3 Core Damage and Related Concepts

INTRODUCTION

This chapter discusses reactor core states, radioactive material inventories, severe accidents, the world experience with reactor accidents, accident sequences, the sources of radiological releases, mechanisms for depleting radioactive source terms, and plant monitoring systems used for assessing accidents. The purpose of emergency planning for commercial nuclear reactors is to avoid as much radiation dose to the public as possible under the circumstances of a specific accident. An important aspect of preplanning is establishing the means to detect abnormal amounts of radioactive material and analyze its likely consequences. Predicting future consequences from releases of radioactive material involves both the direct measurement of radiation and the indirect modeling of its movement through the environment. The resulting dose analysis is not accurate in every detail but when done properly provides a sound technical basis for selecting from among the available protective action strategies. This chapter primarily discusses the assessment of damage to the reactor core and aspects of modeling the behavior of radioactive material in the environment.

The total effective dose equivalent radiation dose to a member of the public from a commercial power reactor accident is the sum of the external doses and the committed dose equivalents received from internally deposited radioactive material. The total effective dose equivalent is estimated at predetermined locations and distances in the environment and does not represent the assumed actual dose of any individual member of the public. Dose estimates (or estimates using different core damage or meteorological assumptions) are intended to provide offsite authorities a basis for selecting among options for taking protective actions. Although the goal is to provide as accurate a radiological analysis as possible, in practice the underlying assumptions often tend more toward "worst case" than "most likely"; therefore, members of the public will likely receive less actual radiation dose than is estimated for their location.

The deep-dose equivalent (external dose) is primarily delivered by gamma (γ) radiation originating in airborne radioactive material within the range of the highest energy gamma radiation in the radioactive mixture, with a small contribution from higher-energy beta (β) radiation. This effective distance is typically between 100 and 150 yards.* The maximum effective range in air of beta radiation originating from the radioactive material typically present in a power reactor accident is about 35 feet. The deep-dose equivalent at a point in the environment is calculated by multiplying the concentration of each individual radioisotope by an external dose conversion factor;† the conversion factor takes into account all of the radiation energies from the isotope and their effective distances. A uniform distribution of radioactive material throughout the effective radius is assumed; this assumption overestimates the likely external dose at locations very near the edges of a radioactive plume. The deep-dose equivalent is the sum of all individual isotope deep-dose equivalents for both gamma and beta radiation. If a direct air sample has been collected, then an actual distribution of radioactive material can be used; otherwise, the analysis uses predetermined distributions of radioactive materials for the different stages of core damage.

* The intensity of 3.0-meV gamma radiation in air at standard temperature and pressure (STP) conditions is reduced by 99% at about 328 feet.

† External dose conversion factors can be found in USEPA (1993).

An individual does not have to be surrounded by radioactive material (i.e., in the plume) to receive an external dose from the radioactive material in the plume. Given the range of gamma radiation in air, a member of the public could be as far as 150 yards away from the plume and still be irradiated at a low dose rate. Gamma radiation originating in radioactive material not near the person receiving the dose is referred to as *shine*. Models of the behavior of radioactive material in the environment vary in their treatment of external exposure along the physical edge of a radioactive plume; the model output is usually an exposure rate or total integrated dose estimate map that rarely distinguishes the physical location of radioactive material from its associated gamma shine. Some models simply do not calculate radiation exposure rates or deep-dose equivalent for locations outside the plume of radioactive material.

The estimation of dose from an internal uptake of radioactive material is complicated and involves many assumptions, some of which are not accurate for specific individuals. Internal dose is estimated by multiplying the concentration of each individual radioactive isotope by a dose conversion factor* and by an exposure time. The dose conversion factor includes models about breathing rates, the biological efficiency of absorbing radioactive material into the body, the destination organs into which radioactive material is ultimately deposited, radioactive decay, and biological processes for eliminating isotopes from the body. As with external radiation, either a measured or a calculated distribution of radioactive isotopes can be used to perform the analysis, and the overall internal dose is the sum of the individual internal doses from each isotope in the distribution.

Because radioactive material enters the body more effectively by breathing than through other pathways, generally only the inhalation dose is calculated. In general, the radioactive isotopes released from a damaged power reactor core are not radioactive species or in chemical forms that have significant direct absorption through the skin.

GENERAL METHOD

The general method for generating a protective action decision is as follows:

- Estimate the radioactive source term based on the core damage state.
- Multiply the source term by a release rate determined from an effluent radiation monitor.
- Apply meteorological dispersion factors to estimate radioactive material concentrations.
- Multiply concentration by isotopic dose conversion factors to obtain gamma exposure and dose rates.
- Multiply dose rates by exposure time to get dose.
- Compare the dose estimate to established Protective Action Guide action thresholds.
- Adjust the release rate as needed to match the estimated gamma exposure rates to measured gamma exposure rates as observed by environmental monitoring teams.

DEFINITIONS

Fuel Damage

Fuel damage is perforation of the fuel cladding that permits the release of fission products into reactor coolant.

Core Damage Frequency

Core damage frequency expresses the likelihood that, given the way a reactor is designed and operated, an accident could cause the fuel to be damaged. The estimated frequency of core damage accidents in the U.S. reactor fleet is 1 in 10,000 reactor-years (Deutch et al., 2003). A single

* Internal dose conversion factors can be found in USEPA (1988).

reactor operating at power for 12 continuous months represents one reactor-year; with an average U.S. fleet capacity factor of 89.6% (NEI, 2006a) and 165 operating reactors, each calendar year currently represents approximately 148 reactor-years. The theoretical expected core damage frequency is approximately once every 67.7 years. WASH-1400 (NRC, 1975b) estimated the frequency of a worst-case reactor accident at approximately one per billion reactor-years, or with the current U.S. fleet once every 600,000 years. Through the end of 2005, the worldwide reactor fleet has operated for about 12,000 reactor-years (UIC, 2006). Commercial reactors are allowed to operate with very small amounts of fission products in the coolant but are required to shut down the reactor when fission product concentrations reach or exceed critical values that are set far below concentrations that could threaten public health and safety. Core damage represents a loss of the fuel integrity fission product barrier but by itself is not sufficient to threaten the public as long as the coolant system and containment barrier remain intact. In most reactor events, core damage would be a problem for the reactor's operators and owners but not for the public living near the plant.

REACTOR FUEL

Commercial power reactors are fueled with a large number of upright tubes (*fuel rods*) that are generally fabricated of stainless steel alloys containing zirconium; the alloy is often referred to as *Zircaloy*. Each tube is filled with ceramic uranium fuel pellets that are enriched in the isotope ^{235}U. For ease of handling, these individual tubes are bundled together in fuel assemblies, and each assembly has a square matrix of fuel rods. The metal alloy tube surrounding and confining the fuel pellets is the *cladding*. The cladding holds the fuel pellets in a rigid geometry, collects the fission gases produced during power operation, and provides an effective surface to transfer heat from the fuel pellets to the moderator (water). The fuel rods in boiling water reactors are designed to facilitate the formation of steam bubbles along the surface of the rods so the upper third of each rod is in contact with steam rather than water; these fuel rods are designed to produce most of their nuclear power and heat at the bottom of the rod. Pressurized water reactors are operated to prevent boiling in the reactor vessel, and the fuel rods are designed to transfer heat more evenly along their length.

A new, unused fuel assembly contains refined uranium and a small amount of natural uranium, an alpha emitter whose isotopes all have very long (millions of years) half-lives. Within its metal cladding tube, the unused fuel presents essentially no radiation protection hazard. When placed in an operating reactor, the fission process produces neutrons, causing the transmutation of nonradioactive atoms in structures and components into radioactive isotopes and also producing fission fragments, the remains of split uranium atoms. The curve describing fission fragment formation has two peaks, one between atomic numbers 85 and 105, the other between atomic numbers 130 and 155, both peaks having abundances (conversion efficiency) between 0.1% and about 8%. Some of these fission fragments are gases, primarily isotopes of krypton and xenon, although there are also volatile nuclides such as iodine and cesium. The gases escape the individual fuel pellets and collect in empty spaces engineered into the fuel rods; the remaining nuclides are particulates that remain trapped in the pellet ceramic or in the metal of reactor components.

ISOTOPES OF INTEREST IN REACTOR ACCIDENTS*

A very large number of radioisotopes are produced in an operating power reactor. Although theoretically one could analyze the contributions of each one to radiation dose, in practice the set is too large to work with conveniently. The usual practice is to concentrate on the most significant isotopes; however, there are various ways to define *significant*.

* This section was developed using data from Lin (1996), Neeb (1997), McKenna and Glitter (1988), Unterweger et al. (1992), USDHEW (1970), and www.periodictable.com. The dispersion calculations were performed by the author using a simplified straight-line Gaussian model.

Total Core Inventory

One way to rank isotopes is by the total activity in the core. In this respect, the most significant isotope is ^{239}Np at about 1.63E6 Ci (for a 1000-MWe reactor), followed by approximately equal activities of ^{133}I, ^{133}Xe^{m}, ^{135}Xe, ^{99}Mo, ^{140}Ba, ^{140}La, and ^{134}I, all at about 1.6E5 Ci, then a group consisting of ^{91}Y, ^{132}Te, ^{132}I, ^{91}Sr, and ^{103}Ru, all at approximately 1.2E5 Ci. A group of isotopes consisting of ^{89}Sr, ^{131}I, ^{144}Ce, ^{88}Kr, ^{87}Kr, ^{133}Xe^{m}, ^{129}Sb, ^{106}Ru, ^{85}Kr^{m}, and ^{131}Te^{m} have core inventories of between 1E3 and 1E4 Ci. The isotopes ^{134}Cs, ^{90}Sr, ^{136}Cs, ^{95}Zr, and ^{135}I have core inventories between 1E3 and 1E4 Ci. The isotopes ^{60}Co, ^{85}Kr, ^{54}Mn, ^{125}Sb, ^{3}H, ^{99}Tc, and ^{129}I have core inventories of less than 1E3 Ci.

Half-Life

One can rank isotopes by their half-lives, either longest to shortest or shortest to longest. Some might rank the longest lived isotopes as more significant because they are present the longest. Others consider the shorter lived isotopes as being more important because they tend to contribute more to dose. Beginning with the shortest lived, the most significant is ^{134}I, with a half-life of 0.875 hours. The isotopes ^{87}Kr, ^{132}I, ^{56}Mn, ^{88}Kr, ^{129}Sb, ^{85}Kr^{m}, ^{135}I, ^{135}Xe, and ^{91}Sr have half-lives between 1 hour and 10 hours. Of this group, ^{135}Xe^{m} has the largest initial inventory, and it takes about 20 days for the entire core inventory to decay. However, even though ^{91}Sr has a somewhat lower initial inventory (1.1E6 Ci vs. 1.7E6 Ci for ^{135}Xe^{m}), it takes longer to completely decay away. A total core inventory less than or equal to 1E-10 Ci was chosen as representing the complete decay of the initial isotopic inventory.

The isotopes ^{24}Na, ^{133}I, ^{131}Te^{m}, ^{140}La, ^{239}Np, ^{99}Mo, ^{132}Te, and ^{127}Sb have half-lives of between 10 and 100 hours. For this group, ^{239}Np has the largest initial inventory, at approximately 1.64E6 Ci; it takes about 125 days for the entire inventory to decay away. In this group, ^{127}Sb is present the longest, taking about 175 days to completely decay. The isotopes ^{133}Xe, ^{133}Xe^{m}, ^{131}I, ^{131}Xe^{m}, ^{140}Ba, ^{136}Cs, ^{51}Cr, ^{129}Te^{m}, and ^{103}Ru have half-lives of between 100 hours and 1000 hours. For this group, ^{131}Xe^{m} has the largest initial inventory, at approximately 1.7E5 Ci; it takes about 600 days to completely decay away. In this group, ^{103}Ru is the longest lasting; approximately 0.0025 Ci is still present after 1000 days.

The isotopes ^{89}Sr, ^{91}Y, ^{95}Zr, ^{58}Co, ^{57}Co, ^{144}Ce, ^{54}Mn, and ^{106}Ru have half-lives of between 1000 hours and 10,000 hours. In this group, the isotope with the highest initial inventory is ^{91}Y, at 1.2E5 Ci; approximately 1 Ci is left after 1000 days. The isotope in this group that is present the longest is ^{106}Ru, which is reduced to approximately 3900 Ci after 1000 days from an initial inventory of 2.5E4 Ci; it takes about 5500 days to decay to 1 Ci.

The isotopes ^{3}H, ^{134}Cs, ^{22}Na, ^{125}Sb, and ^{60}Co have half-lives of between 10,000 and 100,000 hours. In this group, ^{134}Cs has the highest initial inventory at approximately 7500 Ci; after 6500 days, the inventory is reduced to approximately 19.2 Ci, and after 9750 days the inventory is approximately 1 Ci. In this group, there is approximately 58.5 Ci ^{60}Co remaining after 6500 days; the inventory of ^{60}Co is reduced to approximately 1 Ci after 18,000 days. The isotopes ^{85}Kr, ^{90}Sr, ^{137}Cs, ^{99}Tc, ^{135}Cs, and ^{129}I, have half-lives greater than 100,000 hours. In this group, ^{137}Cs has the highest initial inventory at 4700 Ci; after 40,000 days, there is still an inventory of approximately 380 Ci, and the total inventory of ^{137}Cs is reduced to 1 Ci after about 135,000 days. This is also the isotope having the largest remaining inventory of the group after 40,000 days. ^{90}Sr is the other significant isotope in this group, remaining at 40,000 days and being reduced to 1 Ci after about 125,000 days.

External Radiation Dose Factor

Direct external radiation from a plume of radioactive material is a major contributor to the radiation dose to the public during a release of radioactive material. Seven isotopes deliver greater than or equal to 1 R/h external radiation exposure from a total activity of 1 Ci: ^{24}Na at 1.919 R/h, ^{134}I at 1.573 R/h, ^{132}I at 1.421 R/h, ^{60}Co at 1.369 R/h, ^{136}Cs at 1.273 R/h, and ^{88}Kr at 1.025 R/h. The

isotopes ^{135}I, ^{59}Fe, ^{58}Co, ^{54}Mn, ^{87}Kr, ^{91}Sr, ^{133}I, ^{137}Cs, ^{103}Ru, ^{131}I, ^{132}Te, ^{135}Xe, ^{140}Ba, ^{57}Co, ^{106}Ru, ^{99}Tc, ^{99}Mo, ^{133}Xe^{m}, and ^{133}Xe each deliver between 0.1 and 1.0 R/h external exposure, with ^{135}I having the greatest exposure rate at 0.862 R/h. The isotopes ^{134}Cs, ^{95}Zr, ^{144}Ce, and ^{51}Cr each deliver between 0.01 and 0.1 R/h external exposure, with ^{134}Cs delivering the most exposure at 0.07 R/h.

