migrates from the fuel pellet to the environment. Identifying when the release into the environment starts is an important input to dose assessment because source term, applicable reduction factors, and exposure duration are dependent upon this time.

B. Release Rate

Release rate, measured in Curies per second (Ci/sec), is the radionuclide concentration released to the environment taking into account the flowrate. Licensees installed radiation monitors capable of measuring the noble gases released through plant vents after the Three Mile Island (TMI) accident. For a monitored release pathway, the radiation monitor provides a real-time activity measurement (for example counts/min, microcuries/cc, mR/hr) of the release pathway.

The dose assessment model uses the measured activity value and applies assumptions about the applicable radiation monitor efficiency (based on site-specific calibration standards) and isotopic mixture (based on selected source term) to estimate the total nuclide concentration of the release. The dose assessment model then uses applicable flow indications such as plant vent flow, steam generator flow, or containment leak rate to assign a value for the radioactive material released to the environment, per unit of time. Using these inputs for the radiation monitor data and the flowrate, the dose assessment model calculates the estimated release rate.

Since most radiation monitors used for the purposes of dose assessment only measure one family of radionuclides (noble gases, iodines, or particulates), radioiodine and particulate release rates are not typically available as real-time values under accident conditions. There may be a delay in getting the radioiodine and particulate release rates due to the time it takes to collect a sample, transport it to the onsite lab, and count the sample. To account for this, many dose assessment models use conversion constants assigned to the noble gas radiation monitor to provide a radioiodine to noble gas ratio to ensure a timely initial dose assessment.

There are some disadvantages to using radiation monitor data. For example, the radionuclide mixture released is likely not identical to that assumed in the calibration for the selected radiation monitor, and various accident conditions may cause the selected radiation monitor to report erroneous readings (for example contamination of the detector or shine). For these reasons, and others discussed later, use of field survey results is necessary for validating dose projections.
Changes in the release rate occur primarily because of changes the accident that affect escape rates of fission products such as temperature changes, pressure changes, and new transient conditions. Ongoing monitoring and interpretation of key plant safety parameters is critical to detecting potential changes in release rates.

Another factor affecting the release rate is the reactor shutdown time. Because the reactor is shutdown, it is not producing any new fission products. Radiological decay and the release of these decay products (in-growth) to the containment deplete the radionuclide inventory of the fuel during each time step. As expected, the activity fractions for short-lived isotopes decrease with increasing time after shutdown. Consequently, the fractions of long-lived isotopes increase. These changes have implications that may be counterintuitive. For a given total activity release, increasing the time after shutdown increases the inhalation doses because of the larger fraction of certain isotopes, such as I-131. In contrast, increasing the time after shutdown decreases the cloud shine because the activity fraction of isotopes, such as xenon (Xe)-135m and Xe-138, decreases.

C. Release Duration

Dose assessments are forward-looking because dose avoidance is the goal of protective action decisions (PADs) for the public. Release rate multiplied by release duration provides an estimate of the total quantity of radiological material released to the environment. The dose assessment model then applies meteorological data to figure plume dispersion. Plume dispersion contributes to where the plume travels, how far the plume travels, how much of the available radioactive material deposits at various distances, and what the dose impact is at these various locations. Plume dispersion is discussed in detail later on.

During the first dose assessment runs of an actual event, the dose assessor may find it difficult to determine what the release duration will be. For this reason, many dose assessment methods use default release durations based on the type of accident and meteorological persistence studies. These default release durations may range from 15 minutes for a puff release, to multiple hours for a severe accident.

Some dose assessment models assume that regardless of actual release duration, the public will only be exposed to the plume during the evacuation time frame using the time of exposure (time to evacuate) as the release duration. Use of these default release durations allows for timely completion of initial dose assessments while emergency responders estimate a more accurate release duration.

The key point is that the dose assessment process should not rely on default release durations for the entirety of an event. The dose assessor should actively assess the situation and try to get refined release durations from engineering or operations, based on response actions in progress such as repair team status and plant cool down rates.
9. Plume Transport and Diffusion

Up until this point, we have discussed the impacts of release pathways, source term, and release rates and durations on dose assessment results. This section will focus on the many factors that influence the transport and dispersion of the plume.

Plume transport is where the radioactive material travels once it reaches the environment. Once the plume reaches the environment a number of forces interact with the plume resulting in the movement of radioactive material from regions of higher concentrations to regions of lower concentrations. This movement of radioactive material within the environment is known as diffusion. Transport, along with diffusion, will dictate the footprint on the ground of deposited radioactive material. As the terrain gets more complex, these concepts take on more significance.

Transport and diffusion of radioactive materials from a release are directly influenced by the state of the atmosphere along the plume transport path, the topography of the region, and the characteristics of the effluents themselves. Radioactive material concentrations in the surrounding region depends on the:

- amount of activity released
- height of the release
- momentum and buoyancy of the emitted plume
- wind speed, atmospheric stability, and airflow patterns of the site
- various removal mechanisms

Geographic features such as hills, valleys, and large bodies of water greatly influence dispersion and airflow patterns. Surface roughness, including vegetative cover, affects the degree of turbulent mixing. Sites with similar topographical and climatological features can have similar dispersion and airflow patterns, but detailed dispersion patterns are usually unique for each site.

A. Defining the Plume

The term “plume” most commonly characterizes radioactive materials released to the environment. The plume often appears as a cone, which expands with distance as transport and diffusion processes interact with the radioactive material. The "centerline" of the plume is the direction along which the largest concentration of radioactive material exists. Field monitoring teams use plume centerline as a target for identifying the plume location to take decay and immersion measurements.
B. Plume Release Height

The height of the release point affects the meteorological data used. Ground-level release height typically equates to 10 meters or less. Elevated release heights vary between models. However, meteorological data used for elevated releases should correlate to the release height.

For elevated plumes, the plume will rise above the stack height due to thermal buoyancy and other forces such as the speed of the effluent. For example, PWR steam generator safety relief valves that discharge vertically would have a higher plume rise than a BWR standby gas treatment system stack that has much lower volumetric flowrates. Increases in terrain height change the affected areas and, if high enough, can obstruct the plume. Elevated releases generally produce peak ground-level air concentrations near or beyond the site boundary. Near-ground level releases usually produce concentrations that consistently decrease from the release point to all locations downwind.

Many sites have several potential release pathways that are close enough to each other that discreet plumes could merge. The reduced turbulence and increased upward pressure of the merged plumes may increase plume rise significantly. General plume height considerations:

- Height of release above the ground moves release away from the ground and allows for more dilution.
- Elevated releases require stack heights of two to two and a half times the height of adjacent structures. Most NRC guidance suggests two and half times, but there are some guidance documents that suggest two times the height of adjacent structures depending on site specific considerations.

\[ h_e = h_s + h_r - h_t \]

- \( h_e \): Effective Height
- \( h_s \): Stack Height
- \( h_r \): Plume Rise
- \( h_t \): Terrain Height
• Elevated releases can travel some distance before mixing down to ground level such that the plume may “skip over” receptors close to release point. In this scenario, exposure would be based solely on external sky shine.

