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HANDBOOK TO SUPPORT ASSESSMENT OF RADIOLOGICAL RISK ARISING FROM MANAGEMENT OF SPENT NUCLEAR FUEL

by
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31 January 2013

Prepared under the sponsorship of
Nautilus Institute for Security and Sustainability

Abstract

Commercial nuclear power plants around the world harness nuclear fission to produce electricity. At each plant, a fission reactor receives fresh nuclear fuel and discharges spent nuclear fuel (SNF). Although the SNF is "spent", it contains a large amount of radioactive material. Some of that material could be released to the environment by an accident or an attack, causing harm to humans by exposing them to ionizing radiation. The potential for such harm is the "radiological risk" associated with SNF. Independent assessment of this risk could help societies to manage the risk. This report is designed as a handbook that could be used to support such independent assessment. The report has two main parts. The first part provides introductory material, and the second part sets forth a seven-step approach to assessing SNF radiological risk.

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Acknowledgements

This report was prepared by IRSS under the sponsorship of Nautilus Institute for Security and Sustainability. The author gratefully acknowledges comments on a draft version of the report from David Lochbaum, David von Hippel, Peter Hayes, and John Ahearne. However, the author, Gordon Thompson, is solely responsible for the content of the report.

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I. Introductory Material

I.1 Overview

This report is designed as a handbook that could be used to support independent assessment of a particular risk associated with the world's nuclear-power industry. The industry operates about 440 nuclear power plants (NPPs) that harness nuclear fission to provide about 14 percent of world electricity production.[1]

Harnessing nuclear fission creates various types of risk. This handbook focuses on a particular type of risk with two major features. First, the risk is associated with spent nuclear fuel (SNF) discharged from the fission reactors at NPPs. Although the fuel is "spent", it contains some fissionable material – uranium and plutonium – and a large amount of radioactive material. Second, the risk is "radiological". Section I.2, below, defines radiological risk in the context of this handbook. In brief, this term refers to the potential for harm to humans as a result of their exposure to ionizing radiation due to an unplanned release of radioactive material.

The radiological risk posed by SNF has existed since fission reactors first operated. Over the intervening decades, the risk has increased due to growth in SNF inventories, changed properties of nuclear fuel, and design choices regarding modes of SNF storage. These factors are discussed in Section I.4, below.

Public awareness of SNF radiological risk was low before the 2011 accident at the Fukushima #1 nuclear site in Japan. Awareness grew during that accident, as citizens learned that SNF was stored in pools adjacent to the affected reactors, and that there was a potential for a large release of radioactive material from this SNF to the atmosphere.

This handbook addresses a range of technical issues. Each issue is complex, and is associated with a substantial technical literature and body of practical experience. Here, much of the complexity is avoided. A comparatively simple approach to assessing SNF radiological risk is set forth, involving various assumptions and simplifications. With this approach, analysts can assess the risk using a sequence of hand calculations and judgments that is easy to follow. The findings could be used for a variety of public-policy purposes. However, the findings should not be used in situations where a more detailed analysis is required. This matter is discussed further in Section I.5, below, which addresses the purpose and scope of the handbook.

The present level of SNF radiological risk is not inevitable. Instead, it reflects choices made by the nuclear industry and accepted by regulatory organizations. Options are available whereby the risk could be substantially reduced. Some options would affect the operation of NPPs, while others would not. In the analytic approach set forth in this

[1] Nuclear fission's share of world electricity production was about 14 percent in 2010, down from a peak of about 17 percent in the early 1990s. (See: World Nuclear Association website, http://www.world-nuclear.org/info/inf01.html, accessed on 16 February 2012.)

handbook, risk-reduction options are considered as part of an assessment of the collateral implications of SNF radiological risk. Various collateral implications are important.

Users of this handbook are advised to inform themselves about SNF radiological risk from as many sources as possible. The handbook should not be used uncritically, but as a framework to guide a user's exercise of informed judgment. Good judgment can be cultivated by reading technical literature, visiting nuclear facilities, and pursuing dialogue with colleagues. In addition, hand calculations using simple models can build understanding of the basic physics and chemistry, and appreciation of the magnitudes of relevant indicators. Computer-based models are available to address many of the technical issues discussed here, but a computer model should only be employed when the user is thoroughly familiar with its underlying principles.

The structure of this handbook is explained in Section I.6, below. In brief, there are two main parts of the handbook. Section I provides introductory material. Section II describes a seven-step approach to assessing SNF radiological risk.

I.2 Defining and Estimating Radiological Risk

As stated above, in this handbook the term "radiological risk" refers to the potential for harm to humans as a result of their exposure to ionizing radiation due to an unplanned release of radioactive material. There is no single indicator of this risk. Instead, the potential for harm can be assessed by compiling a set of qualitative and quantitative information about the likelihood and characteristics of the harm. Our terminology is consistent with a generic definition of "risk" as the potential for harm due to an unplanned event. The US Nuclear Regulatory Commission (NRC) has articulated a similar definition.[2]

Other perspectives on risk

In the nuclear industry and elsewhere, one often encounters a more limited definition, in which risk is the arithmetic product of a numerical indicator of harmful impact and a numerical indicator of the impact's probability.[3] That definition is hereafter designated as the "arithmetic" definition of risk. As explained below, the arithmetic definition can be seriously misleading in two respects. First, the full spectrum of impact and/or probability may not be susceptible to numerical estimation, and numerical estimates may be incomplete or highly uncertain. Second, many subscribers to the arithmetic definition argue that equal levels of the numerically-estimated risk should be equally acceptable to

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[2] The NRC Glossary defines risk as: "The combined answer to three questions that consider (1) what can go wrong, (2) how likely it is, and (3) what its consequences might be. These three questions allow the NRC to understand likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty, which can be used to identify risk-significant scenarios." (See: http://www.nrc.gov/reading-rm/basic-ref/glossary/risk.html, accessed on 16 February 2012.)

[3] Often, the arithmetic product will be calculated for each of a range of impact scenarios, and these products will be summed across the scenarios.

citizens. Their argument may be given a scientific gloss, but is actually a statement laden with subjective values and interests.

Quantitative analysis is essential to science, engineering, and other fields. Yet, the limitations of quantitative analysis should be recognized. Analysts should be especially careful to avoid the intellectual trap of ignoring issues that are difficult to quantify. Many practitioners of radiological risk assessment fall into that trap. Thus, important risk factors are ignored. Prominent examples include: (i) acts of malice or insanity; and (ii) gross errors in design, construction, and operation of facilities. Risk assessments for nuclear facilities routinely ignore these and other factors that may be major determinants of risk.[4]

A nuclear facility – such as a reactor, or a spent-fuel storage installation – typically has the potential to experience unplanned releases of radioactive material across a spectrum ranging from small releases to large releases. Risk analysts who subscribe to the arithmetic definition often conclude that small releases are more probable. With their arithmetic approach, it then appears that large releases with low probability are equivalent to small releases with high probability. Often, these analysts leap to the assumption that the apparent equivalence is "scientific". Thus, they argue, equal levels of the numerically-estimated risk should be equally acceptable to citizens.

In fact, the assumption of equivalence lacks a scientific basis. It is a subjective statement that reflects the values and interests of this group of analysts. From the perspective of a citizen, the potential for a large release may be much less acceptable than the potential for a small release, regardless of probability. That perspective could have a solid, rational basis, because a large release could have effects that are qualitatively different from the effects of a small release. Moreover, a prudent citizen will be skeptical of the probability findings generated by arithmetic risk analysts, given the propensity of these analysts to ignore important risk factors.

Probabilistic risk assessment

The preceding paragraphs provide a basis for critical examination of an analytic art known as probabilistic risk assessment (PRA). This art can be useful in radiological risk assessment, provided that its limitations are kept firmly in mind.

PRA techniques have been developed to estimate the probabilities and impacts of potential unplanned releases of radioactive material from nuclear facilities. Similar techniques can be used to examine other types of risk, such as the potential for harm due to unplanned releases from chemical plants.

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[4] For example, there is evidence that a major risk factor underlying the 1986 Chernobyl reactor accident was endemic secrecy in the USSR. (See: Shlyakhter and Wilson, 1992.) Also, there is evidence that a major risk factor underlying the 2011 Fukushima accident was collusion among government, the regulators, and the licensee (TEPCO). (See: Diet, 2012, page 16.) Radiological-risk studies performed by the nuclear industry and its regulators do not consider secrecy or collusion as risk factors.

In the nuclear-facility context, most PRAs have been done for nuclear power plants. The first PRA for an NPP was known as the Reactor Safety Study, and was published by NRC in 1975.[5] A PRA for a nuclear power plant considers a range of scenarios (event sequences) that involve damage to the reactor core. The initiating events are categorized as "internal" events (human error, equipment failure, etc.) or "external" events (earthquakes, fires, strong winds, etc.). The core-damage scenarios that arise from these events are termed "accidents".

PRAs typically do not consider initiating events that involve intentional, malevolent acts, although PRA techniques can be adapted to estimate the outcomes of such acts. For example, NRC adapted PRA techniques in developing its 1994 rule requiring protection of a nuclear power plant against attack using a vehicle bomb.[6]

PRAs for NPPs are conducted at Levels 1, 2 and 3, in increasing order of completeness, as discussed below. A thorough, full-scope PRA would be conducted at Level 3, and would consider internal and external initiating events. The findings of such a PRA would be expressed in terms of the magnitudes and probabilities of a set of adverse impacts, and the uncertainty and variability of those indicators. Typically, PRAs focus on atmospheric releases of radioactive material originating in the reactor core.[7] The adverse impacts of such releases at downwind locations would include:

> (i) "early" human fatalities or morbidities (illnesses) that arise during the first weeks and months after the release;
> (ii) "latent" fatalities or morbidities (e.g., cancers) that arise years after the release;
> (iii) short- or long-term abandonment of land, buildings, etc.;
> (iv) short- or long-term interruption of agriculture, water supplies, etc.; and
> (v) social and economic impacts of the above-listed consequences.

The magnitudes and probabilities of such adverse impacts would be estimated in three steps. First, a Level 1 PRA analysis would be performed. In that analysis, a set of event sequences (accident scenarios) leading to damage to the reactor core would be identified, and the probability (frequency) of each member of the set would be estimated. The sum of those probabilities across the set would be the total estimated core-damage probability. That indicator is often known as core-damage frequency (CDF), expressed as a number per reactor-year (RY) of reactor operation.

Second, a Level 2 PRA analysis would be performed. In that analysis, the potential for release of radioactive material to the atmosphere would be examined across the set of core-damage sequences. The findings would be expressed in terms of a group of release

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[5] NRC, 1975.
[6] NRC, 1994.
[7] A release could also occur to ground water or surface water (e.g., river, lake, or ocean). For a given size and composition of release, human exposure to radiation would typically be much larger for an atmospheric release than for a water release.

categories characterized by magnitude, probability, timing, isotopic composition, and other characteristics.

Third, a Level 3 PRA analysis would be performed, to yield the findings described above. In that analysis, the atmospheric dispersion, deposition and subsequent movement of the released radioactive material would be modeled for each of the release groups determined by the Level 2 analysis. The dispersion modeling would account for meteorological variation over the course of a year. Then, the adverse impacts of the released material would be estimated, accounting for the material's distribution in the biosphere. As mentioned above, the impacts would include adverse health effects and socio-economic impacts.

If done thoroughly, this three-step estimation process would account for uncertainty and variability at each stage of the process. A thorough, full-scope, Level 3 PRA is expensive and time-consuming. It yields estimated impacts expressed as statistical distributions of magnitude and probability, not as single numbers. Even after such a thorough effort, there are substantial, irreducible uncertainties in the findings.[8] PRA findings rely on numerous assumptions and judgments. There is no certainty that all of the relevant factors are captured. Findings of very low probability cannot be validated by direct experience. Moreover, a PRA cannot estimate the probabilities of malevolent acts, because there is no statistical basis for doing so. A PRA that considered malevolent acts would have to postulate the occurrence of a set of such acts and then estimate their impacts, accounting for variable factors such as wind speed and direction.

NUREG-1150

The high point of PRA practice worldwide was reached in 1990 with publication by NRC of its NUREG-1150 study, which examined five different US nuclear power plants using a common methodology.[9] The study was well funded, involved many experts, was conducted in an open and transparent manner, was done at Level 3, considered internal and external initiating events, explicitly propagated uncertainty through its chain of analysis, was subjected to peer review, and left behind a large body of published documentation. Each of those features is necessary if the findings of a PRA are to be credible. There are deficiencies in the NUREG-1150 findings, which can be corrected by fresh analysis and the use of new information. The process of correction is possible because the NUREG-1150 study was conducted openly and left a documentary record.

PRA practice in the USA has degenerated since the NUREG-1150 study. Now, PRAs are conducted almost entirely by the nuclear industry, and the only published documentation is a summary statement of findings. NRC formerly sponsored independent reviews of industry PRAs, but no longer does so. Thus, contemporary PRA findings have limited credibility. Outside the USA, PRA practice never attained the combined scope and transparency of NUREG-1150.

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[8] Hirsch et al, 1989.
[9] NRC, 1990.

Figures I.2-1 through I.2-3 show some findings from the NUREG-1150 study that are relevant to this report. The findings are for a pressurized-water reactor (PWR) plant at the Surry site (Virginia, USA), and a boiling-water reactor (BWR) plant at the Peach Bottom site (Pennsylvania, USA). Those plants typify many of the Generation II plants in the present worldwide fleet of NPPs. In viewing the CDF findings in Figures I.2-1 and I.2-2, it should be noted that NUREG-1150 itself warns that estimated core-damage probabilities lower than 1 per 100,000 RY should be viewed with caution because of limitations in PRA. NRC has published for comment a draft report from its SOARCA program, describing new analysis of some core-damage sequences at the Surry and Peach Bottom plants.[10] The merit of this analysis, and its implications regarding the NUREG-1150 findings, are unclear at present.

Estimating core-damage probability from direct experience

Severe fuel damage at an NPP is often thought of as a rare event. Yet, a recent inventory lists twelve events involving severe damage to fuel in the reactor core of an NPP.[11] This inventory excludes similar events at non-power reactors. For example, it excludes the core fire and radioactive release experienced in 1957 by a reactor at the Windscale site in the UK. That reactor was used to produce plutonium and other materials for nuclear weapons.

Of the twelve core-damage events at NPPs, five have both: (i) occurred at a Generation II plant; and (ii) involved substantial fuel melting. These five events were at Three Mile Island (TMI) Unit 2 (a PWR plant in the USA) in 1979, Chernobyl Unit 4 (an RBMK plant in the USSR) in 1986, and Fukushima #1 Units 1 through 3 (BWR plants in Japan) in 2011.

These five events occurred in a worldwide fleet of commercial NPPs, of which about 440 plants are currently operable. These plants and previous plants in the fleet had accrued 14,760 RY of operating experience as of 17 February 2012.[12] The five events provide a data set that is comparatively sparse and therefore does not provide a statistical basis for a high-confidence estimate of CDF. Nevertheless, this data set does provide a reality check for PRA estimates of CDF. From this data set – five core-damage events over a worldwide experience base of about 15,000 RY – one observes a CDF of 3.3E-04 per RY (1 event per 3,000 RY).[13]

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[10] NRC, 2012.
[11] Cochran, 2011.
[12] World Nuclear Association website, http://www.world-nuclear.org/, accessed on 17 February 2012.
[13] This simple estimate of CDF might be criticized because the three core-damage events at Fukushima #1 had a common cause. However, there are some design differences between the three affected NPPs at Fukushima #1, and it appears that there were differences in the accident sequences at these plants. Also, multiple core-damage events with a common cause could occur in the future, potentially involving plants at single-unit sites.

Confidence in this reality check is enhanced by noting that the five events occurred in three different countries at three different types of NPP, involved differing initiating events, and happened on three distinct occasions over a period of 32 calendar years. This spread of accident characteristics is consistent with the diversity of circumstances that PRA analysis seeks to address.

Application of PRA techniques to SNF

This author is unaware of any study, in any country, that has systematically applied PRA techniques to examine the radiological risk posed by SNF. Various studies related to this risk have been performed, but none has the systematic scope of a thorough PRA. The lack of attention to SNF risk is notable because a spent-fuel pool containing SNF is adjacent to every commercial reactor. Moreover, current practice for storing SNF in these pools creates a substantial risk that is linked to the reactor risk, as discussed in Section I.4, below.

I.3 Public Attention to SNF Radiological Risk

The radiological risk posed by SNF has continued growing over the past several decades, due to the factors discussed in Section I.4, below. During most of that period, public awareness of the risk was low. This situation was altered by the 2011 accident at the Fukushima #1 nuclear site. From the publicity accompanying the accident, citizens learned that SNF was stored in pools adjacent to the affected reactors, and that a large amount of radioactive material could have been released to atmosphere if a pool lost water and SNF became exposed to air.

Figure I.3-1, which shows Unit 4 at the Fukushima #1 site during the 2011 accident, exemplifies the information that has become available to citizens. The Unit 4 reactor building exhibits severe damage from a hydrogen explosion. A concrete-pumping truck next to the building is spraying water, through the damaged roof of the building, into the Unit 4 spent-fuel pool. Before concrete-pumping trucks arrived at the site, unsuccessful attempts to add water to the spent-fuel pools at Units 1-4 involved the use of fire trucks and riot control vehicles to spray water, and the dropping of seawater from bags carried by helicopters. Television and press coverage of these activities gave citizens around the world an introduction to the risk posed by SNF.

The explosion in the Unit 4 reactor building involved the combustion of hydrogen in air. The only plausible source of this hydrogen was a reaction between steam and the zirconium alloy (zircaloy) cladding of overheated nuclear fuel.[14] This reaction could not have occurred in the Unit 4 reactor core, because the reactor had been defueled prior to the accident. Thus, when the explosion occurred, many analysts theorized that water had been lost from the Unit 4 spent-fuel pool, leading to overheating of SNF in the pool, culminating in steam-zircaloy reaction. A more recent, alternative theory is that the

[14] The steam-zirconium reaction is exothermic and proceeds as follows: Zr + 2H20 -> ZrO2 + 2H2

hydrogen traveled from Unit 3 to Unit 4 through a ventilation system.[15] This alternative theory seems to be a better fit with the facts known to date.[16] From the perspective of public awareness of SNF risk, what may be most significant about this experience is the visual demonstration – through violent hydrogen explosions – of the latent chemical energy in the zircaloy cladding of nuclear fuel.

Figure I.3-2 represents another demonstration of SNF risk. This figure shows the contamination of land in Japan by radioactive Cesium released to atmosphere during the Fukushima accident. Various effects of the contamination – such as limits on the use of land for agriculture – will be evident to Japanese citizens for decades to come. Japanese officials have conceded that the release of Cesium would have been substantially greater if water had been lost from spent-fuel pools, causing SNF to burn (i.e., react exothermically with steam or air). In a February 2012 interview, Japan Atomic Energy Commission chair Shunsuke Kondo described a "worst-case" release scenario that he delivered to the Japanese government on 25 March 2011. The scenario envisioned an atmospheric release from burning SNF that would be "the radiation equivalent of two reactor cores".[17]

Thus, in an age of instant communication, citizens in Japan and around the world have received vivid illustrations of the radiological risk posed by SNF. Yet, almost two years after the Fukushima accident, relatively little has been done, anywhere in the world, to reduce this risk at operating reactors.[18]

The lack of action to reduce SNF risk is consistent with the high degree of socio-technical inertia that characterizes the nuclear industry worldwide. One indicator of this inertia is the fact that all operating NPPs worldwide embody basic designs that were established half a century ago. Also, comparatively few NPPs have entered service in the past two decades, and the average age of the worldwide fleet is 27 years.[19] Factors reinforcing the nuclear industry's inertia include international conventions and national laws that limit the industry's liability for damages.[20]

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[15] Unit 3 also experienced a hydrogen explosion, the hydrogen in that case being created by steam-zircaloy reaction involving overheated fuel in the Unit 3 reactor core.
[16] INPO, 2011.
[17] Associated Press, 2012.
[18] The portion of the SNF risk that is directly linked with reactor operation has been eliminated at reactors that have been shut down as a policy outcome of the Fukushima accident, either by an explicit policy decision (as in Germany) or by policy default (as in Japan).
[19] Schneider and Froggatt, 2012.
[20] See: "Liability for Nuclear Damage", World Nuclear Association website, http://www.world-nuclear.org/info/inf67.html, accessed on 29 September 2012.

I.4 Technical Basis of SNF Radiological Risk

Human-constructed fission reactors first began operating in the 1940s.[21] The radiological risk posed by SNF has existed since that time. Over the intervening decades, the risk has increased due to: (i) growth in SNF inventories; (ii) changed properties of nuclear fuel; and (iii) design choices regarding modes of SNF storage. These factors are discussed here, with a primary focus on SNF from PWRs and BWRs. This handbook applies exclusively to SNF from water-cooled reactors, as explained in Section I.5, below. Within the context of water-cooled reactors, the handbook focuses primarily on SNF from PWRs and BWRs, with some attention to SNF from pressurized-heavy-water reactors (PHWRs). The reasons for this focus are discussed later in Section I.4.

Growth in SNF inventories

Table I.4-1 shows the inventory and broad characteristics of SNF discharged from commercial reactors in the USA through 2010.[22] About three-quarters of that inventory is stored in spent-fuel pools adjacent to operating reactors, the remainder being stored in dry casks.[23] Other countries have accumulated smaller inventories of SNF, determined in each instance by the size, type, and history of operation of the country's fleet of NPPs.[24] The International Panel on Fissile Materials has published a useful review of worldwide experience in managing SNF.[25]

The units shown in Table I.4-1 deserve an explanation. The mass of fuel is expressed in Mg (metric tons) of total initial uranium (Mg U), where "initial" refers to uranium in the fresh fuel inserted into a reactor. For uranium fuel, this mass is identical to the indicator "metric tons of heavy metal" (MTHM). However, MTHM is a more general indicator, because it encompasses situations in which uranium, plutonium, and other heavy metals are present in fresh fuel. The indicator Mg HM, which is equivalent to MTHM, is used at points in this handbook. Note that the indicators Mg U, Mg HM, and MTHM all refer to elemental mass in fresh fuel.

Table I.4-1 shows the "burnup" of a spent-fuel assembly. This indicator is the cumulative thermal energy – in GWt-days per Mg U – released by fissions while the assembly is present in a reactor. The power unit GWt contrasts with GWe, which refers to electricity output from the NPP.

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[21] Natural, geological fission reactors are known to have operated in uranium deposits at Oklo, in Gabon, Africa.
[22] For an overview of practices and regulations regarding SNF storage in the USA, see: EPRI, 2010.
[23] The NRC states that, as of the end of 2009, pools in the USA contained 48,818 Mg of commercial SNF while dry casks contained 13,856 Mg. (See: http://www.nrc.gov/waste/spent-fuel-storage/faqs.html, accessed on 22 February 2012.) Almost all of the SNF in pools is in pools adjacent to operating reactors.
[24] Choi, 2010.
[25] IPFM, 2011.

The growth in SNF inventories around the world reflects a long-term trend away from the reprocessing of spent fuel. When the nuclear fission industry was launched in the 1950s and 1960s, the industry's managers assumed that SNF would be reprocessed. One outcome of that assumption is that the spent-fuel pools at NPPs were originally designed to hold only a few years' discharge of spent fuel from the reactors. Over time, countries have turned away from reprocessing. For example, commercial SNF in the USA has not been reprocessed since 1972.

Growth in SNF inventories would, other factors remaining equal, have yielded a proportional increase in SNF radiological risk. The risk has actually grown at a faster, disproportionate rate, as a result of design decisions by the nuclear industry. One set of these decisions relates to the properties of nuclear fuel, and the other to choices regarding modes of SNF storage.

Properties of nuclear fuel

Figures I.4-1 and I.4-2 show schematic views of PWR and BWR fuel assemblies. Supporting data are shown in Table I.4-2. The active portion of the assemblies consists of uranium oxide pellets – or, in some instances, mixed plutonium and uranium oxide (MOX) pellets – inside thin-walled metal tubes. When the fuel is fresh, the uranium is low-enriched (up to 5% U-235).[26] The tubes are typically known as "cladding". In contemporary NPPs the cladding is made of zircaloy, whose primary ingredient is zirconium.

Zircaloy is not the only material that can be used for fuel cladding. Stainless steel is an alternative cladding material, and was used in a number of water-cooled reactors during the early years of development of this type of reactor. As of mid-1979, about 7% (about 1,500 fuel assemblies) of the commercial SNF inventory in the USA was fuel with stainless steel cladding.[27] Generally, this fuel performed well. In illustration, a thorough examination was made of a stainless-steel-clad PWR fuel assembly that was driven to a burnup of 32 GWt-days per Mg U in the Connecticut Yankee reactor and then stored for 5 years in a spent-fuel pool.[28] No degradation was observed.

Zircaloy and stainless steel performed about equally well as a cladding material, in terms of durability under the conditions experienced in a water-cooled reactor.[29] However, zircaloy was superior in terms of its lower absorption of neutrons, which improved the economics of NPP operation. Thus:[30] "By 1966 economic considerations had led to the selection of zirconium alloy fuel cladding for all water-cooled reactors." This outcome had been anticipated in a 1958 study that stated:[31]

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[26] In a PHWR, the fresh fuel contains natural uranium (0.7% U-235).
[27] Langstaff et al, 1982, page v.
[28] Langstaff et al, 1982.
[29] Gurinsky and Isserow, 1973.
[30] Gurinsky and Isserow, 1973, page 63.
[31] Benedict, 1958, page 1.

> "In most of the nuclear reactors being designed today for commercial power production, it is technically feasible to use either stainless steel or zirconium or one of its alloys as structural material, fuel cladding or fuel diluent. When used within the neutron flux of the reactor the low neutron-absorption cross section of zirconium gives that material an important economic advantage over stainless steel. Use of zirconium instead of stainless steel makes possible savings through the use of uranium of lower enrichment, through reduction in the critical mass of uranium, or through some combination of these cost-saving features."