Although it is useful to consider only the magnitude of the direct radiation exposure factor, the actual exposure (dose) delivered also depends on initial core inventory, decay rate, total release rate to the environment, and downwind dilution factor (i.e., dispersion). Consider a hypothetical release that is measured 0.5 miles from the release point in an atmosphere with C stability class (i.e., a dispersion factor of 6E-5 using data from *Turner's Workbook*). The total release rate is 1E6 μCi/s, and the hypothetical member of the public is exposed for 1 hour. All isotopes are present in the ratios found in the initial reactor core inventory (a very simplified and unrealistic assumption), and in-flight decay and other removal mechanisms are not considered. In this example, six isotopes deliver between 1E-6 and 1E-5 R/h: ^{134}I at approximately 6.5E-6 R/h, ^{140}La at 5.6E-6 R/h, ^{132}I at 4.7E-6 R/h, ^{131}I and ^{88}Kr at approximately 2E-6 R/h, and ^{91}Sr at 1.3E-6 R/h. Ten isotopes each deliver between 1E-6 and 1E-7 R/h, seven isotopes deliver between 1E-8 and 1E-7 R/h, and five isotopes deliver less than or equal to 1E-8 R/h. The total external exposure rate for this release is approximately 2.8E-5 R/h (note that this value does not include the submersion dose from the eight Kr and Xe isotopes).

Inhalation Dose Factor

A significant amount of the dose delivered to the public in a hypothetical reactor accident is due to the inhalation of radioactive materials. ^{144}Ce, ^{131}I, ^{106}Ru, and ^{91}Sr have inhalation dose conversion factors (in rem/μCi breathed) between 0.1 and 1.0 rem/μCi, with ^{144}Ce delivering the highest dose at 0.256 rem/μCi. ^{134}Cs, ^{60}Co, ^{137}Cs, ^{103}Ru, ^{95}Zr, and ^{59}Fe have inhalation dose factors between 0.01 and 0.1 rem/μCi breathed, with ^{134}Cs delivering the highest dose at 0.0318 rem/μCi. ^{32}Te, ^{133}I, ^{125}Sb, ^{129}Te^{m}, ^{22}Na, ^{136}Cs, ^{140}Ba, ^{90}Sr, ^{238}Np, ^{131}Te^{m}, ^{54}Mn, ^{135}Cs, ^{140}La, ^{58}Co, ^{127}Sb, ^{135}I, ^{99}Tc, ^{24}Na, and ^{99}Mo have inhalation dose factors between 0.001 and 0.01 rem/μCi breathed, with ^{132}Te delivering the highest dose at 8.68E-3 rem/μCi. ^{57}Co, ^{129}Sb, ^{132}I, ^{56}Mn, ^{134}I, ^{51}Cr, and ^{3}H have inhalation dose factors between 1E-4 and 1E-3 (0.001), with ^{57}Co delivering the highest dose at 9.17E-4 rem/μCi. The eight Kr and Xe isotopes are not considered because they each have essentially zero uptake fractions in the lung.

Although it is useful to consider only the magnitude of the inhalation dose conversion factor, the actual dose delivered also depends on the initial core inventory, decay rate, total release rate to the environment, downwind dilution factor (i.e., dispersion), and order in which isotopes are released from the fuel. Consider the previous 1E6-μCi/s release rate with all isotopes present in the ratios found in the initial reactor core inventory. In this case, three isotopes deliver greater than 0.1 rem: ^{144}Ce at approximately 0.57 rem, ^{91}Sr at 0.34 rem, and ^{91}Y at 0.11 rem, totaling 1.02 rem/h. ^{106}Ru, ^{131}I, ^{103}Ru, ^{133}I, ^{132}Te, ^{140}Ba, and ^{140}La each deliver between 0.01 and 0.1 rem/h, with ^{106}Ru delivering the most at approximately 0.09 rem. The total dose from this group is approximately 0.3 rem/h. The five isotopes ^{134}Cs, ^{99}Mo, ^{137}Cs, ^{132}I, and ^{131}Te^{m} deliver inhalation doses between 1E-3 and 1E-2 rem/h; the total dose from this group is approximately 0.016 rem. Eight isotopes each deliver inhalation doses between 1E-4 and 1E-3 rem/h: ^{129}Te^{m}, ^{134}I, ^{95}Zr, ^{90}Sr, ^{136}Cs, ^{129}Sb, ^{60}Co, and ^{127}Sb. The total dose for this group is approximately 4.4E-3 rem/h. The remaining six isotopes each deliver between 4E-5 and 5.4E-11 rem/h, for a total dose of 6.4E-5 rem/h. In this example, the total inhalation dose rate is approximately 1.5 rem/h.[†]

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[*] The distance of 0.5 miles is a representative distance for the nearest member of the public to a reactor at most sites; a release rate of 1E6 μCi/s under most conditions will produce a radiation exposure rate of about 1 mR/h.

[†] An accident producing this dose rate would require declaration of a General Emergency classification based on exceeding a projected 1 rem/h at the site boundary and would require protective actions to be taken offsite. However, the expected inhalation dose at 2 miles would be 0.15 rem/h, which does not exceed Protective Action Guides. The dispersion factor at 2 miles is 5E-6 compared to 6E-5 at 0.5 miles.

The most likely reactor accidents will release only the radioactive gases stored in fuel assemblies and will not release radioactive materials from within the ceramic fuel pellets. Correcting the above example to remove the refractory fission products, three isotopes deliver inhalation doses greater than 0.01 rem/h: ^{131}I at 0.17 rem, ^{133}I at 0.07 rem, and ^{91}Sr at 0.036 rem. If significance is defined as delivering dose at greater than or equal to 1E-4 rem/h, the remaining significant isotopes are ^{132}I at 1.86E-3 rem, ^{134}Cs at 6.8E-4 rem, ^{137}Cs at 3.13E-4 rem, and ^{91}Y at 1.16E-4 rem. The remaining 17 isotopes each deliver from 8E-15 to 9E-5 rem, or a total dose for the group of approximately 2E-4 rem/h. The above calculations used a release rate of 1E6 μCi/s; all dose values simply increase by a factor of 10 for rates of 1E7 μCi/s, by 100 for rates of 1E8 μCi/s, etc.

Limitations on Measurement

The previous discussion assumed that an analyst had information about all of the isotopes that are present in a reactor. In practice, this is not the case during at least the initial several days of an accident. In an actual event, information will only be readily available for γ-emitting isotopes, and only those with characteristic energies above 100 to 200 keV. This partially explains why ^{131}I is considered to be such a key isotope, while ^{133}I, which has about the same dose impact, is essentially ignored. ^{133}I is a γ emitter with a characteristic energy of 56 keV, which is very difficult to measure. Analysts will depend on plant effluent radiation monitors and on in-field exposure rate (e.g., mR/h) measurements; in either case, the measured values are input into dispersion models that have a pre-established, assumed distribution of isotopes. The isotopes that are carefully tracked during an event and considered key are those that rank highly in all of the above attributes, total core inventory, half-life, external exposure, inhalation dose, and ease of measurement. Therefore, the key isotopes are typically considered to be ^{60}Co, ^{91}Sr, ^{131}I, ^{134}Cs, ^{136}Cs, and ^{137}Cs.

SOURCE TERM AND CORE INVENTORY

The starting point for analyzing the consequences of a damaged power reactor is understanding the accident source term. NUREG-1228 (McKenna and Glitter, 1988, p. 1-2) states that, "This characterization of radionuclides that may be released to the environment, in conjunction with release rate and height, is referred to as the 'source term.'"* The source term is the collection or distribution of the radioisotopes that exists at the transition from the reactor site to the environment, along with their initial concentrations. The transition point to the environment could be the top of a chimney, vent, or stack; it could be a hole in the side of a plant building such as an engineered blowout panel; or it could be a damaged seal, such as the gasket surrounding the external hatch of a reactor containment building. The source term is not necessarily the distribution and concentration of radioisotopes measured in reactor coolant or in containment structure atmosphere samples, because both natural and engineered processes act to change and deplete the reactor coolant distribution before it reaches the environment.

The isotopes with the most significant contribution to early health effects were identified in WASH-1400 (NRC, 1975b) and are shown in Table 3.1. These values are similar to those used in the accident models described in NUREG-0925 (NRC, 1983e) and NUREG/CR-3108 (Wilson et al., 1983). These assumptions are considered by the Nuclear Regulatory Commission (NRC) to have a probable error of less than 25%. The inventory of long-lived (half-lives of 10 years or more) isotopes is proportional to the burn-up factor or the length of time the core has been in operation. The inventory of short-lived (half-lives of 30 days or less) isotopes is proportional to the core power density, and the inventory of activation products found in core components is proportional to the average reactor power level.

* NUREG-1228 (p. 1-2) also states, "To be useful, these [radiological consequence] projections should be based on a best-estimate assessment of the source term and not on artificial assumptions intended only for licensing purposes. … These assumptions should not be used to characterize an actual accident."

TABLE 3.1
Total Curies of Radioactive Isotopes Available in a Reactor Core (While Operating, per Megawatt Electric)

Isotope Group	Total Activity (Ci)
Noble gases	520,000
Halogens (iodines)	715,000
Alkali metals (cesium)	15,200
Tellurium group	177,500
Strontium group	367,700
Noble metals (Ru, Rh, Mo, Co)	295,000
Cerium and transuranic metals	1,760,000

Source: Data from NRC, *Source Term Estimation During Incident Response to Severe Nuclear Power Plant Accidents*, NUREG-1228, U.S. Nuclear Regulatory Commission, Washington, DC, 1988.

TABLE 3.2
Total Curies of Radioactive Isotopes Available in a Reactor Core (Immediately After Reactor Trip/Shutdown)

Group	Class	Isotopes	Total Activity (Ci)
1	Noble gases	^{85}Kr, ^{85}Kr^{m}, ^{87}Kr, ^{88}Kr, ^{133}Xe, ^{133}Xe^{m}, ^{135}Xe	3.84E8
2	Halogens	^{131}I, ^{132}I, ^{133}I, ^{134}I, ^{135}I	7.71E8
3	Alkali metals	^{134}Cs, ^{136}Cs, ^{137}Cs, ^{86}Rb	2.18E7
4	Tellurium	^{127}Sb, ^{129}Sb, ^{127}Te, ^{127}Te^{m}, ^{129}Te, ^{129}Te^{m}, ^{131}Te^{m}, ^{132}Te	2.13E8
5	Strontium	^{89}Sr, ^{90}Sr, ^{91}Sr, ^{92}Sr	3.57E8
6	Noble metals	^{58}Co, ^{60}Co, ^{99}Mo, ^{105}Rh, ^{103}Ru, ^{105}Ru, ^{99}Tc^{m}	5.94E8
7	Lanthanides	^{241}Am, ^{242}Cm, ^{244}Cm, ^{140}La, ^{141}La, ^{142}La, ^{95}Nb, ^{147}Nd, ^{143}Pr, ^{90}Y, ^{91}Y, ^{92}Y, ^{93}Y, ^{95}Zr, ^{97}Zr	1.54E9
8	Cerium	^{141}Ce, ^{143}Ce, ^{144}Ce, ^{239}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu	2.15E9
9	Barium	^{139}Ba, ^{140}Ba	3.38E8

Note: The group numbers, classes, and isotopes are as given by NRC, *Accident Source Terms for Light Water Nuclear Power Plants*, NUREG-1465, U.S. Nuclear Regulatory Commission, Washington, DC, 1995.

Although there is variation and some minor disagreement among the different sources regarding the total radioactive inventory of a reactor, a 1000-MWe reactor core contains about 4.2E9 Ci of total activity, with inventories of each critical isotope being between 5E4 and 1E9 Ci. One source* proposed the data shown in Table 3.2 as the typical core inventory at 30 minutes after reactor shutdown (taking radioactive decay into account). If 100% of this inventory was released at a constant rate over 24 hours under constant meteorology, the author calculates that a hypothetical person located 1 mile downwind would receive approximately 8.18E18 rem deep-dose equivalent (DDE) and 2.4E4 rem committed effective dose equivalent (CEDE).†

* Presented in training material for NRC Course P-300, Accident Progression Analysis, Idaho National Engineering and Environmental Laboratory, 2001. The table applies to a reactor of 3300 MW-thermal. The approximate decay periods (in days) for each group are (1) 0.118, (2) 8.04, (3) 0.121, (4) 3.21, (5) 1.06E4, (6) 7.29E-3, (7) 1.58E5, (8) 1.38, and (9) 12.8.

† Calculated using a straight-line Gaussian model with a C stability class.

The isotopes that contribute most to external dose are ^{24}Na, $^{110}Ag^{m}$, ^{134}I, ^{132}I, ^{60}Co, ^{136}Cs, ^{124}Sb, ^{88}Kr, $^{131}Te^{m}$, and ^{54}Mn. Those contributing most to internal doses are ^{154}Eu, ^{152}Eu, ^{90}Sr, ^{129}I, ^{144}Ce, ^{106}Ru, ^{60}Co, ^{91}Y, $^{110}Ag^{m}$, and ^{131}I.* The normal concentration of most nuclides in reactor coolant water during power operation is between 0.1 and 0.001 μCi/g, with pressurized water reactors having somewhat higher concentrations on average than boiling water reactors of the same power rating.† The normal coolant concentrations of the transuranic metals may be as low as 1.0E-8 μCi/g (their concentration is not readily determined because alpha emitters are not detected by gamma spectroscopy).

