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NUREG-2122

Glossary of Risk-Related Terms in Support of Risk-Informed Decisionmaking

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Prepared by:
M. Drouin[1], M. Gonzalez[1],
S. Herrick[1], J. S. Hyslop[1], D. Stroup[1]
J. Lehner[2], T. Pratt[2]
M. Dennis[3], J. LaChance[3], T. Wheeler[3]

[1]U.S. Nuclear Regulatory Commission
Washington, DC 20555

[2]Brookhaven National Laboratory
Upton, NY 11973

[3]Sandia National Laboratories
Albuquerque, NM 87185

M. Drouin, NRC Project Manager

Office of Nuclear Regulatory Research

ABSTRACT

The policy statement on the "Use of Probabilistic Risk Assessment (PRA) Methods in Nuclear Regulatory Activities" (Ref. 31) expressed the Commission's belief that the use of PRA technology in U.S. Nuclear Regulatory Commission (NRC) regulatory activities should be increased. Consequently, the NRC carried out numerous risk-informed activities in all areas of NRC regulation. With increased risk-informed activities came the recognition that the agency could enhance regulatory stability and efficiency if it implemented the many potential applications of risk information in a consistent and predictable manner. An essential part of consistent and predictable implementation is the use of consistent terminology to ensure accurate communication and transfer of information. Further, the NRC recognizes that some risk-related terms have been used in ambiguous ways by practitioners. The increased development of guidance documents, regulations, and procedures related to risk-informed activities makes the fundamental understanding of these risk-related terms more imperative. Consistent terminology is essential to the appropriate implementation of risk-informed activities and the communication between the NRC and its stakeholders. It allows practitioners to eliminate communication issues and avoid unnecessary discussions that may have been erroneously perceived as technical issues. Therefore, a glossary with agreed-upon definitions of risk-informed related terms is an essential tool for future risk-informed activities. This glossary addresses risk-related terms used in the context of risk associated with a reactor of a nuclear power plant.

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The final policy statement on the "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Ref. 33) expressed the U.S. Nuclear Regulatory Commission's (NRC's) belief that the use of probabilistic risk assessment (PRA) technology in NRC regulatory activities should be increased. Since the PRA policy statement, the staff has issued several PRA or risk-informed plans detailing various risk-informed activities.

With increased risk-informed activities comes the recognition that regulatory stability and efficiency would be enhanced if the various risk-information activities are implemented consistently and predictably. An essential part of implementation is the use of consistent terminology to ensure a common understanding of information. A common understanding of information provides increased assurance that the analyses being performed are technically adequate to facilitate better risk-informed decisionmaking.

A glossary with definitions of risk-informed-related terms is an essential tool for risk-informed activities. A glossary provides clarity on the meaning of many terms. For terms that are context or scope dependent, a single definition may not be appropriate, but a discussion on the use of these terms in different contexts will be helpful.

This NUREG report identifies and defines terms used in risk-informed activities related to commercial nuclear power plants. It provides a single source in which these terms can be found. A major goal of the glossary is to reduce ambiguity in the definition of terms as much as possible, so that a common understanding can be achieved that will facilitate communication on risk-informed activities.

ACKNOWLEDGMENTS

Numerous individuals contributed to the development of this glossary. The majority of the definitions are based on definitions from numerous sources (see Section 5 for a list of references); as such, the authors do not claim sole authorship. However, it was an immense and challenging effort to perform the necessary research, identify the terms to be included in the glossary, develop definitions understandable to individuals regardless of their level of risk expertise and experience, and provide the necessary discussion on the usage of terms. Therefore, the following acknowledgements are made:

NRC Project Manager and Technical Lead: Mary Drouin

Main contributors or authors by organizations include:

U.S. Nuclear Regulatory Commission
- Mary Drouin
- Michelle Gonzalez
- J.S. Hyslop
- Sandra Herrick
- David Stroup

Brookhaven National Laboratory
- John Lehner
- Trevor Pratt

Sandia National Laboratories
- Matthew Dennis
- Jeffery LaChance
- Timothy Wheeler

Energy Math Solutions
- Zoran Musicki

OMICRON Safety and Risk Technologies
- Pier Carrara
- Willard Thomas

ACRONYMS AND ABBREVIATIONS

AOOs	anticipated operational occurrences
ATD	atmospheric transport and diffusion
BDBAs	beyond-design-basis accidents
BDBEs	beyond-design-basis events
ECCS	emergency core cooling system
FTR	fails to run
FTS	fails to start
IAEA	International Atomic Energy Agency
IPEEE	individual plant examinations for external events
IPEs	individual plant examinations
LOCCW	loss of the component cooling water
LWR	light-water reactor
NRC	U.S. Nuclear Regulatory Commission
POS	plant operating states
PRA IP	Probabilistic Risk Assessment Implementation Plan
PRA	probabilistic risk assessment
RCPs	reactor coolant pumps
RG	regulatory guide
RIRIP	Risk-Informed Regulatory Implementation Plan
RPP	Risk-Informed Performance-Based Plan
RPS	reactor protection system
SSC	structures, systems, and components
SSE	safe-shutdown earthquake