Exothermic reaction of zircaloy cladding

Although the economic advantage of zircaloy cladding during routine operation of an NPP is clear, there is a price to be paid in terms of radiological risk. Zircaloy, like zirconium, is a chemically reactive material that will react vigorously and exothermically with either air or steam if its temperature reaches the ignition point – about 1,000 deg. C. This temperature is well above the operating temperature of a water-cooled reactor, where zircaloy exhibits good corrosion resistance.[32]

The potential for ignition of zircaloy is well known in the field of reactor risk, and has been observed in practice on a number of occasions. For example, during the TMI reactor accident of 1979, steam-zirconium reaction occurred in the reactor vessel, generating a substantial amount of hydrogen. Some of that hydrogen escaped into the reactor containment, mixed with air, and exploded.[33] Fortunately, the resulting pressure pulse did not rupture the containment. Similar explosions during the Fukushima #1 accident of 2011 caused severe damage to the reactor buildings of Units 1, 3, and 4.

Table I.4-3 illustrates the significance of zircaloy's chemical reactivity in the context of SNF radiological risk. The calculation presented in this table assumes that a PWR fuel assembly surrounded by air experiences a rise in temperature to the point where the zircaloy cladding ignites and burns. Then, it is assumed, 50% of the heat from complete combustion of the cladding enters the adjacent fuel pellets. This amount of heat would raise the pellet temperature to well above the boiling point of Cesium. Thus, a substantial fraction of the pellet's inventory of Cesium would be released. A similar result is obtained if the fuel assembly is surrounded by steam, even though the heat of reaction of zirconium with steam (6.53 MJ per kg Zr) is smaller than the heat of reaction with air (11.9 MJ per kg Zr). These findings provide useful insight into the behavior of SNF in risk-relevant circumstances, despite the simplicity of the calculation.

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[32] Formation of a thin film of oxide on the water-facing surface of the zircaloy enhances corrosion resistance. This film becomes less effective in suppressing oxidation as the zircaloy temperature approaches the ignition point of about 1,000 deg. C. Moreover, as the temperature of zircaloy-clad fuel rises toward that point, the cladding will swell and burst from internal pressure, thus exposing un-oxidized interior surfaces to air or steam.

[33] About 10 hours into the TMI accident, a hydrogen deflagration in the reactor containment caused a containment pressure pulse of about 1.8 bar (26 psi). (See: Camp et al, 1983, Figure I-15.)

Replacing zircaloy with alternative cladding materials

As mentioned above, stainless steel could substitute for zircaloy as a cladding material. The nuclear industry would undoubtedly resist this substitution, which would adversely affect the economics of NPP operation and would disrupt long-established practices in the industry.[34] Also, stainless steel can react exothermically with air or steam, although with a lower heat of reaction than is exhibited by zircaloy.[35]

During the past two decades, there have been efforts to develop ceramic cladding as a replacement for zircaloy. Two major objectives drive these efforts. First, ceramic cladding may allow higher burnup of fuel, which would reduce NPP operating cost. Second, ceramic cladding may behave better in accident conditions, in part by avoiding the heat production and hydrogen generation that are unleashed in the steam-zircaloy reaction.

Currently, efforts to develop ceramic cladding appear to be focused on a "triplex" silicon carbide cladding. The developers hope to begin a prototype test program – in which complete fuel assemblies made with the triplex cladding are placed in commercial reactors – by about 2020.[36] If they keep to this schedule and the tests are successful, then reactors might be routinely fueled with ceramic-clad fuel by about 2030. In that event, ceramic-clad spent fuel would begin adding to SNF inventories in significant quantity by about 2040. Thus, for at least the next three decades, worldwide inventories of SNF will be dominated by fuel using zircaloy cladding.

Re-racking of spent-fuel pools, and its risk implications

At every NPP with a water-cooled reactor, a spent-fuel pool is located adjacent to the reactor. Fresh fuel enters the reactor via the pool, and spent fuel is discharged into the pool. As mentioned above, the pools were originally designed to hold only a few years' discharge of spent fuel from the reactors. As part of that design, the pools were equipped with low-density, open-frame racks into which fuel assemblies were placed. Figure I.4-3 shows a PWR fuel rack of this type. Similar racks were used for BWR fuel.

If water were lost from a pool equipped with low-density racks, there would be vigorous, natural convection of air and steam throughout the racks, providing cooling to the SNF.[37] Thus, in most situations, the temperature of the zircaloy cladding of SNF in the racks would not rise to the ignition point. Exceptional circumstances that could lead to ignition include the presence of SNF very recently discharged from a reactor, and deformation of

[34] The economic impact of switching from zircaloy to stainless steel as a cladding material could be calculated, but that is a task beyond the scope of this handbook.
[35] The heat of reaction of stainless steel with air is 5.98 MJ per kg SS, and the heat of reaction with water is 1.06 MJ per kg SS. (See: Baker and Liimatainen, 1973, Table 3-1.)
[36] Yueh et al, 2010.
[37] Convective cooling of BWR spent fuel would be improved by separating the channel boxes from the fuel assemblies.

the racks. Even then, propagation of combustion to other fuel assemblies would be comparatively ineffective, and the total release of radioactive material would be limited to the comparatively small inventory in the pool.

Faced with the problem of growing inventories of SNF, the nuclear industry could have continued using low-density racks in the pools while placing excess fuel in dry casks. That approach would have limited SNF radiological risk. Instead, the industry adopted a cheaper option. Beginning in the 1970s, the industry re-equipped its pools with higher-density racks. In the high-density racks that are now routinely used around the world, the center-center spacing of fuel assemblies approaches the spacing in a reactor. (See Table I.4-2 for the reactor spacing.) To suppress criticality, the assemblies are separated by plates containing neutron-absorbing material such as boral (boron carbide particles in an aluminum matrix). Figure I.4-4 illustrates the use of high-density racks, in this instance at Unit 4 at the Fukushima #1 site.

The neutron-absorbing plates divide the racks into long, narrow, vertical cells, open only at the top and bottom. If water were lost from a pool, this arrangement would suppress heat transfer by convection and radiation. The presence of residual water in the lower portion of the pool, which would occur in many water-loss situations, would limit heat transfer to only one effective mechanism – convective cooling by steam rising from the residual water. Over a range of water-loss scenarios, radioactive decay heating in the SNF would cause cladding temperature to rise toward the ignition point.[38]

Table I.4-4 sets forth a simple calculation that illustrates the timeframe for cladding temperature to reach the ignition point (about 1,000 deg. C). The calculation assumes adiabatic conditions, which would be approached in the situation where a pool contains residual water. It will be seen that the fuel temperature rises at a rate of (8.5)R deg. C per hour, where R is the fuel assembly's output of radioactive decay heat in kW per Mg HM. Various values of R are shown in Table I.4-5. Consider, for example PWR-U fuel with a burnup of 50 GWt-days per Mg HM, aged 100 days after reactor shutdown. In that case, R = 28. Thus, under adiabatic conditions, fuel temperature would rise at a rate of 238 deg. C per hour.

The preceding discussion sets the scene for considering the attributes of a "pool fire". This incident would involve the following sequence of events:

(i) loss of water from a spent-fuel pool due to leakage, boiling away, siphoning, or other mechanism;
(ii) failure to provide water makeup or cooling;
(iii) uncovering of SNF assemblies;
(iv) heat-up of some SNF assemblies to the ignition point of zircaloy, followed by combustion of these assemblies in steam and/or air;
(v) a hydrogen explosion (not inevitable, but likely) that damages the building surrounding the pool;

[38] For supporting information, see: Alvarez et al, 2003.

(vi) release of radioactive material from affected SNF assemblies to the atmosphere; and
(vii) propagation of combustion to other SNF assemblies.

A pool-fire event sequence would unfold over a timeframe ranging from a few hours to a number of days. During this timeframe, there would be opportunities for personnel to halt or mitigate the event sequence through actions such as plugging holes in a pool, or adding water. However, addition of water after zircaloy ignites could be counter-productive, because the water could feed combustion. Circumstances accompanying the pool-fire event sequence, such as a core-damage event sequence at an adjacent reactor, could preclude mitigating actions. This matter is discussed later in Section I.4.

Recognition of the pool-fire risk

Two studies completed in March 1979 independently identified the potential for a pool fire. One study was by members of a scientific panel assembled by the state government of Lower Saxony, Germany, to review a proposal for a nuclear fuel cycle center at Gorleben.[39] After a public hearing where the study was presented, the Lower Saxony government ruled in May 1979, as part of a broader decision, that high-density pool storage of spent fuel would not be acceptable at Gorleben.[40] Subsequently, new facilities built in Germany to store SNF used dry casks exclusively.

The second study was done by Sandia Laboratories for NRC.[41] In light of knowledge that has accumulated since 1979, the Sandia report generally stands up well, provided that one reads the report in its entirety. However, the report's introduction contains an erroneous statement that complete drainage of the pool is the most severe situation. The body of the report clearly shows that partial drainage can be a more severe case, as was recognized in the Gorleben study.

After receiving the Sandia report, NRC conducted and sponsored a number of studies related to pool-fire risk, which were published over a period of two decades. Unfortunately, those studies employed the erroneous assumption that complete drainage is the most severe case, until NRC partially corrected this error in October 2000. After September 2001, NRC ceased publishing analysis on pool-fire risk, but claims to have done some secret studies.[42] Overall, NRC's work to assess pool-fire risk has useful elements but is deficient in several important respects.[43]

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[39] Beyea et al, 1979.
[40] Albrecht, 1979.
[41] Benjamin et al, 1979.
[42] According to the US Government Accountability Office (GAO), the NRC has lost track of its secret studies. An August 2012 GAO report stated (GAO, 2012, Highlights): "Because a decision on a permanent means of disposing of spent fuel may not be made for years, NRC officials and others may need to make interim decisions, which could be informed by past studies on stored spent fuel. In response to GAO requests, however, NRC could not easily identify, locate, or access studies it had conducted or commissioned because it does not have an agencywide mechanism to ensure that it can identify and locate such classified studies."
[43] NRC studies on pool-fire risk have been identified, summarized, and critiqued in: Thompson, 2009.

NRC's published findings support the analysis presented in this handbook. NRC concedes that a fire could spontaneously break out in a spent-fuel pool following a loss of water, and that radioactive material released to the atmosphere during the fire would have significant, adverse impacts on the environment. To offset those concessions, NRC argues that the probability of a pool fire is very low. NRC has attributed the alleged low probability, in part, to secret security measures and damage-control preparations that were implemented at NPPs in the USA after September 2001.[44] After the Fukushima #1 accident of 2011, NRC released some information about the secret damage-control preparations. This author's review and NRC's own analysis revealed major deficiencies in those preparations.[45]

Although NRC argues that the probability of a pool fire is very low, the agency does acknowledge this event in its planning for emergencies. For example, a workbook used to train personnel in use of NRC's dose-projection code RASCAL contains an exercise in which trainees are asked to calculate offsite radiation doses in the event of a pool fire. The exercise is introduced with the following description of the event:[46]

> "The plant staff are calling you from San Onofre, Unit 2 because there has been an earthquake in the vicinity. The spent fuel pool has lost much of its water due to a large crack possibly flowing into a sink hole. Due to a malfunctioning pump, it has not been possible to provide enough water to make up for the loss. The water dropped to the top of the fuel at 8:49 A.M., and appears likely to continue dropping. Estimates are that the fuel will be fully uncovered by 11:00 A.M. The pool has high density racking and contains one batch of fuel that was unloaded from the reactor only 2 weeks earlier. (A batch is defined as one-third of a core) Another batch was unloaded about a year before that, and 8 batches have been in the pool for longer than 2 years. The spent fuel building has been severely damaged and is in many places directly open to the atmosphere."

Independent studies on pool-fire risk have been performed. In 2003, eight authors published a paper on pool-fire risk and the options for reducing this risk.[47] That paper aroused vigorous comment, and its findings were disputed by NRC officials and others. In an effort to resolve this controversy, the US Congress requested the National Research Council (an affiliate of the National Academy of Sciences) to conduct a study on the safety and security of SNF storage. The Council submitted a classified (secret) report to Congress in 2004, and in 2005 released an unclassified version that was formally published in 2006.[48] Press reports described considerable tension between the Council and NRC regarding the inclusion of material in the unclassified report.[49] That report and

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[44] Thompson, 2009.
[45] NRC, 2011; Thompson, 2011.
[46] Athey et al, 2007, page 116.
[47] Alvarez et al, 2003.
[48] National Research Council, 2006.
[49] Wald, 2005.

the eight-author paper described above are mutually consistent, and both support the analysis in this handbook.

Linkage between pool-fire risk and reactor core-damage risk

At NPPs, a spent-fuel pool is located adjacent to each reactor. Figure I.4-5 shows the respective locations of the reactor and pool in the case of a BWR reactor with a Mark I containment. At PWR plants, the pool is typically located in a separate building that is outside the reactor containment but immediately adjacent to it. The pool shown in Figure I.4-5 is elevated above ground level. At PWR plants in the USA, the floor of the pool is typically at ground level or a few meters below it. There may, however, be open spaces (e.g., rooms, corridors) below the pool floor, into which water could drain.

Systems to cool the water in the pool, and to provide makeup water, are integrated with similar systems that support reactor operation. Thus, cooling and water makeup to the pool would be interrupted during many of the potential event sequences that could lead to reactor core damage. This interruption could initiate – or contribute to – a sequence of events that lead to a pool fire. As mentioned above, that sequence would unfold over a timeframe ranging from a few hours to a number of days.

There would be opportunities during this period for personnel to halt or mitigate the event sequence. In some cases, simply adding water to the pool would be sufficient to prevent a pool fire. However, accompanying circumstances could prevent personnel from taking the necessary actions. For example, the site could be contaminated by radioactive material released from one or more reactors, and structures and equipment could be damaged by hydrogen explosion and/or the influence (e.g., an earthquake) that initiated the event sequence. Indeed, these circumstances arose during the Fukushima #1 accident, and substantially impeded mitigating actions by onsite personnel.[50]

A reactor and its adjacent pool (if filled with SNF at high density) can be thought of as a coupled risk system. The reactor and the pool can affect each other in ways that increase the total risk posed by the system. To illustrate, consider the following hypothetical sequence of events. First, a reactor experiences core damage and a breach of containment. These events lead to severe contamination of the site by short-lived radio-isotopes that are released from the reactor. Intense radiation fields from this contamination, together with damage from a hydrogen explosion, preclude onsite mitigating actions by personnel. The pool then boils dry, or drains due to a related influence. That outcome initiates a pool fire that leads to another hydrogen explosion and a large release of longer-lived radio-isotopes (especially Cesium-137) from the pool. Those phenomena further preclude onsite mitigating actions by personnel, thus prolonging the reactor release and, potentially, initiating releases from other reactors and pools on the site.

[50] INPO, 2011.

This hypothetical sequence of events is not far-fetched. The Fukushima #1 accident could have followed a similar course, given a few changes in site preconditions, in the initiating earthquake/tsunami, and/or in site management during the accident.[51] In that case, the accident would have involved a much larger release of radioactive material than was actually experienced.

The potential for a linked sequence of reactor and pool events is especially ominous when one considers the possibility that a malevolent group of people would deliberately trigger the sequence. A technically knowledgeable and operationally capable group could focus and time an attack in such a manner that both a reactor release and a pool fire would be likely outcomes.[52] The group's investment of resources would be small by comparison with the damage inflicted on the attacked country. Thus, from a military-strategic perspective, a reactor and an adjacent pool filled with SNF at high density are, taken together, a large, pre-emplaced radiological weapon awaiting activation by an enemy.

Linkage between pool-fire risk and reprocessing plant risk

A reprocessing plant is typically co-located with one or more spent-fuel pools. These pools provide buffer storage for SNF that has been received at the reprocessing site but has not yet been fed to the reprocessing plant. At Japan's Rokkasho site, for example, this buffer-storage function is performed by three pools, each with a storage capacity of 1,000 Mg HM of SNF.

While a reprocessing plant does not contain the short-lived radio-isotopes that are present in a reactor, it does contain large amounts of long-lived radioactive material in liquid form. Notably, liquid high-level waste (LHLW) is stored in tanks where it awaits vitrification. Experience and studies show that an accident or attack could release radioactive material from these tanks to the environment.[53] Such a release could prevent personnel access to adjacent areas. Depending on the arrangement of buildings on the site and the nature of the release from LHLW tanks, this prevention of personnel access could contribute to an event sequence that leads to a fire in a spent-fuel pool.

At the Rokkasho site, LHLW tanks are located in a building adjacent to the spent-fuel pools. This fact suggests that an LHLW tank release might contribute to a pool-fire event sequence at this site. Further investigation is needed to examine potential connections between an LHLW tank release and a pool fire at Rokkasho.

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[51] Funabashi and Kitazawa, 2012.
[52] This handbook is intended for general publication. Thus, the optimal foci and timing of an attack are not discussed here. However, technically knowledgeable attackers could readily determine these factors without external advice.
[53] See, for example: Thompson, 1998.

Release of radioactive material from a dry cask

Dry casks are used for storing and transporting SNF. Figure I.4-6 shows a type of cask (the Holtec HI-STORM 100 cask system) that is popular in the USA. The SNF is housed in a sealed, multi-purpose canister (MPC) made of stainless steel and filled with helium. During storage, the MPC is located inside a concrete-and-steel storage overpack as shown in Figure I.4-6. During transportation, the MPC is located inside a transportation overpack.

The MPC-plus-overpack concept is one approach to the design of a dry cask. Another approach is the "monolithic" cask that consists of a single structure. Some monolithic casks are designed solely for storage use, some are designed solely for transportation use, and some are dual-purpose.

The nuclear industry and regulators around the world have given some attention to the radiological risk posed by a dry cask. With a few exceptions, the attention has focused on the potential for humans to be exposed by inhalation of radioactive gases and small particles.

Calculations summarized in Table I.4-6 illustrate the potential for inhalation exposure. These calculations postulate the occurrence of a small hole (equivalent diameter of 2.3 to 36 mm) in a multi-purpose canister, accompanied by severe shaking of the canister. SNF inside the canister would be damaged by shock loading. The canister would experience "blowdown" through the hole, driven by the pressure of helium in the canister plus gases released from SNF rods as a result of damage to their cladding. This event would be slightly more severe than a "design basis" accident. It could, for example, represent the accidental crash of a fighter aircraft on a HI-STORM 100 cask system.

One sees from Table I.4-6 that the fractional release of Cesium-137 would be small. The Cesium-137 release would be somewhat greater if the cask were engulfed by a fire, during and/or after blowdown. If the event were an aircraft crash, a fire could arise from combustion of jet fuel.

There has been some regulatory consideration of attack scenarios, both for transport casks and for storage casks. Various analyses and experiments have been done to estimate the characteristics and radiological consequences of a radioactive release from a dry cask if a shaped-charge warhead were to penetrate the cask.[54] In a typical attack scenario considered in these studies, some SNF rods would experience cladding rupture, and some fuel pellets would be pulverized, creating a radioactive "dust". The warhead would create a pressure pulse inside the cask, helping to drive the radioactive dust into the external atmosphere. The resulting inhalation dose to a nearby, downwind person could exceed the levels shown in Table I.4-6.

[54] Luna et al, 2001.

These industry and regulatory studies have typically not considered the initiation of a zirconium-air reaction inside the cask. Thus, they do not predict a significant fractional release of Cesium-137. Clearly, these studies have not addressed a full spectrum of potential attacks. The rationale for this incomplete investigation is unclear. A few studies have gone against the general trend and considered the potential for cladding ignition during an attack. Unsurprisingly, they have identified a potential for a substantial fractional release of Cesium-137.[55]

Table I.4-3 shows that a zircaloy-air reaction, once initiated, could generate a substantial release of Cesium-137 from SNF rods. Thus, the basic mechanisms of a "cask fire" – analogous to a pool fire – are in place. In order for a cask fire to occur in an actual situation, three conditions must be satisfied. First, a circulating pathway between SNF and the atmosphere must exist, so that air can reach the SNF and combustion products (and Cesium-137) can reach the atmosphere. Second, circulation of fluid through this pathway must be driven by natural convection. Third, the temperature of the cladding of a portion of the SNF in the cask must be raised to the ignition point, so that a self-sustaining reaction can begin.

These conditions are unlikely to be satisfied in an accident situation. They could be satisfied, however, during an attack by knowledgeable, well-resourced actors. This matter is discussed further in Section II.3, below.

Particular attributes of SNF discharged from PHWRs

This handbook applies exclusively to SNF from water-cooled reactors. Within that context, the handbook focuses primarily on SNF from PWRs and BWRs, with some attention to SNF from pressurized-heavy-water reactors. This focus reflects some particular attributes of SNF discharged from PHWRs.

PHWR fuel consists of uranium oxide pellets inside zircaloy tubes. In that sense it is much like PWR and BWR fuel. However, fresh PHWR fuel contains natural uranium (0.7% U-235) rather than the low-enriched uranium (up to 5% U-235) used in PWR and BWR fuel.[56] Also, PHWR fuel is driven to a much lower burnup – typically about 7 GWt-days per Mg HM – than is PWR or BWR fuel.[57] (See Table I.4-1 for typical PWR and BWR burnups.) Finally, a PHWR fuel assembly is comparatively small – about 10 cm in diameter and 50 cm in length, with a mass of about 20 kg U.[58] (See Table I.4-2 for comparative data on PWR and BWR fuel assemblies.)

SNF from a PHWR fueled with natural uranium cannot attain criticality in a light-water environment. Thus, this SNF can be stored in a spent-fuel pool in a compact

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[55] See, for example: Mannan, 2007.
[56] In principle, PHWRs could be fueled with a variety of types of fuel. (See: Whitlock, 2000.) In practice, PHWRs are currently fueled with natural uranium, and this situation is unlikely to change soon.
[57] IPFM, 2011, Table 1.1.
[58] IPFM, 2011, Figure 1.1.

configuration without the presence of the neutron-absorbing plates that are needed for high-density pool storage of PWR or BWR spent fuel. Similarly, the low burnup of PHWR spent fuel results in a comparatively low rate of heat generation (in kW per Mg HM).

Various configurations have been used for PHWR spent-fuel pools. One configuration is to place 60 SNF assemblies in a vertical-axis, hexagonal basket made of stainless steel, and to stack these baskets six-high in the pool.[59]

This author is unaware of any study of the outcome of a loss of water from a PHWR spent-fuel pool, for any configuration of the pool.[60] Accordingly, this handbook does not address the potential for a PHWR pool fire. Further investigation is warranted, in order to determine if a PHWR pool fire could occur.

PHWR spent fuel is stored and transported in dry casks of various configurations.[61] In principle, an attack could initiate a cask fire – as discussed above – in some of these configurations. A cask-specific investigation could assess the potential for a cask fire in each configuration, and the release of Cesium-137 if such a fire were determined to be credible.

I.5 Purpose and Scope of the Handbook

This handbook is designed to support independent assessment of the radiological risk arising from management of SNF from commercial NPPs. Table I.5-1 shows the present worldwide inventory of reactors at these NPPs. One sees that 96% of the operational reactors are water-cooled. Also, 82% are PWRs or BWRs, and 10% are PHWRs.

Types of SNF considered here

As stated earlier, this handbook applies exclusively to SNF from water-cooled reactors. This limitation is for two reasons. First, water-cooled reactors dominate the worldwide reactor inventory. Second, there is a lack of technical literature about the radiological risk posed by SNF from gas-cooled reactors and sodium-cooled fast breeder reactors, and filling that gap is a task beyond the scope of this handbook.

Within the context of water-cooled reactors, this handbook focuses primarily on SNF from PWRs and BWRs, with some attention to SNF from PHWRs. Section I.4, above, discusses the particular attributes of SNF discharged from PHWRs. SNF discharged from LWGRs is not addressed here because: (i) this reactor type is comparatively rare; and (ii) there is a lack of technical literature about the radiological risk posed by LWGR spent fuel.

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[59] Allan and Dormuth, 1999.
[60] Staff of the Canadian Nuclear Safety Commission have recognized that loss of water from a PHWR (CANDU) spent-fuel pool is an event to be feared. (See: CNSC-FTF, 2011.) However, the Commission has not published any study on the outcome of this event.
[61] Allan and Dormuth, 1999; IPFM, 2011, Chapter 2.

Scope of radiological risk

Section I.2, above, defines the concept of radiological risk that is used in this handbook. To be consistent with that definition, users of the handbook should attempt to consider all the major factors that determine the potential for an unplanned release of radioactive material, and the impacts of such a release. Thus, in considering potential initiators of an unplanned release, one should not be limited to the "internal" and "external" initiators that are examined in a typical PRA. Also important are deliberate, malevolent acts of various types, together with gross errors and acts of insanity. Underlying risk factors, such as endemic secrecy or political corruption, may be significant.

A risk assessment should consider the potential for war. Given human history, it would be imprudent to ignore this potential. Yet, the nuclear industry has, in effect, ignored the potential for war.[62] No NPP in the world has been explicitly designed to withstand stressful events – such as aerial bombing, or rocket attack – that routinely accompany modern war. The industry's policy of ignoring a particular class of initiating events should not be followed by risk analysts. An assessment of SNF radiological risk should explicitly consider the likelihood and effects of war.

When nuclear risk analysts consider the potential for acts of war, or for malevolent acts in a non-war situation, they will necessarily examine the vulnerabilities of particular nuclear facilities to attack. In a published risk assessment, these vulnerabilities can be discussed in a general way. However, detailed discussion of vulnerability is inappropriate, because this discussion could assist malevolent actors. In this handbook, there is no detailed discussion of a facility's vulnerability to attack.

At the same time, risk analysts should be aware that secrecy is a factor that increases risk, because secrecy tends to suppress rational discussion of risk and the opportunities for risk reduction. Thus, while a risk analyst has a social responsibility to withhold detailed information about a facility's vulnerability to attack, the analyst has a countervailing responsibility to discuss the vulnerability in general terms, and to discuss the adverse societal impacts of secrecy.