C. Meteorological Conditions

Dose assessment models use meteorological data to estimate the transport, diffusion, and deposition of radioactive material following a release to the environment. Meteorological data essentially accounts for the mixing and movement of the released radioactive material with the ambient environmental conditions.

In order to see rapidly changing meteorological conditions for use in performing dose assessments, the NRC established criteria in Regulatory Guide 1.23 for 15-minute average values compiled for real-time display in the emergency response facilities (control room, technical support center, and emergency operations facility). The NRC expects that all the meteorological channels required for manual input to the dose assessment models are available and presented in a format compatible for input to the model (wind speed is displayed in the proper units such as meters per second or miles per hour).

At some sites, because of complex flow patterns in non-uniform terrain, more wind and temperature instrumentation and more comprehensive programs are necessary. For example, the representation of circulation for a hill-valley complex or a site near a large body of water may need more measuring points to assess airflow patterns and spatial variations of atmospheric stability.

Dose assessors should also have provisions in place to get representative meteorological data (wind speed and direction representative of the 10-meter level and an estimate of atmospheric stability that is not necessarily based on Delta T) from alternative sources during an emergency if the site meteorological monitoring system is unavailable.

Wind Direction

Dose assessment models use wind direction reported as the direction the wind is blowing from, not the direction in which the wind is blowing towards. Wind direction determines which sectors or planning areas are downwind of the release and therefore potentially affected. Sigma Theta (θ) is the standard deviation of horizontal wind direction. It is primarily used to calculate atmospheric stability, but may also be used as an indicator of wind direction sensor problems.

Wind Speed

Wind speed determines the plume arrival time and plume departure time for a given place or planning area. The lower the wind speed, the higher ground deposition and projected dose at any given point within the plume pathway.
Wind speed also affects the diameter of the plume. High wind speeds tend to result in a more straight line or tight cone shape, and low wind speeds tend to result in wider plumes when stability class is neutral to stable.

Atmospheric Stability

Meteorologists distinguish three states of the atmospheric surface layer: unstable, neutral, and stable. These adjectives refer to a parcel of air displaced adiabatically in the vertical and horizontal direction. Adiabatically refers to changes in temperature caused by the expansion (cooling) or compression (warming) of a body of air as it rises or descends in the atmosphere, with no exchange of heat with the surrounding air.

These conditions use temperature differences (DT) between various heights, as measured by instrumentation on the site’s meteorological tower, to calculate the stability of the air (how turbulent it is) in both the vertical and horizontal directions. Large differences in temperature between the measured heights equal unstable conditions, where smaller differences in temperature between the measured heights equal more stable conditions.

The Pasquill Stability Classes (Regulatory Guide 1.23, Rev.1), identify seven stability classes based on surface wind speeds, classes of daytime insolation (amount of heat from the sun reaching a certain area), and classes of nighttime cloudiness:

- A - Extremely Unstable
- B - Moderately Unstable
- C - Slightly Unstable
- D - Neutral
- E - Slightly Stable
- F - Moderately Stable
- G - Extremely Stable
The following image depicts the effect of atmospheric stability on plume shape. Varying levels of stability in both the horizontal and vertical axis can alter the shape of the plume and result in higher or lower exposure and deposition at various locations.

**Precipitation**

Precipitation affects ground deposition of radionuclides. There are two components of wet deposition – washout and rainout. Washout occurs when the plume is lower than a rain producing cloud, which results in the precipitation knocking down the gaseous and particulate material.

Rainout occurs when the plume mixes with a rain producing cloud, which results in the gaseous and particulate material being trapped in the cloud followed by dispersion through rainfall.
Precipitation is usually measured near ground level near the base of the mast or tower. Precipitation is any of the forms of water particles, whether liquid or solid, that fall from the atmosphere and reach the ground.

**Mixing Height**

Mixing height refers to the height of vertical mixing of air and suspended particles above the ground. As long as the air released is warmer than the ambient temperature, the released air will continue to rise. However, once the released air cools down to a temperature close to that of the ambient air around it, the released air will slow down and eventually stop. At night, a deep 100 to 200m stable layer usually forms near the ground with other layers of varying stability above it. In the daytime, there is usually a well-mixed layer with a depth of 500 to 2000m.

**Inversion layers**

Inversion layers are an increase of temperature as altitude increases because of the vertical temperature profile of air. Temperature normally decreases as altitude increases in the troposphere. In an inversion layer, the colder air layer is below the warmer air, resulting in a stable temperature profile that restricts vertical mixing. If a plume can penetrate an elevated inversion layer, the inversion layer may act as a strong barrier of downward diffusion resulting in lower ground-level concentrations near the site and greater dispersion into the upper atmosphere.
D. Terrain / Topography

Complex terrain influences the plume trajectory and diffusion. Complex terrain also causes changes in surface-layer wind speed and direction, which affect radioactive material concentrations. For
example, radioactive material released near the ground into the nighttime drainage layer over sloping terrain may follow the drainage flow downhill. However, that same ground level release during daytime may follow a completely different route. During the morning, the sun heats east and south-facing topography. The warmer air rises and then cools, but as the air cools it sinks over lower lying areas near mountains and hills. 

Many sites only have meteorological measurement stations on-site, which may not properly reflect the effects of topography once the plume leaves the site boundary. Even though it is possible that high radionuclide concentrations may occur in complex terrain (where plumes intercept hillsides), several physical processes tend to lower concentrations such as the tendency of wind to favor the natural direction of the terrain. 

Another favorable meteorological effect in complex terrain is turbulence due to eddies of air passing over and around terrain obstacles. Wind speed and direction shears, and time and space variation in wind velocity, become very important in long-range downwind distances greater than about 10 km.

E. Building Wake

Building wake is the combined effect that buildings and other structures within the plume path may have on the trajectory of the plume. Building wake is another form of turbulence. The overall effect of the building is to increase the plume dispersion, although locally high ground-level concentrations may result due to aerodynamic effects of the building. Numerous studies and wind tunnel experiments provide the basis for some general “thumb rules.”

- If the source is a distance greater than twice the height upstream of the building and its height is greater than 2/3 the building height, the plume will rise over the building face.
- Parts of the plume at heights less than 2/3 the building height catches in the down wash in the frontal eddy over the lower part of the building. Some of this material catches in the horseshoe vortex trailing off along the edge of the wake.
- If the flow re-attaches to the building roof and sides, high concentrations may result at these points.
- If the building length is small, the flow will not re-attach, and the plume deflects above the cavity.
- Any radioactive material caught in the wake cavity will mix thoroughly to the ground.
F. Removal Mechanisms

As the effluent travels from its release point, several mechanisms can work to lower its concentration beyond that achieved by diffusion alone. Such removal mechanisms include radioactive decay and dry and wet deposition.

Radioactive decay is dependent on the half-life and the travel time (often referred to as hold-up time) of each radioactive isotope contained within the release. Some isotopes have such a short half-life that by the time they reach a populated area the dose impact is negligible.

All effluents can undergo dry deposition to the ground surface through either a physical or chemical process. However, the dry deposition rate for noble gases, tritium, carbon-14, and non-elemental radioiodines is so slow that depletion is negligible within 50 miles of the release point. Elemental radioiodines and other particulates are much more readily deposited.