1. INTRODUCTION

1.1 Background

The final policy statement on the "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (Ref. 33) expressed the U.S. Nuclear Regulatory Commission's (NRC's) belief that the use of probabilistic risk assessment (PRA) technology in NRC regulatory activities should be increased. Since the PRA policy statement, the staff has issued several PRA or risk-informed plans detailing various risk-informed activities. The NRC used the first plan, the PRA Implementation Plan (PRA IP), until 1999. This plan identified the initial risk-informed activities undertaken as a result of the PRA policy statement. As the use of risk information in regulation increased further, the NRC replaced the PRA IP with the Risk-Informed Regulation Implementation Plan (RIRIP) in 2000. This plan reflected the increased sophistication and experience in the use of risk assessment methods that included not just PRA, but also integrated safety assessments and other risk-related techniques. The RIRIP was improved and became the Risk-Informed and Performance-Based Plan (RPP), submitted with SECY-06-0217, "Improvement to and Update of the Risk-Informed Regulation Implementation Plan," dated October 25, 2006 (Ref. 99). The NRC implemented the RPP in response to a June 2006 Commission-issued staff requirements memorandum, "Briefing on Status of Risk-Informed and Performance-Based Reactor Regulation," dated June 1, 2006, (Ref. 101), which directed the staff to (1) improve the RIRIP so that it is an integrated master plan for activities designed to help the agency achieve the Commission's goal of a holistic, risk-informed and performance-based regulatory structure, and (2) seek ways to communicate more transparently the purpose and use of PRAs in the NRC's reactor regulatory program to the public and stakeholders. The RPP is updated annually.

As these plans indicate, risk information is used in every aspect of the NRC's work (e.g., regulation and guidance, licensing and certification, oversight, and operational experience). Examples of these include the following:

- Regulation and guidance—Recent risk-informed rules include Title 10 of the *Code of Federal Regulations* (10 CFR) 50.44, "Combustible Gas Control for Nuclear Power Reactors" (Ref. 16); 10 CFR 50.48(c), "Fire Protection" (Ref. 17); and 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems, and Component for Nuclear Power Reactors" (Ref. 21).

- Licensing and certification—Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, issued November 2002 (Ref. 84) provides general guidance on what is needed for risk-informed applications for licensing -basis changes, while specific risk-informed guidance is offered in RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," issued August 1998 (Ref. 86); RG 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," issued August 1998 (Ref. 85); and in RG 1.178, "An Approach for Plant-Specific, Risk-Informed Decisionmaking for Inservice Inspection of Piping," issued September 2003 (Ref. 87).

- Plant oversight—the Reactor Oversight Process uses risk-informed performance indicators and inspections, as well as a risk-informed significance determination process, for inspection findings.

- Operational experience—Management Directive 8.3, “NRC Incident Investigation Program,” dated March 27, 2001, (Ref. 43), provides risk-informed incident investigation direction. The Office of Nuclear Reactor Regulation instruction LIC-504 focuses on risk-informed decisionmaking for event assessment, and the Accident Sequence Precursor Program (Ref. 96) calculates a conditional core damage probability to assess event significance.

With increased risk-informed activities comes the recognition that regulatory stability and efficiency would be enhanced if the various applications of risk-information are implemented consistently and predictably. An essential part of implementation is the use of consistent terminology to ensure a common understanding of information. A common understanding of information provides increased assurance that the analyses being performed are technically adequate and, therefore, produce correct results. This assurance, in turn, leads to better risk-informed decisionmaking.

Historically, some risk-related terms have been used somewhat differently by different practitioners, but the increased development of guidance documents, regulations, and procedures makes a common understanding of fundamental terms in these areas more imperative. A common understanding is fundamental to communication in the risk-informed arena for the consistent and appropriate treatment of risk-informed applications by industry as well as risk-informed regulatory actions by the NRC, and for communication between the NRC and its stakeholders. There are a variety of reasons why consistent use of terminology is not always found in the area of risk-informed activities: multiple definitions exist for the same term, terms are used interchangeably when they are not synonymous, or the definition is scope or context dependent.

A glossary of risk terms is needed for appropriate interpretation of terminology to support a risk-informed regulatory structure (e.g., licensee applications, NRC regulations). A good illustration of this need is the use of the term “internal events.” The term is used to refer to potential events resulting from equipment failures and human errors that can result in a plant disturbance. In some instances, “internal events” has been defined as events occurring within the plant and includes internal fires or internal floods or both; however, in other instances it has been defined as events occurring within the component boundary and does not include internal fires and internal floods. Further, there are instances in which the term has been used without a definition and it is not clear which definition is intended. In the treatment of an application or implementation of a regulation, it is important to know if internal events include internal fires and/or floods, or not.

Therefore, a glossary with definitions of risk-informed-related terms is an essential tool for risk-informed activities. A glossary provides clarity on the meaning of many terms. For terms that are context or scope dependent, a single definition may not be appropriate, but a discussion on the use of these terms in different contexts will be helpful.

1.2 Objective

The objective of this glossary is to identify and define terms used in risk-informed activities related to commercial nuclear power plants. This glossary provides a single source in which these terms can be found. A major goal of the glossary is also to reduce ambiguity in the definition of terms as much as possible, so that a common understanding can be achieved that will facilitate communication on risk-informed activities. Among other things, this glossary will allow individuals to distinguish communication issues—erroneously perceived as technical issues—from actual technical discussions. Where terms are found to have a justifiable variety of definitions, depending on the context in which they are used, the objective of this glossary is to explain the individual definitions, along with the context, to ensure proper context-specific use of the term. Whenever possible, existing definitions are used and redefining terms is avoided.