Use of simplified analysis

This handbook addresses a range of complex, technical issues. For each issue, there is a substantial technical literature and a diverse body of practical experience. A risk analyst could devote a professional lifetime to one of these issues.

As long as nuclear facilities continue to operate, there will be a need for specialized, detailed attention to each issue that is relevant to the radiological risk posed by these facilities. At the same time, however, there will be a need for a broad overview of the

[62] Some analysts associated with the nuclear industry have recognized that war is a risk factor for NPPs and related nuclear facilities. (See, for example: Forsberg and Kress, 1997.) However, this recognition is not evident at decision-making levels of the industry.

risk from a policy perspective. This handbook is designed to meet the latter need, with a focus on SNF risk.

A seven-step approach to assessing SNF radiological risk is set forth here, as summarized in Section I.6, below. This approach avoids much of the complexity that would be involved in a comprehensive assessment of risk. The approach involves various assumptions and simplifications, and allows analysts to assess risk using a sequence of hand calculations and judgments that is easy to follow. The findings could be useful for public-policy purposes including:

> (i) comparative assessment of SNF risk with or without major design changes at nuclear facilities (Note: Possible design changes include re-equipping spent-fuel pools with low-density racks, substituting zircaloy with an alternative cladding material, and/or incorporating attack resistance into facility designs);
> (ii) comparative assessment of SNF risk associated with various scenarios for deployment of nuclear facilities (Note: Scenario characteristics would include the number and type of facilities, facility sites, and the concentration of facilities at sites);
> (iii) examination of the strategic implications of the potential involvement of SNF in war or attack by non-State actors; and
> (iv) estimation of the "externality" cost of various options for providing electricity services (Note: The externality cost attributable to SNF radiological risk would be a component of the externality cost attributable to radiological risk across the nuclear fuel cycle).

For various other purposes, a more detailed analysis would be required. For example, suppose that a government agency is tasked with developing an offsite emergency response plan to accommodate potential, unplanned releases of radioactive material from a nuclear facility. That agency would examine essentially the same risk issues as are discussed here, but would need to conduct a more detailed, site-specific study in order to develop a functional plan for emergency response.

Use of Cesium-137 to represent a radioactive release

SNF contains a variety of radioactive isotopes. In this handbook, attention is focused on a single isotope – the fission product Cesium-137. This radio-isotope is used here to represent an atmospheric release from SNF. Other studies of SNF radiological risk have also focused on Cesium-137, for five reasons.[63] First, Cesium is a comparatively volatile material that is readily released from overheated nuclear fuel, as is evident from its release to atmosphere during the Fukushima #1 accident.[64] Second, when released to

[63] See, for example: Alvarez et al, 2003.
[64] Cesium-137 was not the only radio-isotope released to atmosphere during the Fukushima #1 accident. For example, soil and litter samples collected at some downwind locations contained plutonium isotopes deposited on the ground from the Fukushima plume. (See: Zheng et al, 2012.) These authors estimated that the dose from Plutonium-241 to a person living for 50 years in the vicinity of one sampling site (S2) would be 0.44 mSv.

atmosphere, Cesium forms small particles that travel downwind and are deposited on the ground and other surfaces, from which they can be difficult to remove.[65] Third, the radioactive decay of Cesium-137 creates penetrating gamma radiation.[66] Fourth, Cesium-137 has a 30-year half-life, so its radiological impact is of concern over a typical human lifetime and beyond. Fifth, because of the four preceding reasons, Cesium-137 accounts for most of the offsite radiation exposure that is attributable to the 1986 Chernobyl accident.[67]

I.6 Structure of the Handbook

This handbook has two main parts. Section I provides introductory material. Section II describes a seven-step approach to assessing SNF radiological risk. These two main sections are supported by some other material. Section III sets forth a recommended process for using the handbook. Section IV provides a bibliography. Documents and websites cited here are identified in the bibliography or the footnotes. An Appendix outlines some options to reduce the radiological risk posed by a nuclear facility. Those options are relevant to the seventh step of risk assessment, as explained below.

Table I.6-1 summarizes our seven-step approach to risk assessment, which is described more fully in Section II. The seven steps are:

- Step 1: Specify the system
- Step 2: Characterize SNF in the system
- Step 3: Assess the potential for atmospheric release of radioactive material
- Step 4: Estimate the behavior of a radioactive plume
- Step 5: Characterize downwind assets
- Step 6: Assess harm to downwind assets
- Step 7: Assess collateral implications of SNF radiological risk

Steps 1 through 6 are comparatively straightforward. Step 7 deserves some explanation. This section addresses the "collateral implications" of SNF radiological risk. These are inter-related issues and outcomes that are separate from, but closely connected to, SNF radiological risk. Policy decisions or major technical decisions related to SNF risk may affect these issues and outcomes, and vice versa.

One of the key questions to be addressed in assessing collateral implications is the availability of options for reducing SNF radiological risk. There are various options of this type. Some options would affect the operation of NPPs, while others would not. Some options would affect a facility's mode of operation, while others would affect its design. The Appendix outlines some risk-reducing options, with a primary focus on design options.

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[65] Radioactive Cesium can also contaminate food and water supplies.
[66] Most (95%) of Cesium-137 decays are to an excited state of Barium-137 that decays with a half-life of about 150 seconds, yielding a gamma ray with an energy of 0.66 MeV.
[67] DOE, 1987, page x.

As mentioned in Section I.5, above, this handbook could be used to conduct comparative assessments of SNF radiological risk associated with: (i) various design options for nuclear facilities; and (ii) various scenarios for deployment of nuclear facilities. Each design option or deployment scenario would have its particular costs and other characteristics, in addition to its particular profile of SNF risk. Indeed, SNF radiological risk may be only one of a number of salient characteristics of a nuclear scenario. Thus, it should be possible to perform Steps 1 through 7 quickly and easily for a particular scenario. This handbook is intended to serve that objective.

II. Steps in Assessing SNF Radiological Risk, and Relevant Technical Background

II.1 Step 1: Specify the System

The major task in Step 1 is to identify the components, boundaries (in space and time), linkages, and special features of the system that is being studied. In performing this task, an analyst will address questions including:

- Where is the spent fuel (e.g., geographically, by type of facility)?
- How are spent-fuel facilities in the system linked to related facilities (e.g., linkages between spent-fuel pools and reactors)?
- What are the special features of this system (e.g., seismic risk, potential to be attacked)?

An SNF risk system could be a set of fixed facilities, or a transport operation. In either case, the system would consist of one or more discrete components that are linked to each other in various ways. In the case of fixed facilities, the major components would be spent-fuel pools, reactors, and/or dry-cask storage facilities. For a transport operation, the major components would be spent-fuel transport casks.

In order to constitute a single risk system, the components must have geographic proximity **and** a shared exposure to potential common-mode threats (e.g., earthquake, tsunami, attack). If both conditions are not satisfied, the set of facilities should be divided into two or more risk systems.

Given this definition, all components of a system are linked by their geographic proximity and shared exposure to threats. A typical system also exhibits other linkages among system components, as discussed in Section I.4, above. These linkages differ in their significance regarding radiological risk. For example, if a reactor and spent-fuel pool are physically adjacent and share support functions (e.g., cooling systems), and the pool contains SNF in high-density racks, then the reactor and pool are strongly linked from the perspective of radiological risk. A radioactive release from one could initiate or exacerbate a release from the other. By contrast, a typical dry-cask storage facility has comparatively weak linkages to other components of a system, because of its robust and simple design, relying on passive cooling.

Some examples of SNF risk systems

The Fukushima #1 site in Japan provides a useful illustration of an SNF risk system and its components. Table II.1-1 shows the status of SNF at this site. One sees that the site houses seven spent-fuel pools and a dry-cask storage facility. Six of the pools are adjacent to reactors, and the seventh pool is a stand-alone, common pool that can receive SNF from any of the reactors.[68] Thus, all of the present options for storing SNF are represented at the site. Also, SNF transport operations, on the site or between the site and other locations, occur intermittently.

For the purposes of assessing SNF radiological risk, the overall Fukushima #1 site is the system of interest. Within that system, each of the six reactors is strongly linked to its adjacent pool, and the six reactor-poll dyads are linked to each other in various ways. The common pool and the dry-cask storage facility are also components of the system. As stated above, the dry-cask storage facility has comparatively weak linkages to other components. The common pool is more closely linked to other components, because it relies on active machinery for water makeup and cooling, and incidents at other components could adversely affect the operation of that machinery.

Two other nuclear sites in Japan provide additional illustrations of SNF systems. At the Mutsu site, an independent, stand-alone facility for dry-cask storage of SNF is being constructed. At the Rokkasho site, SNF is received in casks and may be stored for an interim period in those casks. Then, the SNF is transferred to one of three spent-fuel pools, where it is stored and from which it may subsequently be removed and fed into a reprocessing plant. The pools are linked to the reprocessing plant in two senses: (i) functionally, in that the pools are intended to provide a buffer stock of SNF to feed into the reprocessing plant; and (ii) by proximity, in that the pools are adjacent to the reprocessing plant.

II.2 Step 2: Characterize SNF in the System

The major task in Step 2 is to characterize SNF in the system, across space and time. In performing this task, an analyst will address questions including:

- What amounts and types of SNF are in the system?
- What are the design features of SNF facilities (e.g., pools, casks) and linked facilities (e.g., reactors) in the system?
- What are the properties (e.g., age, burnup) of SNF in the system?
- What is the inventory of Cesium-137 in the system, either in SNF facilities or in linked facilities?

The purpose in addressing these questions is to develop a basis for assessing, in Step 3, the potential for atmospheric release of radioactive material, with a focus on Cesium-137.

[68] For further detail, see: INPO, 2011.

(Section I.5, above, explains why Cesium-137 is used to represent a radioactive release.) Analysts should keep this purpose in mind when they collect SNF data, and when they examine the design features of SNF facilities and linked facilities.

Determining the density of SNF storage in a pool

In the case of pool storage of SNF, it is especially important to determine if the pool is equipped with high-density racks. Ideally, an analyst would make this determination from technical drawings. In practice, drawings may not be available, and indirect information must be used. To illustrate the use of indirect information, consider the reactor pools at the Fukushima #1 site.

Table II.1-1 shows that the storage capacity of the six reactor pools at Fukushima #1 is 8,310 fuel assemblies, which averages to 1,385 assemblies per pool.[69] For comparison, consider the Pilgrim NPP in Massachusetts, USA. That facility is a BWR plant with a Mark I containment, and its design is very similar to the design of the Fukushima #1 plants.[70] At Pilgrim, the reactor pool has a licensed capacity of 3,859 fuel assemblies, and is equipped with high-density, closed-form racks.[71] Assuming that the Pilgrim and Fukushima #1 pools have a comparable floor area, one finds that the average storage density in the Fukushima #1 pools is substantially lower than at Pilgrim.[72] Yet, Figure I.4-4 clearly shows the presence of high-density racks in the Unit 4 pool at Fukushima #1. Thus, although high-density racks are used in the reactor pools at Fukushima #1, it is evident that racks of this type do not fill the pools from wall to wall, as is the case at Pilgrim and other plants in the USA.[73]

As another illustration of the use of indirect information, consider the three spent-fuel pools at the Rokkasho reprocessing plant in Japan. Each of these pools is said to have a capacity of 1,000 Mg HM of SNF, and floor dimensions of 11 m by 27 m, which translates to a storage density of 3.37 Mg HM per m^2 of floor area.[74] For comparison, the proposed reprocessing plant at Gorleben, Germany, would have had six spent-fuel pools. Each pool would have had a capacity of 500 Mg HM and floor dimensions of 16.3 m by 9.2 m, which translates to a storage density of 3.33 Mg HM per m^2 of floor area.[75] The Gorleben pools would have been equipped with high-density, closed-form racks, with a

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[69] At the time of the 2011 accident, the Unit 4 reactor pool at Fukushima #1 held 1,535 fuel assemblies, including 548 assemblies temporarily discharged from the reactor core. (See: Stohl et al, 2011.)
[70] The Pilgrim plant has a rated power of 2,028 MWt, has 580 fuel assemblies in its reactor core, and began commercial operation in 1972. (See: Thompson, 2006.)
[71] Thompson, 2006.
[72] The floor dimensions of the Pilgrim pool are 12.3 m by 9.3 m. (See: Thompson, 2006.) Thus, this pool's licensed capacity is 3,859/(12.3x9.3) = 33.7 fuel assemblies per m^2 of floor area, which translates to an average center-center distance of 17.2 cm. As shown in Table I.4-2, the average center-center distance between fuel assemblies in the core of a representative BWR reactor (using an 8x8 assembly array) is 15.9 cm. Given an assembly mass of 0.18 Mg HM, a packing density of 33.7 fuel assemblies per m^2 of floor area is equivalent to 6.1 Mg HM per m^2.
[73] Published photographs indicate that a substantial portion of the floor area of the Unit 4 pool at Fukushima #1 was not occupied by fuel racks at the time of the accident.
[74] Data from author's files.
[75] Data from: Beyea et al, 1979.

center-center distance of 28.5 cm for PWR fuel. It follows that the Rokkasho pools employ high-density racks, which can be confirmed from photographs of the pools.

Determining a pool's propensity for loss of water

If a spent-fuel pool is determined to be equipped with high-density racks, the next step in its characterization is to determine its propensity to experience a loss of water. One of the indicators of this propensity is the existence of potential pathways for water loss via leakage. In the case of an elevated pool, as shown in Figure I.4-5, such pathways are easy to identify. Leakage pathways may also exist, however, if the pool is below ground level. For example, the pools at the Rokkasho site are arranged so that the normal water level is approximately at ground level. At first sight, one might conclude that this arrangement would greatly reduce the potential for leakage. Closer examination reveals that the below-ground portion of the pool building contains voids (rooms, corridors, etc.) into which water could drain.

Determining inventories of Cesium-137

In some instances, inventories of Cesium-137 can be found in the technical literature. Table II.2-1 provides an example. That table shows inventories of Cesium-137 in the reactor cores of some NPPs in the Generation III category. The variation between the normalized core inventories (PBq per GWe) of these plants is (assuming there is no error in calculation) primarily due to the differing burnups to which their fuel is driven before it is discharged.

Table II.2-1 also shows inventories of Iodine-131. This radio-isotope exemplifies the short-lived isotopes that are present in abundance in the core of an operating reactor. These isotopes can be significant for SNF radiological risk if their release from a reactor could contaminate a nuclear site and thereby prevent risk-mitigating actions by personnel. (See Section I.4, above.)

Table II.2-2 provides another example from the literature. That table shows the estimated atmospheric release of Cesium-137 from the Fukushima #1 site during the 2011 accident, and the amount available for release from Units 1-4.

If the inventory of Cesium-137 in an SNF system is not available from technical literature, this inventory can be easily calculated. Table II.2-3 shows the basis for the calculation. The key result in Table II.2-3 is that 1 GWt-day of fission will yield 0.117 PBq of Cesium-137. That number can be adjusted to a spent fuel assembly whose burnup (GWt-days per Mg HM) and mass (Mg HM) are known, and the result can be decay-corrected to account for the age of the assembly after reactor shutdown.[76]

[76] Decay correction proceeds as follows:

Inventory of Cesium-137 (Bq) after t yr = exp((-tln2)/30)(Inventory for t = 0)

II.3 Step 3: Assess the Potential for Atmospheric Release of Radioactive Material

The major tasks in Step 3 are: (i) identify scenarios for release of radioactive material to the atmosphere; and (ii) assess the characteristics and feasibility of the scenarios. In performing these tasks, an analyst will address questions including:

- What types of accident-initiated scenario are significant?
- What types of attack-initiated scenario are significant?
- Could scenarios involve linked facilities?
- What timeframes would be typical for significant scenarios?
- What atmospheric release of Cesium-137 could occur?

As stated above, the radio-isotope Cesium-137 is used here to represent the released radioactive material. Section I.4, above, explains the potential for an atmospheric release of Cesium from SNF due to a fire in a spent-fuel pool or a fire in a dry cask. In either case, the release scenario would have the following attributes: (i) the fuel would experience overheating that triggers an exothermic reaction of zircaloy fuel cladding with steam and/or air, thus liberating Cesium from the fuel pellets; (ii) there would be a pathway from the fuel to the external atmosphere; and (iii) there would be a flow of fluid along this pathway, whereby Cesium in vapor or particulate form is carried into the atmosphere.

A pool fire with these attributes could be caused by an accident, or by an attack. The event sequence leading to this fire could involve linked facilities – such as an adjacent reactor or reprocessing plant – or could be limited to the pool itself. A fire in a dry cask – either a storage cask or a transportation cask – could be caused by an attack, but is unlikely to be caused by an accident.

A pool fire caused by an accident

Section I.2, above, discusses the use of PRA techniques to examine the radiological risk posed by commercial reactors. To this author's knowledge, these techniques have never been systematically applied to examine the radiological risk posed by SNF. There is, however, a literature that can support independent assessments of the potential for a pool fire.[77]

The key event in a pool-fire event sequence would be a loss of water from the pool. Accidental causes of this loss could include: (i) leakage (attributable to factors such as earthquake, cask drop, aircraft impact, or errors during reactor refueling); (ii)

[77] See, for example: Alvarez et al, 2003; National Research Council, 2006; Thompson, 2011; Thompson, 2009; Thompson, 2007; Thompson, 2006; Benjamin et al, 1979. These documents cite additional sources.

displacement by falling objects; (iii) sloshing during an earthquake; and/or (iv) boiling off after a failure of the pool cooling system.[78]

Given a loss of water from a pool equipped with high-density racks, a pool fire would ensue unless water makeup and/or cooling is provided. Factors that could prevent this provision include: (i) plant damage from the initiating event (e.g., earthquake) or from a hydrogen explosion during the event sequence; and/or (ii) contamination of the site by radioactive material released from an adjacent facility (e.g, reactor, reprocessing plant).

For any particular SNF system that includes spent-fuel pools, there could be many accident-initiated, system-specific event sequences that lead to a pool fire. PRA techniques could be used to identify and characterize these sequences, and to estimate their probabilities. That exercise would be expensive, and the findings would be the subject to the various uncertainties that are associated with PRA analysis.

If the SNF system includes operating reactors, the probability of a particular type of pool fire can be estimated in a comparatively simple manner. The particular type of pool fire is one that would arise from, or be exacerbated by, the occurrence of a core-damage event at an adjacent reactor. As shown in Section I.2, above, the worldwide experience base of five core-damage events at NPPs in the Generation II category suggests a core-damage probability (CDF) of 3.3E-04 per RY.

Four of these events (at Chernobyl and Fukushima #1) involved substantial release of radioactive material, damage to equipment, and radioactive contamination of the site. A site afflicted in this manner is described hereafter as an "operationally degraded" site. Given the base of experience, one could reasonably say that the conditional probability of an operationally degraded site, given a core-damage event, is 50%, where that number is assigned to represent the qualitative finding that an outcome is as likely to occur as to not occur.

One can then ask: If a site is operationally degraded, and SNF is stored at the site in reactor-adjacent pools at high density, what is the conditional probability of a pool fire? Experience at Fukushima #1 suggests that a pool fire would be "as likely as not" to occur in these circumstances, so its conditional probability could be assigned a value of 50%.[79]

Thus, one could estimate the probability of this type of accident-initiated pool fire as follows:

Probability = (3.3E-04)x0.5x0.5 = 8.3E-05 per RY

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[78] Pool-cooling systems in the USA typically extract water from the pool at a level just below the normal surface water level. Thus, removal of the top layer of water (e.g., by sloshing during an earthquake) would disable pool cooling. Rectifying this design problem by extracting water at a lower level would create new pathways for loss of water by siphoning or leakage.

[79] The Chernobyl experience is set aside here, because: (i) only a small amount of spent fuel was stored in a pool adjacent to the affected reactor; and (ii) there is a lack of available literature on the potential for a fire in that pool.

Entities that own and operate nuclear sites might argue that such a calculation is too simplistic, ignores insights from PRA analyses, and yields a pool-fire probability that is too high. As a counter-argument, one could say that this simple calculation reflects practical experience, and that the site owner/operator should bear the burden of proof in arguing for a lower probability.[80]

A pool fire caused by an attack

In assessing the potential for an attack-induced pool fire, an analyst would consider phenomena that are similar to the phenomena involved in an accident-induced pool fire. Additionally, the analyst would consider the application of focused violence at times and places chosen by the attackers. Knowledgeable, well-resourced attackers could focus their actions in such a way that the conditional probability of a pool fire, given an attack, could be high – potentially, greater than 50%.

Table II.3-1 categorizes potential types of attack on a nuclear facility, where the facility could be: (i) a reactor; (ii) a spent-fuel pool; (iii) a spent-fuel pool adjacent to a reactor or a reprocessing plant; or (iv) an SNF storage facility employing dry casks. Four types of attack are identified. They are not explained here in detail, in order to avoid giving guidance to malevolent actors. However, some general observations are appropriate. Supporting literature is available.[81]

The scale of energy released by the instruments of attack would decrease as one moves from a Type 1 attack to a Type 4 attack. Indeed, a Type 4 attack might employ instruments whose effects appear comparatively mild in terms of the noise, fire, and smoke that they generate. The magnitude of the radioactive release would be greatest for a Type 1 release. Interestingly, however, the magnitude of radioactive release for a Type 3 event would typically exceed that for a Type 2 event. This outcome would occur because the Type 3 event would preserve a tight configuration of nuclear fuel assemblies, thus facilitating the overheating of these assemblies and the initiation of exothermic reaction of zircaloy cladding. A Type 4 event would unfold in a similar manner, and its radioactive release could match or exceed the release for a Type 3 event. The Type 4 event would require a comparatively high level of knowledge, and precision in execution, but could be more economical in resource requirements than a Type 3 event. Thus, a Type 4 event might be the preferred option for non-State actors.

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[80] Commercial NPPs worldwide have accumulated about 15,000 RY of operating experience. Thus, a pool-fire probability of 8.3E-05 per RY might imply the expected occurrence of a pool fire since NPPs began operating. (15,000x8.3E-05 = 1.2) To date, there has been no pool fire, assuming that industry descriptions (e.g., INPO, 2011) of pool conditions during the Fukushima #1 accident are correct. Nevertheless, for planning and policy purposes, it is reasonable to assume a probability of 8.3E-05 per RY.

[81] There is a body of publicly-available technical literature about attacks on commercial nuclear facilities. See, for example: Ramberg, 1984; Ramberg, 1980; Rotblat, 1981; Fetter, 1982; Fetter and Tsipis, 1980; Knox, 1983; Thompson, 2005; Thompson, 1996; Sdouz, 2007; Morris et al, 2006; Honnellio and Rydell, 2007; POST, 2004. The categorization in Table II.3-1 is attributable to the author of this handbook.

A Type 4 attack on a spent-fuel pool would not necessarily involve direct physical damage to structures by the attackers. For example, attackers might be able to take control of a nuclear site, or a portion of the site where a pool is located. Then, they could siphon or pump water from the pool. Uncovering of the SNF would lead to production of hydrogen, which would explode in the upper part of the pool building. That explosion would create a pathway for Cesium to travel directly from damaged fuel pellets to the atmosphere. Also, the explosion would hinder efforts by site personnel to regain control of the pool from the attackers.

Table II.3-2 shows some potential modes and instruments of attack on an NPP, and the present defenses at US plants. One sees that the defenses are limited in scope. In other countries, NPP defenses are typically no more robust than in the USA. Also, SNF systems that are not co-located with NPPs typically have less robust defenses than do NPPs. Thus, attacks of Types 1 through 4 are credible.

One of the instruments of attack that could be used against SNF systems is a shaped charge. Table II.3-3 summarizes the properties of this instrument. Table II.3-4, Figure II.3-1, and Figure II.3-2 provide supporting information. Expertise in the design and use of shaped charges is widely available around the world. Arms manufacturers are actively developing tandem warheads that employ shaped charges. For example, in January 2008 Raytheon successfully tested the shaped-charge penetrating stage for its Tandem Warhead System.[82] The shaped charge penetrated 5.9 m into steel-reinforced concrete with a compressive strength of 870 bar.

Table II.3-5 shows some characteristics of the containments of selected NPPs. This table gives particular attention to the materials, configurations, and thicknesses of the containment walls, which are indicators of a containment's ability to resist external attack. Clearly, these containments vary in their ability to resist attack, but each of them could be penetrated by instruments that are available to well-resourced attackers. Most spent-fuel pools are similarly vulnerable. For example, at the Pilgrim NPP in the USA, the outward-facing (reinforced concrete) walls of the spent-fuel pool have thicknesses ranging from 1.2 m to 1.9 m, and the pool floor (also reinforced concrete) has a thickness of 1.7 m.[83]

As shown in Table II.3-1, a Type 1 attack would involve the use of a nuclear weapon. Such an attack is perhaps unlikely, but cannot be excluded as a possibility. An attacker could greatly amplify the long-term fallout from a nuclear weapon by using it against a commercial nuclear facility. Table II.3-6 shows the dimensions of the crater that would be caused by ground-level detonation of a nuclear weapon. Parts of a nuclear facility that are within the crater footprint would be vaporized or pulverized.

[82] Warwick, 2008; Raytheon, 2008.
[83] Thompson, 2006, Table 3-2. The inward-facing wall of the pool is integrated with the reactor shield wall.

Indirect effects of violence and disorder

The preceding discussion assumes a deliberate attack on a nuclear facility.[84] There may also be situations in which a nuclear facility could be indirectly threatened by war or other forms of political violence, and/or by societal disorder. Events of this type could, for example: (i) interrupt the provision of electricity, water, and other services to a facility; and/or (ii) prevent personnel from performing their duties at the facility. Those influences could, in turn, initiate an event sequence that leads to an outcome such as a pool fire. The potential for such event sequences could be examined using the same analytic approach as would be used to examine accident-initiated event sequences.