While a reactor is in operation, the noncompressible radioactive gases produced by the fission process must be continually removed. In boiling water reactors, this occurs in the condenser when steam is cooled back to water; the radioactive gases are drawn off and discharged through a chimney (*stack*) or roof vent. In pressurized water reactors, small amounts of gases pass through defects in the steam generator tubing, but most of the gases are removed by the chemical volume and control system. They are then discharged through the gaseous radioactive waste system either through a vent at the top of the containment structure or from a small site stack. Depending on the reactor design, site-specific systems, and specifics about the fuel being used, a typical plant discharges between 800 and 1000 μCi/s to the environment; boiling water designs generally have higher discharge rates than pressurized types of the same power rating. Commercial reactors have operating limits for the maximum amount of radioactive material released offsite and maximum release rates‡ that are generally about 5.0E4 μCi/s (measured by a beta–gamma detector calibrated to noble gases, often to $^{133}Xe^{m}$), although a few plants have limits in the range of 1.0E5. Discharges at rates exceeding these limits for more than a few minutes (often, 15 minutes) indicate emergency conditions.

The normal environmental radiation background rate around most commercial reactors is between 20 and 40 μR/h. During normal operations, if radioactive water or steam lines are breached, the radioactive material released to the environment would not be detectable at the site boundary using most portable survey equipment (generally limited to about 0.25 mR/h or 250 μR/h); this includes reactor coolant water with a normal concentration of nuclides.

CORE STATES

Fuel damage is defined as any perforation of the cladding that permits the release of fission gases or fission products. It can result from physical strain of the fuel assembly or from fuel overheating. A radioactive release cannot occur unless the physical barriers that isolate the activity from the environment are all breached. These are generally considered to be the integrity of fuel cladding,§ the integrity of the reactor coolant system, and the integrity of the primary containment. Some texts also describe fuel pellet integrity as a fourth barrier, but the ceramic matrix only acts as a barrier for some particulate fission products and for heavy metals. The overall core state is the state of fuel cladding averaged across all of the fuel assemblies. The status of each fission product barrier is determined by several predefined, measurable, parameters; when a threshold value is reached in any parameter, the associated barrier is considered to be challenged. A barrier may be *potentially lost* or *lost*, depending on the parameter and value; any potential loss or loss of a barrier means the barrier is outside of its design parameters and constitutes an emergency condition. Simultaneous potential losses or losses in multiple barriers are more serious, meriting a higher emergency classification, because the plant's margin for preventing a radioactive release is more seriously degraded.

* Data sources are *Radiochemistry in Nuclear Power Reactors* (Lin, 1996) and Health Canada (1999). The isotopes listed have the highest individual gamma exposure factors and internal organ doses; for specific accident sequences, other nuclides may contribute more to the projected dose because of their greater abundance in the reactor's inventory.

† Also see ANSI (1984b).

‡ Typically found in the Radiological Environmental Technical Specifications (RETS), *Technical Requirements Manual*, or *Offsite Dose Calculation Manual Guidance* (Meinke and Essig, 1991).

§ *Cladding* refers to the zirconium–aluminum alloy tubing that surrounds the fuel pellets and holds them in a rigid configuration. The cladding provides a smooth surface, generally free of defects, to facilitate heat transfer from the fuel to the surrounding coolant.

The four generally used terms that describe the condition of a reactor core (by increasing severity) are *normal*, *gap release*, *in-vessel fuel melt*, and *ex-vessel fuel melt*. NUREG-1465 (Soffer et al., 1995) defines the phases of a severe accident as *coolant activity release*, *gap activity release*, *early in-vessel release*, *ex-vessel release*, and *late ex-vessel release*.

In normal power operations, the peak centerline fuel pellet temperature is between 1700°F and 3500°F, depending on reactor type and fuel design, whereas the outer clad skin temperature is about 600°F. Although the cladding may contain or develop pinhole leaks from manufacturing, at this temperature no additional cladding defects are expected to form. The permissible concentration of nuclides in coolant is controlled by the site technical specifications; because of the importance of iodine in Protective Action Guides, one common measure of overall activity in coolant is *dose equivalent iodine* (DEI), a calculation that takes the concentrations of all iodine isotopes present and converts them to an effective concentration of ^{131}I having the same dose consequence. A typical limit on activity in coolant (for boiling water reactors) is a DEI less than 4.0 μCi/g; if this limit is exceeded, operators have 48 hours to correct the condition; otherwise, operators are required to have the reactor subcritical within the next 12 hours.

When fuel cladding exterior temperatures approach 1300°F, defects begin to form in the metal fuel cladding, and these defects are widespread when surface temperatures reach 2100°F; these temperatures are not sufficient to damage or deform the ceramic fuel pellet. Cracks and pinholes in the cladding allow the built-up radioactive gases to escape the fuel rod into the surrounding reactor coolant; essentially all of the gases have escaped by the time surface temperature reaches 2000°F. In addition to the noble gases Kr and Xe, some volatile elements, such as cesium and iodine, that are at the surface of the fuel pellet may also escape. Between 1000°F and 2200°F, the noble gas, iodine, cesium, and tellurium isotope groups are released from the fuel. The fuel condition that allows radioactive gases and volatile elements to escape the fuel rod into the coolant is referred to as a *gap release*. Following a gap release into coolant, the typical concentration of noble gases, iodine isotopes, cesium, and volatile species is between 1.0E2 and 5.0E4 μCi/g, with between 3% and 5% of the total available isotopic inventory being released; up to 80% of the inventory of specific radioactive gases and volatile isotopes may be released. High cladding temperatures tend to develop in regions or zones of the core and do not usually occur homogeneously across the core at the same time; however, the discussion in NUREG-1228 (McKenna and Glitter, 1988) assumes rupture of 100% of the fuel rods during a gap release. Table 3.3 compares the fraction of total core inventory of some important groups of nuclides released from fuel rods during a gap release, as described in several Nuclear Regulatory Commission reports: NUREG-1228 (McKenna and Glitter, 1988), NUREG-1465 (Soffer et al., 1995), and NUREG-5942 (Carbajo, 1993).

After the fuel cladding fails at about 2200°F, the rate of release of fission products from the ceramic fuel increases with temperature, the rate doubling with every 180°F to 200°F. Between fuel cladding temperatures of 2200°F and 2800°F, the strontium and barium isotope groups are released from the fuel; above about 2800°F, the noble metal, lanthanide, and cerium isotope groups are released. Also, when the fuel cladding exterior temperature increases to 3000°F, the fuel pellets begin to physically deform, with melting (liquefaction) occurring by 4500°F. NUREG-0772 (Lorenz et al., 1980) and NUREG-1228 (McKenna and Glitter, 1988) assume that after the cladding exterior temperature

TABLE 3.3
Comparison of Gap Release Fractions Used in Regulatory Guidance

Nuclide Group	NUREG-1228	NUREG-1465	NUREG-5942
Xenon, krypton	0.03	0.05	0.022–0.05
Iodine	0.02	0.05	0.03–0.05
Cesium, rubidium	0.05	0.05	0.028–0.058
Tellurium, antimony, selenium	1E-4	0	0

TABLE 3.4
Comparison of Fuel Melt Release Fractions Used in Regulatory Guidance

Nuclide Group	NUREG-1228	NUREG-1465	NUREG-5942
Xenon, krypton	1.0	0.95	0.73–0.95
Iodine	1.0	0.25	Not discussed
Cesium, rubidium	1.0	0.2 BWR, 0.25 PWR	0.21–0.44
Tellurium group	0.3	0.05	0.005–0.1
Barium, strontium	0.2	0.02	0.03
Ruthenium, molybdenum, cobalt	7E-3	2.5E-3	1.1–1.5E-3
Lanthium group	1E-4	2E-3	0
Cerium group	1E-4	5E-3	0

Note: Table values for NUREG-1465 and NUREG-5942 are for the *early in-vessel* melt category. NUREG-5942 has lanthium and cerium only released during late in-vessel and ex-vessel melt sequences. NUREG-1228 also has the category *grain boundary release*, which begins at local surface temperatures of 3000°F (this category has release fractions of 50% for Xe, Kr, I, and Cs; about 1% for Sb, Ba, and Mo; and less than 1E-3 for Ru and Sr). According to NUREG-1228, the uncertainty regarding release rates from melting fuel is no better than a factor of 10. *Abbreviations:* BWR, boiling water reactor; PWR, pressurized water reactor.

reaches 3600°F the remaining inventory is released at 10% per minute. Regions of the core where melting occurs collapse, forming a molten mixture called *corium*. When corium has formed, a corium–water steam explosion becomes a major concern for operators, as is a hydrogen explosion inside containment (the hydrogen is liberated during a zirconium–water reaction at the metal cladding surface). Table 3.4 compares the fraction of total core inventory of some important nuclides released from fuel during a fuel melt event, as described in several NRC reports: NUREG-1228 (McKenna and Glitter, 1988), NUREG-1465 (Soffer et al., 1995), and NUREG-5942 (Carbajo, 1993).

After water is removed from a fuel assembly, the cladding may reach about 4000°F in about 15 minutes, which forms the basis for several emergency preparedness decision and notification requirements. There is no essential difference between in-vessel and ex-vessel core melt, except that in the latter case the reactor vessel also fails, allowing the corium to escape the vessel.

Fuel damage can only be directly assessed by sampling liquid reactor coolant, in normal operations and during accidents. A sample of normal reactor coolant has a contact radiation exposure of at most a few mrem/h, depending on sample volume. Process radiation monitors provide secondary indications. In boiling water reactors, these include main steam line radiation monitors and reactor water cleanup system radiation monitors; in pressurized water reactors, they include main steam line radiation monitors (to detect steam generator leaks) and letdown radiation monitors. Process radiation monitors are effective in detecting small leaks (slow releases from one or two fuel rods) but are not very effective in assessing more general fuel damage, because (1) the detector saturates at a relatively low dose rate, (2) there is an automatic isolation signal at a relatively dose rate, (3) abnormal operating procedures direct operators to manually isolate the line monitored by the radiation detector, or (4) some other isolation signal causes an isolation of the line being measured. The containment or drywell atmosphere of many reactors can be sampled, but this may require resetting interlock signals, placing jumpers across circuit boards, and manually opening valves in the reactor buildings; automatic containment atmosphere process monitors are physically located outside of the containment structure and are automatically isolated on containment isolation signals. Atmosphere samples and monitors are very indirect indicators of the core state because some radioactive isotopes released from the core remain mostly in liquid coolant and do not readily become airborne, because containment sprays and other mitigation systems change the airborne isotope mixture

compared to what was released from the fuel and atmosphere samples can only be collected for loss-of-coolant accidents. The core state can also be inferred from reactor water level, primarily in boiling water reactors, using the rule of thumb that gap releases occur 15 minutes after the core is uncovered and fuel melting begins about one-half hour after core uncovery (see above discussion); rules of thumb may be highly inexact and are not descriptive of any actual accident sequence, but they may represent all the information that is available.

Most reactor sites have developed site-specific, accident-range radiation monitor curves to estimate core damage based on drywell and containment radiation dose. Section A.4, *Evaluation of Containment Radiation*, of NUREG/BR-0150 (NRC, 2002h) contains similar generic curves. Fuel damage can be inferred from the response of accident range monitors only when the reactor coolant system is breached (a loss-of-coolant accident) because the calibrations are valid only for a uniform or well-mixed submersion in a site-specific noble gas/iodine mixture. Often, curves are given by the licensee that differentiate between gap release and fuel melt conditions but the average radiation energies and decay characteristics of these mixtures are significantly different. The monitors can only be accurately calibrated for one level of core damage, and care must be taken when assessing core damage based on monitor responses to understand the basis for the monitor calibration. Although the accident range monitor curves read into the mR/h range, they cannot be used to evaluate fuel damage below about 1% of the core because high-range monitors utilize built-in *keep-alive* radiation sources that prevent them from reading less than about 1000 mR/h. Reactor containments and drywells also have two installed operating-range radiation monitors, which typically have ranges limited to between 100 and 1000 mR/h, depending on their physical location in primary containment; these monitors respond (increase) to fuel damage even when the reactor coolant system is intact but cannot be used to evaluate the damage. Operating-range containment radiation monitors also respond to reactor coolant system breaches with routine levels of radioactive material but will become saturated (read off-scale) before the activity in the coolant reaches the high end of the allowed operating band. A response on operating-range radiation monitors cannot be used to differentiate between an intact and breached radiation coolant system, and additional confirming indications are required such as the response of containment atmospheric monitors, humidity or hydrogen sensors, containment temperature sensors, or the filling of containment sumps. Because of the keep-alive radiation sources, any response on accident-range monitors above their baseline indicates a breach of the radiation coolant system.

CORE DAMAGE FREQUENCY

Core damage frequency is a term for a mathematical construct used by reactor licensees and the Commission to characterize the safety of a particular configuration of plant equipment. The configuration describes the mode of reactor operation, and the equipment (both safety-related and balance of plant) currently available to respond to an emergency. The core damage referred to is not actual core damage but rather a probability function that the reactor will reach the beginning of core degradation (the start of a core damage sequence); the analysis assumes that, once started, a core damage sequence cannot be mitigated before the damage occurs. For example, in boiling water reactors, the core damage is assumed to occur if the reactor vessel water level lowers to the top of active fuel, even though this level would have to be sustained for 15 to 30 minutes before fuel would actually rupture. The Commission's *Reactor Safety Goal Policy Statement* (NRC, 2000e) includes a secondary goal of maintaining the core damage frequency of each reactor at less than 1E-4 per reactor-year of operation.* It should be emphasized that core damage frequency is a mathematical concept not tied directly to any physical quantity, that core damage frequency is used to determine the acceptability or risk of changes to the reactor plant, and that the core damage frequency value does not drive any emergency preparedness function or decision-making process.

* Also see NRC (2002a).

In practice, the *conditional* or *marginal* core damage frequency is used in regulatory analyses, safety analyses, and real-time risk management computer software to control the allowed reactor plant configuration. The conditional core damage frequency represents the difference in risk between the planned or actual plant configuration and an ideal plant configuration with all systems operational. The plant is allowed to remain operating as long as the conditional core damage frequency remains within a prescribed band. Typically, conditional core damage frequencies in the range of 1E-5 or lower are permissible as long as the resulting overall core damage frequency remains less than 1E-3 to 1E-4.

It is not possible to always maintain the reactor plant in the configuration giving the lowest core damage frequency because plant equipment requires scheduled preventative maintenance and must be periodically run to demonstrate its continuing ability to operate. Some of the riskiest plant configurations are during plant startup and shutdown and while fuel is being loaded and unloaded from the reactor, all of which must be done to operate the reactor.