This NUREG glossary is a supporting document for documents and procedures that include risk terminology. As such, it is a reference for NRC staff, as well as other stakeholders in the risk-informed arena.

1.3 Scope and Limitations

In developing the list of terms to be included in the glossary, it was not possible to identify every applicable source nor was it practical to review every available source. For this glossary, the sources are limited to internal NRC sources (e.g., regulations, NUREG reports, technical reports, regulatory guides, standard review plans, Commission documents) that are risk-related. Except for certain PRA standards and International Atomic Energy Agency (IAEA) documents, industry sources were not included.

This glossary is intended to provide a common understanding of the identified terms for users with different backgrounds, skills, and experience. Therefore, this glossary contains terms that are likely to be familiar even to users with only limited exposure to risk-informed activities (although they may not have a common understanding of the term), as well as terms that only experienced practitioners would use in their work. Consistent with the objective to make this glossary also of use to analysts and regulators inexperienced in the area of risk-informed activities, certain “core” terms that are fundamental to the basic understanding of risk concepts and risk analyses are included.

This glossary covers terms used in all three levels of a light-water reactor (LWR) PRA: core damage frequency analysis (Level 1 PRA), radionuclide release frequency analysis (Level 2 PRA), and consequence analysis and risk integration (Level 3 PRA). It also includes terms used in other quantitative analyses (i.e., terms such as “seismic margin”) and in qualitative risk considerations (i.e., terms such as “dependency”).

Terms that are common across all hazard groups—internal hazards (e.g., internal events, internal floods, and internal fires) and external hazards (e.g., earthquakes, external floods, high winds)—are included. For specific hazards (i.e., internal fires), a hazard-specific glossary is provided. At this time, the only hazard-specific glossary included in this NUREG is one for internal fires.

The terminology in this glossary is meant to cover risk-related terms used for an at-power PRA, as well as for the other plant states (i.e., low power and shutdown). This glossary also addresses risk terms that may come up in the different life stages of a power reactor: design, licensing, operation, and decommissioning.

There are general scientific terms that do not take on an additional meaning for risk-related activities; in general, these terms are not included in this glossary. However, in some cases, some of these terms are included because they are fundamental to understanding the results or insights used in risk-related activities.

The terms defined in this glossary are those used in risk-informed activities related to commercial nuclear power reactors. As such, risk terms related to nonnuclear industries, such as the chemical industry, are not included. In addition, nuclear technology-related risk terms specific to nonpower reactor parts of the industry (e.g., nuclear material and waste, research reactors) also are excluded.

This glossary contains many generic terms used throughout the risk field whose definitions are broadly applicable to areas outside of the area of power reactors. However, some of these generic terms may have different meanings when used in nonreactor or nonnuclear fields; therefore, these other definitions are not included in the glossary.

In the power reactor area, this glossary is meant to be broad and wide-ranging. The terms included are meant to support all risk-informed activities related to LWRs (including advanced LWRs) that produce electric power. However, some risk-related reactor terms may not be included (e.g., terms that are non-LWR specific).

In summary, this NUREG provides a single broad source for terms used in risk-informed activities related to LWR power reactors.

1.4 Approach Summary

A major challenge in developing such a glossary is selecting the terms to include in the glossary. The defined scope provides general boundaries for inclusion in the glossary. Although the focus of the glossary is on terms for risk communication, the list still could be exhaustive. Consequently, guidance was developed to identify and select the terms. The second major challenge is developing the definitions. Providing just a definition for each term would not completely accomplish the glossary's objective. To meet the objective, both a definition of each term and insights on its meaning and use were needed. Consequently, guidance also was developed for developing the definitions. The approach is summarized below and discussed in detail in Section 2.

1.4.1 Identification and Selection of Terms

A major challenge in developing a glossary is in how to identify and select specific terms to include. An initial list was developed that was meant to be as broad as possible to help ensure a term was not prematurely excluded from consideration. The general guidance used to identify this list was that the term should be related to risk communication. For example, if the term is used to communicate what is meant by the term "risk," then the authors considered such issues as what is a risk analysis, what are risk results and insights, and how risk results and insights are used in decisionmaking. Based on this guidance, an initial list of terms was compiled in a two-step process.

Terms were identified by reviewing documents related to or that support risk-informed activities. The types of documents selected for review included PRA standards, NUREGs and technical methodology documents, regulatory guides and standard review plans for risk-informed applications, risk-informed regulations, and Commission documents on risk-informed activities. Section 5 provides the list of sources.

The NRC staff and management also were asked to augment the initial list. Participants in this step included individuals with and without risk expertise and both junior and senior staff.

Although the initial list was meant to be broad, it resulted in a list of more than 1,000 terms. Table 1-1 gives a small sample of this initial list of terms.

Table 1-1 Sample List of Initial Terms for Inclusion in Glossary

Accident Consequences	Containment Failure Mode	External Hazards
Accident Mitigation	Core Cooling Water System	Failure Probability
Assumption	Core Damage	Frequentist
At-Power	Defense-in-Depth	Fussell-Vesely Importance Measure
ATHEANA	Decision Model	Human Error
Atmospheric Transport and Dispersion	Deterministic	Infrequent Event
Authority Having Jurisdiction	Diagnosis	Initiating Event
Backstop	Dominant Accident Sequence	Key Assumption
Basic Event	Dominant Contributor	Living PRA
Bayesian	Dose Conversion Factor	Mean Value
Certified Seismic Design	Dose Response Model	Operational Risk
Circuit Failure Mode	Double Contingency Principle	Performance Shaping Factor
Classical Statistics	Dynamic Probabilistic Risk Assessment	Recovery Action
Common Cause Failure	Dynamic Risk Level	Significant Accident Sequence
Common Mode Failure	Early Containment Failure	Technical Adequacy
Containment Event Tree	Event Scenario	Uncertainty Distribution
	Event Sequence	Zone Of Influence

With such a large list, it was necessary to prioritize the selection of the final list. The authors developed guidelines consistent with the objectives of this project for ensuring that terms were appropriately selected for the glossary.