A cask fire caused by an attack

As shown in Section I.4, above, SNF in a dry cask used for storage or transportation could experience an exothermic reaction of zircaloy cladding with air. Such an event would be a "cask fire". A substantial fraction of the cask's inventory of Cesium-137 could be released to atmosphere.

In this handbook, the potential for an accident-caused cask fire is assumed to be negligible, and only attack-caused cask fires are considered. For the latter class of events, the principles set forth in Table II.3-1 are applicable. Thus, for the reasons discussed above in the context of pool fires, a Type 4 attack might be the option preferred by non-State actors.

Attackers seeking to cause a pool fire do not need to use incendiary instruments. High-density racks in the pool could serve the function of raising cladding temperature to the ignition point, after which the exothermic reactions could become self-sustaining. In a dry cask, however, a breach of the cask would, by itself, typically not cause cladding temperature to rise to the ignition point. Informed attackers would equip themselves with incendiary instruments to achieve this outcome. Also, informed attackers would probably breach the cask in a manner that encourages a "chimney" effect, whereby air flows through the cask interior and feeds a zircaloy-air reaction. A brief examination of Figure I.4-6 might suggest how this could be done.

The dry cask shown in Figure I.4-6 is the Holtec HI-STORM 100 cask system. This vertical-axis cask is typically placed in an array of identical casks on a concrete pad in the open air. The overpack has an outer diameter of about 3.7 meters and a height of about 5.9 meters. Its outer, carbon steel shell is about 2 cm thick, the inner shell is about 3 cm thick, and the space between these shells is filled by about 69 cm of concrete. Inside the overpack is a multi-purpose canister made of stainless steel, with a wall thickness of

[84] A facility might be attacked inadvertently or contrary to the wishes of high commanders, as a result of a communication failure or other factor. For the purposes of this handbook, such an attack can be regarded as deliberate.

about 1.3 cm. After SNF is loaded into this canister, the canister is filled with helium and sealed. Details vary in particular applications.[85]

Release of Cesium-137 during an accident or attack

Given the potential for strong (risk-related) linkage between a reactor and an adjacent spent-fuel pool, an atmospheric release of Cesium-137 from a pool fire would, in many instances, be accompanied by a radioactive release to atmosphere from an adjacent reactor. Both types of release (i.e., from a pool, and from a reactor) are considered here. The reactor release would include a number of radio-isotopes. However, Cesium-137 would be the dominant isotope in regard to long-term land contamination. Thus, in considering reactor releases, this handbook focuses entirely on Cesium-137.

For the range of events considered here, thorough estimation of the atmospheric release of Cesium-137 from a pool, reactor, or dry cask would be a time-consuming undertaking, involving complex models. There would be substantial variation of the release fraction across a spectrum of hypothesized events, and substantial uncertainty in the estimates. The empirical basis for estimating releases during pool fires and cask fires is limited. Recently, as illustrated in Figure II.3-3, tests have been performed to investigate the burning of SNF. Although useful, such tests do not provide direct empirical evidence regarding the atmospheric release of Cesium and other risk-relevant elements.

In this handbook, estimation of the release of Cesium-137 employs a simple but physically reasonable approach. Table II.3-7 provides background regarding the approach.

The fractional release (to atmosphere) of the Cesium-137 inventory of an affected pool, reactor, or dry cask is assumed here to range from 0.1 to 0.5 (10% to 50%). This range applies to a pool fire, a reactor core melt with containment failure, or a cask fire. <u>For a pool fire</u>, the upper end of the range would apply to a situation in which racks cover most of the pool's floor area, and the majority of the rack slots are full. Those conditions would favor fire propagation across the pool. <u>For a reactor core melt</u>, the upper end of the range would apply to a situation in which there is substantial breach or bypass of the reactor containment. <u>For a cask fire</u>, the upper end of the range would apply to a situation where the modes of cask breach and zircaloy ignition facilitate fire propagation within the cask.

For the pool or reactor events, a Cesium-137 release fraction of 0.1 to 0.5 would be applicable to events involving either accident or attack. For the dry-cask events, this release fraction would apply only to events involving attack. Accidents affecting dry casks would yield a smaller atmospheric release of Cesium-137. Table 1.4-6 provides estimates of the release fractions of Cesium-137 and other radio-isotopes for a particular class of potential accidents affecting a dry cask.

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[85] Thompson, 2009, Section 6.

II.4 Step 4: Estimate the Behavior of a Radioactive Plume

Step 3 identifies potential atmospheric releases of radioactive material, focusing on the radio-isotope Cesium-137. A particular release would create a radioactive plume that travels downwind. The major tasks in Step 4 are: (i) estimate the track of the radioactive plume; (ii) estimate the deposition of radioactive material on the ground and other surfaces; and (iii) estimate the radiation dose resulting from this deposition. In performing these tasks, an analyst will address questions including:

- What are typical weather data at this location?
- In what manner would a plume transport and deposit Cesium-137?
- What ground contamination and external radiation dose would arise from deposition of Cesium-137?

In an actual event, the behavior of the radioactive plume could be complex. Radioactive material could be released to the atmosphere over a period of hours or days, potentially in a series of "puffs" during that period. Some puffs could place radioactive material higher into the atmosphere than others. Wind direction, wind speed, atmospheric stability, and other atmospheric characteristics could change substantially during the release period and along the plume path. Topographic features could affect plume behavior. The plume could encounter zones of precipitation (rain, snow, etc.) in which plume "washout" substantially increases the deposition of radioactive material on the ground. Factors such as these could lead to a highly uneven pattern of ground contamination.

Experience from Chernobyl and Fukushima #1

Each of the factors mentioned above was operative during the Chernobyl accident of 1986. There were six distinct atmospheric plumes between 26 April and 4 May 1986, heading in differing directions and with differing shapes.[86] The resulting pattern of land contamination by Cesium-137 was highly uneven.[87]

These factors were also operative during the Fukushima #1 accident of 2011. One estimate of the atmospheric release of Cesium-137 during that event shows a series of eight large puffs during the period 12-19 March 2011, when the release rate was highest. Each of those puffs exceeded a release rate of 10 GBq per second, and two reached about 400 GBq per second.[88] Wind direction and other atmospheric characteristics varied considerably during the Fukushima release, and precipitation caused substantial washout of Cesium-137 from the plume at some locations.[89] The resulting pattern of ground contamination of Japanese territory by Cesium-137 is uneven, as shown in Figure I.3-2. According to one estimate, as summarized in Table II.2-2, about 6.4 PBq of Cesium-137 was deposited on Japan, from a total atmospheric release of about 36 PBq. Most of the

[86] UNSCEAR, 2011, Figure I.
[87] UNSCEAR, 2011, Figure II.
[88] Stohl et al, 2011, Figure 5.
[89] Stohl et al, 2011.

Cesium-137 fell over the North Pacific Ocean, and about 0.7 PBq fell on land areas other than Japan.[90]

The wedge model of plume behavior

This handbook offers a simplified model of plume behavior – the "wedge" model. Formulation of the model is generally attributed to a Study Group convened by the American Physical Society.[91] With the wedge model, an analyst can perform hand calculations that are completely transparent and reproducible. The findings can be valuable for policy-related analyses such as those set forth in Section I.5, above.

Table II.4-1 (which has two parts) shows how the wedge model can be used to estimate external radiation dose from contamination of land by Cesium-137. An illustrative calculation of this kind is set forth in Table II.4-2.

A user of the wedge model must provide several parameters. The most basic of these parameters are wind speed and direction. Data on these parameters are often presented in a "wind rose", such as the example shown in Figure II.4-1. Wind roses are published for many locations around the world, including nuclear-facility sites.[92]

Other parameters in the wedge model include: wedge angle (θ); deposition velocity (V); and mixing-layer height (H). Table II.4-1 shows typical values of these parameters. Table II.4-3 shows how atmospheric stability – indicated by "Pasquill category" – varies with weather conditions. Then, Table II.4-4 shows how atmospheric stability relates to the variability of wind direction. These tables could help to guide a user's choice of wedge angle. Figure II.4-2 shows estimated deposition velocity (under dry conditions) for a range of atmospheric stability categories and wind speeds.[93] Further guidance regarding wedge-model parameters can be obtained from a large body of relevant technical literature.[94]

If necessary, the wedge model could simulate an atmospheric release involving a series of puffs over time, with variation of parameters such as wind direction. Such a multi-puff release could be simulated by a set of wedge models whose results would be superposed.

As described above, atmospheric plumes from the Chernobyl and Fukushima #1 accidents encountered zones of precipitation, leading to enhanced deposition of Cesium-

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[90] Stohl et al, 2011.
[91] Lewis et al, 1975.
[92] In the USA and many other countries, preparation of a Safety Analysis Report (SAR) or similar document is part of the licensing process for any commercial nuclear facility. A well-prepared SAR will contain one or more wind roses for the facility site. Wind roses can also be obtained from various other sources. For example, Enviroware offers wind roses for about 3,000 locations worldwide. (See: http://www.enviroware.com/metar-wind-roses-for-year-2012/)
[93] Aerosol deposition velocity can range over several orders of magnitude, depending on particle size and other factors. Figure II.4-2 assumes a particle diameter of 5 micron.
[94] See, for example: Lewis et al, 1975; Hanna et al, 1982; Slade, 1968; Jow et al, 1990; McGuire et al, 2007.

137 (washout) at some ground locations. Given the purpose and scope of this handbook, as described in Section I.5, above, it is not clear how washout can be readily accommodated in the analysis. Also, Table II.4-5 shows that rain is comparatively infrequent at NPP sites, at least in the USA. Thus, this handbook does not currently address plume washout, and focuses on dry deposition.[95]

II.5 Step 5: Characterize Downwind Assets

Step 4 estimates the potential radiation exposure of people who are downwind from the point of atmospheric release. Step 6 assesses the harm arising from this radiation exposure, focusing on the effects on human health and land use. Step 5 is a bridge between these two steps. The major task in Step 5 is to characterize assets – including population and property (e.g. buildings, agricultural land) – at downwind locations. In performing that task, an analyst will address questions including:

- What are the relevant sub-national and national boundaries?
- What are the concentrations of population?
- What are the major land uses (e.g., agriculture)?
- What is the value of potentially affected assets (e.g., buildings)?

Answering these questions will require the collection and compilation of data from various sources. Table II.5-1 illustrates some relevant data. If this handbook is used to support a number of assessments of SNF radiological risk, with the intent of comparing the findings, it would be useful to develop a standardized framework for collecting data of this kind. Section III, below, recommends an iterative process to develop standardization across all the aspects of an SNF risk assessment.

An analyst characterizing downwind assets would typically consider both: (i) the value of each asset; and (ii) the probability that a radioactive plume would pass over a particular asset (i.e., the probability that the asset would actually be downwind if a release occurred). To take a simple example, a large city might be relatively near to the point of release, but the wind might blow toward that city only on relatively infrequent occasions. The analyst must exercise judgment when considering these factors in a risk assessment. This handbook is not a substitute for judgment, but aims to ensure that judgment is well informed.

II.6 Step 6: Assess Harm to Downwind Assets

The major tasks in Step 6 are: (i) estimate the effects of the radioactive plume on human health and land use, focusing on the effects of ground contamination by Cesium-137; and (ii) assign monetized values to these effects. In performing these tasks, an analyst will address questions including:

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[95] For an introduction to issues involved in modeling wet deposition, see: Jones, 1986.

- What is the collective radiation dose?
- What mortality can be attributed to this radiation dose?
- What is the monetized value of human health effects?
- What area of land would be contaminated above a specified dose threshold?
- What is the value of abandoned or damaged assets?

Step 4 uses the wedge model to examine plume behavior, ground contamination, and external radiation dose. Step 5 characterizes downwind assets that could be affected by the plume. The findings from these two steps can be used here, in Step 6, to address questions such as those stated above.

Collective dose and resulting effects on health

Table II.6-1 shows how the wedge model can be used to estimate collective radiation dose from external exposure due to ground contamination by Cesium-137. In this case, the plume intersects a populated region between two specified radii from the point of release.

Table II.6-2 then examines an illustrative example in which the two radii are 50 km and 60 km, 1 PBq of Cesium-137 is released, dry deposition is assumed, and the populated region has a population density of 5,000 persons per square km. The resulting collective dose, over 30 years, would be 4,240 person-Sv. Assuming a dose-effect coefficient of 0.051 excess solid-cancer deaths per person-Sv, as discussed below, this collective dose would translate to 216 excess deaths within an affected population of 687,500 persons. Note that the "excess" deaths would, in fact, be "premature" deaths, because the affected persons would eventually die from some cause.

Table II.6-3 examines a more severe example in which the release of Cesium-137 is increased to 100 PBq, and the collective dose from ingestion equals the collective dose from external radiation. In this example, there would be 43,250 excess deaths, representing 6.3 percent of the affected population.

Table II.6-4 shows the estimated collective dose attributable to the 1986 Chernobyl accident, for various regions. These estimates provide a background of experience for comparison with the calculations described above.

Table II.6-5 summarizes findings in the US National Academies' BEIR VII report regarding the health effects from radiation exposure. These findings show that a population with an age distribution similar to that of the US population would experience 0.051 excess solid-cancer deaths per person-Sv of collective dose. The BEIR VII findings rely upon a "linear no-threshold" (LNT) model of dose response for solid cancer, and a "linear-quadratic" model for leukemia. The LNT model has been criticized, but is the prevailing hypothesis at this time.[96]

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[96] Beyea et al, 2012.

Monetized value of health effects

Regulatory agencies often assign monetized values to adverse health effects that arise from human-created hazards. In the context of commercial nuclear facilities, monetized values are assigned to health effects attributable to radiation arising from occupational exposure, routine releases of radioactive material, and unplanned releases. Those monetized values can be assigned directly, by assigning a value to a premature death, or indirectly, by assigning a value to a person-Sv of collective dose.

The assigned values vary greatly, between countries, between regulatory agencies within a country, and between regulatory functions of a particular agency.[97] Governments occasionally attempt to standardize these values. For example, in 2004 the US Office of Management and Budget advised US federal agencies that they should assign a value of between US$1 million and US$10 million to a premature death, and more recently supplemented this advice by warning agencies that a value below US$5 million would be difficult to justify.[98]

Assuming a dose-effect coefficient of 0.051 excess solid-cancer deaths per person-Sv of collective dose, assigning a value of US$10 million to a premature death (from solid cancer) would be equivalent to a value of US$510,000 per person-Sv. For comparison, in the early 1990s the owners of NPPs in the USA were spending an average amount of US$1,130,000 per person-Sv to reduce occupational exposure.[99]

Area of land contaminated above a specified dose threshold

The wedge model can be used to calculate the area of land that would be contaminated above a specified dose threshold, for a given atmospheric release of Cesium-137. Different thresholds could trigger different policy actions. For example, a particular threshold could call for abandonment of the contaminated land. The abandoned land would contain assets, whose value would be identified in Step 5.

Table II.6-6 shows a calculation of the area of land that would be contaminated by Cesium-137 to a level such that the external radiation dose to an individual over the first year would exceed N (Sv). In this particular calculation, N is set to 0.02 Sv. Note that the US Environmental Protection Agency (EPA) has recommended population relocation if the 1st-year dose is projected to exceed 0.02 Sv, although EPA uses a more conservative definition of dose than is used in Table II.6-6.[100]

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[97] See, for example: Baum et al, 1994.
[98] Appelbaum, 2011.
[99] The average expenditure was $734,000 per person-Sv in 1990$. (See: Baum et al, 1994, Figure 4 and Table 15.) That amount is adjusted here to 2010$ using a relative GDP deflator of 1.54.
[100] EPA states that the projected 1st-year dose should account for external gamma radiation and inhalation of re-suspended radioactive material, but not for shielding from structures or the application of dose-reduction techniques. (See: EPA, 1992, page 4-4.) This handbook's definition of N neglects inhalation and accounts for shielding on an average-exposure basis. EPA's definition could yield a larger land area than is calculated in Table II.6-6, other factors being equal.

From Table II.6-6 one sees, for example, that a release of 100 PBq of Cesium-137 would lead to contamination of 2,500 square km of land above a 1st-year dose threshold of 0.02 mSv. This contaminated land would extend to a distance of 140 km from the point of release.

EPA has recommended that the cumulative 50-year dose to an individual from living on contaminated land should not exceed 0.05 Sv. Application of that criterion would mean that a 1st-year dose of 0.02 Sv, calculated as in Table II.6-6, would require population relocation for at least 50 years.[101] In that case, from the perspective of a resident or property owner, the affected area could be regarded as permanently abandoned.

Value of abandoned or damaged assets

If physical assets such as land and buildings are regarded as permanently abandoned, it could be relatively straightforward to determine their lost value. A calculation of economic impact would be more difficult if land or buildings are designated as damaged, with a potential for restoration of their use through decontamination. Analysts who calculate the economic impacts of radioactive releases have considered the role of decontamination.[102] Studies of that kind require a range of assumptions.

Land, buildings, and other physical items are not the only non-human assets that may be damaged by a radioactive release. Damage to non-physical assets could also be significant. For example, suppose that a radioactively-contaminated region is part of a larger region that is a major agricultural producer. The market for crops grown across the entire, larger region could be adversely affected, even if the sale of crops from the contaminated part of the region is banned. In that instance, the damage would be to a non-physical asset – the reputation of the larger region as an agricultural producer. Indirect economic impacts of this type might exceed the direct impacts.

II.7 Step 7: Assess Collateral Implications of SNF Radiological Risk

The major task in Step 7 is to examine the wider context and implications of SNF radiological risk. In performing this task, an analyst will address questions including:

- What are the societal and strategic implications of radiological risk?
- What are the opportunities to reduce radiological risk?
- How is radiological risk linked with other risks (e.g., nuclear-weapon proliferation)?
- How does radiological risk relate to the sustainability of a nuclear infrastructure investment?

[101] See Table II.6-6, note (g).
[102] See, for example: Beyea et al, 2004.

From Steps 1 through 6 it is clear that SNF systems could experience a large release of radioactive material to atmosphere, caused by an accident or an attack. The release could have substantial adverse impacts of various types, including health, economic, and political impacts. In the context of potential conflicts between States, or between a State and non-State actors, SNF radiological risk could have strategic implications. From a military-strategic perspective, a reactor and an adjacent pool filled with SNF at high density are, taken together, a large, pre-emplaced radiological weapon awaiting activation by an enemy.

Risk-reducing options and their implications

The present level of SNF radiological risk is not inevitable. Instead, it reflects choices made by the nuclear industry and accepted by regulatory organizations. Options are available whereby the risk could be substantially reduced. The Appendix discusses some options of this kind, with a primary focus on design options.

Choices about risk-reducing options can have wide-ranging societal implications. This matter is explored in Table II.7-1, which discusses some approaches to protecting a country's critical infrastructure from attack by non-State actors. From the table, it is clear that substantial benefits could arise from designing infrastructure facilities to be robust and inherently safe. SNF facilities could be designed with that objective.

SNF and plutonium

Worldwide inventories of SNF contain large amounts of plutonium. As shown in Table II.7-2, it has been estimated that SNF discharged from commercial reactors worldwide through 2010 would contain about 2.1 million kg of plutonium.

For comparison with the quantities shown in Table II.7-2, note that the critical mass of a bare sphere of plutonium (pure Plutonium-239, alpha-phase) is about 10 kg. The radius of that sphere would be about 5 cm. With addition of a natural uranium reflector about 10 cm thick, the critical mass would be reduced to about 4.4 kg, comprising a sphere with a radius of about 3.6 cm, the size of an orange. The critical mass could be further reduced using implosion techniques. An implosion device built to a modern design could achieve a nuclear explosion using 2 to 3 kg of plutonium.[103]

Nuclear warheads deployed by the nuclear-weapon states each contain, on average, about 3 to 4 kg of plutonium.[104] The world's inventory of military plutonium, at the end of 1994, was about 249,000 kg, mostly held by the former USSR and the USA. About 70,000 kg of that plutonium was in operational warheads.[105]

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[103] Barnaby, 1992.
[104] Albright et al, 1997, page 34.
[105] Albright et al, 1997, Table 14.2.

The presence of plutonium in SNF creates a risk of nuclear-weapon proliferation. That risk can intersect with SNF radiological risk, especially in the context of reprocessing. For example, at the Rokkasho site in Japan, spent-fuel pools equipped with high-density racks have the capacity to store 3,000 Mg HM of SNF. This mode of storage poses a significant radiological risk. At the same time, the purpose of these pools is to feed a reprocessing plant. Operation of that plant increases the risk of nuclear-weapon proliferation by separating plutonium from SNF, thus making the plutonium more accessible for weapons use.

Radiological risk and sustainability

Nuclear facilities such as NPPs are large, long-lived items of infrastructure. In the current era, there is growing acceptance of the idea that such items of infrastructure should be designed and operated according to the imperatives of sustainability. This idea is discussed within the nuclear industry.[106] At present, there is no consensus about principles and practices to account for sustainability when planning and designing infrastructure. In the context of nuclear infrastructure, however, it is clear that one of the major issues to be examined in an assessment of sustainability would be radiological risk.

III. A Recommended Process for Using this Handbook

This handbook sets forth a particular approach to assessing SNF radiological risk, and explains (see Section I.5, above) the public-policy purposes for which this approach is appropriate. Thus, to some extent, the handbook provides a template for producing standardized risk assessments. Such standardization could be valuable, as discussed below.

Yet, while the approach set forth here has some standardized features, it relies heavily on the judgment of analysts using the approach. As a result, different analysts working in isolation to assess the same system could generate differing findings.

To some degree, the reliance on analysts' judgment reflects the simplified analysis described here, but informed judgment would remain essential even if the analysis were more elaborate. Sophisticated calculations would require the use of computer models, but there is no suite of models that could represent all of the issues addressed here. Moreover, computer models rely on numerous assumptions that are often hidden.

Given the reliance of this handbook's approach on informed judgment, the approach will not automatically yield standardized findings. Yet, a standardized approach to assessing SNF radiological risk could be valuable in many applications. Suppose, for example, that the approach is used to assess the risk implications of a design change across a number of nuclear facilities. The policy value of the findings would be greatly enhanced if the analysts examining these facilities employed similar assumptions and methodology.

[106] NEA, 2000.

To achieve the necessary degree of standardization, and to build analytic capability, a phased process for using this handbook is recommended. The process would seek to: (i) enhance the skills of participating analysts; (ii) refine the analytic approach; (iii) develop consensus among the analysts regarding appropriate assumptions and methodology; and (iv) improve the handbook.

During the initial phases, participating analysts would be engaged in parallel assessments of the same systems, and would share their findings through several iterations. Over time, this exercise should lead to convergence of assumptions and methodology. Then, in later phases, separate teams of analysts would assess different systems.

Three features of the process would be constant throughout all phases. First, analysts would be in regular communication, comparing and critiquing each other's work. Second, the analysts' work would be exposed to independent review. Third, the analysts would strive to make their analyses transparent and their assumptions explicit.

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Table I.4-1
Inventory and Characteristics of Spent Fuel Discharged from US Commercial Reactors through 2010

Reactor Type	Total Number of Spent Fuel Assemblies	Total Initial Uranium (Mg U)	Average Enrichment when Fresh (% U-235)	Average Burnup (GWt-days per Mg U)	Average Age After Discharge (yr)
PWR	97,400	42,300	3.74	39.6	14.9
BWR	128,600	23,000	3.12	33.3	15.4
Total	226,000	65,200	N/A	N/A	N/A

Notes:
(a) Data are from: Carter et al, 2011, Sections 2.1 and 2.2.
(b) Almost all fuel currently being discharged from US reactors has a burnup exceeding 45 GWt-days per Mg U, and some fuel approaches 60 GWt-days per Mg U. Burnup is currently limited in the USA by the reactor licensing basis of 62.5 GWt-days per Mg U, and by the 5% U-235 licensing basis for enrichment and fuel fabrication plants.

Table I.4-2
Selected Characteristics of Representative PWR and BWR Reactors

Characteristic	Value	
	PWR	BWR
Rated thermal power	3,411 MWt	3,579 MWt
Rated electrical output	1,100 MWe	1,220 MWe
Core (or fuel rod) active length	3.7 m	3.8 m
Number of fuel assemblies	193 (15x15 assembly array)	748 (8x8 assembly array)
Av. thermal power per assembly	17.7 MWt	4.78 MWt
Total number of fuel rods	39,372	46,376
Fuel cladding material	Zircaloy-4	Zircaloy-2
Cladding diameter (OD)	1.07 cm	1.23 cm
Cladding thickness	0.06 cm	0.08 cm
Cladding temperature (typical)	347 °C	304 °C
Fuel material	UO_2	UO_2
Pellet diameter	0.9 cm	1.04 cm
Pellet height	1.5 cm	1.04 cm
Total mass of fuel (UO_2)	98.4 Mg	155 Mg
Total mass of fuel (U)	86.7 Mg	137 Mg
Av. mass of fuel (U) per assembly	449 kg	183 kg
Core diameter	3.4 m	4.9 m
Av. area density of fuel mass (U) over core footprint	9.55 Mg per m^2	7.27 Mg per m^2
Av. center-center spacing of fuel assemblies	21.7 cm	15.9 cm
Design fuel burnup	32 GWt-days per Mg U	28.4 GWt-days per Mg U
Fresh fuel assay	3.2% U-235	2.8% U-235
Spent fuel assay (design)	0.9% U-235, 0.6% Pu-239 & 241	0.8% U-235, 0.6% Pu-239 & 241

Notes:
(a) Data are from: Nero, 1979, Tables 5-1 and 6-1.
(b) The PWR represents Westinghouse plants, and the BWR represents General Electric plants.
(c) The values shown are correct only for the specific, representative reactors. Other reactors have somewhat different values.
(d) Typical fuel burnup has increased substantially since these data were compiled. Almost all fuel currently being discharged from US reactors has a burnup exceeding 45 GWt-days per Mg U, and some fuel approaches 60 GWt-days per Mg U. (See: Carter et al, 2011, Section 2.2.)