Dry deposition is a continuous process while wet deposition only occurs during periods of precipitation. The dry removal process is not as efficient as the wet removal process. At most sites,
precipitation occurs during a small percentage of the hours in a year so that, despite the greater efficiency of the wet removal process, dose calculations for long-term averages (typically performed for the ingestion pathway during the intermediate phase) consider only dry deposition. However, wet deposition can be a significant factor in dose calculations for releases from stacks at sites where a well-defined rainy season corresponds to the local grazing season.

Most dose assessment models apply deposition correction factors based on wind speed, stability class, topography, and surface roughness assumptions. Wet deposition correction factors typically use rainfall rate as a basis.
10. Plume Models

Most computerized dose assessment programs include a dispersion model which accounts for the various transport and diffusion factors that were just covered. A variety of models exist, however, most models fall into one of two categories: gradient-transport theory or statistical theory. Gradient-transport theory holds that diffusion at a fixed point in the atmosphere is proportional to the local concentration gradient. This theory attempts to figure out momentum or material fluxes at fixed points. The statistical approach attempts to figure out the histories of discrete particles and the statistical properties necessary to represent diffusion.

Several basic models use these approaches. These models vary according to their treatment of the spatial changes of input data and how their use of either a variable trajectory model or a constant mean wind direction model.

Determining which model is the best comes down to identifying the model that best simulates atmospheric transport and diffusion in the area of interest from the source to the various receptor locations. This determination will require consideration of the meteorological characteristics of the region, the topography, the characteristics of the effluent source and the effluent, as well as the receptor, the availability and representativeness of input data, source to receptor distance, and the ease of application.

A. Dispersion Factors and Deposition Coefficients

To calculate the atmosphere’s dilution and dispersal effect on a radioactive release, a quantitative representation is needed. This atmospheric dispersion factor is known as relative concentration or $x/Q$, which is the measured concentration (uCi/cc) divided by the source strength at a given distance and direction from the source of the release. These values represent the atmosphere's ability to dilute and disperse effluents over the time chosen. All operating nuclear power plants developed default $x/Q$ values as part of the Offsite Dose Calculation Manual (ODCM). These default $x/Q$ values typically use long-term average wind speed and stability class information to form a general representation of site-specific dispersion characteristics. Most computerized dose assessment models calculate the $x/Q$ values based on the data inputs provided by the dose assessor. Use of default $x/Q$ values is typically reserved for scenarios where the computerized model is unavailable or input data is unavailable such as during a loss of all meteorological data.

Another value that is typically calculated by computerized dose assessment models is the deposition coefficient, or $D/Q$. Calculating deposition coefficients is similar to calculating $x/Q$, with the addition of particle settling or dry deposition of material from the plume. To do this, the $D/Q$ calculation multiplies the $x/Q$ value by a particle settling velocity factor. $D/Q$ determines radioactivity deposition per unit area of ground or vegetation. $D/Q$ follows the same
general patterns of $x/Q$ with respect to behaviors for ground-level and elevated releases, general reduction with increasing distance, “skip over” phenomenon for elevated releases.

B. Class A and Class B Models

Class A models, such as Gaussian models, are simple in design and produce transport and diffusion estimates for the plume exposure emergency planning zone (EPZ) after the classification of an incident involving a radiological release. Class A models use simplifying assumptions to reduce time-dependent variables to manageable quantities. Class A models:

- Use actual 15-minute average meteorological data from the onsite meteorological measurements systems.
- Provide calculations or relative concentrations ($x/Q$) and transit times within the plume exposure EPZ.
- Use diffusion rates based on atmospheric stability as a function of site-specific terrain conditions.
- Use site-specific local climatological effects on the trajectories, such as seasonal, diurnal, and terrain-induced flows.
- Account for source term characteristics (release mode, and building complex influence).
- Assume conditions defined at the time of release remain constant for the duration of each analysis.

The output from the Class A model includes the plume dimensions and position, and the location, magnitude, and arrival time of the peak relative concentration and the relative concentrations at different locations. Limitations of Class A models include:

- Users need to understand the assumptions and the limits of the model so that conditions beyond these limits are identifiable.
- The model correlates local conditions and provides reliable information with conservative results.
- Considerable error is possible for dose rate estimates for a specific point in the plume.

Class B models, such as Particle in Cell (PIC) models, represent the actual spatial and temporal variations of plume distribution and can give estimates of deposition and relative concentration of radioactivity within the plume exposure and ingestion EPZs for the release duration. The greater complexities of these models permit calculating the effects of changing field conditions on plume boundaries.

- Data input to the model is more complex with some plants using real-time inputs from integrated data systems.
- These larger models usually require more user involvement and more time to complete a projection.
- Verifying model predictions is more difficult due to greater complexity and the design goal being more accurate.
• These models predict specific dose rates at a place using projected or actual field conditions.

C. Straight-Line Simple Gaussian Model

Perhaps the most prevalent dispersion model used in dose assessment is the Gaussian model. Gaussian models assume that the plume spread has a Gaussian distribution (standard bell curve) in both the horizontal and vertical directions and, therefore, uses the standard deviations of plume concentration distribution in the horizontal ($\sigma_y$) and vertical ($\sigma_z$). The Gaussian model is popular because it:

• Produces results that agree with data gathered during in-field experiments
• Is fairly easy to perform the mathematical equations
• Is consistent with the random nature of turbulence
• Does not contain large amounts of empiricism in its final stages
• Has found its way into most federal guidance documents

The two basic elements represented in a simple straight-line Gaussian model are material transport and atmospheric dispersion. The model depicts transport as a straight line extending from the point of release in the direction of the prevailing wind. The downwind transport rate (plume travel time) is equal to the wind speed. The wind direction and speed are constants used in the model to define the centerline of the plume and to estimate the travel time to downwind locations. This model uses average 15-minute wind speed and direction.

The model depicts atmospheric dispersion as a Gaussian distribution, which is the familiar bell-shaped curve in statistics. Normal or Gaussian distribution theory gives density or concentrations of radioactive material over time. The average density is greatest at the centerline. Density decreases with distance from the centerline and is symmetrical about the centerline.

The wind carries released material which mixes with the atmosphere based on meteorological forces previously discussed and the rules of chemical dispersion that define the rate of mixing based on intermolecular forces. The combined effect of wind-induced and chemical-induced dispersion is a spreading of the released material over a larger area as the material moves downwind.