Terms not relevant to risk-informed activities were excluded. For example, self-evident terms, such as names of organizations, names of structures, systems, and components (SSCs), and units of measure were excluded from the glossary (e.g., auxiliary feedwater system, becquerel).

Terms necessary to understand risk analysis initially were retained as potential candidates for the glossary (i.e., terms needed to understand the term "risk," to understand what constitutes a risk analysis, to understand the different kinds of risk analyses, including their associated terminology, were not excluded). Further, terms that play a role in risk-informed decisionmaking were not necessarily excluded from the list. While a term may not relate to the understanding of risk analysis, the term may often be used or associated with risk-informed decisionmaking. These terms also were retained as potential candidates for inclusion in the glossary.

After the initial screening, although some terms were identified as candidates to be screened from the list, they were, nevertheless, retained. These terms were appropriately identified to be screened from the list because they are not used in the broad definition of risk communication; however, they are related to risk communication for a specific hazard. It was decided that these previously screened terms which were identified as being related to risk communication for a specific hazard should be retained in the glossary. Consequently, it was determined that there should be two types of glossaries:

1. A main glossary that focuses on terms used in communicating risk at a high level.

2. Hazard-specific glossaries that focus on terms used in communicating the details in performing the hazard-specific risk analysis.

The main glossary is documented in Section 4, and the hazard-specific glossaries are documented in the appendices. At this time, only an internal fire hazard-specific glossary has been developed.

Although a term is important to risk communication (whether at a high level or hazard-specific level), there still may be other reasons for excluding the term from the glossary. The term may not have a meaning that is unique in a risk context (i.e., a term's definition may be well established throughout the community without a basis for possible miscommunication). These terms are potential candidates for exclusion from the glossary. However, the term may be fundamental to a basic understanding of risk. Such a fundamental term is not excluded and is retained as a potential candidate for the glossary. For example, "probability" is a term whose definition is well established and does not change when used in a risk context. As such, the term becomes a candidate for exclusion. However, the term "probability" is fundamental in understanding risk, and it is also frequently used in discussing both inputs to a PRA model and results from the PRA. Consequently, "probability" is a term that should not be excluded, but should be retained and included in the glossary.

The last guideline involves determining if the term has policy implications and requires approval from the Commission to issue an officially documented definition. A good example of such a term is "defense-in-depth." This term can be important in risk communication. There are many different understandings of the term "defense-in-depth" and there is no consensus on its meaning. Development of a definition, however, does have policy implications and, therefore, would require Commission approval. These types of terms are included in the glossary; however, a single "official" definition is not developed. The glossary discusses the various definitions that have been used and notes the policy implications.

1.4.2 Development of Definitions

This glossary does not recreate definitions. Consequently, where the definition of the term already exists, and there is consistency among the various sources, that definition is used as the basis for the definition provided in the glossary. However, there are terms with multiple definitions among different sources that are not in agreement. For terms that have multiple (legitimate) definitions, each definition is included in the glossary, and an explanation describing the differences and bases is provided. For terms that have multiple, conflicting definitions, an appropriate definition is selected, and an explanation is provided for the basis for the definition provided.

As noted above, it is not the intent of this glossary to recreate definitions. In some cases, it was determined that (1) there may not be disagreement about the definition, (2) the definition was appropriate, or (3) for multiple definitions, there is consistency. However, it was determined that for some terms, an experienced risk analyst may be needed to understand the definition. In these cases, although the definition from the sources is included, a definition is developed in "plain language" (i.e., a definition is provided that does not rely on technical jargon). The reason the definition is written in plain language is to minimize any misunderstanding of the definitions. Furthermore, plain language helps PRA practitioners, including those who are not native English speakers, to understand the definitions with minimum language barriers. The use of plain language is an NRC policy to improve communication with the general public and other stakeholders.

To help the user, numerous terms are cross-referenced in the glossary. The authors used these cross-references when they thought that to completely understand the term, related terms also needed to be understood. These cross-references are for terms, for example, that meet the following criteria:

- They are similar but have different meanings and there are instances in which they are used incorrectly (e.g., “probability” versus “frequency,” “core damage” versus “core melt”)

- They are the opposite of another term (e.g., “deterministic” and “probabilistic”).

- They are closely related in meaning and may be a subset of the related term or an example of the related term (e.g., “risk analysis” and “probabilistic risk analysis”, “accident consequence” and “health effects”).

- They are closely related in that understanding one term depends on understanding the related term (e.g., “aleatory uncertainty” and “uncertainty,” “core damage” and “core damage frequency” and “core damage probability”).