Table I.4-3
Illustrative Calculation of Heat-Up of a Fuel Rod in a PWR Fuel Assembly Due to Combustion in Air

Calculation Step	Properties and Behavior of Rod Components	
	Zircaloy Cladding	**UO_2 Pellets**
Solid volume, per m length	1.90E-05 m^3 (OD = 1.07 cm; thickness = 0.06 cm)	6.36E-05 m^3 (OD = 0.9 cm)
Mass, per m length	0.124 kg (@ 6.55 Mg per m^3)	0.700 kg (@ 11.0 Mg per m^3)
Heat output from complete combustion of material in air, per m length	1.48 MJ (@ 2,850 cal per g Zr, where 1 cal = 4.184 J)	Neglected
Heat input if material receives 50% of heat output from adjacent combustion, and if heat loss from material is neglected	Neglected	(1.48)(0.5) = 0.74 MJ (i.e., 1.06 MJ per kg UO_2)
Equilibrium temperature rise due to heat input	Neglected	approx. 2,700 deg. C (The enthalpy rise if UO_2 temp. rises from 300 K to 3,000 K = 1.05 MJ per kg UO_2)

Notes:
(a) This table is adapted from Table 6-2 of: Thompson, 2009.
(b) Melting point of UO_2 is 2,850 deg. C (3,123 K), and boiling point of elemental Cesium is 685 deg. C.
(c) Boiling point of CsI is 1,280 deg. C, and boiling point of CsOH is 990 deg. C. (See: Silberberg et al, 1986, Table 3.2.)
(d) Data compiled by the UK Atomic Energy Authority indicate that the fractional release of Cesium from LWR fuel over a 5-minute period would be 0.45 at 1,600 deg. C, 0.73 at 1,700 deg. C, and 0.94 at 1,800 deg. C. (See: Gittus et al, 1982, page 546.)
(e) Average enthalpy rise per deg. C as UO_2 temperature rises from 300 K to 3,000 K = $(1.05 \times 10^3)/(2{,}700)$ = 0.39 kJ per kg UO_2 per deg. C. (See also: Popov et al, 2000.)
(f) An analogous table could be prepared for combustion of the zircaloy cladding in steam. In that case the heat of reaction would be 1,560 cal per g Zr = 6.53 MJ per kg Zr. (See: Baker and Liimatainen, 1973, Table 3-1.) As shown above, the heat of reaction in air would be 2,850 cal per g Zr = 11.9 MJ per kg Zr. Both values are approximate.
(g) Oxidized Zr will form a liquefied two-phase mixture with UO_2 at about 1,900 deg. C. (See: Silberberg et al, 1986, Table 3.2.)

Table I.4-4
Illustrative Calculation of Heat-Up of a Fuel Rod in a PWR Fuel Assembly Due to Decay Heat, in an Adiabatic Situation

Calculation Step	Properties and Behavior of Rod Components	
	Zircaloy Cladding	**UO_2 Pellets**
Solid volume, per m length	1.90E-05 m^3 (OD = 1.07 cm; thickness = 0.06 cm)	6.36E-05 m^3 (OD = 0.9 cm)
Mass, per m length	0.124 kg (@ 6.55 Mg per m^3)	0.700 kg (@ 11.0 Mg per m^3)
Specific heat (average, approx.)	400 J/kg/K	300 J/kg/K
Heat output from radioactive decay (assembly)	R = decay heat in kW per Mg U	
Heat output from radioactive decay (rod)	0 (W per kg Zr)	(238/270)R = (0.88)R (W per kg UO_2)
Rate of temperature rise from decay heat, if pellets and cladding are a tightly coupled adiabatic system	(0.88)R(0.7)/((0.7)(300) + (0.124)(400)) = (2.4E-03)R (K per second) or (8.5)R (K per hr)	

Notes:
(a) Data are from: Thompson, 2009, Table 6-2; Popov et al, 2000, Figure 4.2; CRC, 1986; IAEA, 1997, Figure 4.2.1.1.
(b) As an example, consider PWR fuel with a burnup of 50 GWt-days per Mg U, aged 100 days after reactor shutdown. In this case, R = 28 kW per Mg U. Thus, the adiabatic rate of temperature rise would be 8.5x28 = 238 K per hr (deg. C per hr).

Table I.4-5
Radioactive Decay Heat in Spent Fuel at Selected Times After Reactor Shutdown, with a Fuel Burnup of 50 GWt-days per Mg HM

Type of Fuel	Decay Heat (kW per Mg HM) at Selected Times After Reactor Shutdown				
	1 day	10 days	100 days	1,000 days	10,000 days
PWR-U	182	78	28	5.1	1.3
PWR-MOX	187	93	41	7.7	2.9
BWR-U	180	77	27	4.9	1.2
BWR-MOX	180	91	40	7.3	2.7

Notes:
(a) Data are from: Ade and Gauld, 2011. These data were estimated using the SCALE code system. Decay heat was estimated for burnups of 35, 40, 45 and 50 GWt-days per Mg HM, and for times from 0.01 to 19,300 days after reactor shutdown.
(b) PWR-U and BWR-U fuel pellets contain only uranium oxide when fresh. PWR-MOX and BWR-MOX fuel pellets contain a mixture of uranium oxide and plutonium oxide when fresh. ("MOX" refers to mixed-oxide fuel.) The decay heats shown for MOX fuel are for fuel made from reactor-grade plutonium.
(c) "HM" refers to heavy metal (uranium and plutonium) in fresh fuel.

Table I.4-6
Estimated Atmospheric Release of Radioactive Material and Downwind Inhalation Dose for an Event Involving Fuel Damage and Breach of the Multi-Purpose Canister in a Spent-Fuel-Storage Module (Cask)

Indicator		Release Characteristics for Selected Values of MPC Leakage Area		
		4 sq. mm (equiv. dia. = 2.3 mm)	**100 sq. mm (equiv. dia. = 11 mm)**	**1,000 sq. mm (equiv. dia. = 36 mm)**
Fuel Release Fraction	Gases	3.0E-01	3.0E-01	3.0E-01
	Crud	1.0E+00	1.0E+00	1.0E+00
	Volatiles	2.0E-04	2.0E-04	2.0E-04
	Fines	3.0E-05	3.0E-05	3.0E-05
MPC Blowdown Fraction		9.0E-01	9.0E-01	9.0E-01
MPC Escape Fraction	Gases	1.0E+00	1.0E+00	1.0E+00
	Crud	7.0E-02	5.0E-01	8.0E-01
	Volatiles	4.0E-03	3.0E-01	6.0E-01
	Fines	7.0E-02	5.0E-01	8.0E-01
Inhalation Dose (CEDE) to a Person at a Distance of 900 m		0.063 Sv	0.48 Sv	0.79 Sv

Notes:
(a) This table is adapted from Table 6-1 of: Thompson, 2009.
(b) The assumed multi-purpose canister (MPC) contains 24 PWR spent fuel assemblies with a burnup of 40 MWt-days per kgU, aged 10 years after discharge.
(c) The assumed event involves shock damage to spent fuel in the MPC, and a breach of the MPC envelope. This event might, for example, be the impact of an aircraft on a storage module. Radioactive material is released to atmosphere by blowdown of the MPC. There is no fire inside or outside the cask.
(d) The following radioisotopes are considered: Gases (H-3, I-129, Kr-85); Crud (Co-60); Volatiles (Sr-90, Ru-106, Cs-134, Cs-137); Fines (Y-90 and 22 other isotopes).
(e) The calculation follows NRC guidance for calculating radiation dose from a design-basis accident, except that the MPC Escape Fraction is drawn from a study by Sandia National Laboratories that used the MELCOR code package.
(f) CEDE = committed effective dose equivalent. In this scenario, CEDE makes up most of the total dose (TEDE) and is a sufficient approximation to it.
(g) The overall fractional release of a radioisotope from fuel to atmosphere is the product of Fuel Release Fraction, MPC Blowdown Fraction, and MPC Escape Fraction. The Fuel Release Fraction is the same in all three cases because the shock damage to spent fuel is assumed to be the same. The only change in each case is the MPC leakage area.
(h) For an MPC leakage area of 1,000 square mm, the overall fractional release is: Gases (0.27); Crud (0.72); Volatiles (1.1E-04); Fines (2.2E-05).

Table I.5-1
Number of Commercial Nuclear Reactors Worldwide, by Type

Type Code	Description	Number of Reactors as of 31 December 2010	
		Operational	**In Construction**
PWR	Pressurized Light-Water-Moderated and Cooled Reactor	269	56
BWR	Boiling Light-Water-Moderated and Cooled Reactor	92	4
PHWR	Pressurized Heavy-Water-Moderated and Cooled Reactor	46	4
GCR	Gas-Cooled, Graphite-Moderated Reactor	18	
LWGR	Light-Water-Cooled, Graphite-Moderated Reactor	15	1
FBR	Fast Breeder Reactor	1	2
TOTAL		441	67

Notes:
(a) This table is adapted from: IAEA, 2011, Table 23.
(b) PHWR reactors are in Argentina, Canada, China, India, South Korea, Pakistan, and Romania. The PHWR reactors built by the Canadian nuclear industry are known as CANDU reactors.
(c) All GCR reactors are in the UK.
(d) LWGR reactors were constructed only in the former USSR, where they were known as RBMK reactors.
(e) The fast breeder reactors listed in this table are cooled by sodium.

Table I.6-1
Steps in Assessing the Radiological Risk Posed by a Spent-Fuel System

Step Name	Major Tasks	Key Questions
Step 1: Specify the system	Identify components, boundaries (in space and time), linkages, and special features of the system	• Where is the SNF (e.g., geographically, type of facility)? • How is the SNF linked to related facilities (e.g., reactors)? • Special features of system (e.g., seismic risk, attack potential)?
Step 2: Characterize SNF	Characterize SNF in the system (across space and time)	• Amount and type of SNF? • Design features of SNF facilities (e.g., pools, casks) and linked facilities? • Properties of SNF (e.g., age, burnup)? • Cesium-137 inventory?
Step 3: Assess release potential	Identify scenarios for radioactive release to atmosphere; assess their characteristics and feasibility	• Significant accident scenarios? • Significant attack scenarios? • Scenarios involving linked facilities? • Timeframes of scenarios? • Atmospheric release of Cesium-137?
Step 4: Estimate plume behavior	Estimate track of radioactive plume, deposition of material, and resulting radiation dose	• Local weather data? • Plume transport and deposition of Cesium-137? • Ground contamination and external radiation dose?
Step 5: Characterize downwind assets	Characterize population and property (e.g. buildings, agricultural land) at downwind locations	• Sub-national and national boundaries? • Concentrations of population? • Land uses (e.g., agriculture)? • Value of assets (e.g., buildings)?
Step 6: Assess harm to downwind assets	Estimate effects of the radioactive plume on human health and land use; assign monetized values to these effects	• Collective radiation dose? • Radiation-caused mortality? • Monetized value of health effects? • Area of abandoned land? • Value of abandoned or damaged assets?
Step 7: Assess collateral implications	Examine the wider context and implications of SNF radiological risk	• Societal and strategic implications of radiological risk? • Opportunities to reduce rad. risk? • Linkage of rad. risk with other risks (e.g., nuclear-weapon proliferation)? • Role of risk in the sustainability of a nuclear infrastructure investment?

Table II.1-1
Storage Status of Spent Fuel at Fukushima #1 Nuclear Site in Japan, as of March 2010

Storage Method	Storage Capacity (number of fuel assemblies)	Inventory (number of fuel assemblies)
Spent-fuel pools at six reactors	8,310	3,450
Common spent-fuel pool	6,840	6,291
Dry casks	408	408
Total	15,558	10,149

Notes:
(a) These data are from: Kumano, 2010.
(b) Six reactors were operational at the Fukushima #1 site prior to the accident of March 2011. These reactors discharged about 700 spent fuel assemblies each year. The site's total spent-fuel storage capacity of 15,558 assemblies was approximately 450% of the total core capacity of the six reactors.
(c) The six reactors entered commercial service between March 1971 (Unit 1) and October 1979 (Unit 6).

Table II.2-1
Estimated Core Inventories of Iodine-131 and Cesium-137 at Three Types of NPP in the Generation III Category

Plant Type	Core Inventory (PBq)		Normalized Core Inventory (PBq per GWe)	
	Iodine-131	Cesium-137	Iodine-131	Cesium-137
ACR-1000	3,640	172	3,640	172
US-EPR	5,140	914	3,210	571
AP1000	3,560	418	3,560	418

Notes:
(a) This table is adapted from Table 3-2 of: Thompson, 2008. Core inventories are estimates by Bruce Power, which operates NPPs in Ontario, Canada. It can be presumed that the core inventories were estimated for full-power, steady-state operation.
(b) According to Bruce Power, the nominal electricity-generating capacities of the three plant types are:

- ACR-1000: 1,000 MWe
- US-EPR: 1,600 MWe
- AP1000: 1,000 MWe

(c) These plants are generic, proposed, Generation III plants. The ACR-1000 is an "advanced CANDU" plant. The US-EPR and AP1000 are PWR plants. Data for specific plants that are built may differ from the values shown here.
(d) The half-lives of Iodine-131 and Cesium-137 are 8 days and 30 years, respectively.

Table II.2-2
Amounts of Cesium-137 Related to the Chernobyl and Fukushima #1 Accidents

Category	Amount of Cesium-137 (PBq)
Chernobyl release to atmosphere (1986)	85
Fukushima #1 release to atmosphere (2011)	36
Deposition on Japan due to the Fukushima #1 atmospheric release	6.4
Pre-release inventory in reactor cores of Fukushima #1, Units 1-3 (total for 3 cores)	940
Pre-release inventory in spent-fuel pools of Fukushima #1, Units 1-4 (total for 4 pools)	2,200

Notes:
(a) This table shows estimated amounts of Cesium-137 from: Stohl et al, 2011. The estimates for release from Fukushima #1 and deposition on Japan may change as new information becomes available.
(b) Stohl et al, 2011, provide the following data and estimates for Fukushima #1, Units 1-4, just prior to the March 2011 accident:

Indicator	Unit 1	Unit 2	Unit 3	Unit 4
Number of fuel assemblies in reactor core	400	548	548	0
Number of fuel assemblies in reactor spent-fuel pool	392	615	566	1,535
Cesium-137 inventory in reactor core (Bq)	2.40E+17	3.49E+17	3.49E+17	0
Cesium-137 inventory in reactor pool (Bq)	2.21E+17	4.49E+17	3.96E+17	1.11E+18

(The core capacity of Unit 4 was 548 assemblies. The core of Unit 3 contained some MOX fuel assemblies at the time of the accident.)
(c) Assuming a total Cesium-137 release to atmosphere of 36 PBq, originating entirely from the reactor cores of Units 1, 2, and 3, which contained 940 PBq, the overall release fraction to atmosphere for Cesium-137 was 36/940 = 0.038 = 3.8 percent.

Table II.2-3
Calculation of Fission-Product Yield in a Uranium-Fueled Nuclear Reactor

Assumptions and data

- Steady-state thermal fission of U-235 and Pu-239 in a reactor at a power level S (GWt) for a period M (days), where the Pu-239 is created during reactor operation
- Available energy yield from fission is 200 MeV (3.20E-11 J) per fission of a U-235 nucleus or 210 MeV (3.36E-11 J) per fission of a Pu-239 nucleus
- Fraction of total fissions that are Pu-239 fissions = 0.33 (typical for LWR fuel)
- Consider a fission product with a half-life τ (seconds) and a fission yield Z (percent), where Z is assumed to be identical for U-235 and Pu-239 fission (this is a reasonably accurate assumption for Cesium-137)
- Neglect radioactive decay of the fission product during reactor operation
- The fission product is not produced as a daughter or by activation

Calculations for a generic fission product

- Total energy yield = (power)(time) = (energy yield per fission)(no. of fissions)
- Total energy yield (J) = S(1E+09)M(60x60x24) = SM(8.64E+13)
- Energy yield (J) per av. fission = 0.67(3.20E-11) + 0.33(3.36E-11) = 3.25E-11
- Number of fissions = SM(8.64E+13)/(3.25E-11) = SM(2.66E+24)
- Number of fission-product atoms created = SM(2.66E+24)(Z/100)
- Number of fission-product atoms after t sec = (number created)(exp -(tln2)/τ)
- Thus, radioactive decay (Bq) per fission-product atom created = (ln2)/τ
- Thus, reactor yield (Bq) of fission product = SMZ(2.66E+22)(ln2)/τ

Calculations for the fission product Cesium-137

- For Cesium-137, τ = 9.46E+08 seconds (30 yr), and Z = 6 (approx.)
- Set SM = 1 (i.e., the reactor yields 1 GWt-day of energy)
- Thus, yield of Cesium-137 (Bq per GWt-day) = 6(2.66E+22)(ln2)/(9.46E+08) = 1.17E+14
- Inventory of Cesium-137 (Bq) after t yr = exp((-tln2)/30)(Inventory for t = 0)

Notes:
(a) For background see: Parker and Barton, 1973.
(b) The key result in this table is that 1 GWt-day of fission energy will yield 1.17E+14 Bq of Cesium-137.
(c) 1 Ci = 3.7E+10 Bq, thus 1.17E+14 Bq = 3.16E+03 Ci

Table II.3-1
Potential Types of Attack on a Nuclear Facility (a Reactor and/or Spent-Fuel-Storage Installation) Leading to Atmospheric Release of Radioactive Material

Type of Event	Facility Behavior	Some Relevant Instruments and Modes of Attack	Characteristics of Atmospheric Release
Type 1: Vaporization or Pulverization	• All or part of facility is vaporized or pulverized	• Facility is within the fireball of a nuclear-weapon explosion	• Radioactive material in facility is lofted into the atmosphere and amplifies fallout from nuc. explosion
Type 2: Rupture and Dispersal (Large)	• Facility structures are broken open • Fuel is dislodged from facility and broken apart • Some ignition of zircaloy fuel cladding may occur, typically without sustained combustion	• Aerial bombing • Artillery, rockets, etc. • Effects of blast etc. outside the fireball of a nuclear-weapon explosion	• Solid pieces of various sizes are scattered in vicinity • Gases and small particles form an aerial plume that travels downwind • Some release of volatile species (esp. Cesium-137) if zirc. combustion occurs
Type 3: Rupture and Dispersal (Small)	• Facility structures are penetrated but retain basic shape • Fuel may be damaged but most rods retain basic shape • Damage to cooling systems could lead to zirc. combustion	• Vehicle bomb • Impact by commercial aircraft • Perforation by shaped charge	• Scattering and plume formation as in Type 2 event, but involving smaller amounts of material • Substantial release of volatile species if zirc. combustion occurs
Type 4: Precise, Informed Targeting	• Facility structures are penetrated, creating a release pathway • Zirc. combustion is initiated indirectly by damage to cooling systems, or by direct ignition	• Missiles (military or improvised) with tandem warheads • Close-up use of attack instruments (e.g., shaped charge, incendiary, thermic lance)	• Scattering and plume formation as in Type 3 event • Substantial release of volatile species, potentially exceeding amount in Type 3 release

Table II.3-2
Some Potential Modes and Instruments of Attack on a Nuclear Power Plant

Attack Mode/Instrument	Characteristics	Present Defenses at US Plants
Commando-style attack	• Could involve heavy weapons and sophisticated tactics • Successful attack would require substantial planning and resources	Alarms, fences and lightly-armed guards, with offsite backup
Land-vehicle bomb	• Readily obtainable • Highly destructive if detonated at target	Vehicle barriers at entry points to Protected Area
Small guided missile (anti-tank, etc.)	• Readily obtainable • Highly destructive at point of impact	None if missile launched from offsite
Commercial aircraft	• More difficult to obtain than pre-9/11 • Can destroy larger, softer targets	None
Explosive-laden smaller aircraft	• Readily obtainable • Can destroy smaller, harder targets	None
10-kilotonne nuclear weapon	• Difficult to obtain • Assured destruction if detonated at target	None

Notes:
(a) This table is adapted from: Thompson, 2007, Table 7-4. Further citations are provided in that table and its supporting narrative. For additional, supporting information of more recent vintage, see: Ahearne et al, 2012, Chapter 5.
(b) Defenses at nuclear power plants around the world are typically no more robust than at US plants.

Table II.3-3
The Shaped Charge as a Potential Instrument of Attack

Category of Information	Selected Information in Category
General information	• Shaped charges have many civilian and military applications, and have been used for decades • Applications include human-carried demolition charges or warheads for anti-tank missiles • Construction and use does not require assistance from a government or access to classified information
Use in World War II	• The German MISTEL, designed to be carried in the nose of an un-manned bomber aircraft, is the largest known shaped charge • Japan used a smaller version of this device, the SAKURA bomb, for kamikaze attacks against US warships
A large, contemporary device	• Developed by a US government laboratory for mounting in the nose of a cruise missile • Described in detail in an unclassified, published report (citation is voluntarily withheld here) • Purpose is to penetrate large thicknesses of rock or concrete as the first stage of a “tandem” warhead • Configuration is a cylinder with a diameter of 71 cm and a length of 72 cm • When tested in November 2002, created a hole of 25 cm diameter in tuff rock to a depth of 5.9 m • Device has a mass of 410 kg; would be within the payload capacity of many general-aviation aircraft
A potential delivery vehicle	• A Beechcraft King Air 90 general-aviation aircraft can carry a payload of up to 990 kg at a speed of up to 460 km/hr • The price of a used King Air 90 in the USA can be as low as $0.4 million

Source:
This table is adapted from Table 7-6 of: Thompson, 2009.

Table II.3-4
Performance of US Army Shaped Charges, M3 and M2A3

Target Material	Indicator	Value for Stated Type of Shaped Charge	
		Type: M3	Type: M2A3
Reinforced concrete	Maximum wall thickness that can be perforated	150 cm	90 cm
	Depth of penetration in thick walls	150 cm	75 cm
	Diameter of hole	• 13 cm at entrance • 5 cm minimum	• 9 cm at entrance • 5 cm minimum
	Depth of hole with second charge placed over first hole	210 cm	110 cm
Armor plate	Perforation	At least 50 cm	30 cm
	Average diameter of hole	6 cm	4 cm

Notes:
(a) This table is adapted from Table 7-7 of: Thompson, 2009. The data are from US Army Field Manual FM 5-25, published May 1967.
(b) The M2A3 charge has a mass of 5 kg, a maximum diameter of 18 cm, and a total length of 38 cm including the standoff ring.
(c) The M3 charge has a mass of 14 kg, a maximum diameter of 23 cm, a charge length of 39 cm, and a standoff pedestal 38 cm long.

Table II.3-5
Some Characteristics of Containments of Selected NPPs in the Generation II and Generation III Categories

Plant Name or Type	Containment Characteristics
Indian Point Units 2 and 3	• The containment is a reinforced concrete vertical cylinder topped by a hemispherical dome made of the same material. The cylinder walls are 1.4 m thick with a 1.0 cm thick steel liner, and the dome is 1.1 m thick with a 1.3 cm thick steel liner. • There is no shield building.
ACR-1000	• The containment is a vertical cylinder with a domed top, made of pre-stressed (cable-tensioned) concrete and equipped with a steel liner. The wall thickness of the cylinder is 1.8 m. According to Bruce Power: "The containment structure is designed for tornado conditions, including tornado missiles, and aircraft crashes." • There is no shield building.
US-EPR	• The containment is a vertical cylinder with a domed top, made of pre-stressed (cable-tensioned) concrete and equipped with a steel liner. The wall of the cylinder is 1.3 m thick, and the dome is 1.0 m thick. • Surrounding the containment is a shield building (with a configuration similar to that of the containment) made of reinforced concrete. This building's wall and dome are each 1.8 m thick.
AP1000	• The containment is a vertical, steel cylinder with a wall thickness of 4.4 cm. • Surrounding the containment is a cylindrical shield building made of reinforced concrete, with a wall thickness of 0.9 m.

Notes:
(a) Data are from: Thompson, 2007, Section 7.5; Thompson, 2008, Section 5.
(b) Indian Point Units 2 and 3 are Generation II PWR plants operating in New York State, USA, and are located on the Hudson River upstream of New York City.
(c) The other three plants are generic, proposed, Generation III plants. The ACR-1000 is an "advanced CANDU" plant. The US-EPR and AP1000 are PWR plants. Data for specific plants that are built may differ from the values shown here.
(d) These characteristics provide an indication of each containment's ability to resist external attack. Other characteristics would also be relevant to a full-scope assessment of the radiological risk posed by each plant.

Table II.3-6
Predicted Size of Crater Caused by Ground Burst of a Nuclear Weapon

Indicator	Value of Indicator for Stated Weapon Yield			
	1 kt	**10 kt**	**100 kt**	**1,000 kt**
Crater diameter (m)	32	71	150	320
Crater depth (m)	7.3	16	35	73

Notes:
(a) Predictions are from: Glasstone, 1964, pp 292-293.
(b) The ground surface is assumed here to be rock such as granite or sandstone. Consistent with that assumption, the predictions for dry soil are adjusted here by a multiplier of 0.8.
(c) A weapon yield of X kt indicates that the weapon releases energy equivalent to that from exploding X kiloton (X Gg) of TNT.
(d) Below the crater would be a "rupture zone" in which the soil or rock has numerous cracks. The diameter of the rupture zone would be roughly one and one-half the diameter of the crater.
(e) A "lip" of excavated material would extend beyond the crater.
(f) The crater would be created by the combined effects of vaporization, scouring, and compression.