SEVERE ACCIDENTS

NUREG-1228 (McKenna and Glitter, 1988) states that the fuel cladding surfaces in an uncovered core heat up at 1°F/s because air and steam are much poorer conductors of heat than water. As long as neutrons continue to flow through the core, fissions continue and undissipated heat builds up in the cladding. NUREG-1228 states

> The consensus is that even for the worst accident analyzed, if the plant safety systems work as designed, less than 20% of the fuel pin cladding will fail, releasing a large fraction of the gap in those pins. … This type of accident (within plant design limits) would result in release of considerably less than 20% of the gap from the reactor coolant system. Therefore, any accident that releases more than 20% of the gap from the reactor coolant system is considered a *severe accident*.

The terminology used to quantify fuel damage during a severe accident is site specific, imprecise, and often confusing. The most frequent unit used to express core damage is *percent fuel damage*, which correlates most closely with the percentage of the core that has sustained damage beyond a gap release. Another common convention is that a release (into reactor coolant, not necessarily to the atmosphere or the environment) of 10% of the gap inventory is equivalent to the release of 1% of the fuel inventory.

Reactor licensee emergency operating procedures are designed to save the core and prevent significant fuel damage; abnormal and emergency operating procedures are primarily designed for analyzed accidents and maintain the plant within its design basis. Since 1990, reactor owner groups and licensees have developed severe accident management guidelines (SAMGs) and strategies* that address issues beyond-design-basis events. The severe accident strategies assume a substantial loss of reactor water level and significant (greater than 20%) fuel damage (i.e., loss of reactor coolant system integrity and loss of fuel cladding integrity), and their goal is to save primary containment and to prevent large radioactive release to the environment. The Commission has identified three broad categories of severe accident strategies:

- Replenishing batteries, borated water, compressed air, and other resources that support operation of safety systems
- Innovative uses of plant systems not designed to supply water to the core or to remove decay heat (e.g., fire suppression pumps)
- Defeating interlocks and signals meant to protect plant equipment

These categories can be applied to the functions of reactivity control, reactor coolant inventory, heat removal, and containment integrity. Current industry efforts are aimed at developing additional low-power, shutdown (decay heat only), and fuel building strategies.

* Also see IAEA (2004).

The Commission identified a need for licensees to prepare for severe accidents in an NRC policy statement (1985e). This was followed by Generic Letter 88-20 (NRC, 1988c), which states

> The NRC will evaluate licensee IPE submittals to obtain reasonable assurance that the licensee has adequately analyzed the plant design and operations to discover instances of particular vulnerability to core melt or unusually poor containment performance given a core melt accident. Furthermore, the NRC will assess whether the conclusions the licensee draws from the IPE regarding changes to the plant systems, components, or accident management procedures are adequate.

Licensees were initially expected to submit Individual Plant Examination (IPE) reports by the end of 1991. Individual Plant Examination reports were only required to assess the implications of failures and events internal to the reactor plants. The Commission required an additional assessment of severe accident vulnerabilities due to external events (primarily hurricanes, tornadoes, and flooding) in a 1991 supplement to Generic Letter 88-20. All assessments of external events were expected to be submitted by the end of 1998; none of the severe accident analyses was required to address malevolent acts or site attack scenarios. In the author's opinion, from 1985 to 1991, the Commission clearly intended to develop and promulgate additional regulations requiring licensees to implement a severe accident response program and include these elements in its inspection program.

In 1994, the Nuclear Energy Institute proposed to the Commission that reactor licensees commit to implementing a voluntary severe accident management program. This approach was accepted by the Commission in a 1995 letter to the Nuclear Energy Institute and confirmed in SECY-95-004 (NRC, 1995i).* The Nuclear Energy Institute committed to the Commission that every reactor site would implement such a program no later than December 1998. The reasons why the Commission accepted a voluntary industry initiative to implement severe accident management instead of the formal regulatory program it previously had intended are unclear, but by 1993 (and continuing throughout the remainder of the decade) the Commission was facing budget reductions, and it had already reduced the scope of its evaluation of Individual Plant Examinations of External Events (IPEEEs). Reactor licensees benefited from this approach because the Commission did not impose new regulations and did not incorporate severe accident management into its inspection and drill programs. In particular, licensees did not want the inspection of drills to verify a licensee's ability to implement severe accident procedures, because the plant reference simulators in use during the mid- to late-1990s were incapable of modeling severe accident core melt sequences and simply ceased operating outside of design-basis plant conditions. Although the core thermodynamic and heat transfer models incorporated into current plant reference simulators are much improved compared to those of the 1990s, current simulators generally are not designed to operate in the severe accident regime because the NRC agreed not to require demonstration of these capabilities. As of 2015, reactor operators were not being tested by the Commission regarding their knowledge of severe accident procedures as part of their individual license examination, and reactor licensees were not required to demonstrate implementation of their severe accident management strategies as part of their routine drill and exercise program.

ACCIDENTS INVOLVING STORED IRRADIATED REACTOR FUEL

All currently operating U.S. power reactors store the most recently irradiated fuel rods in deep (about 40 to 50 feet) water pools, which resemble large open rectangular swimming pools. These fuel pools generally contain between two and four complete core loads. The oldest fuel assemblies are 8 to 10

* See also *Integration Plan for Closure of Severe Accident Issues* (SECY-88-147); *Status of Implementation Plan for Closure of Severe Accident Issues, Status of the Individual Plant Examinations, and Status of Severe Accident Research* (SECY 94-166, SECY-95-004); *Status of the IPE and IPEEE Programs* (SECY-96-051); *Status of the Integration Plan for Closure of Severe Accident Issues and the Status of Severe Accident Research* (SECY-96-088, SECY-97-132, SECY-98-131); and *Severe Accident Issue Closure Guidelines* (NEI, 1994).

TABLE 3.5
Inventory of Some Isotopes in a 1000-MWe BWR One Year After Adding Fuel

Nuclide	Curies *1E7	Nuclide	Curies *1E6	Nuclide	Curies *1E5
^{241}Pu	2.19	^{106}Ru	9.31	^{95}Zr	5.10
^{137}Cs	1.97	^{134}Cs	5.80	^{238}Pu	4.54
^{90}Y	1.39	^{85}Kr	1.33	^{242}Cm	3.50
^{90}Sr	1.38	^{95}Nb	1.11	^{241}Am	3.21
^{144}Ce	1.13			^{60}Co	2.85
				^{91}Y	2.21
				^{244}Cm	2.19
				^{240}Pu	1.30

Source: NRC, *Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants*, NUREG-1738, U.S. Nuclear Regulatory Commission, Washington, DC, 2001, p. A3-4.

years old; because of delays in providing a permanent fuel repository, many licensees have reconfigured their fuel pools to hold more used fuel than they were originally designed for (referred to as *re-racking*). Most licensees have now moved, or are in the process of moving, their oldest fuel rods, which date back to original operation, to air-cooled semipermanent storage casks (independent spent fuel storage installations, or ISFSIs) located outside elsewhere on the licensee's property.

Accidents that result in damage to the fuel stored in fuel pools are possible. These accidents develop more slowly than in operating reactors because the latent heat in fuel rods drops rapidly after reactor shutdown. Catastrophic events are unlikely but possible, primarily including natural phenomena such as earthquakes and tornadoes; beginning in 2001, concerns have been raised about aircraft impacts against fuel pools. The worst-case analyzed accidents can have dose consequences comparable to operating reactor accidents (that is, protective actions are required in the emergency planning zone), and the accident probability is about the same as for operating reactors. The major concern for fuel in its initial 36 months of storage is a zirconium fire; this condition can occur if cooling water is removed and the rod is steam or air cooled. In either case, the fuel surface temperature becomes very high (greater than 2200°F) because the heat transfer coefficients of air and steam are so much lower than that of liquid water, and the metal rapidly oxidizes. The zirconium reaction is exothermic (i.e., produces more heat than the heat of combustion) and produces flammable free hydrogen.

The radioactive isotopes with the highest dose rates also tend to be short lived (i.e., 100% of all iodine in fuel rods has decayed by 7 months after reactor shutdown). Fuel pool accidents are dominated by isotopes that are longer lived (greater than 3-month half-lives).* A typical fuel pool has an inventory of more than 40 isotopes with a total radioactive material inventory of about 7E7 Ci. A typical fuel pool source term is given in Table 3.5. Some expected release fractions for a severe fuel pool accident are shown in Table 3.6. These release fractions would be expected for a fuel pool where the stored fuel assemblies remain mostly to fully covered by water. Radioactive gases and volatile particulates are not effectively absorbed into water, and virtually all of these isotopes escape the pool.

The least likely fuel damage accident is to fuel transferred from fuel pools into independent spent fuel storage installations. This is primarily because the fuel rod internal decay heat has decayed enough that external plant cooling systems are not required to maintain fuel integrity. Fuel

* Most power reactor licensees do not have dose assessment computer software that correctly models the fuel pool source term. Although a case could be made that this is not in compliance with 10 CFR 50.47(b)(9) in that the licensee may not possess "adequate methods … for assessing … actual or potential offsite consequences" of a fuel pool not in use, the historical NRC practice has been that this does not constitute a deficient dose analysis program.

TABLE 3.6
Expected Isotopic Release Fractions from a Fuel Pool Accident

Nuclides	Fraction	Nuclides	Fraction
Krypton	1.0	Strontium, barium	0.002
Iodine[a]	1.0	Ruthenium, molybdenum, cobalt	2E-5
Cesium, rubidium	1.0	Lanthium, zirconium, niobium, curium, americium, cerium, plutonium	6E-5
Tellurium, antimony	0.02		

Source: NRC, *Technical Study of Spent Fuel Accident Risk at Decommissioning Nuclear Power Plants*, NUREG-1738, U.S. Nuclear Regulatory Commission, Washington, DC, 2001, p. A4-6; NRC, *A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants*, NUREG/CR-6451, U.S. Nuclear Regulatory Commission, Washington, DC, 1997.

[a] Assuming an initial inventory of 5.65E8 Ci of all iodine isotopes, less than 1 Ci of ^{131}I (half-life of 3.1 days) remains after 7 months, and all other iodine isotopes have half-lives at least 9 times shorter than ^{131}I.

transferred into spent fuel storage installations must have decayed for at least one year; in practice, they have generally decayed at least 5 and often more than 10 years. All fuel located in spent fuel storage installations can be cooled indefinitely by air* without any expected structural degradation, though the on-contact dose rates remain dangerous to personnel health and safety, and so the fuel must be isolated and shielded. Although there are several designs for spent fuel storage installations, all of them feature a multi-ton semipermanent, steel-lined, sealed (inaccessible) concrete storage cask that contains from one to four fuel assemblies; the concrete is sufficiently thick that exterior gamma radiation dose rates require only minimal radiological controls. The storage casks are then located on reactor licensee property away from the operating reactor to eliminate the potential for the cask to be damaged by failures of plant equipment;† at some sites, the storage casks themselves are placed inside reinforced bunkers to further increase their security and reduce accessibility.

Because of their structural strength, individual storage casks are most likely to be affected by only the most powerful earthquakes. The design criteria for casks and storage bunkers make it very unlikely that floods, hurricanes, tornadoes, or other kinds of catastrophic natural events would have any impact. The confined fuel assemblies have generally already lost their most volatile isotopes prior to being loaded into casks

WORLD REACTOR ACCIDENT EXPERIENCE

It is difficult to concisely describe the world's experience with reactor accidents to date. This is partially because of philosophical and technical differences regarding the meaning of the term *accident* and partially because there are no reliable reference documents that summarize either national or international accident reports for any but single-year periods. What can be generally said is that (1) radiation injuries and accidents involving x-rays and radioactive materials (primarily isotopes of radium) first occurred somewhere around 1905 (including events related to medical treatment);

* The need for storing reactor fuel outside of water-filled fuel pools had been anticipated by at least 1981; see ANSI/ANS-18.1-1984, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)*, and Regulatory Guide 3.60, *Design of an Independent Spent Fuel Installation (Dry Storage)*. Although guidance and regulation have provided for independent wet storage—such as ANSI/ANS-2.19-1981, *Nuclear Fuel Facilities—Spent Fuel Storage Installation, Site Selection and Design of an Independent Spent Fuel Storage Installation (Water-Pool Type)*, and Regulatory Guide 3.49, *Design of an Independent Spent Fuel Storage Installation (Water-Basin Type)*—all licensees to date have installed dry or air-cooled storage facilities. The first licensed spent fuel storage installation was in 1986 at the Surry Plant in Virginia.

† Although there have been proposals to establish semipermanent, independently operated or consortium-operated spent fuel storage installations away from specific licensee properties that would receive and store fuel as an interim step before final geological disposal, no such interim repositories have reached the siting or licensing stages.

TABLE 3.7
Summary of Radiation-Related Accidents through 2005

Event	Number	Fatalities	Injuries
Power reactor accidents	2	41	438
Research reactor accidents	4	6	9
Naval reactor accidents	3	18	80
Non-reactor criticality	19	15	27
Irradiator accidents	31	8	39
Criminal use of radioactive material	5	4	1
Totals	64	92	594

Note: Reactor accidents only include those that resulted in abnormal radiation dose; for example, they do not include the Fermi Unit 1 accident of 1966 or the Browns Ferry Unit 1 fire in 1975. A large number of lost radioactive sources, accidental dispersals of radioactive material (e.g., incident in Goinna, Brazil), and other accidents with medical radiation-producing machines are not included. The term *injuries* refers only to non-fatal examples of acute radiation syndrome and does not include non-radiological burn or mechanical injury.

(2) the first accidents involving experimental, test, or military reactors occurred somewhere around 1950; (3) the first accident in a civilian commercial power reactor resulting in fuel damage was in 1966; and (4) there have been four reactor accidents that resulted in radiologically significant offsite releases—Windscale, Three Mile Island, Chernobyl, and Fukushima Daiichi. One unofficial source has provided the breakdown of radiation accidents through 2005 shown in Table 3.7.