The glossary also combines terms. Instead of appearing as individual terms, they are defined together in the glossary as a single term or a group of terms. These groups include terms, for example, which are:

- composed of multiple words with the same “adjective” (e.g., “significant,” “significant contributor,” “significant basic event”; “significant accident sequence”)

- similar and convey the same meaning (e.g., “general transient” and “transient”)

- complementary of each other (e.g., “availability” and “unavailability”)

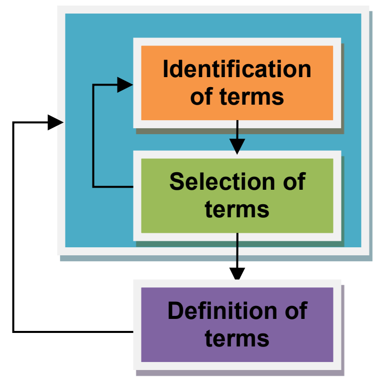
For each term in the glossary, the definition(s) stated in the source documents are collected. Examples of the source documents are the PRA standards, NUREGs associated with risk-informed activities or risk methodologies, tools or data, regulatory guides and standard review plans for risk-informed applications, risk-informed regulations, and Commission documents on risk-informed activities (e.g., PRA policy statement). Section 5 provides a complete list of the source documents.

2. APPROACH

2.1 Introduction

Two major tasks were performed while developing this glossary. The first task was identifying and selecting terms. The second task was developing the actual definitions of the selected terms.

The approach used for these two tasks are discussed in detail in Sections 2.2 and 2.3.



Both the identification and development of the definitions is an iterative process. Additional terms may need to be included in the glossary as terms are identified, selected, and defined, as shown in the illustration at right. However, the steps associated with the guidance are discussed sequentially, although each step in the two tasks also may result in identifying a new term to be included in the glossary.

2.2 Task 1: Identification and Selection of Terms

The list of terms was meant to be as extensive as possible. However, for practical purposes, guidelines were necessary to keep the list to a manageable size. Therefore, the list was developed by identifying potential candidate terms and then selecting the actual terms for the glossary. Guidance consistent with the objectives of this project was developed to ensure that terms were appropriately selected for the glossary.

A set of eight steps was developed with the sole purpose of determining if a term should 1) be considered as a candidate term, and 2) if the candidate term should remain as part of the glossary. The process for identifying and selecting terms for the glossary consisted of the following eight steps:

1-1 ***Initial Potential Candidate*** – A list of candidate terms is first developed by reviewing a set of risk-related documents and interviewing staff. A high-level screening is performed to exclude such terms as "auxiliary feed water."

1-2 ***Used in Risk Communication or Specific Hazard Risk*** – Not all candidate terms from Step 1 are essential to risk communication and perhaps should be excluded from the glossary. However, although a term is not considered essential in communicating risk at a high level, it may be essential in communicating the details of performing a hazard-specific risk analysis (e.g., internal fire). These terms should not be excluded.

1-3 ***Risk Context Specific Definition*** – Terms that have a risk context specific meaning are terms whose meaning may be consistent with the actual definition in a dictionary, but their meaning in risk communication has a risk connotation (e.g., internal events). These terms should be retained as potential candidates.

1-4 ***Availability of Definitions*** – Terms with readily available definitions are terms whose definition can be easily found in technical documents. These terms should be considered for potential exclusion.

1-5 ***Multiple Definitions*** – Terms may have multiple definitions. For some terms, these definitions may be unrelated, some may be legitimate, and others may not be.

1-6 ***Consensually Established Definitions*** – Not all sources necessarily have the same (or similar) definition for a term. However, if there is agreement among the different sources on a term's meaning, then it can be assumed it has been consensually established. These terms should be considered for potential exclusion.

1-7 ***Term Fundamental to Risk Communication*** – Some terms have been identified for potential exclusion, yet they may be essential to a basic understanding of risk-informed activities. These terms should be retained and included in the glossary.

1-8 ***Policy Implications*** – If it is determined that a term has policy implications, its definition may require Commission approval. These terms are identified and a "formal" definition is not developed.

This eight-step process is illustrated in Figure 2-1 and discussed in detail below.

2.2.1 Step 1-1: Initial Potential Candidates

This step identifies the initial list of potential candidates for inclusion in the glossary. This initial list of candidate terms is as broad as possible to ensure that a term is not prematurely excluded.

The guidelines to develop an initial list of terms include:

1. identifying initial list of candidate terms
2. performing a high-level screening