Table II.3-7
Estimation of Atmospheric Release Fraction of Cesium-137 for Selected Fuel-Damage Events

Fuel-Damage Event	Release Fraction from Affected Fuel (RF_1)	Multiplier for Release Inhibition along Escape Pathway (RF_2)	Release Fraction to Atmosphere (RF_1)(RF_2)
Melting of a reactor core	See note (d)	See note (e)	See note (f)
Fire in a spent-fuel pool	See note (g)	See note (h)	See note (i)
Fire in a dry cask containing spent fuel	See note (j)	See note (h)	See note (i)

Notes:
(a) Background information is available from many sources, including: NRC, 1990; McGuire et al, 2007.
(b) For the purposes of this handbook, the "affected" fuel is typically the entire fuel inventory in each reactor core, spent-fuel pool, or dry cask. Event scenarios can be envisioned in which a portion of the fuel inventory is not affected (i.e., not significantly damaged). Consideration of those scenarios would require a more sophisticated analysis than is presented here.
(c) The "release fraction to atmosphere" in the fourth column is the fraction of Cesium-137 in affected fuel that reaches the external atmosphere, and is the product of entries in the two preceding columns (i.e., RF_1 and RF_2).
(d) McGuire et al, 2007, Tables 1.3 and 1.4, suggest a value of 0.6 (60%).
(e) McGuire et al, 2007, Table 1.9, suggest values for various scenarios, ranging from 0.001 to 1.0, where the upper end of the range reflects breach or bypass of the reactor containment.
(f) Given a substantial breach of reactor containment, a range of 0.1 to 0.5 is reasonable.
(g) McGuire et al, 2007, Table 2.1, suggest a value of 0.3 (30%).
(h) There is no containment surrounding a spent-fuel pool or dry cask. Thus, mitigation of the release would be comparatively ineffective. A value of 1.0 would typically be appropriate.
(i) A range of 0.1 to 0.5 is reasonable for a pool or cask fire.
(j) This value would be comparable to the value for a pool fire – see note (g).

Table II.4-1 (Part 1 of 2)
Estimation of External Radiation Dose from Cesium-137 Contamination of Land, Using the Wedge Model

Background
The wedge model is a simple representation of the atmospheric dispersion of radioactive material. Formulation of the wedge model is generally attributed to a Study Group convened by the American Physical Society (Lewis et al, 1975, Appendix II). An advantage of the model is that it does not require computer calculations.

Assumptions underlying the wedge model
An amount Q (Bq) of radioactive material is released to the atmosphere from a single point, over a time period when the wind velocity is constant. The air concentration of the material in the resulting airborne plume at a downwind distance r (m) is circumferentially uniform across a wedge of angular width θ (radians), and vertically uniform across a mixing layer of height H (m) above the ground. The wind velocity has magnitude W (m/s). The aerosol deposition velocity is constant, with magnitude V (m/s). Radioactive decay of the released material can be considered in the wedge model, but in the case of Cesium-137 this decay would be negligible over the duration of plume passage.

Aerosol deposition in the wedge model
Assumption of a constant deposition velocity, literally interpreted, could imply that the radioactive material would be fully deposited on the ground when the plume has traveled a distance L = WH/V where L is known as the "deposition length" (m). However, the wedge model assumes that: (i) aerosol concentration remains vertically uniform up to height H; and (ii) ground deposition at a given distance is proportional to the amount of material remaining airborne in the plume as it passes that distance.

Thus, if the amount (in Bq) of radioactive material remaining airborne at distance r is designated $Q_{remaining}\{r\}$, the wedge model says:

$$Q_{remaining}\{r\} = (Q)\exp(-r/L) \quad \text{or} \quad dQ_{remaining}/dr = (-1/L)Q_{remaining}$$

Ground contamination
During plume passage, the ground at radius r accumulates radioactive material at the spatial rate $-dQ_{remaining}/dr$ across an arc (the plume front) of length θr. Thus, if the initial ground contamination at distance r, from deposition of radioactive material, is designated σ (Bq per m^2), we see from the preceding discussion that:

$$\sigma\{r\} = (1/L\theta r)Q_{remaining} = (Q/L\theta r)\exp(-r/L)$$

Table II.4-1 (Part 2 of 2)
Estimation of External Radiation Dose from Cesium-137 Contamination of Land, Using the Wedge Model

External dose rate from Cesium-137 contamination
If ground is contaminated by deposition of Cesium-137 in the amount σ (Bq per m^2), then the whole-body dose rate to a local human individual from external gamma radiation, designated D' (Sv per yr), can be estimated, accounting for both weathering and radioactive decay, by the formula (Beyea et al, 1979, Section 3.13, Note 9a; NRC, 1975, Appendix VI, page E-4 and Table VI C-2):

$$D'\{t\} = (1.83E\text{-}08)(0.63(\exp{-1.15t}) + 0.37(\exp{-0.03t}))\sigma F$$

where t is time in yr and F is the average shielding factor

Cumulative external dose from Cesium-137 contamination
The external dose rate to an individual at time t (yr) is D'{t} (Sv/yr), calculated as above. The cumulative external dose until time T (yr), designated $D_{cum}\{T\}$ (Sv), is determined by integrating D'{t} from t = 0 to t = T, yielding the equation:

$$D_{cum}\{T\} = (D'\{0\})(0.548(1-\exp(-1.15T)) + 12.3(1-\exp(-0.03T)))$$

Typical values (dry deposition)
θ = 0.25 radians (stability class "D"); W = 5 m/s; V = 0.01m/s; H = 1,000 m; F = 0.33

Notes:
(a) Relevant sources include: Alvarez et al, 2003; Beyea et al, 1979, Section 3.13; Lewis et al, 1975, Appendix II; NRC, 1975, Appendix VI.
(b) The variation of D' and D_{cum} over time, calculated as shown above, can be illustrated by the following examples:

Time, t (yr)	1	10	20	30	40	50
D'{t}/D'{0}	0.56	0.27	0.20	0.15	0.11	0.08
D_{cum}\{t\}/(D'{0}	0.74	3.7	6.1	7.8	9.2	10.2

Table II.4-2
Illustrative Calculation: Downwind Ground Contamination and External Radiation Dose to a Local Human after Atmospheric Release of 1 PBq of Cesium-137

Indicator	**Value at Downwind Distance, r (m)**			
	r = 1.0E+03 (1 km)	**r = 1.0E+04 (10 km)**	**r = 1.0E+05 (100 km)**	**r = 1.0E+06 (1,000 km)**
Ground contamination, σ{r} (Bq per m^2)	7.98E+06	7.84E+05	6.55E+04	1.08E+03
Initial whole-body dose rate, D'{0} (Sv per yr)	4.8E-02	4.7E-03	4.0E-04	6.5E-06
Wedge area (m^2) within radius r	1.25E+05	1.25E+07	1.25E+09	1.25E+11

Notes:
(a) Assumptions: use of the wedge model; point release of Q = 1.0E+15 Bq (1 PBq) of Cesium-137; θ = 0.25 radians (stability class "D"); W = 5 m/s; V = 0.01m/s; H = 1,000 m; F = 0.33
(b) σ{r} = (Q/Lθr)exp(-r/L) where L = WH/V
(c) D'{t} = (1.83E-08)(0.63(exp-1.15t) + 0.37(exp-0.03t))σF where t is time in yr
(d) σ{r} and D'{t} scale linearly as a function of Q
(e) Wedge area = $(\theta r^2)/2$

Table II.4-3
Relation of Weather Conditions to Pasquill Category of Atmospheric Stability

Surface Wind Speed (m/s)	**Day-time: Pasquill Category for Stated Insolation (Sun Intensity)**			**Night-time: Pasquill Category for Stated Conditions**	
	Strong	**Moderate**	**Slight**	**Thin overcast or 4/8 or more cloudiness**	**3/8 or less cloudiness**
<2	A	A-B	B		
2	A-B	B	C	E	F
4	B	B-C	C	D	E
6	C	C-D	D	D	D
>6	C	D	D	D	D

Notes:
(a) This table is adapted from: Slade, 1968, Table 3.3.
(b) The Pasquill categories of atmospheric stability are:
A: Extremely unstable
B: Moderately unstable
C: Slightly unstable
D: Neutral
E: Slightly stable
F: Moderately stable
(c) Category D applies to heavy overcast, day or night.
(d) The degree of cloudiness is defined as the fraction of the sky above the local apparent horizon that is covered by clouds.
(e) Insolation (sun intensity) is the solar radiation flux (W/m^2) on a horizontal surface.

Table II.4-4
Typical Variability of Wind Direction According to Pasquill Category of Atmospheric Stability

Pasquill Category of Atmospheric Stability	Standard Deviation of Wind Direction (radian)	1.28 x Standard Deviation of Wind Direction (radian)
A: Extremely unstable	0.44	0.56
B: Moderately unstable	0.35	0.45
C: Slightly unstable	0.26	0.34
D: Neutral	0.17	0.22
E: Slightly stable	0.09	0.11
F: Moderately stable	0.04	0.06

Notes:
(a) The standard deviations shown are from: Slade, 1968, Section 3-3.4.
(b) These standard deviations are from experiments with a sampling time of 10-60 minutes.
(c) Assuming a normal distribution, one standard deviation encompasses 68% of data values, 1.28 x standard deviation encompasses 80% of data values, and 1.64 x standard deviation encompasses 90% of data values.

Table II.4-5
Frequency of Rain Conditions at Representative NPP Sites in the USA

Site	Av. Fraction of Time During Which Rain Occurs (percent)
Northeast river valley	5.5
Great Lakes shore	9.9
Dry Western desert	0.6
Central Midwest plain	6.0
Pacific coast	1.0
Atlantic coast	4.9
Southeast river valley	7.5

Source:
Adapted from Table 3 of: Sprung, 1978.

Table II.5-1
Population Density and Related Characteristics for Selected Cities

City	Population (million)	Land Area (square km)	Side of a Square (km) with the Same Area	Population Density (people per square km)
Mumbai, India	14.35	484	22.0	29,650
Seoul/Incheon, South Korea	17.50	1,049	32.4	16,700
Shanghai, China	10.00	746	27.3	13,400
Manila, Philippines	14.75	1,399	37.4	10,550
Jakarta, Indonesia	14.25	1,360	36.9	10,500
Ho Chi Minh City, Vietnam	4.90	518	22.8	9,450
Singapore	4.00	479	21.9	8,350
Osaka/Kobe/Kyoto, Japan	16.43	2,564	50.6	6,400
Paris, France	9.65	2,723	52.2	3,550
New York, USA	17.80	8,683	93.2	2,050

Source:
City Mayors Foundation, 2007

Table II.6-1
Estimation of Collective External Radiation Dose from Cesium-137 Contamination of a Populated Area

Assumptions

The wedge model is used with assumptions as stated above. Also, the wedge intersects a region containing a population with a uniform density of P persons per m^2. In the direction of plume travel, the intersected portion of the populated region is bounded by radii r = A and r = B (both in m) where B>A. The radioactive release consists of Q Bq of Cesium-137. External radiation from the deposited Cesium-137 provides a collective (population) dose, C, measured in person-Sv.

Intersection between wedge and populated region

- Area (m^2) of intersection = $(\theta/2)(B^2 - A^2)$
- Population (number of persons) in this area = $(P\theta/2)(B^2 - A^2)$

Collective (population) dose across intersected area

- Ground contamination $\sigma\{r\} = (Q/L\theta r)\exp(-r/L)$ (Bq per m^2)
- Initial external dose rate $D'\{0\} = (1.83E-08)(\sigma F)$ (Sv per yr)
- Initial collective dose rate $C'\{0\} = (1.83E-08)(F)$(Integral $P\theta r\sigma\{r\}dr$, A to B)
- Thus $C'\{0\} = (1.83E-08)(QFP)(\exp(-A/L) - \exp(-B/L))$ (person-Sv per yr)
- $C_{cum}\{T\}$ (person-Sv) is the cumulative collective dose up to time T (yr), determined by integrating $C'\{t\}$ from t = 0 to t = T
- $C_{cum}\{T\}/C'\{0\} = D_{cum}\{T\}/D'\{0\} = 0.548(1-\exp(-1.15T)) + 12.3(1-\exp(-0.03T))$

Typical values (dry deposition)

F = 0.33; W = 5 m/s; V = 0.01 m/s; H = 1,000 m; L = WH/V = 5E+05 m

Table II.6-2
Illustrative Calculation #1: Collective Dose and Health Effects in a Populated Region, Attributable to External Radiation following Atmospheric Release of 1 PBq of Cesium-137

Assumptions

The wedge model is used. A region has a population density P = 5E-03 persons per m^2 (5,000 persons per km^2). The plume of Cesium-137 intersects this region between radius A = 5E+04 m (50 km) and radius B = 6E+04 m (60 km). Collective (population) dose from external radiation is accrued over a period of 30 yr. Also: F = 0.33; W = 5 m/s; V = 0.01 m/s; H = 1,000 m; L = WH/V = 5E+05 m; Q = 1E+15 Bq; θ = 0.25 radians (stability class "D").

Collective dose

- C'{0} = (1.83E-08)(QFP)(exp(-A/L) – exp(-B/L)) (person-Sv per yr)
- With the stated assumptions, C'{0} = 544 person-Sv per yr
- C_{cum} {30}/C'{0} = D_{cum} {30}/D'{0} = 7.8
- Thus, 30-year collective dose, C_{cum} {30} = (544)(7.8) = 4,240 person-Sv
- The collective dose scales linearly in Q and P
- The preceding calculations are for external, whole-body dose only
- Chernobyl experience suggests that ingestion dose will be comparable to external dose
- Total collective dose is the sum of dose from external exposure, ingestion, and inhalation (In the case of Cesium-137 deposition, inhalation dose will typically be negligible.)

Health effects

- BEIR VII, 2006, provides an estimate of 0.051 excess, solid-cancer deaths per person-Sv of collective dose (The "excess" deaths are, in fact, "premature" deaths, because the affected persons would eventually die from some cause.)
- Thus, a collective dose of 4,240 person-Sv would yield (0.051)(4,240) = 216 excess deaths
- Exposed population = $(P\theta/2)(B^2 - A^2)$
- Exposed population in this example = 687,500 persons
- Percent of the exposed population experiencing excess death from solid cancer (due to external radiation) = (0.051)(C_{cum} {T})(100)/(exposed population)
- Percent of exposed population experiencing excess death in this example, due to 30 years of exposure = (0.051)(4,240)(100)/(687,500) = 0.03 percent

Table II.6-3
Illustrative Calculation #2: Collective Dose and Health Effects in a Populated Region, Attributable to External Radiation and Ingestion following Atmospheric Release of 100 PBq of Cesium-137

Assumptions

- Assume a Cesium-137 release, Q = 1E+17 Bq (100 PBq)
- Assume that the collective dose from ingestion equals the collective dose from external exposure
- Assume other factors and methods as in Illustrative Calculation #1 (see above)

Collective dose

- Collective dose over 30 yr from external radiation = $C_{cum}\{30\}$ = (4,240)(100) = 424,000 person-Sv
- Total collective dose from external radiation and ingestion = (2)(424,000) = 848,000 person-Sv

Health effects

- Exposed population = 687,500 persons
- Number of excess (premature), solid-cancer deaths = (0.051)(total collective dose) = (0.051)(848,000) = 43,250
- Percent of exposed population experiencing excess (premature), solid-cancer death = (43,250)(100)/(687,500) = 6.3 percent

Table II.6-4
Estimated Human Dose Commitment from 1986 Chernobyl Release of Radioactive Material to Atmosphere

Region	50-Year Collective Dose Commitment (person-Gy)	50-Year Average Individual Dose Commitment (mGy)
USSR (European)	4.7E+05	6.1E+00
USSR (Asian)	1.1E+05	Not available
Europe (non-USSR)	5.8E+05	1.2E+00
Asia (non-USSR)	2.7E+04	1.4E-02
USA	1.1E+03	4.6E-03
Northern Hemisphere Total	1.2E+06	Not available

Notes:
(a) These estimated doses are whole-body doses, from: DOE, 1987, Table 5.16, "preferred estimate".
(b) Most of the dose is attributable to Cesium-137 (see: DOE, 1987, page x).
(c) Estimates for non-USSR countries show that, on average, about 50% of the collective dose is attributable to external exposure, and about 50% is attributable to ingestion (see: DOE, 1987, Table 5.14). Uncertainty in these estimates is greater for ingestion than for external exposure.
(d) In this instance, 1 Gy is equivalent to 1 Sv.

Table II.6-5
Estimated Lifetime Incidence and Mortality of Cancer and Leukemia per 100,000 People, With or Without Exposure to 0.1 Gy of Radiation

Category of Health Effect	All Solid Cancer		Leukemia	
	Males	**Females**	**Males**	**Females**
Number of cases in the absence of radiation exposure (i.e., Incidence)	45,500	36,900	830	590
Number of deaths in the absence of radiation exposure (i.e., Mortality)	22,100	17,500	710	530
Number of excess cases from exposure to 0.1 Gy	800	1,300	100	70
Number of excess deaths from exposure to 0.1 Gy	410	610	70	50

Notes:
(a) Estimates are from: BEIR VII, 2006, Table 12-13. That table shows confidence intervals for the numbers of radiation-induced health effects.
(b) These health effects arise in a population of 100,000 with an age distribution typical of the USA.
(c) These BEIR VII estimates of radiation-induced health effects rely upon a "linear no-threshold" (LNT) model of dose response for solid cancer, and a "linear-quadratic" model for leukemia.
(d) The radiation exposure considered here is a one-time dose of 0.1 Gy. Where exposure is attributable to Cesium-137, 1 Gy is equivalent to 1 Sv.
(e) Individual risk of a radiation-induced health effect varies with age at exposure. From Table 12D-2 of BEIR VII, 2006, one sees that the lifetime, radiation-induced, solid-cancer mortality risk for males (females) is 1,028 (1,717) per 100,000 people if radiation exposure occurs at birth, declining to 102 (152) per 100,000 people if the exposure occurs at age 80 years.
(f) The LNT model allows individual risk of exposure to be estimated as follows:

- For a typical (average) person, the lifetime, radiation-induced, solid-cancer mortality risk is: (((410+610)/2)/100,000)/(0.1) = 0.051 per Gy (or Sv) of exposure.
- For a newborn, the lifetime, radiation-induced, solid-cancer mortality risk is: (((1,028+1,717)/2)/100,000)/(0.1) = 0.14 per Gy (or Sv) of exposure.

(g) The first-listed finding in note (f) can be re-stated as follows:

- Across a steady-state population with an age distribution similar to that of the USA, there will be 0.051 excess, solid-cancer deaths per person-Sv of cumulative, collective radiation exposure, where the exposure can occur at any point(s) in time, or across any time period(s).

Table II.6-6
Estimation of Downwind Distance Within Which 1st Year External Radiation Dose Exceeds a Specified Threshold, Following an Atmospheric Release of Cesium-137

Indicator	Value of Indicator for Atmospheric Release of Cesium-137 in the Amount Q (Bq)			
	Q = 1E+15 (1 PBq)	Q = 1E+16 (10 PBq)	Q = 1E+17 (100 PBq)	Q = 1E+18 (1,000 PBq)
Downwind distance (m) within which 1st year external radiation dose exceeds N (Sv), for the case in which N = 0.02 Sv	1.8E+03 (1.8 km)	1.7E+04 (17 km)	1.4E+05 (140 km)	5.7E+05 (570 km)
Area (m^2) of land in wedge contaminated by Cesium-137, within this distance	4.1E+05 (0.41 square km)	3.6E+07 (36 square km)	2.5E+09 (2,500 square km)	4.1E+10 (41,000 square km)

Notes:
(a) Assumptions: use of the wedge model; point release of Q Bq of Cesium-137; θ = 0.25 radians (stability class "D"); W = 5 m/s; V = 0.01m/s; H = 1,000 m; F = 0.33; L = WH/V = 5E+05 m
(b) Here, the specified dose threshold, N (Sv), is the cumulative external dose over the 1st year of exposure to Cesium-137 deposited on the ground, accounting for shielding by a factor F. Thus, $N = D_{cum}\{1\} = (0.74)(D'\{0\})$ where $D'\{0\} = (1.83E\text{-}08)(\sigma F)$ and $\sigma\{r\} = (Q/L\theta r)\exp(-r/L)$
(c) Given the preceding assumptions, the downwind distance r (m) within which 1st-year external dose exceeds N (Sv) is determined by: $(1/r)\exp(-r/L) = (NL\theta)/((1.35E\text{-}08)(QF))$
(d) Thus, for example, if N = 0.02 (Sv) and Q = 1E+15 (Bq), then $(1/r)\exp(-r/L)$ = (5.61E-04), and that equation can be solved numerically to yield r = 1.8E+03 (m). The same approach is used here to calculate r for other values of Q, yielding the results shown in the table above.
(e) Wedge area (m^2) within radius r is $(\theta r^2)/2$
(f) In its guidance manual for nuclear incidents, the US Environmental Protection Agency (EPA) recommends that the general population be relocated if the cumulative 1st-year dose to an individual at a radioactively-contaminated location is projected to exceed 0.02 Sv. EPA states that the projected dose should account for external gamma radiation and inhalation of re-suspended material during the 1st year, but should not account for shielding from structures or the application of dose reduction techniques. (See: EPA, 1992, page 4-4.) Thus, EPA's threshold dose is more conservative than the dose, N, used in this table.
(g) EPA also recommends that the cumulative 50-year dose to an exposed individual should not exceed 0.05 Sv. (See: EPA, 1992, page 4-4.) Given the assumptions used in this table, a 1st-year dose of 0.02 Sv from external exposure to deposited Cesium-137 implies a cumulative 50-year dose of (0.02/0.74)(10.2) = 0.28 Sv.

Table II.7-1
Selected Approaches to Protecting a Country's Critical Infrastructure From Attack by Non-State Actors, and Some Strengths and Weaknesses of these Approaches

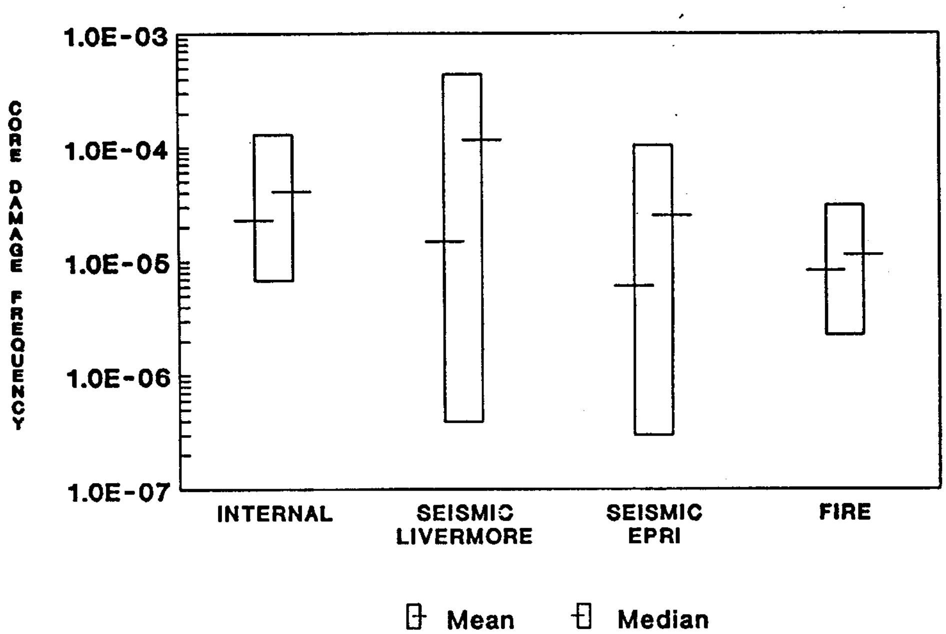
Approach	Strengths	Weaknesses
Offensive military operations internationally	• Could deter or prevent governments from supporting non-State groups hostile to the Country	• Could promote growth of non-State groups hostile to the Country, and build sympathy for these groups in foreign populations • Could be costly in terms of lives, money, etc.
International police cooperation within a legal framework	• Could identify and intercept potential attackers	• Implementation could be slow and/or incomplete • Requires ongoing international cooperation
Surveillance and control of the domestic population	• Could identify and intercept potential attackers	• Could destroy civil liberties, leading to political, social and economic decline
Secrecy about design and operation of infrastructure facilities	• Could prevent attackers from identifying points of vulnerability	• Could suppress a true understanding of risk • Could contribute to political, social and economic decline
Active defense of infrastructure facilities (by use of guards, guns, gates, etc.)	• Could stop attackers before they reach the target	• Requires ongoing expenditure & vigilance • May require military involvement
Robust and inherently-safe design of infrastructure facilities	• Could allow target to survive attack without damage, thereby enhancing protective deterrence • Could substitute for other protective approaches, avoiding their costs and adverse impacts • Could reduce risks from accidents & natural hazards	• Could involve higher capital costs

Table II.7-2
Estimated Discharge of Plutonium from Nuclear Power Reactors, 1961-2010: Selected Countries and World Total

Country	Cumulative Discharge of Plutonium (kg)		
	1961-1993	1994-2010	1961-2010
Argentina	5,970	12,200	18,170
Brazil	520	4,400	4,920
Canada	67,230	99,270	166,500
India	4,500	21,120	25,620
Korea (South)	14,670	49,870	64,540
Pakistan	410	780	1,190
South Africa	2,340	5,600	7,940
WORLD TOTAL	846,200	1,278,760	2,124,960

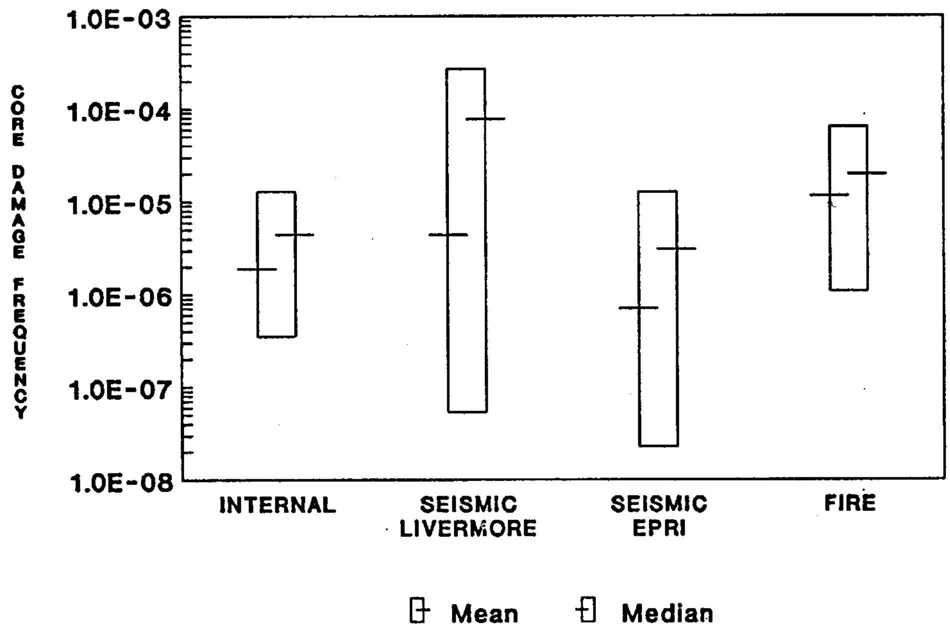
Source:
Albright et al, 1997, Tables 5.3 and 5.4.