NUREG-1437 (NRC, 1996f) discusses the following fuel melt accidents: Fermi Unit 1 (Newport, MI), Saint-Laurent (France), National Research Experimental (NRX) reactor (Chalk River, Canada), Experimental Breeder Reactor (Atomic City, ID), Heat Transfer Reactor Experiments (Idaho Falls, ID), Westinghouse Test Reactor (Waltz Mill, PA), and the Oak Ridge Research Reactor (Oak Ridge, TN). Report LA-3611 (Stratton, 1967) discusses 36 criticality incidents or accidents that occurred worldwide between 1949 and 1965.

A chronological listing of reactor accidents that resulted in damaged fuel, including those at experimental and test reactors, is given below. The list does not include operational events at reactors that did not result in core damage and is not a list of declared emergencies. The list does not discuss the numerous non-reactor radioactive material incidents that resulted in personnel injuries or fatalities.

- *NRX reactor* (Chalk River, Canada; INES 5; December 12, 1952)—An unplanned high-power excursion occurred with a subsequent hydrogen explosion in the balance of plant systems; 1E4 Ci were released into cooling water.
- *Windscale* (Cumbria, England; INES 5; October 10, 1957)—During annealing maintenance on a non-power reactor the graphite moderator caught fire, and the fire was not extinguished for 24 hours. Twenty percent of the core was damaged, and about 8.24E4 Ci were released* (mostly iodine). Worker doses were 150 times occupational limits, and offsite doses were retrospectively calculated at 10 times maximum allowable lifetime dose limits for the public. No evacuation of the public was performed. Restrictions were placed on milk produced in an area of 500 km^2 for about 2 months, and radioactive iodine was found in samples taken outside this area; ^{131}I was measured in milk at concentrations of up to 5.0E4 Bq/L.

* The estimated releases include 18,000 Ci ^{131}I, 850 Ci ^{137}Cs, 240 Ci Po^{210}, 105 Ci ^{89}Sr, and 7.5 Ci ^{90}Sr (the midpoints of the estimate ranges are given) (Eisenbud, 1987). The release duration was 21 hours.

- *NRX reactor* (Chalk River, Canada; May 24, 1958)—A fuel rod fire occurred during refueling, causing building and site contamination.
- *SL-1 reactor* (Idaho National Laboratory, Idaho Falls; January 3, 1961)—An inadvertent criticality during maintenance resulted in a steam explosion. Three fatalities occurred (one victim lived for 90 minutes after rescuers arrived). The deceased workers' bodies measured greater than 100 R/h on contact; 99.9% of the radioactive inventory was released, consisting of about 80 Ci. Measured exposure rates were from 25 R/h at the reactor building wall to greater than 1000 R/h at the reactor vessel; the reactor building was essentially a corrugated tin building not designed as a containment structure. Twenty responders received radiation doses in excess of 1 rem, three of which exceeded 25 rem.
- *Fermi Unit 1* (liquid metal reactor; Newport, MI; October 5, 1966)—A sodium coolant malfunction resulted in melting two fuel assemblies and damaging two more; no radioactive release was detected.
- *Chapelcross reactor* (Dumfries and Galloway, Scotland; May 1967)—A cooling channel blockage (broken-off piece of zirconium) resulted in one fuel assembly catching fire.
- *Swiss research reactor* (Lucens, Vaud, Switzerland; January 21, 1969)—A loss-of-coolant accident resulted in a steam explosion, but no radioactive release was detected. The reactor was located underground, and the contamination was contained in the sealed cavern.
- *KS150 reactor* (Plant A1; Jasloveske, Bohunice, Czechoslovakia; INES 4; February 22, 1977)—Fuel was damaged when the reactor was restarted after fueling because of foreign material left in a fuel assembly.
- *Three Mile Island Unit 2* (Middletown, PA; INES 5; March 28, 1979)—A failed reactor coolant system component followed by mistakes by operators led to a loss of coolant, uncovering of the core, fuel damage, and a hydrogen explosion in containment. About 2E6 Ci of noble gases were intentionally vented (0.9% of inventory), and between 15 and 20 Ci of iodine were vented (3E-5%) of inventory). Three plant workers each received about 4 rem, and members of the public within 8 miles received from 0 to 8 mrem. According to a health study conducted in 1997, there were no detectable health effects in the surrounding population.
- *Reactor A2* (Orleans, France; INES 4; March 13, 1980)—An inadvertent power excursion resulted in fuel damage, and the release of 8E10 Bq to the environment.
- *RA2 facility* (Buenos Aires, Argentina; INES 4; September 23, 1983)—An inadvertent criticality while refueling an experimental test reactor resulted in the release of 3E17 Bq; one operator received a fatal dose of 2000 rads, and 35 other persons each received between 1 and 35 rads.
- *Chernobyl Unit 4* (Prypiat, Ukraine; INES 7; April 26, 1986)—An improperly performed safety test resulted in an uncontrolled power excursion and a steam explosion in the core, followed by complete core meltdown and a graphite moderator fire; operators intentionally disabled some required safety systems prior to conducting the test. The reactor building (not designed as a containment structure) was completely destroyed. The accident resulted in 28 radiation fatalities in 91 days; 238 persons were treated for acute radiation syndrome, 710 persons had measured or estimated doses greater than 50 rads, 20,000 persons had measured or estimated doses between 25 and 50 rads (including stabilization and cleanup activities through 1988), and 130,000 persons received doses in excess of ICRP radiation safety limits for the general public. About 4000 cases of thyroid cancer occurred in children as a result of the accident, with 10 fatal cases.
- *THTR-300 high-temperature gas reactor* (Hamm-Uetrop, Germany; May 4, 1986)—Cladding damage to a fuel pebble that became stuck in a refueling channel caused a radioactive release detectable offsite.
- *Greifswald nuclear power station* (Greifswald, Germany; November 24, 1989)—Equipment failure during a test of safety systems caused overheating in 10 fuel assemblies (similar to Chernobyl in that some safety systems were intentionally defeated prior to the test).

- *Reactor 1, Shika Nuclear Plant* (Ishakawa Prefecture, Japan; June 18, 1999)—An operator error while inserting control rods resulted in three rods being withdrawn, causing an inadvertent power spike.
- *Paks nuclear power plant* (Paks, Hungary; INES 3; April 10, 2003)—Thermal shock to an irradiated fuel assembly being handled after removal from the core caused rods to rupture, spilling fuel pellets into a tank.
- *Fukushima Daiichi Nuclear Power Plant* (Fukushima, Japan; INES 7; March 11–20, 2011)—Extensive core damage in multiple reactors and spent fuel pools following a station blackout caused by a very strong earthquake and a subsequent tsunami. The tsunami flooded the emergency diesel generators and caused the vital emergency power buses to be deenergized. Reactor damage occurred after the station batteries were exhausted.

The four accidents with the most effect on offsite populations were Windscale (the plant was later renamed Sellafield), Three Mile Island, Chernobyl, and Fukushima Daiichi. Table 3.8 compares some nuclides in the radioactive releases to the environment from these accidents.

The *Additional Report of the Japanese Government to the IAEA, Second Report* (Nuclear Emergency Response Headquarters, 2011) estimated that during the first 4 days of the Fukushima accident 4.60E6 Ci ^{131}I were released to the atmosphere, along with 4.87E5 Ci ^{134}Cs and 4.05E5 Ci ^{137}Cs; the total estimated ^{131}I release was later revised to 2.7E4 Ci. An IAEA report (IAEA, 2015b) estimated that the ^{134}Cs and ^{137}Cs releases to the environment from the Fukushima accident were ~14% of the total release from Chernobyl, and the total activity deposited on the ground was 4.6% of the deposition from Chernobyl. The World Nuclear Association estimated in 2015 that 4.33E6 Ci ^{131}I had been released from Fukushima into the atmosphere and 4.05E5 Ci ^{137}Cs. TEPCO, the operator of Fukushima, estimated in 2012 that 2.54E7 Ci ^{131}I had been released into the atmosphere and 2.16E7 Ci of noble gases, primarily ^{133}Xe. Estimates are that 20% of overall releases to the atmosphere from Fukushima came from Unit 1, 40% came from Unit 2, and 40% came from Unit 3, with Unit 4 contributing almost a negligible amount. The calculated Chernobyl release fractions are shown in Tables 3.9 and 3.10, and the estimated Fukushima release fractions are shown in Table 3.11.

The Windscale and Fukushima accidents resulted in significant discharges of radioactivity into the oceans. The primary discharges at Windscale were from water injected into the graphite pile to suppress the fire which was then routed through a discharge canal, into a nearby river, and into the Irish Sea. Discharges from Fukushima were a combination of leaks from the internal structure and discharges from water pumped or sprayed into the reactor core to stabilize the damaged fuel.

TABLE 3.8
Total Airborne Releases (Ci) of Selected Radioisotopes at Four Reactor Accidents

Nuclide	Windscale	Three Mile Island	Chernobyl	Fukushima
^{133}Xe	3.80E5	1.00E7	1.80E8	5.50E8[a]
^{131}I	2.74E4	1.40E2	4.70E7	4.30E6
^{137}Cs	1.40E3	0	2.30E6	4.14E5
^{90}Sr	5.4	0	2.70E5	4.03E3
^{239}Pu	54	0	810	0.09

Source: Data from SCK•CEN (Mol, Belgium) and from Povinec et al. (2013); some unit conversions by the author.

[a] It is estimated that 100% of the ^{133}Xe inventory in Units 1, 2, and 3 escaped to the atmosphere; the value is calculated from NUREG-1228 (McKenna and Glitter, 1988) assuming three 1000-MWe units.

TABLE 3.9
Estimated Release Fractions of Selected Radioisotopes from the Chernobyl Unit 4 Accident

Nuclide	Fraction (Percentage of Core Inventory)	Nuclide	Fraction (Percentage of Core Inventory)
^{85}Kr	100	^{137}Cs	33
^{133}Xe	100	^{140}Ba	4.6
^{131}I	50–60	^{89}Sr	4.6
^{99}Mo	30–50	^{90}Sr	4.6
^{103}Ru	30–50	^{95}Zr	3.5
^{106}Ru	30–50	^{141}Ce	3.5
^{132}Te	25–60	^{144}Ce	3.5
^{134}Cs	33		

Source: OECD, *The Chernobyl Accident Source Term*, Organization for Economic Cooperation and Development, Paris, France, 1996.

TABLE 3.10
Estimated Release Fractions of Iodines, Telluriums, and Cesiums by Reactor Unit

Isotope Group as Defined by NUREG-1465	Unit 1	Unit 2	Unit 3
Iodine	0.7%	0.4%	0.4 –0.8%
Tellurium	1%	0.4–3%	0.3–0.6%
Cesium	0.3%	0.3–6%	0.3–0.6%

TABLE 3.11
Estimated Release Fractions of Selected Radioisotopes from Fukushima Units 1, 2, and 3

Isotope	Release (Percentage of Total Inventory)	Isotope	Release (Percentage of Total Inventory)
^{131}I	2.6%	^{90}Sr	0.03%
^{134}Cs	2.4%	^{238}Pu	1E-6%
^{137}Cs	2.2%	^{240}Pu	1E-6%
^{89}Sr	0.03%	^{241}Pu	1E-6%

Source: Povinec, P.P. et al., *Fukushima Accident: Radioactivity Impact on the Environment*, Elsevier, Amsterdam, 2013.

It is difficult to estimate the total amount of radioactive material discharged into the ocean as a result of the Windscale accident, in part because the majority of the available data are for the larger airborne discharges and in part because the site has continued to be used until the present day. For example, the Sellafield site released 158 Ci of ^{137}Cs into the Irish Sea in 2011, ~52 Ci ^{90}Sr, and ~2.6 Ci ^{134}Cs, according to the Radiological Protection Institute of Ireland (McGinnity et al., 2012). It is

TABLE 3.12
Isotopic-Specific Release Fractions into the Oceans

Stagnant Water		Direct to the Ocean	
Isotope	Percentage (%) of Total Core Inventory (Units 1–3)	Isotope	Percentage (%) of Total Core Inventory (Units 1–3)
^{89}Sr	1.2	^{54}Mn	0.016
^{90}Sr	1.6	^{60}Co	0.11
^{99}Tcm	0.58	^{90}Sr	1E-5
^{115}Sb	0.015	^{125}Sb	2.8E-4
^{131}I	32	^{134}Cs	0.49
^{134}Cs	20	^{137}Cs	0.5
^{136}Cs	17	^{144}Ce	3.0E-5
^{137}Cs	20		

Source: Povinec, P.P. et al., *Fukushima Accident: Radioactivity Impact on the Environment*, Elsevier, Amsterdam, 2013.

estimated that about 6.5 Ci ^{137}Cs were discharged from Windscale during the entire year (1957) in which the fire occurred. Other important isotopes discharged during the accident were ^{95}Zr, ^{95}Nb, and ^{106}Ru/^{106}Rh (Jeffries, 1968; Mauchline and Templeton, 1963).

According to the Japan Atomic Industrial Forum (JAIF) Earthquake Report #250, the Fukushima site released a total of 7.33E5 Ci to the ocean between March 21 and July 15, 2011. The French Radioprotection and Nuclear Safety Institute estimated in 2011 that a total of about 7E5 Ci of ^{137}Cs alone had been released to the ocean, about 5% of the estimate of the total radioactivity released to the atmosphere. The second Japanese Government Report to the IAEA estimated that through April 6, 2011, Fukushima released a total of 1.27E5 Ci of all isotopes to the ocean. TEPCO estimated in 2012 that releases to the ocean totaled 2.97E5 Ci ^{131}I, 9.46E4 Ci ^{134}Cs, and 9.73E4 Ci ^{137}Cs. Other estimates of releases to the ocean are 1.08E5 to 1.07E6 Ci ^{137}Cs (Buesseler, 2014), 2.52E4 Ci to 1.08E5 Ci ^{137}Cs (Dietzel and Kriest, 2012), and 2.7E3 Ci ^{90}Sr (Povinec et al., 2012). Some published values of the fractions of specific isotopic inventories released to the ocean are given in Table 3.12.

DAMAGE TO REACTORS FROM SITE ATTACK

There have been concerns about the possibility of armed attacks against nuclear power plants since at least 1993, when an intruder drove through fences at Three Mile Island, and these concerns were compounded after the 9/11 airliner attacks in the United States. Since 2001, both generic and specific vulnerability studies of nuclear power plant designs have been conducted by the Department of Energy, Department of Homeland Security, and the Nuclear Regulatory Commission, primarily to determine whether containment structures could withstand the impact of large aircraft or missiles. Although the specifics of these studies are classified or sensitive and are not available to the public, the Commission has publicly discussed their conclusions: Containment structures are in general strong and robust, and the likelihood of both damaging the reactor core and releasing sufficient radioactive material into the environment to affect public health and safety is very low.