The illustration below shows these general concepts with a normal distribution curve overlay to show how the simple straight-line Gaussian model accounts for the width and the height of the plume based on plume centerline.
The generalized simple Gaussian equation for ground-level concentration is:

\[ C(x,y,z) = \frac{Q}{\pi \sigma_y \sigma_z u} \exp \left( -\frac{1}{2} \left( \frac{y^2}{\sigma_y^2} + \frac{z^2}{\sigma_z^2} \right) \right) \]

Where:
- \( C \) = concentration at a location
- \( x, y, z \) = Cartesian coordinates of the point of interest (x, y, z coordinates on a three axis graph) with x being the distance from the release point, y being the horizontal distance from the plume centerline at point x and z being the terrain height above ground at point x
- \( Q \) = source release rate
- \( U \) = wind speed
- \( \sigma_y, \sigma_z \) = standard deviation of horizontal and vertical dispersion coefficient
- \( \exp \) = bases of natural logarithms, 2.7183

The ground-level concentration at the centerline for a source located at ground level is:

\[ \frac{Q}{\pi \sigma_y \sigma_z u} \]
The ground-level concentration for an elevated (H) source below the centerline (where, H = height) is:

\[ C = \frac{Q}{\pi \sigma_y \sigma_z u} \exp\left( -\frac{1}{2} \left( \frac{H^2}{\sigma_z^2} \right) \right) \]

Simple straight-line Gaussian models use assumptions similar to the following:

- Meteorological conditions are homogeneous and stationary. The wind direction and speed responsible for transporting the plume from the release point to the downwind locations and the turbulence responsible for diffusion do not change with location.
- Meteorological conditions do not change as a function of time during the release and time required for transport.
- The release rate is constant over time for the duration of the release.
- None of the emitted material washes out, plates out, or falls out from the plume as it moves downwind, and there is complete reflection at the ground.

Together, these assumptions limit the usefulness of the simple straight-line Gaussian model to estimating concentrations and doses at locations near the release point for release durations that are a few minutes to about one hour. The simple straight-line Gaussian model also tends to overestimate concentrations and doses during low wind speed conditions and becomes undefined for calm wind conditions because wind speed is in the denominator. Sites affected by complex terrain or sea breeze conditions should be cautious of using simple straight-line Gaussian models. The most limiting aspect of a simple Gaussian model is the inability to test spatial and temporal differences in model inputs as it assumes all inputs remain the same for the release duration.

D. Segmented Gaussian Model

The segmented Gaussian model is the next step in complexity when attempting to describe the plume transport and diffusion processes (dispersion). The segmented model does this by accounting for the variability of meteorological conditions of a particular location. The use of such a model is not especially important for those locations where little variability of the wind flow exists, allowing released material to travel extended distances in a uniform direction. However, for those locations where wind trajectories may shift over the course of time, the use of a segmented Gaussian model provides an enhanced tool for estimating atmospheric dispersion.

The segmented Gaussian model normally keeps all variables constant with time, except for the wind direction, wind velocity, and atmospheric stability. This model divides the calculation window into equal time steps, usually a 15-minute period. Meteorological and release data are assigned to each time step. Time steps allow the model to show temporal changes not accounted for under the simple Gaussian model. The plume then grows under the new meteorological conditions until the next time
step. The model processes each time step individually, integrating the calculation results. The model approximates the integrated plume by combining a series of straight-line Gaussian plumes, each estimated on the parameter values applicable to each time step.

Visualize the segmented plume model by assuming that a parcel of air captured within an elastic balloon released from a point where dispersion is to begin. As the wind blows this balloon downwind at some constant rate in a particular direction, the balloon disperses (moves about) in a horizontal and vertical direction. As the balloon progresses downwind, it leaves a footprint behind to mark the vertical and horizontal distance from the plume centerline. At the next time interval, the balloon changes direction from the point in time and space when the last meteorological time interval ended. From this new place, the balloon proceeds in a new downwind direction, governed by the new wind direction and speed, which will decide the distance covered from the end of the first time interval to the beginning of the next time interval. The balloon assumes a new horizontal and vertical path under the second interval's new atmospheric stability parameters. This process continues for the desired time steps (release duration). The footprint of the "plume" resembles not a straight-line but a snake-like form based upon the actual meteorological conditions.

Segmented Gaussian models use assumptions similar to the following:

- All locations downwind from the point of the meteorological observation are responding to the observed meteorological change in an identical way.
- When the wind changes from a previous observation, the wind also changes at every other point within the grid of computation.
• The release rate is constant over time for the duration of the release.
• None of the emitted material washes out, plates out, or falls out from the plume as it moves downwind, and there is complete reflection at the ground.

In summary, the segmented Gaussian model allows for a better representation of plume dispersion over time and space and accepts simple conditions including some site-specific data for validation. The segmented plume model may aid in representing those locations where significant wind trajectory deviations exist caused by terrain effects or daily cyclic variations of the wind. A segmented Gaussian model that shows wind trajectory deviations over time may better represent locations such as those along large bodies of water like oceans or large lakes.

E. Puff Model

Puff models use a series of discrete puff releases to depict a continuous plume. Each puff released by the source has its own strength and release height. At finite time intervals, usually 15 minutes, wind direction, wind speed, mixing depth, and stability update. All puffs move during the next time step under the updated meteorological conditions. Use of discrete puffs permits temporal variations in source characteristics and both spatial and temporal variations in meteorological conditions. For instance, the release may stop and material released previously will remain as part of the projection until removed by transport or diffusion conditions. The puff formulation is compatible with both a continuous release over a long period and an instantaneous release occurring over a very short period.

The release rate of puffs varies with meteorological conditions. Wind speed and stability class at the release point influence each puff release rate that helps support sufficient overlapping of successive puffs to make sure an accurate approximation of a continuous plume. The puff model assumes each puff has a circular cross-section. This method ensures that each puff diffuses as with time regardless of its straight-line distance from the release point. The model assumes a Gaussian diffusion (normal distribution about the centerline) of the material in the puffs.

Ground-level concentration of a radionuclide ($j$), at a receptor within a puff ($i$) is:

$$X_{i,j}(r) = \frac{Q_i}{(2\pi)^{1.5}\sigma_z\sigma_y} \exp\left[ -\frac{1}{2} \left( \frac{r}{\sigma_y} \right)^2 \right] \exp\left[ -\frac{1}{2} \left( \frac{H_i}{\sigma_z} \right)^2 \right]$$

Where,
• $X_{ij}(r) = $ ground level concentration at $r$
• $Q_i = $ total quantity of material released during the time interval
• $r_i = $ radial distance from puff center to receptor
Hi = effective puff release height
σyi = lateral diffusion coefficient
σzi = vertical diffusion coefficient

The algorithm identifies locations within the circular footprint of each puff out to distances of three standard deviations from the puff center. Total ground level concentration of a radionuclide at a location is the sum of concentrations from all puffs affecting that location. Diffusion coefficients are account for atmospheric stability and puff travel distance that describe how far each puff spreads horizontally and vertically. The relations differentiate with respect to travel distance to get the rates of change of diffusion coefficients with respect to travel distance. For each time step, the model calculates the diffusion coefficients for the current stability class for each puff.

Puff Gaussian models use assumptions similar to the following:
• A Gaussian distribution of material in the vertical
• When a puff’s vertical extent exceeds the mixing depth (encounters the lid), material in the puff has a uniform vertical distribution from that time interval on

F. Particle-in-Cell (PIC) Model

While Gaussian models are relatively easy to use, they have a number of inherent limitations such as:
• They only address terrain height in determining the effective plume height.
• They typically do not account for the impact of terrain on plume transport.
• Straight-line models cannot “curve” a plume around mountains or follow a river valley.
Advanced models can address terrain impact on plume transport and diffusion. One such model is the particle-in-cell (PIC). Using information from site meteorological instrumentation (at multiple elevations), the model determines a set of x-y-z coordinate wind velocities for each cell in the wind field. These velocities may differ from cell to cell. Wind field cells displaced by terrain have zero speed. Some PIC models divide the wind field into horizontal regions composed of several cells with each region having its own meteorological data input (supplemental towers). In some PIC models, there is more than one wind field vertically. This latter capability provides a means to model conditions in which the wind speed or wind direction is different between elevations (wind shear), such as in valley terrains. In the PIC model used at one power station, the use of acoustic sounding equipment (SODAR) allows seven possible shear layers.