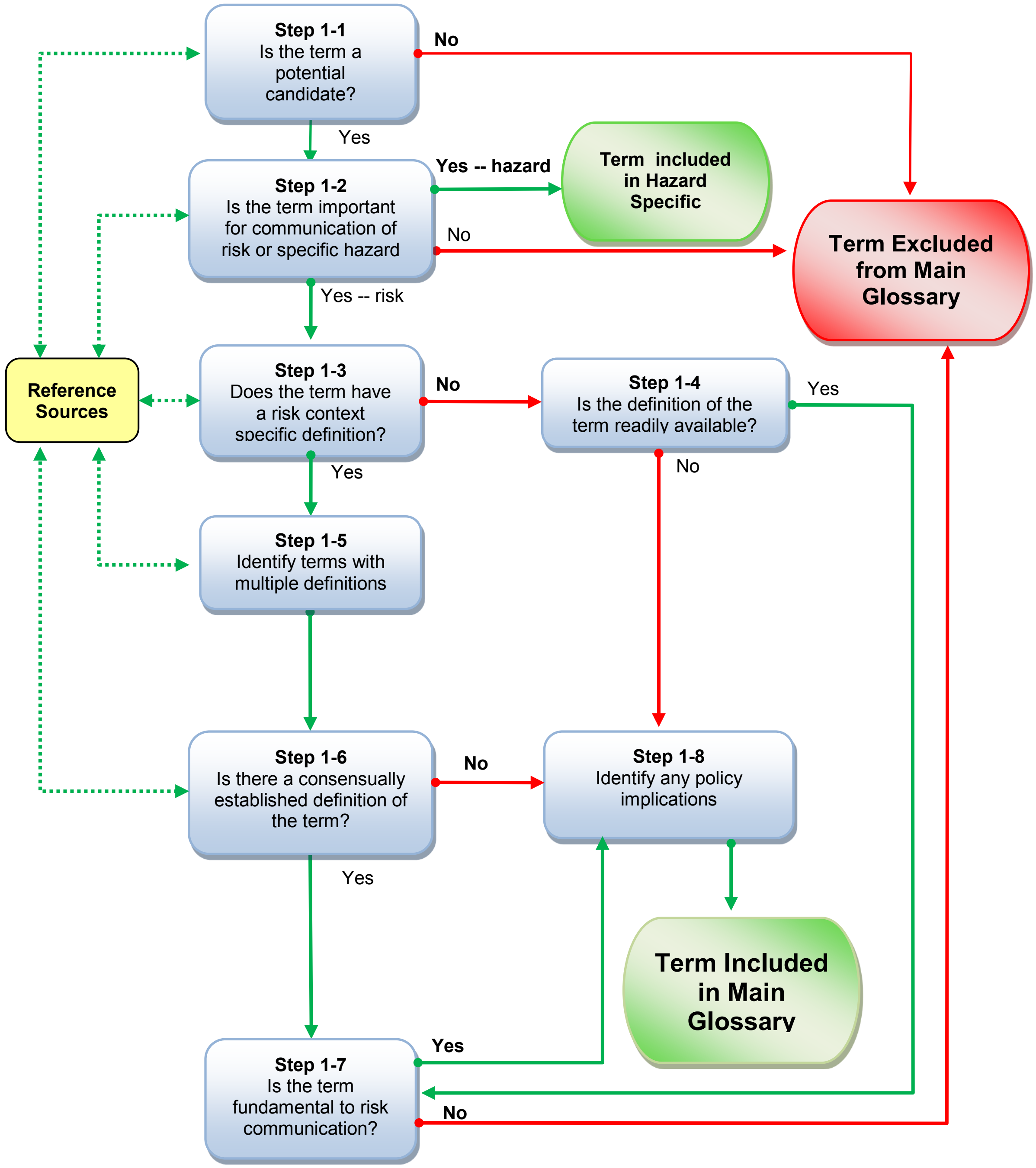


Figure 2-1 Process to identify terms for glossary

Guideline 1-1.1 – Identification of Initial List

This guideline is used to develop an initial list of candidate terms. Terms were identified by reviewing documents related to risk communication. For example, the term may be used to communicate what is meant by the term "risk," what is a risk analysis, what are risk results and insights, or how are risk results and insights used in decisionmaking. The types of documents selected for review included the following:

- PRA standards
- risk-informed related NUREGs
- risk-informed regulatory guides and standard review plans
- risk-informed regulations
- NRC inspection manuals
- Commission documents on risk-informed activities
- selected IAEA documents

The sources reviewed are listed in Section 5.

NRC staff and management also were asked to augment the initial list. Participants included individuals with and without risk expertise and both junior and senior staff. The purpose and objective of the glossary was provided to these participants, along with the guidelines used to identify the initial list. Terms identified by the staff were added to the initial list of terms.

The process discussed above for identifying potential terms resulted in an initial list of more than 1,000 terms. A high-level review indicated many terms that simply do not belong in a "risk communication glossary." Consequently, the second guideline of Step 1-1 involved a high-level screening to identify terms that do not fit the scope of the glossary.

Guideline 1-1.2 – High-Level Screening

This guideline is used to perform a high-level screening. The high-level screening identifies terms that do not have a risk context. These terms are well known and do not need to be defined in the glossary. These types of terms include names of SSCs and operational procedures, units of measure, organizations, chemicals, nuclear safety analyses, and names of computer codes (including acronyms). Examples of terms excluded from the glossary include the following:

- Examples of SSCs and operational procedure:
  - auxiliary feedwater
  - cable
  - heating, ventilation, and air conditioning

- Examples of units of measure:
  - becquerel
  - rem
  - curie

- Examples of organizations:
  - Gesellschaft fuer Anlagen-und Reaktorsicherheit (Germany)
  - IAEA

- Examples of chemical terms:
  - chlorinated polyethylene

- Examples of nuclear safety analyses terms:
  - cladding
  - water hammer
  - departure from nucleate boiling

- Examples of computer code acronyms:
  - CAFTA
  - RISKMAN

These types of terms were excluded from the list of candidate terms.

<u>OUTPUT</u>

Once the guidance in this step has been applied, an initial list of potential candidate terms for the glossary was identified.

Because a term is a potential candidate for inclusion in the glossary does not necessarily mean that it should remain in the glossary. For example, the term may not be important to risk communication; or while it may be important, it is a well-known term whose meaning is widely accepted. In the next steps, each term was further reviewed to determine if it should remain as a candidate for the glossary.

2.2.2 Step 1-2: Important for Risk Communication

This step determines if any of the candidate terms from Step 1-1 should be excluded from the glossary because they are not important to risk communication. If a term is considered not essential to risk communication, it can be excluded from the glossary; conversely, if a term is considered essential, it should remain as a potential candidate for the glossary.