Similar studies have been conducted on behalf of the nuclear industry by the Electric Power Research Institute (EPRI, 2002), which reached similar conclusions. These studies are also not available to the public but the threat basis has been described: a fully fueled (23,980 gallons of aviation fuel) Boeing Model 767-400 weighing approximately 400 tons at an impact speed of 350

mph. The aircraft is 201 feet long, with a wingspan of 170 feet. Each of its two engines weighs 9500 pounds, and the engines are located 15 meters apart so that only the fuselage and one engine are assumed to impact the structure (the other engine being deflected by the curvature of the structure). The study assumed the aircraft and engine struck perpendicular to the containment building centerline. The analysis of fuel pool integrity assumed the aircraft and engine made a perpendicular hit at the mid-point of the fuel pool wall. The models are primarily based on a 1988 study by Sandia National Laboratory in which a 27-ton (54,000-pound) F-4 Phantom fighter plane was propelled into a 3.7-meter (145-inch) thick reinforced concrete wall at 480 mph.*

ACCIDENT SEQUENCES

The four emergency classifications are arranged in a pyramid configuration, with many emergency action levels that result in classifying a Notification of Unusual Event and very few that result in classifying a General Emergency. Although there are almost infinite combinations of plant equipment failures that could result in an emergency classification, there are only three severe accident categories that can result from those combinations: a reactor coolant system break inside containment, a steam line break outside containment with a failure to isolate (stop) steam flow, and a break in a liquid recirculation line outside containment with a failure to close the recirculation line.

The reactor coolant system break inside containment can lead to an unfiltered and unmonitored release directly to the atmosphere if the containment structure breaks or any external hatch seals fail. A filtered and monitored release may occur from containment into plant buildings (secondary containment) if piping or electrical penetration seals fail or some lines fail to isolate (i.e., because all containment penetrations have at least two isolation valves arranged in parallel, both the inboard side and outboard side values must fail). Valve failures in containment purge and vent lines could also create a path for radioactive material to reach the environment.

An unisolated steam line break can lead to an unfiltered and unmonitored release if it occurs in the steam tunnel or if the plant turbine building has lost integrity. A steam break around the turbine generator and condenser leads to a filtered and monitored release through turbine building ventilation. A steam generator tube rupture is a special case of a steam line bypass accident that may be unfiltered and unmonitored if the steam line safety valves lift; otherwise, the release is filtered and monitored.

A reactor coolant system break inside containment causes containment building sump pits (an engineered safety feature) to fill with contaminated water from condensed steam. When the refueling water storage tank (or other large external water sources used to inject water into the core through safety injection systems) are emptied or exhausted, a water circulation path is established from sump pits through heat exchangers external to containment, back into the reactor core, and out the reactor coolant system breach to the sump pit. This water can contain a large inventory of radioactive material if core melting has occurred. A recirculation line break outside of containment puts this contaminated water into other plant buildings and creates a release path to the environment.

SOURCES OF RELEASES

Radioactive releases to the environment can originate from the reactor primary containment, from the spent fuel cooling pool, from systems that process and transfer radioactive waste, or from areas where processed radioactive waste is stored. Although the available radioactive material inventory

* Numerous references state that 96% of the aircraft's kinetic energy was dispersed by the destruction of the aircraft and 4% was absorbed by the concrete wall. Unlike reactor containment structures, the concrete used in this test did not have embedded steel rebar. The Sandia report is not available to the public. See also Sugano et al. (1993) and Hessheimer and Dameron (2006).

and source terms are different for each release pathway, the emergency response to the release component of the accident is the same: (1) maintain situational awareness using available installed radiation monitoring systems, (2) perform local radiation surveys, (3) dispatch environmental survey and monitoring teams, (4) collect site and environmental samples, (5) estimate the resulting potential dose to the public using appropriate radioactive material transport models and real-time meteorology, and, for the licensee, (6) when appropriate, communicate protective action recommendations to offsite authorities.

Processed radioactive waste is generally made up of suspensions, resins, and slurries containing long-lived (half-lives greater than several months) radioactive solids and metals stored in drums, barrels, or a larger volume high-integrity container (often a cylindrical polyethylene container holding about 200 to 300 ft^3 of waste). This waste has undergone weeks to months of radioactive decay, and the source term is greatly reduced from what was present when the waste was collected. Waste of this type presents a local radiation hazard but under normal conditions there is no potential to affect the public. If a container seal fails, the physical form of this waste limits its mobility, although it could cause a local airborne radioactivity area resulting in some localized radioactive material contamination and potentially intakes of radioactive material by nearby workers. Any release of radioactive material from stored waste sufficient to cause a radiation dose to the public (beyond or away from the licensee's property) requires a catastrophic event that simultaneously breaches multiple containers, such as a tornado or major fire. The radioactive material would move directly from the breached containers into the atmosphere without any delay, filtration, or reduction mechanisms and would not be measured by plant radiation monitoring systems. Releases of this kind would not usually be expected to result in offsite protective actions, although detectable offsite contamination may be possible.

Plant systems that collect, process, filter, and hold radioactive waste can contain radioactive gases, liquids, resins, slurry, suspensions, and solids of half-lives greater than a few hours. The radioactive materials are confined in piping, ducts, filters, holding tanks, and processing tanks and have undergone hours to weeks of radioactive decay. Plant areas around these pipes, filters, and tanks present local radiation hazards but under normal conditions there is no potential to affect the public; seal leaks in piping carrying radioactive gases and unsealed open tanks holding radioactive resins, slurry, and solids both may require airborne radioactivity controls. A release of radioactive material sufficient to cause a radiation dose to the public requires more than minor failures or breaches of waste system piping, ducts, and tanks. Radioactive gases are immediately discharged to the environment through plant ventilation systems, while radioactive material from other physical forms are collected and discharged after drying, which allows solid materials to become suspended in air. Most radioactive gases would be delayed but not filtered or reduced, whereas most aerosols and particulates would be both delayed and filtered prior to release. Releases through engineered plant ventilation systems are measured by plant radiation monitoring systems. Most releases from plant radioactive waste systems would be short, less than 30 minutes, because the affected tank or piping volume is small or because automatic systems or plant operators isolate the breach, or both. Releases of this kind would not be expected to result in offsite protective actions, and those involving radioactive gases would not result in offsite contamination.

The spent fuel cooling or holding pool (often just called the *fuel pool*) holds highly radioactive fuel assemblies after they are removed from the reactor, with isotopes of half-lives ranging from hours to millions of years. The pool water provides cooling to dissipate decay heat in the assemblies, provides radiation shielding to personnel working around the pool, and filters radioactive material released through minor defects in fuel cladding. Because of the water and the thickness of concrete used in pool construction, there is not normally a radiation hazard near the pool. Small amounts of radioactive gases may be present near the pool during routine operations because of fuel defects but in insufficient concentrations to be either a local or environmental hazard. Similarly, low concentrations of radioactive solids are found in the pool water but at insufficient concentrations to be a radiation hazard.

A release of radioactive material sufficient to cause a radiation dose to the public requires either direct physical damage to fuel rods or an extended loss of the pool's ability to remove decay heat. Direct physical damage could occur if a fuel assembly drops while being moved,* if the assembly is incorrectly lowered into its holding rack, or if a sufficiently heavy foreign object drops into the pool onto one or more stored assemblies. A fuel pool could lose the ability to remove decay heat if the entire pool cooling system (usually two or more redundant sets of pumps and heat exchangers) fails, allowing the pool water to heat to boiling, which can lead to localized steam cooling of fuel assemblies, a zirconium–water reaction, and rupture of the fuel. A loss of pool cooling is particularly significant within the first months after fuel is removed from the operating reactor. The same result would occur if there is a catastrophic failure of fuel pool integrity, allowing pool level to fall below the top of stored fuel; a loss of pool integrity also causes very high radiation levels around the reactor site because of the loss of radiation shielding material. Direct physical damage would normally affect one or at most only a few fuel assemblies, and the primary hazard would be from radioactive gases not being absorbed by the water; a release of this kind would be expected to be of short duration, a few minutes or less, as the gases are fully discharged and would primarily be a hazard to plant workers. An extended loss of fuel pool cooling, whether from a cooling system or pool integrity failure, affects the entire fuel rod inventory, and all assemblies aged (approximately) 36 months or less are at risk of structural failure; a release of this kind could be similar in both duration and concentration (though not in source term isotopes) to that caused by damaged fuel in the operating reactor, with additional complications from the very high radiation conditions onsite.

All reactors are designed with at least one filtered plant ventilation system that collects radioactive gases and aerosols from the fuel pool area; the filters remove essentially all of the radioactive iodine, volatile gases and aerosols, and radioactive particulates (with an initial removal efficiency of at least 99.99%), leaving only the noble gases. In boiling water reactors, the entire reactor building is kept at negative pressure relative to the environment and acts as a secondary containment structure; the structural integrity of the building is protected by blow-out panels designed to rupture at an interior overpressure of a few psig, creating an unmonitored, unfiltered pathway to the atmosphere; these panels are often located in the fuel pool area. Even though boiling water reactor fuel pool areas are very large spaces, a loss of fuel pool cooling scenario can result in pool boiling, with resulting overpressures sufficient to cause a loss of the blowout panels. Except for designed accident-range radiation detectors, working-level area radiation monitors on the adjacent refuel floor do not have sufficient measuring range and will become saturated (off-scale on the high end) during an accident.

Gaseous releases from one to a few fuel assemblies with an intact pool would not be expected to result in offsite protective actions, and radioactive contamination would not be expected offsite. For loss of fuel pool cooling and loss of pool integrity scenarios, radiation doses could be sufficient to warrant protective actions for the public, and in the case of boiling water reactors where blowout panels are lost protective actions would be expected.

DRYWELL AND CONTAINMENT RADIATION MONITORS

Containment structures for both boiling water and pressurized water reactors generally have two ranges of installed radiation monitors, with at least two instruments in each range (each one powered from a different vital electrical bus). Operating range detectors are usually unshielded quenched Geiger–Mueller detectors with a top detection range of 10,000 mR/h (100,000 mR/h at some plants); in radiation fields above their top range, the detectors become saturated and either

* All operations involving handling spent fuel rods are performed under many (15 to 25) feet of water because of the potentially lethal radiation hazard. Spent fuel rods are never allowed to be exposed to air. Fuel assemblies transferred to dry independent spent fuel storage facilities are loaded into their storage casks underwater, sealed, then drained and dried; the multi-ton concrete and steel casks provide radiation shielding comparable to the depth of water in the fuel pool.

read zero (on older model instruments) or continue to read at their maximum value. Accident range detectors can be shielded Geiger–Mueller instruments but may also be shielded sodium iodide (NaI) crystals sized between 4 and 16 in.^{2}. Although accident range detectors may have lower detection limits less than 1 R/h, they are generally installed with radiation sources that ensure constant readings between 1 and 2 R/h during normal operations; their maximum ranges are typically 10,000 to 100,000 R/h. Normal operating range detectors and accident range detectors are typically installed in different areas of the containment structure to minimize the possibility the detectors will be destroyed; therefore, even when the normal operating range detector response overlaps the accident range detector range, the readings are not directly comparable because of differences in detector geometry, and they cannot be used to validate one another; this is especially true when the monitors are of different physical type (Geiger–Mueller vs. sodium iodide).

Accident range monitors are calibrated to a site-specific average energy that corresponds to the average energy of an assumed nuclide mix at a stated time after the accident (details and assumptions are found in the site's final safety analysis report; see Chapter 5), and the detector does not provide accurate indications for other accident sequences or times. The calibration must consider both the containment geometry and shielding. The average radiation energy of a loss-of-coolant accident (reactor coolant system breach) is nearly constant for the initial 2 to 3 hours after fuel damage, decreases over the following 24 hours, and then as short-lived high-energy nuclides decay off begins to increase again. Between 2 and 24 hours after fuel damage, the accident range monitors tend to under-report actual dose rates; thus, concentrations of airborne radioactive material derived or calculated from containment dose rates are also less than what is actually present.

Accident range detectors typically receive a live source calibration only when they are installed and may not be subsequently live-tested in their installed configuration. Live source calibrations are often done with ^{137}Cs (662 keV) or ^{60}Co (1332 and 1173 keV) sources due to their ready availability at accident ranges; care must be taken with these live calibrations because the dominant energies of cesium and cobalt may not closely approximate the average energies of the site-specific accident source term.

Containment monitors are useful only for loss-of-coolant accident sequences (reactor coolant system breaches inside primary containment). For containment bypass scenarios with severe fuel melting, an increased radiation response is expected on all of the normal and accident range detectors because of the release of radioactive material from the fuel into the coolant system, but this increase cannot be used to diagnose the core status because the accident range calibration assumes an equal concentration of airborne radioactivity throughout the containment volume, which is not the case when the material remains confined in coolant system piping. No significant additional response should be expected on these monitors for containment bypass with low to moderate fuel damage, or no damage at all.

Accident range radiation monitors are used (1) as recognition or entry conditions for emergency action levels, (2) to estimate the amount of core damage, and (3) to bound the public health consequences of a release of airborne radioactive materials to the environment. Most sites have prepared site-specific core state curves based on generic vendor-supplied curves; the Commission uses similar generic curves that are not site specific.* These curves give an analyst a rough idea of core state, but they must be used with care and with an understanding of their limitations: (1) they do not apply when the reactor coolant boundary inside the containment structure is intact, (2) they do not apply to containment bypass scenarios, (3) the minimum monitor reading (in R/h) may be above the low end of the curve (the keep-alive source may mask and overstate the true radiation reading), (4) the curves assume a 100% transfer of nuclides from coolant to atmosphere (although this may not be true), (5) the curves assume a homogeneous nuclide mixture in containment (although this may not be true), (6) when curves are provided that take into account containment or core sprays it is often to determine whether or not the sprays are effective, and (7) the effects of isotopic decay and source depletion are not considered.

* For example, see NUREG-0150, *NRC Response Technical Manual*, Revision 4, Figure A.5, p. A-30.