With the wind field determined, the model releases particles into the wind field for each time step. Each of these particles carries a history of source term data. As the particle moves into a cell, the x-y-z wind speed vectors for that cell act upon it. The particle may speed up, slow down, or change direction depending on the effect of these vectors. Movement is possible in all three coordinate directions. If the particle encounters a terrain cell, its movement halts laterally. It may then change direction or rise vertically. Periodically, the model stops and counts the particles in each cell. Since each cell has the same shape and size, the number of particles in a given cell represents the radioactive material concentration. If the release is longer than one time step (typically 15 minutes), the particles from the first time step decay and continue their movement through the wind field.
When dispersion is complete, the model converts the integrated particles in each cell to relative concentrations and then dose equivalents. The model assigns inhalation dose and ground deposition for particles in cell layers next to the ground.

Because of their temporal and spatial characteristics and their inherent ability to model wind flow around terrain obstructions, PIC models are best at sites characterized by rugged terrain. Obviously, such models are very site-specific and need more meteorological data.

G. Uncertainties

The dose assessment process has thus far included some assumptions and estimates. For example, the assumptions made in the initial core release fractions and all the fission product fractions based on limited experimental data that most likely correspond to a single temperature. Not only will the actual temperature conditions most likely be different from those of the experimentally obtained release fractions, but also the temperatures of the fission products and their environment will continually change with time and with accident conditions. The transport and diffusion of fission products in the reactor coolant system and in the containment also will be affected by pressures, viscosity, densities, hole sizes, material melting and boiling points, heat capacities, flows, time, surface areas, volumes, system availabilities, and chance. Any dose assessment made will have within it an inherent measure of uncertainty ranging between 1 and 1000 (NUREG-0956) depending on the accident sequence.
11. Dose Calculations

The dose calculation is the point in the dose assessment process where all inputs come together to give a projection of potential dose consequences at various locations from the release point. The use of modern computers provides dose assessors with advanced calculation methods, which take into account all factors previously discussed (core inventory, source term, release heights, release duration, plume transport, and plume diffusion).

A. Dose Conversion Factors (DCFs)

Dose conversion factors (DCFs) convert the specific isotopic concentrations in the source term to dose equivalent values. These DCFs include assumptions such as breathing rate, relative biological effectiveness for organ doses, and surface roughness factors in the calculations depending on the exposure pathway.

The table below is an example of DCFs from NUREG-1940. The table lists the applicable isotope and provides a dose equivalency factor for specific organs and tissue that the dose assessment model uses to convert the projected doses (based on source term, release rate, and dispersion) into estimated dose equivalent values. Notice how each nuclide has a different impact for each organ and the impact varies depending upon the type of exposure (inhalation, ground shine, plume immersion). The table also shows how some isotope, like Strontium, have a larger dose consequence compared to isotopes like Zinc.

<table>
<thead>
<tr>
<th>FGR-11 AND FGR-12 DCFs</th>
<th>RBE 30-DAY ACUTE DCFs</th>
<th>ICRP-66° DCFs</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Inhalation (Sv/Bq)</td>
<td>Thyroid (Sv/Bq)</td>
</tr>
<tr>
<td></td>
<td>1.33E+09</td>
<td>1.78E-10</td>
</tr>
<tr>
<td>Ni-63°</td>
<td>2.25E-09</td>
<td>1.21E-10</td>
</tr>
<tr>
<td>Zn-65°</td>
<td>5.76E-10</td>
<td>2.94E-11</td>
</tr>
<tr>
<td>Zn-69m+</td>
<td>2.31E-10</td>
<td>1.35E-11</td>
</tr>
<tr>
<td>Ce-68°</td>
<td>1.40E-08</td>
<td>6.64E-10</td>
</tr>
<tr>
<td>Br-82°</td>
<td>2.41E-11</td>
<td>3.29E-12</td>
</tr>
<tr>
<td>Sr-89°</td>
<td>1.36E-10</td>
<td>1.30E-11</td>
</tr>
<tr>
<td>Sr-90°</td>
<td>1.66E-08</td>
<td>1.21E-09</td>
</tr>
<tr>
<td>Sr-90°</td>
<td>3.53E-07</td>
<td>2.65E-09</td>
</tr>
<tr>
<td>Sr-91°</td>
<td>4.55E-10</td>
<td>4.12E-11</td>
</tr>
</tbody>
</table>

B. Exposure Pathways

During and following a radiological release, varieties of pathways expose people to airborne and deposited radioactive materials. The exposure pathways may also change as the event progresses. In the early phases of an extended radiological release, the exposure pathways of concern is direct.
exposure to plume shine and ground shine, inhalation of noble gas and iodine due to plume immersion, and exposure to deposition of radioactive material on the ground. Delayed exposure pathways may include any of the early phase pathways, as well as ingestion and re-suspension.

External exposure (plume shine and ground shine) stops when the person leaves the area of the source. Internal exposure (inhalation and ingestion) continues until the body’s natural processes flush the radioactive material or enough time elapses for radioactive decay to complete. A person who has ingested radioactive material receives an internal dose to several different organs. The absorbed dose to each organ is different, and the sensitivity of each organ to radiation is different.
Direct Exposure

Direct exposure to radioactive materials in an atmospheric plume, which is also called plume shine, sky shine, or cloud shine is direct external exposure to Gamma emitters (xenon and krypton) in the plume. Calculated doses use either a semi-infinite cloud model or a finite-plume model. The semi-infinite model assumes a uniform distribution of activity distributed through a large volume. The finite model assumes that the activity accumulates in a finite number of points distributed through a volume.

Inhalation Exposure

Inhalation of radionuclides from immersion in a radioactive plume may occur. Immersion exposure is when a radioactive plume completely surrounds an individual. Immersion exposure leads to external exposure to beta and gamma radiation, external contamination from radionuclides in the plume, and internal exposure and contamination due to inhalation.

Re-suspension is inhalation exposure from deposited radioactive materials that goes airborne again through walking or wind gusts. Once airborne, the re-suspended radioactive material once again becomes an inhalation hazard. The dose from inhalation of re-suspended materials is usually not a primary factor in the decision to evacuate during the early or intermediate phases. Re-suspension is a primary concern during the recovery and re-entry planning process.

Some radionuclides become concentrated in a single body organ, with only small amounts going to other organs. For example, a significant fraction of inhaled radioiodines will move through the
bloodstream to the thyroid gland. Stations protect workers from iodine exposure, by evacuating personnel before arrival of a radioiodine plume, use of respiratory protection, or administration of potassium iodide (KI). Radioiodines are present in large quantities because the fission process creates a large inventory available for release. Filtered pathways remove most radioiodines and particulates, leaving noble gas exposure as the dominate source. If the release is through a non-filtered pathway, then noble gases deliver lower integrated doses than the other radionuclides because they do not deposit in the body after inhalation and do not deposit on the ground like radioiodine and particulates.