Figure I.2-1
Core Damage Frequency for Accidents at a Surry PWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



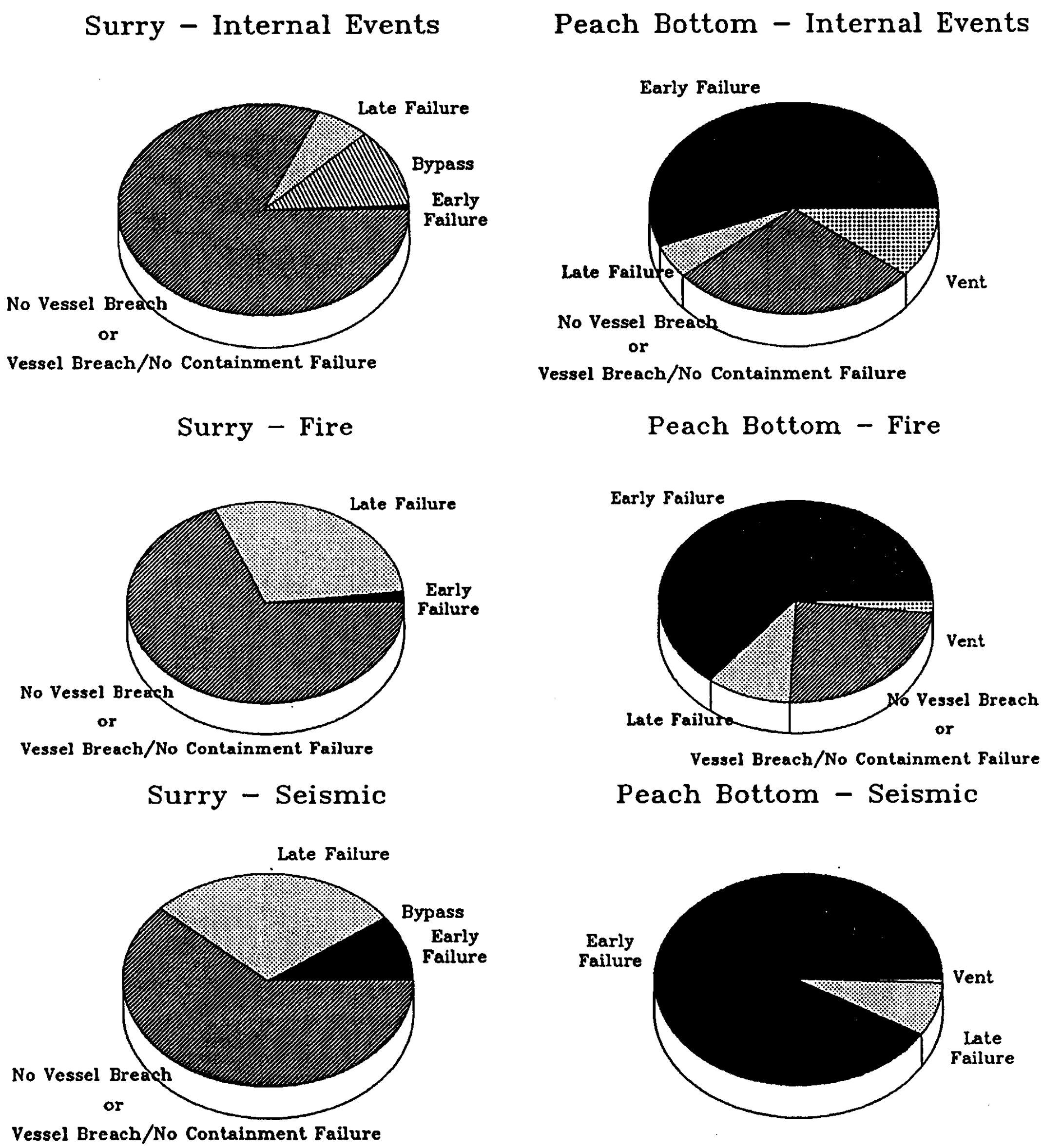
Notes:
(a) This figure is adapted from Figure 8.7 of: NRC, 1990.
(b) The bars range from the 5th percentile (lower bound) to the 95th percentile (upper bound) of the estimated core-damage frequency (CDF). CDF values shown are per reactor-year (RY).
(c) Two estimates are shown for the CDF from earthquakes (seismic effects). One estimate derives from seismic predictions done at Lawrence Livermore National Laboratory (Livermore), the other from predictions done at the Electric Power Research Institute (EPRI).
(d) CDFs are not estimated for external initiating events other than earthquakes and fires.
(e) Malevolent acts and gross errors in design, construction, or operation are not considered.

Figure I.2-2
Core Damage Frequency for Accidents at a Peach Bottom BWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



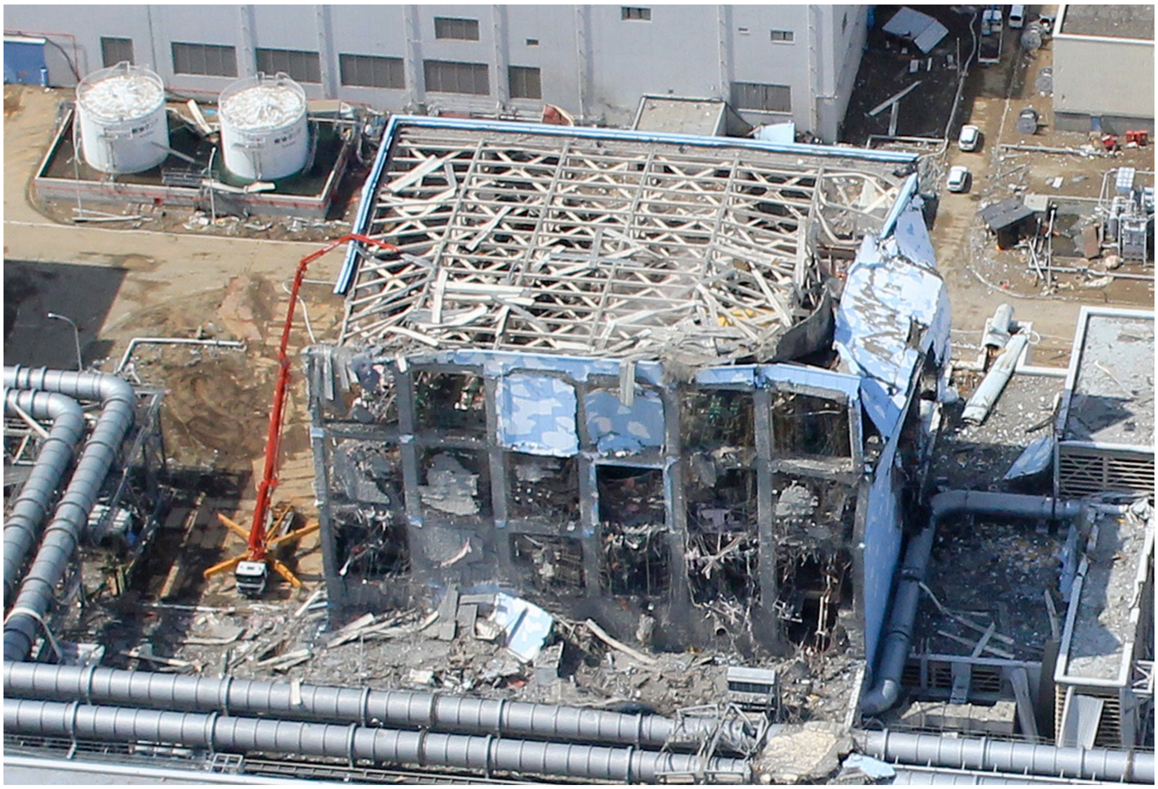
Notes:
(a) This figure is adapted from Figure 8.8 of: NRC, 1990.
(b) The bars range from the 5th percentile (lower bound) to the 95th percentile (upper bound) of the estimated core-damage frequency (CDF). CDF values shown are per reactor-year (RY).
(c) Two estimates are shown for the CDF from earthquakes (seismic effects). One estimate derives from seismic predictions done at Lawrence Livermore National Laboratory (Livermore), the other from predictions done at the Electric Power Research Institute (EPRI).
(d) CDFs are not estimated for external initiating events other than earthquakes and fires.
(e) Malevolent acts and gross errors in design, construction, or operation are not considered.

Figure I.2-3
Conditional Probability of Containment Failure Following a Core-Damage Accident at a Surry PWR or Peach Bottom BWR Nuclear Power Plant, as Estimated in the NRC Study NUREG-1150



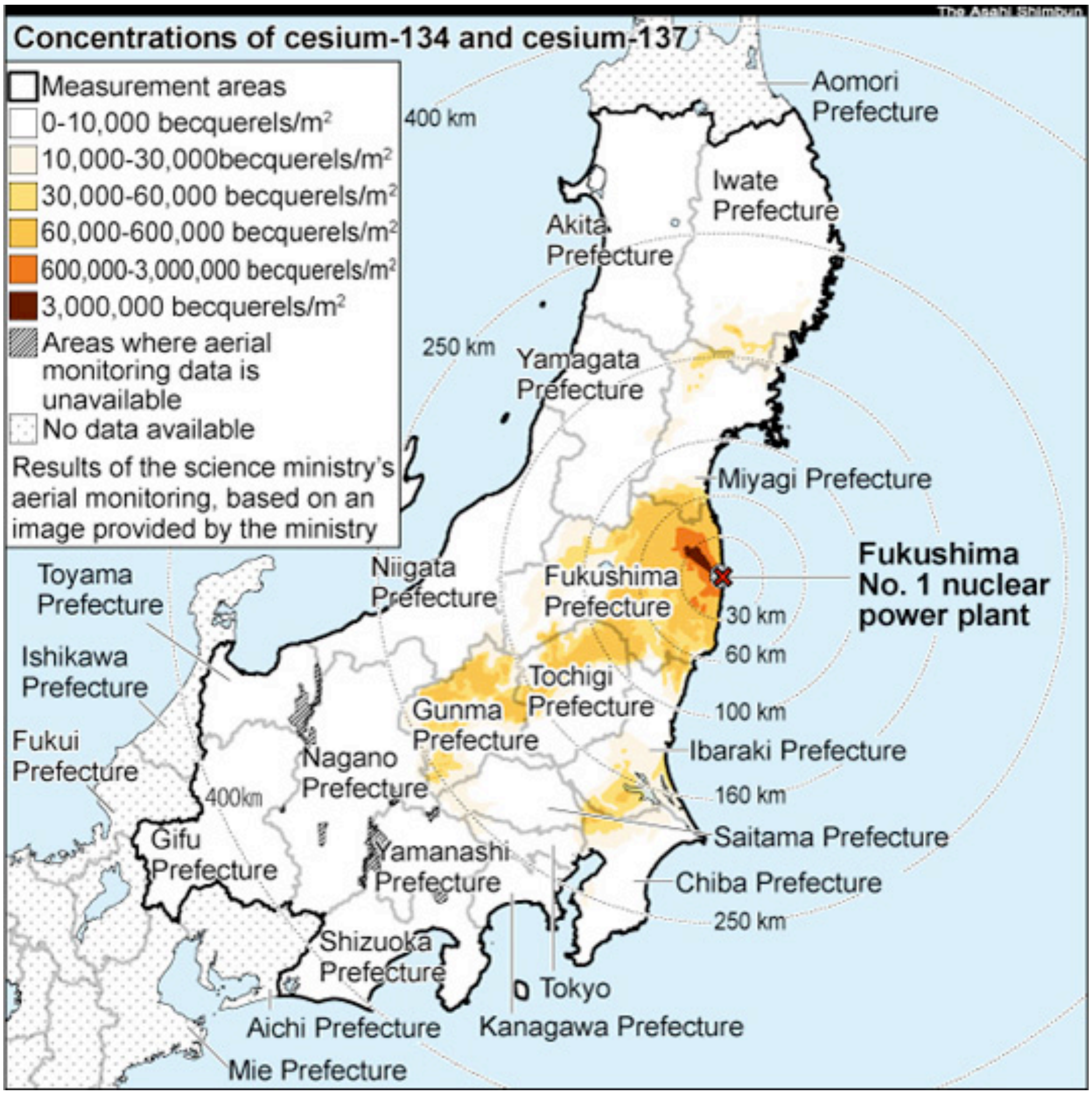
Note:
This figure is adapted from Figure 9.5 of: NRC, 1990.

Figure I.3-1
Unit 4 at the Fukushima #1 Site During the 2011 Accident



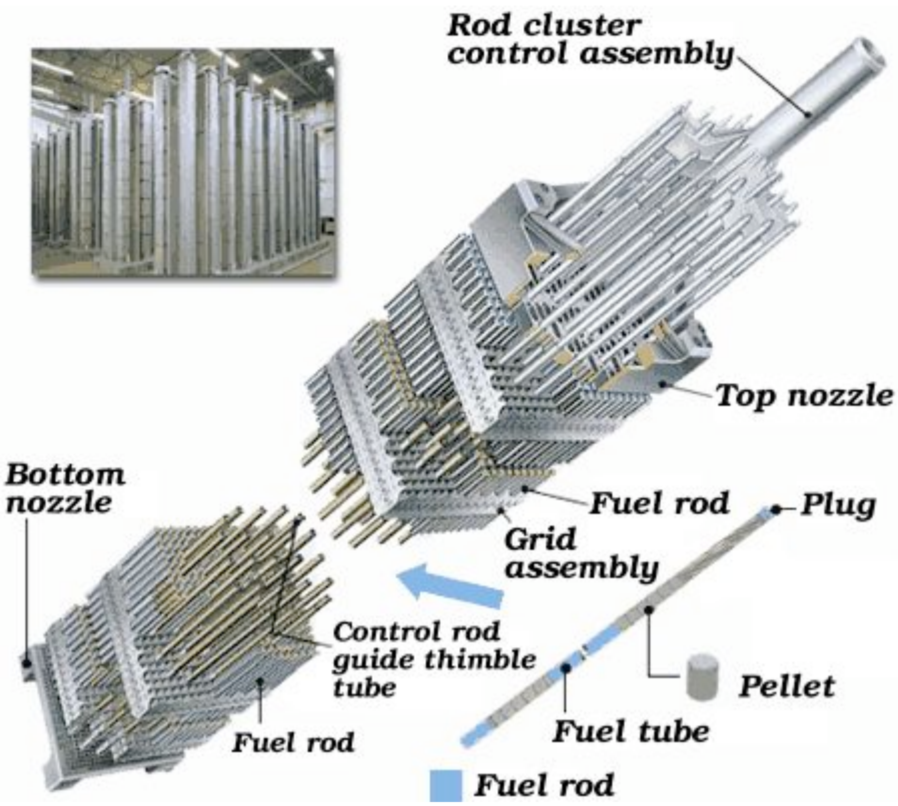
Source:
Accessed on 20 February 2012 from Ria Novosti at:
http://en.rian.ru/analysis/20110426/163701909.html; image by Reuters Air Photo Service.

Figure I.3-2
Contamination of Land in Japan by Radioactive Cesium Released to Atmosphere During the Fukushima #1 Accident of 2011



Source:
Asahi Shimbun, 2011.

Figure I.4-1
Schematic View of a PWR Fuel Assembly (Mitsubishi Nuclear Fuel)

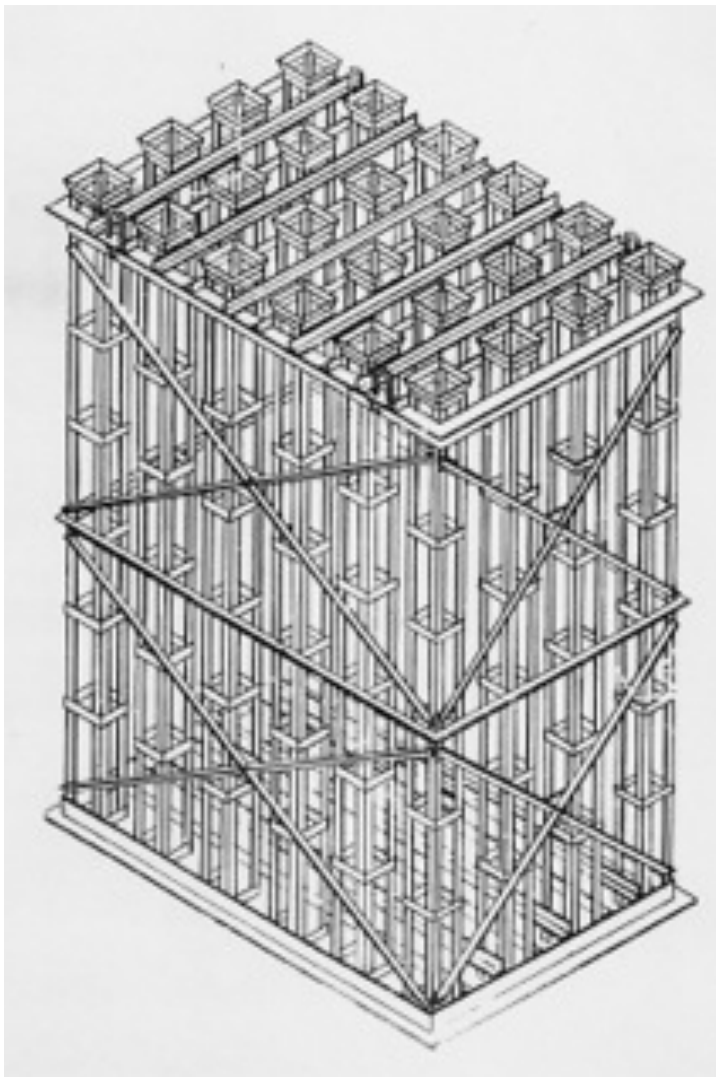


Source:
Accessed on 22 February 2012 from: http://www.world-nuclear.org/info/nuclear_fuel_fabrication-inf127.html

Figure I.4-2
Schematic View of BWR Fuel Assemblies (General Electric)

Source:
Accessed on 22 February 2012 from: http://www.world-nuclear.org/info/nuclear_fuel_fabrication-inf127.html

Figure I.4-3
Typical Low-Density, Open-Frame Rack for Pool Storage of PWR Spent Fuel

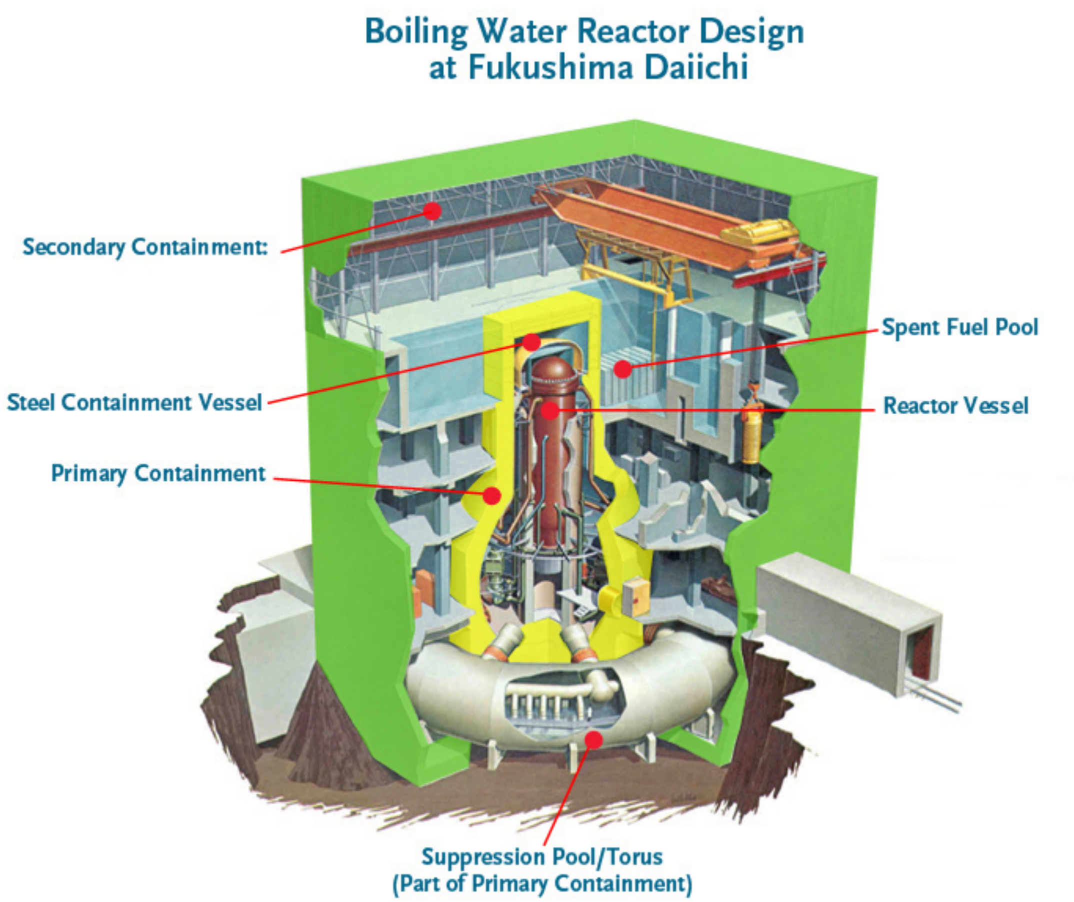


Source:
Adapted from Figure B.2 of: NRC, 1979.

Figure I.4-4
February 2012 View of Spent Fuel in the Unit 4 Pool at Fukushima #1

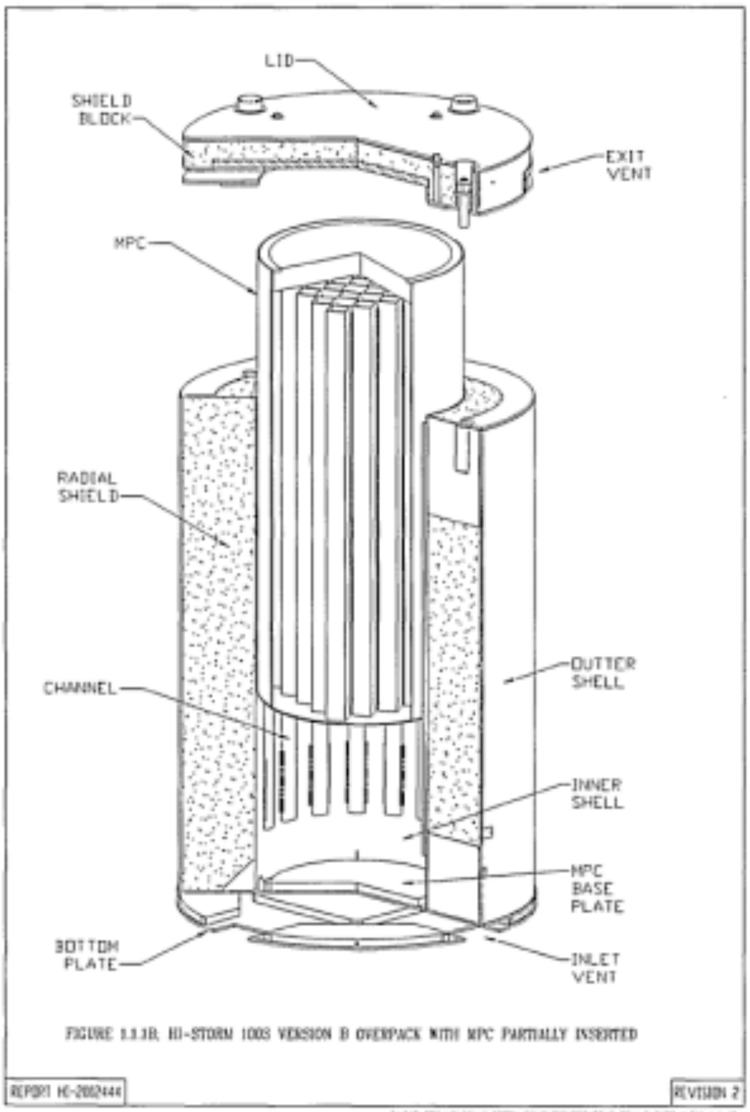
Notes:
(a) This figure is from: Asahi Shimbun, 2012.
(b) The figure is from video footage taken by TEPCO on 9 February 2012
(c) The storage configuration shown here is a high-density, closed-frame rack.
(d) A variety of debris, such as that shown in the figure, is distributed across the pool.