To determine if a term is essential for risk communication, guidelines need to be established to judge a term's importance. The candidate terms are then reviewed against these guidelines. However, developing the guidelines requires an understanding of what is meant by "risk communication," and therefore, what is meant by "term is needed (or essential) to risk communication."

Risk communication can have various meanings, from a very strict interpretation to a wide interpretation. For example, risk communication can mean communicating:

- What is meant by the term "risk"?
- What is a risk analysis?
- What are risk results and insights?
- How are risk results and insights used in decisionmaking?

As discussed in Section 1, for the purposes of this glossary, risk communication includes all of the above. In addition, understanding the audience is equally important. The audience may range from senior executives to subject experts to lay personnel. Consequently, guidelines are developed consistent with these objectives and expectations.

The guidelines developed to retain or screen candidate terms included:

1. The term is related to the science of risk analysis.
2. The term plays a role in risk-informed decisionmaking.

Guideline 1-2.1 – Term Related to Science of Risk Analysis

This guideline is used to identify terms related to the science of risk analysis (i.e., terms needed to understand risk analysis). Understanding risk analysis involves more than providing a definition of the term "risk analysis." Terms used in communicating a risk analysis and those used in understanding the details of performing a risk analysis need to be included.

The former (i.e., communicating a risk analysis) involves understanding the following:

- What constitutes a risk analysis? (i.e., what is a risk analysis?)
- What are the different kinds of risk analyses?
- What are the different terms used to explain risk analyses?
- What are the objectives of a risk analysis?
- What are the inputs and outputs of the analysis?

Terms needed to understand risk analysis (using the above guidance) should be retained as potential candidates for inclusion in the glossary. Examples of terms related to the high-level understanding include:

- consequence
- probability
- core damage frequency
- health effects
- initiating event

Understanding the details of performing a risk analysis involves comprehending the different technical elements of the specific hazards for which a risk analysis is performed (e.g., internal fire risk analysis) and the associated terminology for each technical element. These terms are included and defined in the hazard-specific glossaries. At this time, only one hazard-specific glossary has been developed (internal fires).

Determining if the term should be in the main glossary or the hazard-specific glossary can be subjective. For example, "hot short" is a term generally not used in risk communication, but it is part of the lexicon for internal fire. Consequently, it is defined in the internal fire-hazard glossary.

Guideline 1-2.2 – Term Plays a Role in Risk-Informed Decisionmaking

This guideline is used to identify terms that may not be related to the science of risk analysis, but often are used or associated with risk-informed decisionmaking. These terms are used in nuclear safety activities that are not necessarily risk specific, but are often used when risk-informed issues are discussed and communicated. There are also terms that may have risk aspects included under the umbrella of their broader meaning and should be retained as potential candidates for inclusion in the main glossary.

Examples of terms that play a role in risk-informed decisionmaking include:

- safety margins
- severe accidents
- public health effects

OUTPUT

Once the guidance in this step has been applied, terms are screened as either (1) important at a high level to risk communication for potential inclusion in the main glossary, (2) important for hazard-specific risk communication for potential inclusion in the hazard-specific glossary, or (3) not important to risk communication at any level and excluded from the glossary.

Because a term is important to risk communication, it does not necessarily mean that the term should remain in the main glossary. For example, the term may have no risk context or the term's definition may be well established throughout the community without any potential miscommunication. In the next steps, each candidate term for the main glossary is further reviewed to determine if it should remain as a candidate.

The remaining steps and guidance are discussed relative to the main glossary. However, they are applicable to the hazard-specific glossary and used where appropriate.

2.2.3 Step 1-3: A Risk-Context Specific Definition

This step determines if any of the terms from Step 1-2 have a risk-context specific meaning. Specifically, although a term's meaning may be consistent with the actual definition in a dictionary, its meaning in risk communication has a risk connotation. There are terms, however, whose meaning is the same regardless of their use.

The sources are reviewed to determine if a term's meaning has a risk connotation. At this point, there may be multiple risk meanings for a single term, which may or may not be consistent. However, the purpose of this step is only to identify those terms that do not have a risk connotation and, therefore, may be excluded from the main glossary.

For example, consider the terms "probability" and "internal hazards." The meaning of the term "probability" does not change when used in a risk context. Probability is defined as the "relative possibility that an event will occur as expressed by the ratio of the number of actual occurrences to the total number of possible occurrences." The term "internal hazards," however, does have a risk-context meaning. Outside of a risk context, an internal hazard would be "something dangerous that happens in the interior of something." However, in a risk context, an internal hazard has a very specific meaning. An "internal hazard" can be "an event originating within a nuclear power plant that directly or indirectly disrupts the steady state operation of the plant." For the main glossary, terms such as "probability" should be candidates for exclusion whereas terms such as "internal hazard" should be retained as a possible candidate.

OUTPUT

Once the guidance in this step has been applied, terms that have a risk-context specific meaning have been identified, along with those whose meanings are not risk-context specific.

Because a term has risk-context specific meaning does not necessarily mean it should be included in the main glossary. The term may have a risk-context specific meaning that is consensually established, uniformly known, and consistently used to the extent that it does not need to be explained to potential audiences. Conversely, because a term has no risk-context specific meaning does not necessarily mean it should be excluded from the main glossary. There may be other reasons for including it in the glossary, such as it is fundamental in risk communication. In the next steps, each term is further reviewed to determine if it should remain as a candidate for the main glossary.