**Deposition Exposure**

Deposition (deposited materials) of radioiodine and particulates from a radioactive plume can continue to emit “ground shine” (beta and gamma radiation) after the plume has passed causing continued exposure to skin and internal body organs. Beta radiation emitters may deposit on the ground, but the beta component is not a controlling factor in determining whether an evacuation is required during a nuclear reactor accident.

A plume may deposit materials on surfaces, posing a risk of longer-term exposures via ingestion, direct external exposure, and inhalation pathways. If the release has large quantities of radioiodines or particulates, the resulting long-term exposure to this ground shine has a greater potential dose consequence than external exposure from the passing plume if the exposure time to the ground contamination exceeds the plume passage time.

**Ingestion Exposure**

Ingestion is internal exposure resulting from the ingestion (eating or drinking) of contaminated foods or liquids.

**C. Integrated Exposure Duration**

The projected dose for comparison to the EPA PAGs uses an exposure duration of four days (96 hours) after the start of a release. The aim is to encompass the entire period of exposure to the radioactive plume and deposited material before implementation of any further, longer-term protective action such as relocation. For planning purposes, the EPA chose a four-day period as the exposure duration during the plume phase because it is a reasonable estimate of the time necessary to make measurements, reach decisions, and prepare to start further protective actions.

The EPA PAGs assume that an individual remains in a specific place for the 96-hour exposure period. This makes the PAGs somewhat conservative in nature, as it is unlikely that someone would stay in the same spot for 96 straight hours.
D. Plume Exposure Duration

Another factor affecting projected dose estimates is the duration of the plume at a particular location. For purposes of calculating projected dose from most pathways, exposure starts at a particular location when the plume arrives and ends when the plume is no longer present. Exposure from deposited materials continues for an extended period as long as people are present.

E. Units of Measure

Emergency response staff should understand radiation and its potential hazards to have meaningful conversations with offsite response partners and the public in the aftermath of a radiological emergency. Dose assessors and decision makers should understand what the dose assessment results mean in terms of potential health consequence. Two categories describe the health effects of radiological exposure, deterministic and stochastic.

- Deterministic effects are those that are immediate and observable due to acute exposure such as reddening of the skin and vomiting.
- Stochastic effects are long-term and include things such as cancer and genetic disorders.

The number of atoms disintegrating per unit time measures radioactivity. The most common term used in the United States to measure radioactivity is the Curie (Ci), which is equal to 37 billion disintegrations per second. A disintegrating atom can emit a beta particle, an alpha particle, a gamma ray, or some combination of all these, which is why Curies alone do not give enough information to assess the risk to a person from a radioactive source. In terms of dose assessment, Curies quantify the amount of radioactive material released to the environment in total, or per a unit of time such as Curies per second (release rate).

As radiation moves through the body, it can dislodge electrons from atoms, disrupting molecules. Each time this happens, the radiation loses some energy until it escapes from the body or disappears. The energy the radiation deposits in tissue is called absorbed dose, which is measured in Rads (radiation absorbed dose). A Rad is a basic unit of absorbed dose, or the amount of energy deposited by ionizing radiation in a unit mass of tissue. It is a measure of the amount of energy absorbed by the body, where one Rad equals the dose delivered to an object of 100 ergs of energy per gram of material. The Rad does not describe the biological effects of different radiations, only the amount of energy deposited.

Not all radiation has the same biological effect, even for the same amount of absorbed dose (Rads). The Rem (roentgen equivalent man), relates the absorbed dose (Rads) in human tissue to the effective biological damage of the radiation. Calculating Rem requires multiplying the number of Rads by the quality factor for the type of ionizing radiation (Alpha, Beta, and Gamma) to reflect the potential damage caused.
The total potential health effect of exposure to a radiological plume is the sum of the external dose impacts and the internal dose impacts. Deep dose equivalent (DDE) quantifies the external impact as a measure of biological damage (dose equivalent) caused by the external exposure to the whole-body at a tissue depth of 1 cm. The whole-body impact (head, trunk, arms above the elbow, or legs above the knee) implies that the individual’s exposure was to a uniformly distributed source like a plume going overhead or ground shine.

Committed dose quantifies the internal dose that a person receives from the time the nuclide enters the body until it is gone. The committed dose accounts for continuing exposures expected over a long period of time (such as 30, 50, or 70 years). Radionuclides distribute to different organs and stay there for days, months, or years until they decay or the body’s natural processes remove them. Committed dose equivalent (CDE) is the internal dose equivalent to a specific organ or tissue received from an intake of radioactive material by an individual during the 50-year period following the intake. Each organ has a dose conversion factor based on the type of ionizing radiation and the mechanism of exposure (inhalation, ingestion, immersion). Committed effective dose equivalent (CEDE) is the sum of the products of the individual CDE values for each organ multiplied by the applicable weighting factors for each of the organs or tissue.

The total effective dose equivalent (TEDE) is the sum of the internal dose (noble gas + iodine immersion) measured as CEDE and the external dose (plume shine + ground shine) measured as DDE. The dose assessment calculations for TEDE assume that members of the public within the pathway of the plume take no protective actions, such as evacuation or sheltering. The calculations assume that people are outdoors during plume passage and will stay outdoors exposed to ground shine from deposited radionuclides for the duration selected in the dose assessment. Thus, the TEDE calculated is likely larger than the TEDE that would be expected for people who took protective actions or who continued their normal activities like spending some part of the day indoors.

Decision makers use the calculated exposures to decide whether doses without any protective actions would exceed the EPA PAGs. The EPA PAGs include values for both TEDE and Thyroid CDE that give decision makers thresholds at which the risk of taking protective actions for the public would likely outweigh the risks of doing nothing. Projected TEDE and Thyroid CDE that a person would receive at a specified distance such as at the site boundary are the basis for protective action decisions.

F. Protective Action Guides (PAG)

A PAG is the projected dose to an individual from a release of radioactive material at which the EPA recommends a specific protective action to reduce or avoid that dose. PAGs do not establish an acceptable level of risk for normal, non-emergency conditions, nor do they represent the boundary between safe and unsafe conditions. The EPA established PAGs for the principal phases of a nuclear incident (early, intermediate, and late). During the early phase, OROs make decisions quickly based
on available information, where decision-making in the intermediate and late phases allows for more time and resource inputs. This section mainly focuses on the early phase PAGs as those are the ones licensee dose assessors are most likely to encounter. The EPA PAG Manual has more details on intermediate and late phase PAGs.

The following table shows the PAGs from the latest revision of the EPA PAG Manual (R1, 2017). The PAG thresholds for sheltering and evacuation are solely based on projected TEDE dose, or the sum of the effective dose from external radiation exposure (groundshine and plume submersion) and the committed effective dose from inhaled radioactive material. The PAGs use a range of 1 to 5 Rem with a recommendation from the EPA to begin taking the most effective action, or combination of actions, at 1 Rem (10 mSv) with the goal being to achieve the lowest exposure for most of the population. The previous revision of the EPA PAG Manual (R0, 1992) also used a similar 1 to 5 Rem TEDE range for sheltering and evacuation, however, there was also guidance on use of a threshold based on Thyroid CDE of 5 to 25 rem. Another key difference between the two revisions of the PAG Manual is the basis for the KI PAG. The 2017 revision uses child thyroid dose (one-year age group) where the 1992 revision uses an adult thyroid dose.