Figure I.4-5
Schematic View of a BWR Reactor with a Mark I Containment, as Used at the Fukushima #1 Site and Elsewhere



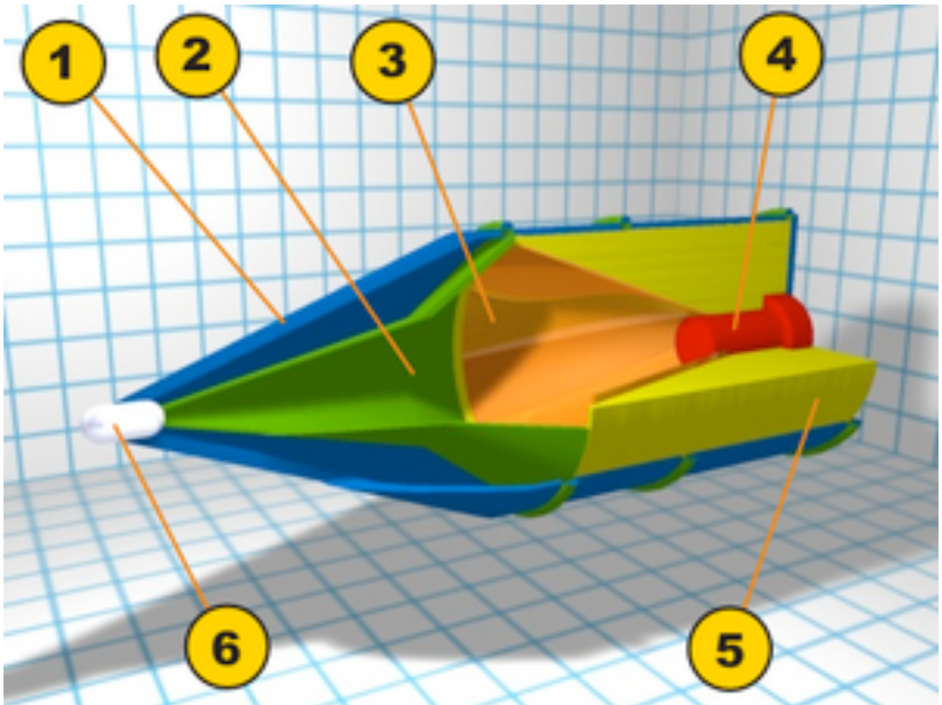
Notes:
(a) This figure accessed on 24 February 2012 from:
http://safetyfirst.nei.org/japan/background-on-fukushima-situation/
(b) All BWR reactors with Mark I containments have the same basic configuration. Details vary for specific reactors.

Figure I.4-6
Schematic View of Dry Cask for Storing PWR or BWR Spent Fuel (Holtec HI-STORM 100 Cask System)



Source:
Accessed on 28 February 2012 from: http://www.nrc.gov/reading-rm/sensitive-info/faq.html

Figure II.3-1
Schematic View of a Generic Shaped-Charge Warhead



Notes:
(a) Figure accessed on 4 March 2012 from: http://en.wikipedia.org/wiki/Shaped_charge
(b) Key:

Item 1: Aerodynamic cover
Item 2: Empty cavity
Item 3: Conical liner (typically made of ductile metal)
Item 4: Detonator
Item 5: Explosive
Item 6: Piezo-electric trigger

(c) Upon detonation, a portion of the conical liner would be formed into a high-velocity jet directed toward the target. The remainder of the liner would form a slower-moving slug of material.

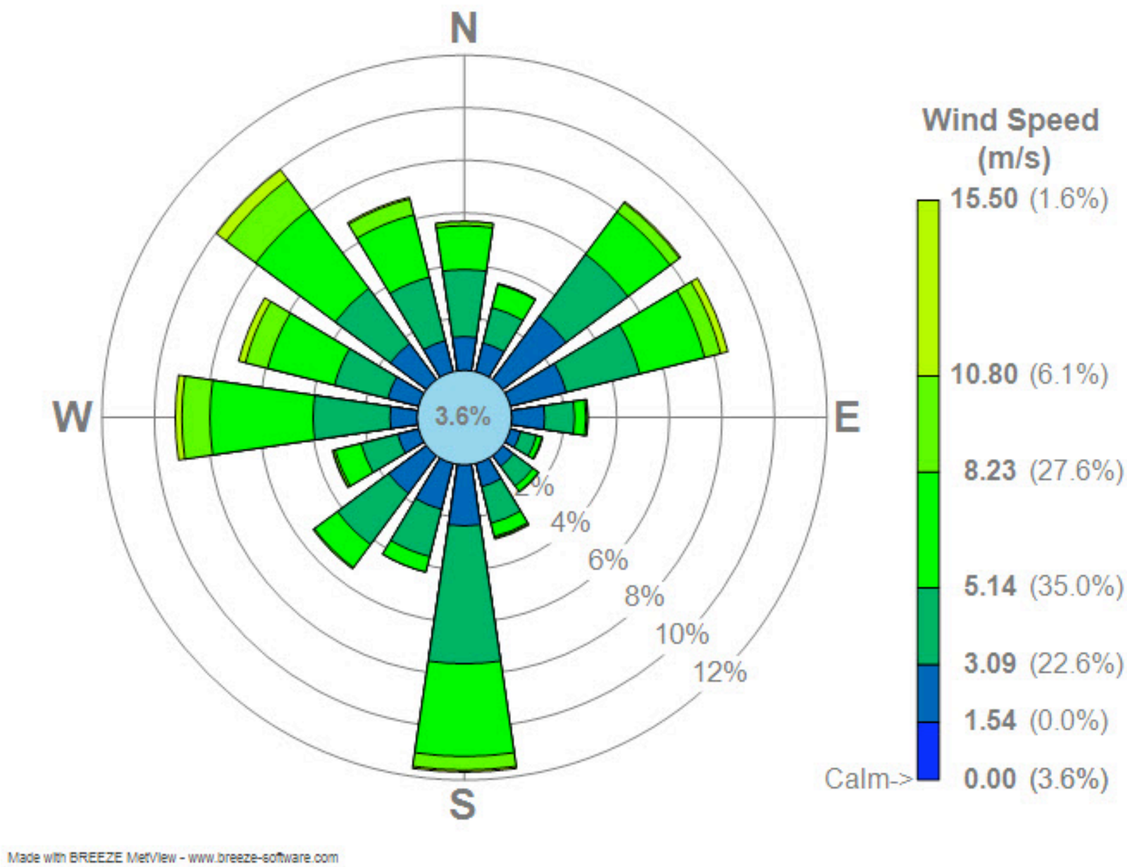
Figure II.3-2
MISTEL System for Aircraft Delivery of a Shaped Charge, World War II

Notes:
(a) Photo accessed on 5 March 2012 from:
http://www.historyofwar.org/Pictures/pictures_Ju_88_mistel.html
(b) A shaped-charge warhead can be seen at the nose of the lower (converted bomber) aircraft, replacing the cockpit. The aerodynamic cover in front of the warhead would have a contact fuse at its tip, to detonate the shaped charge at the appropriate standoff distance.
(c) A human pilot in the upper (fighter) aircraft would control the entire rig, and would point it toward the target. Then, the upper aircraft would separate and move away, and the lower aircraft would be guided to the target by an autopilot.

Figure II.3-3
Outcome of Test Burn of a BWR Fuel Assembly

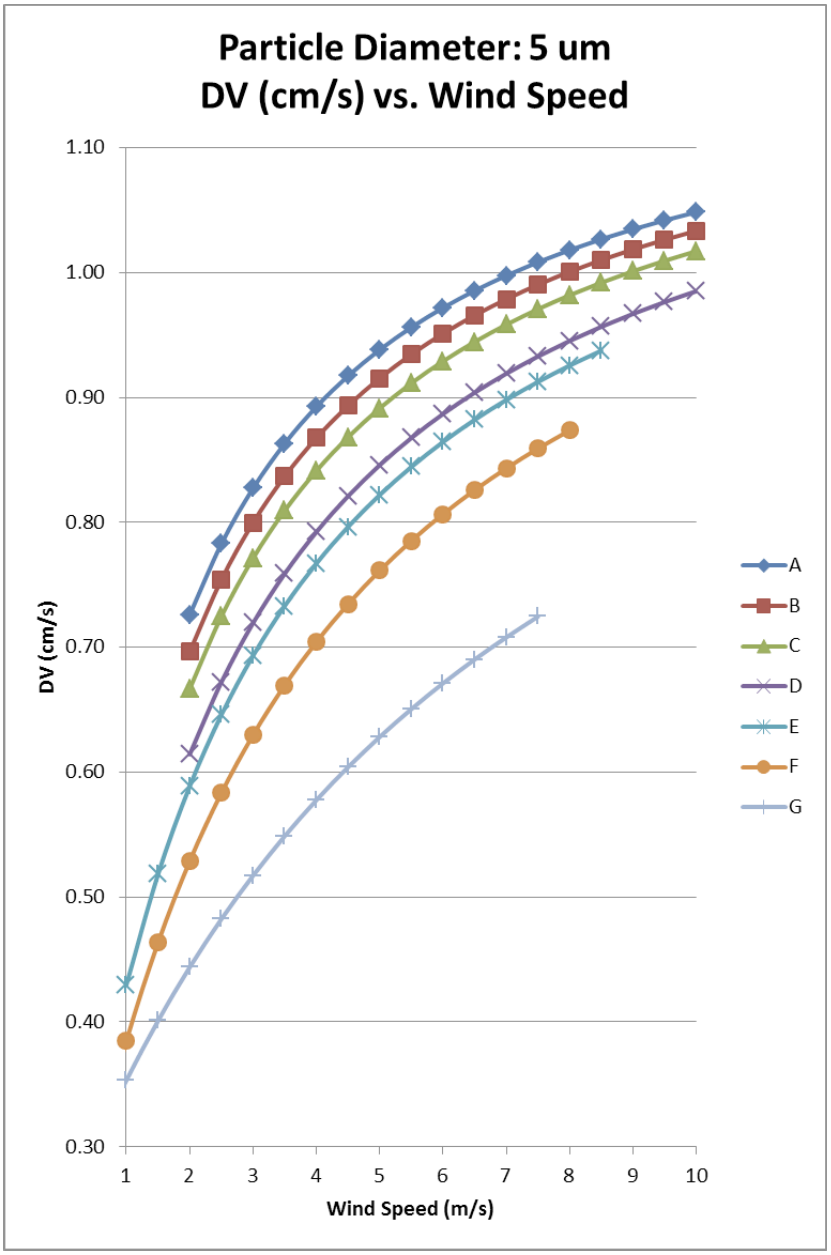
Notes:
(a) This figure is from: Weber, 2011.
(b) The figure shows the outcome of a test to investigate the burning of SNF. An inactive 9x9 BWR fuel assembly with zircaloy-2 cladding was burned in air. The assembly was at reactor scale although not all rods were full length. The assembly was electrically heated (via 74 electric heater rods) at a rate of 5 kW.
(c) The fuel assembly was surrounded by thermal insulation – the white material in the photograph.
(d) This test did not attempt to simulate the release of Cesium or other materials from the damaged fuel.

Figure II.4-1
A Typical Wind Rose (La Guardia Airport, New York, 2008)



Notes:
(a) Figure accessed on 17 August 2012 from: http://en.wikipedia.org/wiki/Wind_rose
(b) The rose shows the average frequency of wind blowing **from** each direction class, and the average frequencies of wind-speed categories within each direction class.
(c) The bar shows the average frequencies of wind-speed categories aggregated across all direction classes.

Figure II.4-2
Selected Values of Dry Deposition Velocity (DV) as Estimated by the GENII Codes



Notes:
(a) This figure is from: Rishel and Napier, 2012.
(b) The letters A through G refer to Pasquill-Gifford (Pasquill) categories of atmospheric stability.

Appendix:

Options to Reduce the Radiological Risk
Posed by a Nuclear Facility

App.1: Introduction

The present level of radiological risk posed by commercial nuclear facilities is not inevitable. Instead, this level of risk reflects choices made by the nuclear industry and accepted by regulatory organizations. The most significant choices relate to facility design, and the designs are strongly influenced by two factors. First, cost minimization is a major driver of the initial design decisions. Second, the nuclear industry is very reluctant to revisit those decisions at a later time, even if evidence accumulates that the initial designs were deficient.[107]

This Appendix discusses options to reduce the radiological risk posed by SNF systems and NPPs, with a primary focus on design options. Section App.2 discusses options to reduce SNF radiological risk at existing facilities, and Section App.3 discusses design options for a new NPP. The design options discussed in Section App.3 could reduce the radiological risk posed by reactors and by SNF stored adjacent to those reactors.

App.2: Options to Reduce SNF Radiological Risk at Existing Facilities

Table App-1 describes some options to reduce the risk of a fire in a spent-fuel pool at an NPP. One can see that the option of re-equipping the pool with low-density, open-frame racks would be the most effective and reliable method of reducing risk. This would be a design option that requires no alteration in the operation of the reactor. Excess spent fuel could be transferred to dry casks located at the plant site or elsewhere.

The cost of introducing this option would be modest. Table App-2 provides a cost estimate for a PWR plant that would now be commencing operation. The plant's pool would be equipped with low-density racks, and spent fuel would be transferred to dry casks after 5 years of storage in the pool. The incremental cost would be 0.04 US cent per kWh of power produced by the plant, beginning in the 11th year of plant operation.[108] If the plant were already in operation, the cost would be greater because previously-accumulated fuel would have to be transferred to dry casks. However, in either case, the true incremental cost, summed over the plant's lifetime, would typically be much smaller. Given the present worldwide trends in SNF reprocessing and development of waste

[107] Specific instances are discussed in: Ford, 1982.
[108] This incremental cost could be an over-estimate, because (i) low-density racks would be cheaper than high-density racks; and (ii) transfer of SNF to dry storage could be cheaper than the assumed rate of US$200 per kg U.

repositories, it is likely that SNF will remain at the sites of many NPPs after these plants are shut down. Given that outcome at an NPP, the SNF in the plant's pool would typically be transferred to dry storage soon after shut-down. Thus, the true incremental cost of transferring SNF to dry storage now, rather than after plant shut-down, would be the time value of the transfer expense.

NRC established a task force of staff members to study the Fukushima #1 accident and make recommendations about incorporating lessons from the accident into NRC regulation. The task force reported in July 2011. Some of its recommendations were intended to reduce the risk of a pool fire. For example, the task force recommended that each NPP owner be required to install fixed pipes that could spray water into each reactor-adjacent pool, with a ground-level connection so that a portable pump could feed water to the pipes.[109] (Table App-1 discusses this option.) The task force did not, however, recommend that the pools be re-equipped with low-density racks.

Dry casks pose a much lower radiological risk than do spent-fuel pools, especially if the pools are equipped with high-density racks. Nevertheless, dry casks could be attacked, and a non-State group could initiate a cask fire as discussed in Section II.3, above. In recognition of the potential for attack, analysts have proposed that dry casks be given additional protection. For example, a researcher at Tokyo University has discussed options for underground placement of dry casks.[110]

Holtec has developed a design for a vertical-axis, dry-cask system in which, for most of its height, the cask would be below ground. The system is known as the HI-STORM 100U, and is a variant of the system shown in Figure I.4-6. Holtec has described the robustness of the 100U system as follows:[111]

> "Release of radioactivity from the HI-STORM 100U by any mechanical means (crashing aircraft, missile, etc.) is virtually impossible. The only access path into the cavity for a missile is vertically downward, which is guarded by an arched, concrete-fortified steel lid weighing in excess of 10 tons. The lid design, at present configured to easily thwart a crashing aircraft, can be further buttressed to withstand more severe battlefield weapons, if required in the future for homeland security considerations. The lid is engineered to be conveniently replaceable by a later model, if the potency of threat is deemed to escalate to levels that are considered non-credible today."

[109] NRC, 2011, Appendix A.
[110] Choi, 2010.
[111] Holtec, 2007. Also, see: Holtec, 2012.

App.3: Design Options for a New NPP

The most reliable option for reducing the risk of an unplanned release of radioactive material from a nuclear power plant would be to design the plant according to highly stringent criteria of safety and security. During the 1970s and 1980s, some plant vendors and other stakeholders sought to develop designs that could meet such criteria. One design approach was to provide a highly robust containment – which might be an underground cavity – to separate nuclear fuel from the environment. Another approach was to incorporate principles of "inherent" or "intrinsic" safety into the design. The two approaches could be complementary.

Underground siting

In the 1970s, there were several studies on constructing NPPs underground.[112] Those studies are exemplified by a report published in 1972 under the auspices of the California Institute of Technology (Caltech).[113] The report identified a number of advantages of underground siting. Those advantages included highly-effective confinement of radioactive material in the event of a core-damage accident, isolation from falling objects such as aircraft, and protection against malevolent acts. Based on experience with underground testing of nuclear weapons, the report concluded that an appropriately designed plant would provide essentially complete containment of the radioactive material liberated from a reactor core during a core-damage event.

The Caltech report described a preliminary design study for underground construction of an LWR power plant with a capacity of 1,000 MWe. The minimum depth of the underground cavities containing the plant components would be 46 to 76 m. The estimated cost penalty for underground siting would be less than 10 percent of the total plant cost.

In an appendix, the Caltech report described four underground nuclear reactors that had been constructed and operated in Europe. Three of those reactors supplied steam to turbo-generators, above or below ground. The largest of those reactors and its above-ground turbo-generator made up the Chooz plant in France, which had a capacity of 270 MWe. In describing the European reactors, the report noted:[114]

> "The motivation for undergrounding the plant appears to be insurance of containment of accidentally released radioactivity and also physical protection from damage due to hostile military action."

Since the 1970s, underground siting of NPPs has been considered by various groups. For example, in 2002 a workshop was held under the auspices of the University of Illinois to discuss a proposed US-wide "supergrid". That grid would transmit electricity – via

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[112] Buchhardt, 1980.
[113] Watson et al, 1972.
[114] Watson et al, 1972, Appendix I.

superconducting DC cables – and liquid hydrogen, which would provide cooling to the DC cables and be distributed as fuel. Much of the energy fed to the grid would be supplied by nuclear power plants, which could be constructed underground. Motives for placing those plants underground would include "reduced vulnerability to attack by nature, man or weather" and "real and perceived reduced public exposure to real or hypothetical accidents".[115]

The PIUS reactor

In the 1980s the reactor vendor ASEA-Atom developed a preliminary design for an "intrinsically safe" commercial reactor known as the Process Inherent Ultimate Safety (PIUS) reactor. An ASEA-Atom official described the company's motives for developing the reactor as follows:[116]

> "The basic designs of today's light water reactors evolved during the 1950s when there was much less emphasis on safety. Those basic designs held certain risks, and the control of those risks led to an increasing proliferation of add-on systems and equipment ending up in the present complex plant designs, the safety of which is nevertheless being questioned. Rather than to continue into this 'blind alley', it is now time to design a truly 'forgiving' light water reactor in which ultimate safety is embodied in the primary heat extraction process itself rather than achieved by add-on systems that have to be activated in emergencies. With such a design, system safety would be completely independent of operator actions and immune to malicious human intervention."

The central goal of the PIUS design was to preserve fuel integrity "under all conceivable conditions". That goal translated to a design specification of "complete protection against core melting or overheating in case of:

- any credible equipment failures;
- natural events, such as earthquakes and tornadoes;
- reasonably credible operator mistakes; and
- combinations of the above;

<u>and</u> against:

- inside sabotage by plant personnel, completely knowledgeable of reactor design (this can be considered an envelope covering all possible mistakes);
- terrorist attacks in collaboration with insiders;
- military attack (e.g., by aircraft with 'off-the-shelf' non-nuclear weapons); and
- abandonment of the plant by the operating personnel".[117]

[115] Overbye et al, 2002.
[116] Hannerz, 1983, pp 1-2.
[117] Hannerz, 1983, page 3.

To meet those requirements, ASEA-Atom designed a light-water reactor – the PIUS reactor – with novel features. The reactor pressure vessel would contain sufficient water to cool the core for at least one week after reactor shut-down. Most of that water would contain dissolved boron, so that its entry into the core would inherently shut down the reactor. The borated water would not enter the core during normal operation, but would enter through inherent mechanisms during off-normal conditions. The reactor pressure vessel would be made of pre-stressed concrete with a thickness of 7.6 m. That vessel could withstand an attack using 450-kg bombs. About two-thirds of the vessel would be below ground.

ASEA-Atom estimated that the construction cost of a four-unit PIUS station with a total capacity of 2,000 MWe would be about the same as the cost of a station equipped with two 1,000 MWe "conventional" light-water reactors. The PIUS station could be constructed more rapidly, which would offset its slightly lower thermal efficiency. Thus, the total generating cost would be about the same for the two stations. ASEA-Atom estimated (in 1983) that the first commercial PIUS plant could enter service in the early 1990s, if a market existed.[118] To date, no PIUS plant has been ordered.

PRIME reactors

In 1991, a study conducted at Oak Ridge National Laboratory examined various types of commercial nuclear reactor that were under development at the time.[119] Some types of reactor represented a comparatively small evolutionary step from existing reactors. Their safety systems tended to be simpler, and to rely more on passive mechanisms, than the safety systems of existing reactors. Other types of reactor were said to have PRIME characteristics. That acronym applied to designs with the features:

- Passive safety systems;
- Resilient safety systems;
- Inherent safety characteristics (no need for safety systems);
- Malevolence resistance; and
- Extended safety (remaining in a safe state for an extended period after an accident or attack).

The Oak Ridge study identified several types of reactor as being in the PRIME category. Those reactors, which were in various stages of development, were: the PIUS reactor; the ISER reactor being developed in Japan; the Advanced CANDU Project; modular, high-temperature, gas-cooled reactors being developed in the USA and Germany; and a molten-salt reactor being developed jointly by the USSR and the USA. The Oak Ridge study did not set forth a framework of indicators and criteria that could be used to assess the comparative merits of those reactors, or to determine if a reactor belonged in the PRIME category.

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[118] Hannerz, 1983, pp 73-76.
[119] Forsberg and Reich, 1991.

Design criteria for substantial reduction of risk

Table App-3 sets forth criteria for designing and siting a nuclear power plant that would pose a risk of unplanned release that is substantially lower than the risk posed by the Generation II plants that are now in use worldwide, and by the Generation III plants that vendors are currently offering. These criteria are similar to ASEA-Atom's design specification for the PIUS plant. Thus, there is evidence that the criteria set forth in Table App-3 are achievable. If ASEA-Atom's cost projections were accurate, there would be no overall cost premium for complying with such criteria.

Table App-1
Selected Options to Reduce the Risk of a Pool Fire at a PWR or BWR Plant

Option	Passive or Active?	Does Option Address Fire Scenarios Arising From: Attack?	Other Events?	Comments
Re-equip pool with low-density, open-frame racks	Passive	Yes	Yes	• Would substantially reduce pool inventory of radioactive material • Would prevent auto-ignition of fuel in almost all cases
Install emergency water sprays above pool	Active	Yes	Yes	• Spray system must be highly robust • Spraying water on overheated fuel could feed Zr-steam reaction • Pool overflow could disable reactor safety systems (especially at BWRs with Mark I and II containments)
Mix hotter (younger) and colder (older) fuel in pool	Passive	Yes	Yes	• Could delay or prevent auto-ignition in some cases • Would be ineffective if debris or residual water blocks air flow • Could promote fire propagation to older fuel
Minimize movement of spent-fuel cask over pool	Active	No (Most cases)	Yes	• Could conflict with adoption of low-density, open-frame racks
Deploy air-defense system (e.g., Sentinel and Phalanx) at site	Active	Yes	No	• Implementation would require presence of military personnel at site
Develop enhanced onsite capability for damage control	Active	Yes	Yes	• Would require new equipment, staff and training • Personnel must function in extreme environments

Table App-2
Estimation of Incremental Cost if Spent Fuel from a Newly Operational PWR is Transferred to Dry Casks After 5 Years of Storage in the Reactor-Adjacent Pool

Estimation Step	Estimate
Average period of use of a fuel assembly in the reactor core	5.4 years
Period of storage of a spent-fuel assembly in the spent-fuel pool, prior to transfer to dry storage	5 years
Point in plant history when transfer of spent fuel to dry storage begins	11^{th} year of plant operation
Average annual transfer of spent fuel from pool to dry storage	36 fuel assemblies
Capital cost of transferring spent fuel from pool to dry storage (assuming a dry-storage cost of US$200 per kg U, and a mass of 450 kg U per fuel assembly)	US$3.2 million per year
Capital cost of transferring spent fuel from pool to dry storage (assuming a plant capacity of 1.08 GWe, and a capacity factor of 0.9)	0.04 US cent per kWh of nuclear generation

Notes:
(a) This calculation employs NPP data that apply to Indian Point Unit 2 in the USA, which typifies PWR plants.
(b) This table is adapted from Table 8-3 of: Thompson, 2009.
(c) The capital cost of transfer to dry storage begins in the 11^{th} year of plant operation, and continues while the plant operates.

Table App-3
Criteria for Design and Siting of an NPP that Would Pose a Radiological Risk Substantially Lower than is Posed by Generation II or III NPPs

Application of Criteria	Criteria
Safety performance of the plant during reactor operation (design-basis criteria)	<u>No significant damage of the reactor core or adjacent stored spent fuel in the event of</u>: • Loss of all electrical power (AC & DC), compressed air, other power sources, and normal heat sinks for an extended period (e.g., 1 week); • Abandonment of the plant by operating personnel for an extended period (e.g., 1 week); • Takeover of the plant by hostile, knowledgeable persons who are equipped with specified explosive devices, for a specified period (e.g., 8 hours); • Military attack by specified means (e.g., 500-kg air-dropped bombs); • An extreme, specified earthquake; • Conceivable erroneous operator actions that could be accomplished in a specified period (e.g., 8 hours); or • Any combination of the above.
Safety performance of the plant during reactor refueling (design-basis criteria)	<u>A specified maximum release of radioactive material to the accessible environment in the event of</u>: • Loss of reactor coolant at a specified time after reactor shut-down, with replacement of the coolant by fluid (e.g., air, steam, or unborated water) creating the chemical and nuclear reactivity that would maximize the release of radioactive material, at a time when the plant's containment is most compromised; and • Any combination of the events specified above, in the context of reactor operation.
Site specification (radiological-impact criteria)	<u>In the event of the maximum release of radioactive material specified above, in the context of reactor refueling, radiological impacts would not exceed specified values regarding</u>: • Individual dose; • Population dose; and • Land areas in various usage categories that would be contaminated above specified levels.

Notes:
(a) The criteria in the first two rows of this table would apply to the reactor core and to spent fuel stored adjacent to the reactor. Separate criteria would apply to an independent facility for storing spent fuel, whether onsite or offsite.
(b) For a more detailed discussion, see: Thompson, 2008, Section 4.3.