FUEL STORAGE POOL RADIATION MONITORS

Boiling water reactors usually have radiation monitors installed on the refuel floor with at least two ranges, one instrument for normal operations and one for accidents; the normal range instrument typically reads from below 1 mR/h to a maximum of 1000 mR/h, whereas the accident range typically reads from 1 to 1E6 mR/h. Pressurized reactors with separate fuel buildings usually have only installed accident range monitors. Both reactor types typically have two instruments in each range, fed from different vital or essential power sources.

Normal ventilation from pressurized reactors has installed process radiation monitors that normally automatically isolate on signals slightly above background (between 5 and 10 mR/h). These detectors are frequently scintillation detectors; high-range detectors may be shielded with calibration adjustments to compensate for shielding absorption. The fuel building release path effluent monitor is generally calibrated to noble gases, as it is assumed that the 25 feet of water over the fuel racks will absorb all particulates (all halogens are either very reactive with water or have such half-lives that there is essentially no inventory left 200 days after they are removed from the reactor). These systems are designed to detect a gap release from a small number of fuel assemblies as scrubbed by the water overlay (e.g., a fuel rod dropped onto a small number of racked assemblies during handling).

Boiling water reactor refuel floors are normally exhausted via the plant chimney, and pressurized water reactor fuel handling buildings normally have their own engineered vent to the environment. Care must be taken not to use main chimney or stack effluent monitor readings when assessing the consequences of a fuel handling accident when the associated radiological release is not exhausted through the chimney; in boiling water reactors where the refuel floor is exhausted through the plant stack, the main stack noble gas monitor results require adjustment to compensate for the different source term (while iodine is present in stored fuel assemblies the iodine monitor channel should not require compensation).

Most plants also have a radiation detector installed on the fuel bridge. This bridge is a moveable platform spanning the fuel storage pool that supports the fuel assembly grapple and the fuel handling operator's grapple control station. It may also be equipped with an array of underwater lights and inspection cameras, as well as remote-telemetry electronic dosimeters. The fuel bridge monitor is generally a mid- to high-range Geiger–Mueller instrument with a local display and a high dose rate alarm. It is designed to alert workers on the fuel bridge when the grapple lifts too high (in case the upper limit switch fails) so they can secure the fuel assembly in a safe position and exit the area before receiving a radiation overexposure. This monitor may not always be connected to the normal plant radiation monitoring system, may not be recorded by the plant computer, and may not feed an alarm in the plant control room.

SOURCE TERM DEPLETION

Source term as used in dose assessment refers to the inventory and concentrations of radioactive material that survive the transit from the core to the environment. Therefore, the source term that is available to produce a radiation dose to the public is reduced from the inventory initially released from reactor fuel. The depletion and reduction mechanisms include the natural processes of diffusion, radioactive decay, particulate settling, and chemical scavenging and the technological processes of filtration, rainout (washout from sprays), and flooding. Although it occurs, credit is not given for the retention of radioactive material in primary coolant as an inventory reduction process.*

* NUREG-1228 (McKenna and Glitter, 1988) states that "a comparison of available computer projections shows dramatically different plate-out (reactor coolant system retention) for different chemical forms and events." NUREG-0956 (Silberberg et al., 1986) states that "primary retention factors cannot be used rigorously as a multiplier of accident source term nor can they be combined linearly with other retention factors." Therefore, reactor coolant system retention generally will not be considered. However, for bypass accidents, system retention is the only reduction mechanism, and a reduction factor typical of those predicted by computer codes for this accident will be used (RDF of 0.4).

Diffusion describes the tendency of a gas or vapor to expand to fill up the available space, which tends to reduce its average density or concentration; this is especially true as superheated water under pressure flashes to steam at normal atmospheric pressure. *Particulate settling* describes the tendency of larger and heavier particulates, especially those of 1-μm diameter or larger, to settle out in the absence of significant air currents. *Chemical scavenging* describes chemical reaction processes and bonding that result in some radioactive species becoming bound to surfaces and airborne dusts because of their chemical properties, rather than because of their radiological properties. This can be particularly important with the halogen species and with chemical forms that readily react with hydrogen, fluorine, boron, and some other highly reactive chemicals typically found as additives to reactor coolant.

Filtration describes any process where the gas or vapor is made to flow over, past, or through a high-efficiency particulate filter, charcoal bed, resin bed, impaction line, or other mechanical device that removes radioactive material by mechanical or chemical means, absorption, or adsorption; a holdup or decay line acts to increase the time required for effluents to reach the plant chimney, creating time for increased radioactive decay; therefore, these system features are not filters. *Rainout* describes the physical interaction of airborne radioactive material in vapors, particulate, and contaminated steam with water; the water spray or mist physically removes the material through impact (agitation), dissolves the radioactive material, and then traps it, or it causes the material to adhere to non-radioactive particles (i.e., dusts) in the air. In plant buildings, the water is supplied by engineered sprays (core sprays, drywell sprays, containment sprays). Water can also be sprayed on external plant vent or release points using *ad hoc* procedures (e.g., fire hoses); when the radioactive plume moves through natural rain (or snow) essentially the same process depletes the plume and concentrates particulates on the ground closer to the plant. When radioactive gases or steam can be exhausted or bubbled under water, the cooler water quenches and absorbs some steam. Some radioactive material dissolves in the water, and some radioactive particulates are ejected from the effluent stream and become trapped in the water (they do not have enough energy to become airborne). The process of removing radioactive material by exhausting it through water is often referred to as *scrubbing.* A similar effect occurs when steam and hot gases are vented through ice condensers, producing water and acting as a physical filter.

Flooding describes the ability to trap radioactive materials, primarily particulates and halogen and not radioactive noble gases, within the damaged reactor core by covering the core with water (unrelated to the water's core cooling function). The inherent decay heat of a damaged core will still drive steam production, which facilities radioactive material becoming airborne, but a layer of water greatly reduces the rate of transfer to air. Some radioactive species will also dissolve in the water.

NUREG-1228 (McKenna and Glitter, 1988) estimated that natural processes (the interaction of diffusion, decay, settle, and plate-out) remove about 60% of aerosols and particulates over 30 to 60 minutes and remove about 94% over 12 or more hours. Most of the benefit of natural processes occur in the initial 24 hours, after which the marginal rate of inventory reduction decreases significantly. Sprays increase the early removal effectiveness from 60% to 97%. Bubbling radioactive steam through water, such as a torus or suppression pool, reduces aerosols and particulates by between 95% and 99%, depending on the water temperature. The water in a steam generator has a 50% reduction factor (the steam generator is less effective at scrubbing than is the torus because of its much smaller water volume and its higher water temperature). These estimates of plume reduction apply only to a single puff or bubble of radioactive effluent that is subsequently not affected by any other process.

Settling and deposition are assumed to not have a significant effect on radioactive material in transit, such as through a fan or ventilation system, due to the relatively fast transit times as compared to the settling velocity of about 1 cm/s. Scavenging can occur while material is in transit but estimates are not available regarding its efficiency, which is assumed by NUREG-1228 and NUREG-1465 (Soffer et al., 1995) to be small. Decay will be significant only for those nuclides whose half-lives are on the order of a few (less than 10) minutes, due to the relatively short transit times as compared to their half-lives.

Sprays achieve an activity reduction by a factor of 20 in the first hour of operation, after which the marginal rate of continued reduction begins to decrease. After 24 hours, the overall reduction may be on the order of 500. When sprays have been secured, the remaining decay heat may be sufficient to cause material previously removed by spraying to become resuspended in containment and available for release.

The initial effectiveness of filtration trains* is very close to 100% for particulates and halogens, particularly for high-efficiency filters (99.999% effective), and effectively zero for radioactive noble gases; as the filter media become coated, flow through the filter decreases and differential pressure increases until the filter ruptures. As charcoal traps halogens, its adsorption efficiency decreases until it has essentially no removal capability left. Temperature and humidity also affect the effectiveness of charcoal, with higher temperatures and more liquid content tending to impair halogen removal and decreasing its overall removal effectiveness.

It is difficult to calculate the depleted environmental source term starting from knowledge of the core state because it is often not known (or knowable) what removal processes are present. Also, interactions and synergies between various processes and the accident chemistry are unknown, there is a continuing input of new radioactive material mixing with previous (partially depleted) material, and there may be an outflow of radioactive material through design basis leakage or an active containment breach. Simply put, the radioactive material inventory in reactor coolant or the containment air space is usually not close enough to steady-state conditions to allow a useful prediction. From an operational perspective, any technological actions that deplete the radioactive source term are desirable, whether or not their results are detectable or quantifiable.

RELEASE POINTS (AIRBORNE PATHWAYS)

Boiling water reactors are generally designed with a single elevated plant stack (chimney) which receives air flow from all of the engineered plant ventilation systems; this stack is the release point for all routine plant effluents. The stack is described as *elevated* when it is at least twice the height of nearby plant buildings; the stack is also considered *isolated* if it is located a distance at least twice the height of plant buildings away from the nearest plant building. A typical boiling water reactor stack is between 60 and 110 meters high and 2 to 3 meters in diameter, and it has a normal operating gas flow of greater than 100,000 cfm.

Pressurized water reactors are generally designed with three routine plant effluent stacks (often called *vents*): one for the combined primary containment and the auxiliary systems building pathway, one from the turbine building, and one from the fuel building. These stacks are typically attached to the building and are of low height (from roof-height vents to chimneys of less than 10 meters), and they have normal operating flows much smaller than in boiling water reactor designs (10,000 to 50,000 cfm). Engineered release points have installed radiation monitors and go through high-efficiency particulate filters (>99.9% effective) and charcoal filtration beds. Boiling water reactors also are designed with two independent low-flow (10,000 cfm) monitored backup filtration systems (*standby gas treatment systems*) that draw from the torus (suppression pool) air space and primary and secondary containments. Release point filtration systems are highly effective in removing radioactive halogens (iodine and cesium) and particulates, but they do not remove radioactive noble gases (krypton and xenon).

The primary source of radioactive gases released from boiling water and pressurized water reactors during normal operations is the steam condenser. During an accident, radioactive effluents continue to flow through the engineered release point while a normal steam-flow path still exists (i.e., main steam line isolation valves are open). If the containment structure cannot be isolated, breaches

* Assumes a train that consists of an electric dryer system to remove small amounts of suspended liquids or vapors, a preheater, a high-efficiency particulate pre-filter, a charcoal filter bed at least 6 feet long, and a high-efficiency particulate post-filter.

in any steam or water system that flows between containment and the plant building may create an open path to the environment (i.e., *containment bypass scenarios*). The inability to close both main steam isolation valves in the same steam line, followed by a steam line rupture in the steam tunnel or turbine building, is one such scenario. Other systems that can create bypasses include exhausts from steam-driven turbines, containment sump recirculation lines, and the shutdown cooling (pressurized water reactors) or residual heat removal (boiling water reactors) systems. A few older boiling water reactors (e.g., Dresden Units 2 and 3) also have a once-through emergency heat exchanger (isolation condenser) located in containment that rejects decay heat directly to the environment—a tube leak in this heat exchanger allows radioactive material in the reactor coolant to flow directly to the environment. Steam line pressure relief valves at boiling water reactors discharge inside the drywell to the water-filled torus or suppression pool, but safety relief valves at pressurized water reactors discharge directly to the environment. During a steam generator tube rupture event, this could provide an unmonitored, unfiltered pathway for contaminated steam.

During loss-of-coolant accidents, failures in engineered penetrations through the containment wall can create unmonitored release paths, such as primary and emergency air lock seals, seals around personnel and equipment hatches, and seals around piping and electrical lines. Pressure-driven, non-specific, design basis leakage from steam-filled containment structures also provides pathways directly to the environment; depending on the containment's design, from 0.5% to 1% of the containment cavity volume may escape per day. NUREG-1228 assumed non-specific leakage rates of 0.10% to 0.25% per day for pressurized water reactor containments (large/dry) and 0.5% leakage per day for boiling water reactor drywells (wet well) and primary containment structures.* Note that at the time NUREG-1228 was written all operating boiling water reactors used either Mark I or Mark II containments, Mark III design plants had not yet gone into operation; NUREG-1465 (Soffer et al., 1995) did not address the subject of containment leakage.

Pressure spikes can occur in drywell or containment structures when high-pressure coolant flashes to steam after a reactor coolant system break, after a steam line break inside containment, or after a hydrogen burn or detonation (the standard curves for hydrogen burns are based on low-pressure or near-atmospheric events and may overestimate the amount of hydrogen required under high-pressure conditions). Pressure spikes may directly damage the containment structure, causing cracks and holes that allow the escape of trapped radioactive gases; depending on containment design, the final safety analysis report safety limit is between 20 and 60 psig, with structural failure expected at pressures approximately double the safety limit value (newer plants typically have lower safety limits than older plants). With containment coolers in operation it is likely that containment safety limits would not be challenged until several hours into a loss-of-coolant accident. A hole in the containment structure 2 ft^2 or larger is considered to be catastrophic containment failure; a typical accident analysis assumption is that following catastrophic damage 100% of the available (existing) radioactive material inventory in containment is discharged to the environment over one hour (not including any additional radioactive inventory added to containment during that hour). A failure to isolate containment due to the failure of seals in a through-wall valve is assumed to result in releasing 100% of the existing radioactive material inventory from containment over the subsequent 24 hours; this is equivalent to a hole of 8 $in.^2$

Boiling water reactors are designed with an unmonitored, unfiltered emergency vent pipe† connecting the primary containment to the environment which is used to relieve containment pressure before it reaches the structural safety limit. It is considered sufficiently important to prevent a structural failure of the containment that pressure is relieved regardless of the radiation dose to the public delivered by the released radioactive material. Pressurized water reactors have monitored and filtered containment purge systems that can be used to relieve containment pressure, but emergency use of these systems requires defeating interlocks and isolation signals or placing jumpers in

* Many computer-based dose assessment programs do not model passive containment leakage. The NRC's radiological assessment model (RASCAL 3.X) uses a default containment leakage value of 0.2% per day.

† At most boiling water reactor sites, this emergency pressure relief is referred to as the *hardened vent*.

control boxes, so they cannot be operated immediately in the same manner as the hardened vent. Emergency venting may cause very high radiation conditions along the ventilation ducts that connect containment to the atmospheric vents because of shine from the radioactive material being exhausted; after the venting is complete, these radiation conditions may persist for an extended period of time because of fission product plate-out on the interior duct surfaces and the collection of radioactive material at duct connections and elbows.