EAL schemes based on NEI 99-01 use varying TEDE and Thyroid CDE values as trigger points for emergency classification purposes. For example, 1 Rem TEDE and 5 Rem Thyroid CDE are the typical General Emergency triggers. These values use the 1992 revision of the EPA PAG Manual as the underlying basis. Most dose assessment methods and licensee protective action recommendation processes follow the 1992 guidance. However, dose assessors should understand which version of the PAG Manual is in use at their site.
Sheltering-in-place or evacuation of the public

- 1 to 5 Rem (10 to 50 mSv) projected dose over four days

Supplementary administration of prophylactic drugs – KI

- 5 Rem (50 mSv) projected child thyroid dose from exposure to radioactive iodine

Limit emergency worker exposure (total dose incurred over entire response)

- 5 Rem (50 mSv)/year (or greater under exceptional circumstances)

The EPA used a risk-benefit balancing process, designed to prevent acute effects, balance protection with other important factors and ensure that actions result in more benefit than harm, and lower risk of chronic effects to derive the PAGs. This risk-benefit balancing process incorporates a level of precaution into the PAGs. For example, early phase derived levels assume that a person is outdoors 24 hours a day for four days being exposed to the plume. Protection of the public from unnecessary exposure to radiation may require some form of intervention that will disrupt normal living. Such intervention is termed a protective action. Examples include:

- Evacuating an area (plume phase)
- Sheltering-in-place within a building or protective structure (plume phase)
- Administering potassium iodide (KI) as a supplemental action (plume phase)
- Relocation (post plume phase)
- Acquiring an alternate source of drinking water (post plume phase)
- Interdiction of food/milk (post plume phase)

Evacuation and sheltering-in-place offer different levels of dose avoidance from the principal exposure pathways: direct gamma exposure and inhalation. Both sheltering-in-place and evacuation are possible during the same response in different areas or timeframes. Evacuation, if completed...
before plume arrival, is 100 percent effective in avoiding radiation exposure. The effectiveness of evacuation will depend on many factors, such as the speed of implementation and the nature of the emergency. For events where the principal source of dose is inhalation, evacuation could increase exposure if performed during the passage of a short-term plume.

Sheltering-in-place is a low-cost, low-risk protective action that gives protection with an efficiency ranging from zero to almost 100 percent, depending on the type of release, the type of shelter available, the plume passage duration, and meteorological conditions. Sheltering-in-place may be preferred for special populations (e.g., those who are not readily mobile) or when environmental, physical, or weather hazards impede evacuation for the general public. It is also comparatively easy to communicate with population’s sheltered-in-place.

KI is effective against uptake of radioiodine, and is best taken before or just after exposure. The protective effect of a single dose of KI lasts about 24 hours. People directed to use KI should do so until the risk of significant exposure to radioiodine (either by inhalation or ingestion) no longer exists. KI is a supplemental action, secondary to evacuation or sheltering.

G. Protective Action Recommendations (PAR)

The person with command and control determines the PAR. The purpose of a PAR is to provide the applicable ORO decision makers with a recommendation for the actions they should consider for public health and safety. The licensee shall include a PAR on the initial General Emergency notification form. The initial PAR may use dose assessment results or be based on plant conditions such as radiation monitor readings if dose assessment results are not yet available. Subsequent PARs are typically determined based on dose assessment results or changes in meteorological conditions such as wind shifts that move the plume over previously unaffected areas.

H. Protective Action Decisions (PAD)

OROs make the final decision on what protective action is necessary to protect public health and safety, and then relay these decisions to the public. The status of the plant and the prognosis for worsening conditions informs the choice of protective action. Some OROs may use precautionary actions based on worst-case scenarios before implementation of protective actions based on PAGs. The decision to tell members of the public to take a protective action during a radiological incident involves a complex judgment which weighs the radiological risk against the protective action’s inherent risks. For example, what are the risks of evacuating a large population under the current conditions? Are roadways accessible? Are projected meteorological conditions stable or likely to change before the plume arrives in a populated area? These decisions will likely occur under stressful emergency conditions, with limited information and little time to analyze options.
Dose assessors should understand how the output of the dose assessment model relates to the EPA PAGs, site-specific PARs, and offsite response organization protective action decisions (PADs). These terms sequentially relate to each other. The first step is for the dose assessor to decide if any area within the projected plume exceeds an EPA PAG. If the answer to that question is yes, then the dose assessor provides that information to licensee decision makers who decide the applicable PAR. Licensees issue PARs to OROs responsible for making the decision for whether or not to order public protective actions. The PAD for the public resides with the OROs, as they must consider the recommended protective action against other risks (impediments to evacuation such as severe weather, damaged roadways, evacuation time estimates). The graphic below depicts the relationship between PAGs, PARs, and PADs.

The relationship between the licensee and the OROs is critical to the timely implementation of PADs. Licensee staff have to communicate accurate information in a clear and understandable way, recognizing that many OROs likely do not have a background in radiological assessment terminology. It is incumbent upon the licensee to discuss the uncertainties inherent within the dose assessment process to make sure the decision-makers have the best possible information. Licensee staff should understand and be ready to discuss:

- The dominant plant conditions that influence the potential risk to the offsite population
- A best estimate of the size of any offsite releases and the assumptions for these estimates
- The level of confidence in the dose assessment including major uncertainties and assumptions
12. Field Team Measurements and Model Validation

As discussed earlier in this document, dose assessment models contain varying levels of uncertainty, which may make the results conservative or non-conservative. Use of field monitoring teams provides an opportunity to collect plume phase sample data and compare it to the dose projection model. These comparisons offer a means to dose assessors and decision makers to develop a level of confidence with respect to the projections.

For example, field monitoring teams dispatched at the onset of a significant radioactive release can easily measure exposure rates from sky shine, immersion dose rates and ground deposition to confirm the location of the plume as well as the plume size and shape characteristics. Some dose assessment models provide projected exposure rates at pre-determined locations for timely comparison to actual field data collected by the monitoring teams.

A. Field Team Measurement Strategies

Placement of field monitoring teams is one of the most important factors in their effectiveness. Field teams should be downwind of the plume and capable of traversing the projected plume pathway so that they can try to locate the centerline of the plume. Finding the centerline of the plume is important because samples from the centerline should provide comparable results to the dose projection model by providing a sampling environment that contains the highest concentration of released material. Typical samples taken by the field team include:

- Arrival dose rates – open and closed window to measure for Gamma and Beta exposure rates to differentiate sky shine and ground shine (plume phase)
- Air samples – a standard volume of air pumps through a cartridge and filter paper to measure iodine and airborne particulate concentrations (plume phase)
- Environmental Samples – soil, vegetation, and water samples measure deposition levels (typically during intermediate phase)

B. Field Team Samples and Dose Assessment

Dose assessors use field team surveys in several ways with respect to dose assessment.

- To compare against dose projections derived using installed effluent radiation monitor data
- As input data to the dose assessment model because installed radiation monitor data is not available
- To confirm the location and size of the plume

Under the first scenario, the dose assessors may input the field sample results directly into the dose assessment model to run a confirmatory projection. Dose assessors then compare the field data projection to the effluent monitor projection to determine if similarity exists between the two.
Some sites use a factor of 10 between field survey data and dose projection data as a thumb rule for establishing a level of confidence in the model. A factor of 10 may seem like a wide margin of error, however, it is important to remember the aforementioned uncertainties associated with the dose assessment model as well as the uncertainties in the field surveys. For example, there is no guarantee that the field teams took measurements under the exact conditions assumed within the dose assessment model. Breathing rates, deposition rates, re-suspension factors, plume travel time, and plume diffusion are all assumptions that weigh in on the dose assessment models projections. A difference in any one of these assumptions could result in disparity between field survey results and the dose assessment model. If the two projections are off by more than the selected factor, then the dose assessor should review assumptions made in the dose assessment model and request confirmatory field survey samples.

Under the second scenario, dose assessors use the field survey results as the input into the dose projection model due to a loss of effluent monitoring ability or an unmonitored release pathway. The dose assessor inputs the measured plume centerline exposure rate and air sample results at a known distance downwind from the release point. The dose assessment model then calculates 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. This method carries a high level of uncertainty because the measurements available at the reference distance may prove unrepresentative, especially if atmospheric conditions are creating erratic plume behavior. In the case of an elevated plume, the ground level concentration increases with distance from the source and then decreases, because any high-energy gamma radiation from the overhead cloud continuously decreases with distance. For these reasons, this method will perform best for surface releases or if the point of measurement for an elevated release is more than one mile from the point of release for the plume to have expanded to ground level. The use of measurements from many locations averaged over time will improve the accuracy of this method.

Lastly, field team samples may confirm the presence and path of the plume. Dose assessment models project plume characteristics using meteorological conditions at a single point; however, wind speed, wind direction and stability class could all change at various locations causing the actual plume path to track in a different direction. Field teams can find the actual centerline (highest dose rate) and indicate if the plume has reached the ground (presence of beta radiation).
13. References

A. Regulatory Basis

Title 10 CFR 50.47

In addition to codifying the 16 Planning Standards, regulation 10 CFR 50.47 states that the NRC cannot issue an initial operating license unless adequate protective measures can and will be taken in the event of a radiological emergency. The NRC decision includes advice from the Federal Emergency Management Agency (FEMA) that ORO emergency plans are adequate and a determination that there is reasonable assurance that the plans are implementable.

Title 10 CFR 50.47(b)(9) states, as a requirement, that “adequate methods, systems, and equipment for assessing and monitoring actual or potential off-site consequences of a radiological emergency condition are in use.”

Title 10 CFR 50, Appendix E - Emergency Planning and Preparedness for Production and Utilization Facilities – Contents of Emergency Plans

10 CFR 50, Appendix E includes all emergency planning and preparedness information that utilities must include in their emergency plan. For the purposes of dose assessment, Appendix E section IV.E.2 describes requirements for “equipment for determining the magnitude of and for continuously assessing the impact of the release of radioactive materials to the environment.”

B. Guidance Documents

The documents listed below are guidance documents. Users of this document should not confuse these documents with regulatory requirements. How each licensee uses these guidance documents is completely up to them based on site specific licensing requirements and E-Plan commitments.

Reg. Guides 1.3 and 1.4

Regulatory Guides 1.3 and 1.4 establish the characteristics of the fission product release from the core into the containment. The NRC added stability class G, “extremely stable”, in RG. 1.23.

Regulatory Guides 1.3 and 1.4 assume that the source term within containment is instantaneously available for release and that the iodine chemical form is assumed as predominantly (91 percent) in elemental form, with 5 percent assumed to be particulate iodine and 4 percent assumed in organic form. These assumptions have affected the design of engineered safety features.

**Reg Guide 1.183 - Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors**

This regulatory guide provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable alternative source term (AST) and identifies the significant attributes of other ASTs that the NRC may find acceptable.

**NUREG-0654 - Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants**

NUREG-0654 Appendix 2 – Meteorological Criteria for Emergency Preparedness at Operating Nuclear Plants provides a means to comply with 10 CFR Part 50.47 requirements that the Emergency Plan provides “adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition ... ”

**NUREG-0737**

This document incorporates all TMI-related items approved for implementation by the Commission. A large number of post-TMI requirements required the installation of a number of control room indications including radiation monitors and meteorological data used within the dose assessment process.

**NUREG-1228 - Source Term Estimation during Incident Response to Severe Nuclear Power Plant Accidents**

This regulatory guidance describes a range of accident conditions from those within the design basis (FSAR Chapter 15) to severe accidents. This guidance presents a method of source term estimation that reflects the understanding of source term behavior. This document also discusses various methods of estimating radionuclide releases to the environment based on release pathways.

**NUREG-1465 - Accident Source Terms for Light-Water Nuclear Power Plants**

Published in 1995, this document updated available information on fission product releases based on significant severe accident research. Current light water reactor (LWR) licensees may voluntarily propose applications based upon it.
**NUREG-1940 - RASCAL 4: Description of Models and Methods**

The methods that the RASCAL 4 source term calculations use for nuclear power plant accidents largely rely on the methods described in NUREG-1228, “Source Term Estimation during Incident Response to Severe Nuclear Power Plant Accidents,” (McKenna and Giitter, 1988). Various aspects of the source term estimation methodology, including release timing, were modified to account for the accident source term insights in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants, Final Report,” (Soffer et al., 1995).

**EPA-400 - EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents**

This document, known as the PAG Manual, has served as a primary reference for development of PARs and PADs since its issuance in 1992 and later revision in 2017. It provides guidance in the form of numerical dose criteria for early and intermediate phase protective actions.

**Response Technical Manual (RTM) - 96**

NRC Report # NUREG/BR-0150 – Published 1996, has methods (procedures) used to do the assessments necessary to meet the public protection goals. The manual is consistent with the guidance in the U.S. Environmental Protection Agency’s May 1992 Manual of Protective Action Guides and Protective Actions for Nuclear Incidents (EPA 400-R-92-001).

**TID-14844 - Calculation of Distance Factors for Power and Test Reactors**

In 1962, the Atomic Energy Commission issued Technical Information Document (IMD) 14844. This document included a postulated a release of fission products from the core of a LWR into the containment atmosphere (“source term”) was for the purpose of calculating off-site doses in accordance with 10 CFR Part 100, “Reactor Site Criteria.” The source term modeled an accident that resulted in the meltdown of the core, and the fission products assumed released into the containment were based on an understanding at that time of fission product behavior. Since the publication of TID-14844, significant advances have been made in understanding the timing, size, and chemical form of fission product releases from severe nuclear power plant accidents. A holder of an operating license issued before January 10, 1997, or a holder of a renewed license under 10 CFR Part 54 whose initial operating license was issued before January 10, 1997, can voluntarily revise the accident source term used in design basis radiological consequence analyses per 10 CFR 50.67, “Accident Source Term.”