LIQUID RELEASES

Liquid and sludge-like radioactive material is generally produced by plant radioactive waste systems that filter and remove radioactive material from reactor coolant or store or process the material downstream of the filter. This radioactive material contains a relatively high concentration of longer lived activation product isotopes, such as cesium, cobalt, iron, manganese, tellurium, and zirconium. At some plants, this material is periodically discharged from the plant into surface waters, to either an onsite settling pond or a lake, river, or ocean. A few plants (e.g., Palo Verde in Arizona) are zero release sites, which do not have any engineered liquid discharge points. The greatest potential radiation dose to humans from liquid plant effluents is from drinking after the direct uptake of effluents by a surface water supply system; washing and bathing with contaminated water produce a much smaller dose. Pure surface water supply systems are relatively rare and are not commonly located within miles of nuclear power plant outfalls. Most water systems are capable of being isolated for up to several hours without appreciable impact on system performance. Another concern is the transfer of radioactive material into home-grown (garden) food after plants are watered by contaminated surface water. Compared to airborne effluents, liquid effluents

- Have a very low total radioactive content.
- Are released at relatively low flow rates and volumes.
- Come from a small number of sources (primarily storage tanks), primarily located below grade with no ready release paths.
- Have high engineered and natural mixing (dilution) rates.
- Present little opportunity for direct radiation exposure (shine) because of the low concentration and high self-shielding.

As with engineered airborne release paths, intentional liquid releases require multiple valves to be aligned to the open position (system valves are normally closed, with at least one valve locked closed), and all release paths must have installed radiation monitors.

PLANT PROCESS RADIATION MONITORS

Process radiation monitors are installed on plant liquid and gas lines, primarily to detect breaks and failures in the system during normal operation. They often are Geiger–Mueller instruments with relatively low ranges, the high values being 100 to 1000 mR/h. These detectors are generally unshielded and are susceptible to radiation shine from nearby pipes, tanks, lines, or equipment rooms; monitors located near external walls also can experience shine from an external radioactive plume. Because many of these monitors are in plant radioactive waste systems handling mostly longer lived nuclides, they are frequently calibrated to either ^{137}Cs or ^{60}Co. Most process monitors input into automatic line or ventilation isolations that occur at low radiation levels. They may not remain representative under accident conditions (calibrations, temperatures, pressures, or flow rates). Some process monitors used in plant emergency action levels (e.g., failed fuel monitor) may automatically isolate, or be isolated manually as directed by emergency operating procedures while still on-scale. As count rates increase, instrument dead-time correction becomes a problem and can cause the detector to under-report actual radiation conditions by as much as 25% to 30%.

Plant effluent monitors are a special case of process monitor. Many plant monitors are high-range Geiger–Mueller instruments with parallel detectors for noble gases, iodine/halogens, and particulates; older scintillation-type detectors are often found abandoned in place because they proved to be unreliable. Low-flow pumps pull from the vent or chimney effluent stream and direct flow in parallel past the separate detector channels. Each channel is calibrated to a marker isotope. Channels may read in counts per minute (requiring a manual conversion to activity units) or in μCi/cc or μCi/s, or the plant computer may provide conversions so that results are simultaneously available in multiple units. The plant computer may also apply an adjustment factor to the raw count rate to correct for differences between the average energy of the marker isotope and the calibration source. Plant effluent monitors can have the following problems: (1) because they are high-count-rate instruments, dead time correction can create a significant under-reporting of the true count rate; (2) the detectors are often unshielded or partially shielded, leaving the detector susceptible to radiation shine from the effluent stream where both the vertical and horizontal plume components act as line sources; (3) because the detectors rely on air pumps, a loss of power to the pump or a pump failure causes the radiation monitor to be unreliable; and (4) because the sampling lines are open to the environment they are susceptible to changes in environmental conditions (low and high temperatures, rain, snow, and dust loading). Plants with chimneys located away from plant buildings can have the associated effluent monitors housed in a remote building away from the actual chimney, resulting in long runs of small-diameter piping with bends and elbows that deplete particulate and halogens from the effluent through impact against the bend walls; they also require a constant effluent temperature and are particularly affected by cold winter temperatures (leading to under-reporting of release rates).

Older plants may be equipped with emergency air sampling systems that automatically redirect flow from the chimney air sample pumps through activated charcoal canisters or cartridges as the concentration of radioactive material in the effluent increases. The systems are not designed to sample containment or drywell atmospheres. Although the systems are essentially identical, they have different names because they were made by different vendors; two common ones are the wide-range gas monitor (WRGM) and separate particulate, iodine, and noble gas (SPING) monitors. These systems automatically isolate the normal (low-range) effluent monitoring channel when the mid-range detector reads on-scale (there is some range overlap between the low- and mid-ranges). The systems feature a mid-range charcoal filter and a separate high-range filter (again, with a range overlap between mid- and high-range detectors). These systems do not have a real-time measurement capability. The charcoal canister must be physically removed and analyzed by spectroscopy; a two-person team requires 30 minutes to exchange sample canisters and under accident conditions team members may incur a substantial radiation dose. The used charcoal cartridge will contain substantial radioactivity, and a heavily shielded transport container is required. There may be flow time restrictions on the system that require a minimum sample time of between 8 and 12 hours. One design quirk is that, as effluent concentration falls, the system cannot automatically realign to its normal detection mode; it requires a local reset. Alignment to the normal operating mode can cross-contaminate the detector with higher level activity from the accident unless the system is well purged (usually with nitrogen).

IN-PLANT AIR SAMPLES

During an accident, radiation protection technicians or chemistry technicians are likely to be dispatched to collect manual in-plant grab samples because process monitors are not designed to sample the atmospheres of plant buildings. This information is primarily used to plan radiation protection actions for emergency workers, although it could provide an input to offsite dose assessment. The most commonly used instrument for air sampling is a low-flow (<5 cfm) sampling pump with a 4- to 5-inch sampling head; all sampler heads have a particulate prefilter (paper or fiberglass) and a cartridge of absorbing media. The most common absorber is activated charcoal, which has a high affinity for noble gases and radioisotopes over a wide variety of environmental conditions.

When iodine is known or suspected, the media may be silver zeolite,* which is much more effective at adsorbing iodine than is charcoal. Activated charcoal is often preferred over zeolite because of cost—charcoal is around $1 a cartridge vs. $40 to $50 per cartridge for zeolite (the price fluctuates with the price of silver). Grab air samples are desirable because they provide the best estimate of the conditions workers will be exposed to and may provide a better estimate of the environmental source term than does a reactor coolant sample or containment/drywell atmosphere sample. Problems with grab air samples include (1) dust or (non-radioactive) particulate loading on the prefilter, which both degrades the absorption media's ability to collect radioactive material and introduces self-shielding for the beta–gamma counting of the filter, leading to an underestimate of the airborne radioactive material concentration; (2) although the sample may be collected in 5 minutes or less, sample analysis requires 30 minutes to a few hours;† (3) the technician collecting the sample may incur a substantial radiation dose; and (4) there may be difficulties in making or obtaining high-activity air sample calibration sources for onsite counters, leading to inaccurate or uncertain measurements.

Neither charcoal nor zeolite are particularly effective in collecting radioactive noble gases. The best method for collecting these gases is a *sample bomb*, which is a vessel evacuated to less than atmospheric pressure. The sample is collected when the collection valve is opened, allowing the empty vessel to be filled with the sample gas. A common collection vessel is a sealed cylindrical Marinelli beaker in sizes between 1 and 5 liters. Collection may be through a particulate prefilter to screen out any halogens, iodines, or fission product particulates, so that only the noble gases are collected. Marinelli beakers are commonly used to count radioactive liquids and solids, but obtaining calibration sources for an accident mix of noble gases may be a problem.

A plant is likely to have a few portable continuous air monitors, which are mostly used during plant outages and have local displays and alarms (i.e., they are not connected to the plant computer system). Emergency response facilities, particularly the ones located onsite, will often have installed continuous air monitors, located in the facility proper or sampling from the facility ventilation system. Portable monitors may be used in the plant under accident conditions but they are bulky and heavy (usually mounted on a large wheeled cart), the technician putting them into service may incur substantial radiation dose not justified by the data acquired, and (as noted) they usually lack a remote data capability. Continuous air monitors intended for routine plant operations may not be capable of measuring accident condition concentrations (i.e., they may go into saturation at concentrations much above the upper alarm threshold).

POST-ACCIDENT MONITORING SYSTEMS

After the Three Mile Island accident, all power reactor stations were required to install or design the capability to collect post-accident reactor coolant (liquid) samples; this requirement was intended to address the difficulties the licensee and Nuclear Regulatory Agency had understanding reactor core damage during the March 1979 accident. The initial requirements to install a post-accident sampling capability were contained in NUREG-0578 (NRC, 1979l) and NUREG-0660 (NRC, 1980g). NUREG-0737 (NRC, 1980a) required that by January 1, 1982, all licensees must have the capability installed to promptly obtain samples under accident conditions without incurring radiation doses to plant workers greater than 3 rem whole body and 18.75 rem to the extremities. Licensees were required to have the capability to sample the primary coolant and containment sump and containment

* For a discussion of historical problems with silver zeolite, see NRC Information Notice 86-043 (NRC, 1986f). The same event is also discussed in Gavila (2003a).

† The minimum measurement sensitivity and the amount of time required to analyze a sample of radioactive material is determined by the square of the background (i.e., no sample present) radiation count rate. The onsite beta–gamma counters require a relatively low radiation background to obtain an accurate measurement, even when located inside a substantial radiation shield; under accident conditions, the external radiation count rate may become too high, effectively making the counters inoperable for all except the highest activity samples.

air. The systems were to be designed so that samples could be collected and analyzed within 3 hours of the decision to initiate sampling. "Accident conditions" in this context referred to Regulatory Guide 1.3 (USAEC, 1974a) or 1.4 (USAEC, 1974b) releases of fission products. Secondary conditions included the following: (1) the design must not depend on operating a system previously isolated from the reactor, (2) a backup grab sample capability must exist when inline instrumentation is used, and (3) the samples obtained must be representative of the reactor coolant and containment atmosphere. Regulatory Guide 1.97, *Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident* (Revision 0, 1975; Revision 2, 1980; see also NRC, 2006a) also has design requirements for post-accident sampling systems. To ensure that licensees maintained and could operate post-accident sampling systems, NUREG-0654 (NRC, 1980c), Planning Standard N(2)(e)(2), required an annual demonstration drill including actual use of the system to collect a liquid sample; site technical specifications required quarterly surveillances to ensure the sampling equipment remained operable.

At lower levels of fuel damage the post-accident sampling system collects an undiluted reactor coolant sample of at least 100-cc volume; at higher fuel damage levels, when large reactor coolant samples pose a radiation hazard for the collecting technician, a diluted sample of 3- to 5-cc total volume is collected (diluted at 50 to 1 or more). Bulky shielded transport containers (referred to as *pigs*) were required for even a small diluted sample of coolant from a severely damaged reactor because of on-contact radiation dose rates calculated at greater than 1 R/h. These sampling systems featured long runs of high-pressure, small-diameter pipe, with significant amounts of lead and concrete shielding. Most licensees experienced problems with dilution systems and in maintaining isolation and flush valves. The calibration of radiation counting systems for non-operating source terms and the extremely high count rates expected from even small-volume samples also caused problems.

The Westinghouse Owners Group assessed the technical basis and effectiveness of post-accident sampling (WOG, 1998), and the owners groups for other reactor vendor types performed similar analyses at about the same time (e.g., CEOG, 1999). The industry generally concluded that relevant direct reactor coolant and drywell/containment atmosphere sample information could not be collected in a sufficiently timely manner during an accident to guide real-time emergency response decision-making and that indirect measures of core damage were sufficient for licensees to carry out their safety responsibilities. In addition to timeliness issues, there were technical questions about whether post-accident sampling systems collected representative reactor coolant and drywell/containment atmosphere samples, about the effectiveness of distilled water flushing, about whether sample materials remained trapped in elbows and turns along piping runs, about uncertainties in measuring flow rates, and about possible cross-contamination between sample runs. After containment isolation occurs, manual actions (such as pulling reactor protection system fuses or installing jumpers on circuit boards) were required to defeat the isolation signals and allow containment isolation values on the sample lines to be opened.

In addition to ongoing valve maintenance issues, the long piping runs required long purge times (15 to 30 minutes). Although the sampling stations were heavily shielded to reduce radiation dose to the chemistry technician collecting the sample, the piping runs were often either not shielded or received much less shielding, so when the system was operated other areas of the plant (reactor building) would also potentially have much higher radiation doses. At many plants, particularly boiling water reactors, the post-accident system sampling room was a permanent contaminated area. The sampling team was required to dress in anti-contamination clothing (sometimes double sets of clothing) and remain in the area for up to an hour.

Both industry and Commission staff questioned whether the post-sampling system results were meaningful for emergency response purposes and their usefulness in site remediation. The primary use of post-accident chemistry reports was to compute the percentage of cladding damage and melted fuel; however, these percentages are not directly used (and often not even indirectly used) in the assessment of potential dose to the public and generation of protective actions, and they do not result in any specific emergency response actions. As discussed previously, even accurate knowledge

of radioactive material concentrations in reactor coolant and the containment atmosphere does not readily translate into accurate knowledge of the radioactive release source term. The Commission agreed, and beginning in 2002 it permitted licensees to remove liquid and atmosphere sampling systems from technical specifications. Most, although not all, licensees have now either abandoned such systems in place or have taken them out of their licensing basis (and emergency plans) but have the systems in standby for possible use.*

* Also see Generic Letters 82-05 (*Post-TMI Requirements*), 83-36 (*NUREG-0737 Technical Specifications*), and 83-37 (*NUREG-0737 Technical Specifications*); Industry/TSTF Standard Technical Specification Change Traveler TSTF-413, *Elimination of Requirements for a Post Accident Sampling System (PASS)*; and the model safety evaluations 65 FR 65018 (2000), 66 FR 66954 (2001), 67 FR 13027 (2002), 68 FR 10052 (2003), and 68 FR 25664 (2003).