2.2.4 Step 1-4: Availability of Definitions

This step determines if the terms identified in Step 1-3 have readily available definitions. The question becomes: Is there a readily available definition for terms whose usage (or definition) is the same whether it is used in a risk context or some other context?

If the term is consistently defined in well-established and authoritative references, such as published governmental regulations, text books, dictionaries, or consensus standards, then its inclusion in the main glossary may not be necessary. Examples of terms used in risk communication that have well-defined definitions, which apply in a risk context as well as in other contexts, include:

- probability
- aleatory/epistemic
- seismic

<u>OUTPUT</u>

Once the guidance in this step has been applied, those terms for which a definition is readily available have been identified along with those that do not have documented sources.

Because a term has consensually established sources for its definition, this does not mean it should be excluded from the main glossary. The term may be essential to understanding risk communication, for example, and on that basis should be included in the main glossary. In step 1-7, these terms are reviewed for their importance to determine if they should remain in the main glossary.

2.2.5 Step 1-5: Multiple Term Definitions

This step determines if the terms identified in Step 1-3 as having a risk-context specific definition have more than one risk-context definition. Using a non-risk term for illustration, the term “fathom” has two meanings. One, a fathom is a nautical term meaning 6 feet; however, second, “to fathom” means to understand. The purpose of this step is to identify such terms. As such, this step does not serve any screening purpose. It is simply a checkpoint to identify if a candidate term has multiple definitions.

The guidelines developed to determine if a term has multiple definitions include:

1. reviewing the definitions
2. performing a peer review

Guideline 1-5.1 – Definition Review

This guideline is used to review the definitions for the candidate terms. For each source, the definitions are reviewed to determine how each term is used. In some documents, this review is straightforward because the document provides a list of definitions or a glossary. In other cases, an actual definition may not be provided and a definition (or possible definitions) must be inferred. In these situations, the usage of the term may be very clear; however, there may be times when the usage is not clear and the intent of the term is ambiguous. For each term, the definitions are reviewed to determine if there are multiple definitions.

Guideline 1-5.2 – Peer Review

This guideline is used to perform a peer review. The objective of this review is to perform a sanity check. Reference documents may have been missed or not reviewed during the definition review resulting in terms with multiple definitions. Consequently, a peer review is performed to identify this gap. For each term, the reviewer identifies if there are multiple definitions based on his or her experience. If multiple definitions are identified, these additional definitions are added to the glossary (unless the term is eventually excluded from the glossary). In addition, the reviewer looks at the associated sources and, based on his or her experience, identifies any additional relevant documents that need to be included as a source. If additional sources are identified, the definitions are reviewed per Guideline1-5.1.

OUTPUT

Once the guidance in this step has been applied, terms with multiple definitions have been identified (terms are not screened in this step). Regardless of whether a term has one or more definitions, it still needs to be reviewed to determine if it should remain in the main glossary. This step ensures the completeness of the definitions of the candidate terms for the main glossary.

2.2.6 Step 1-6: Consensually Established Definitions

This step determines if there is agreement on meaning in a risk context for the terms reviewed in Step 1-5. If there is agreement about a term's meaning, then it can be assumed it has been consensually established. It cannot be assumed a term has been consensually established because it is defined in a standard. The definition has only been agreed upon for how the term is used in that standard. There may be disagreement or controversy in other uses (e.g., different uses in different standards, technical reports, guidance documents, or regulations). The challenge of this step is identifying the different uses of each term.

The guidelines developed to determine if there is agreement include:

1. reviewing the definitions
2. performing a peer review

The two guidelines of Step 1-6 are the same as Step 1-5. In general, determining if a term has a consensually established definition is performed at the same time as determining if a term has multiple definitions.

Guideline 1-6.1 – Definition Review

This guideline is used to review the definitions for the candidate terms. For each source, the definitions are reviewed to determine how each term is used. In some documents, this review is straightforward because the document provides a list of definitions or a glossary. In other cases, an actual definition may not be provided and a definition (or possible definitions) must be inferred. In these situations, the usage of the term may be very clear; however, there may be times when the usage is not clear and the intent of the term is ambiguous. For each term, the definitions are reviewed for consistency. Terms whose usage has been consistent are those whose meanings have been consensually established.

In this step, there also may be terms with multiple legitimate definitions. These multiple definitions are reviewed for consistency. Using the example of "fathom" from Step 1-5, both definitions would be reviewed to determine if both have been consistently used.

Guideline 1-6.2 – Peer Review

This guideline is used to perform a peer review. The objective of this review is to perform a sanity check. Reference documents may have been missed or not reviewed that could result in terms with definitions in disagreement or inconsistent usage. Based on experience, the peer reviewer is aware of disagreements or inconsistencies. Consequently, a peer review is performed to identify this gap. For each term, the reviewer looks at the associated sources, and based on his or her experience and expertise, identifies any additional relevant documents that need to be included as a source. If additional sources are identified, the definitions are reviewed under Guideline 1-6.1.

<u>OUTPUT</u>

Once the guidance in this step has been applied, those terms whose meaning has been consensually established have been identified along with those in disagreement, misunderstanding, or controversy.

If there is no disagreement, misunderstanding, or controversy about a term's meaning, it should not necessarily be rejected from the main glossary. The term's meaning may be essential to understanding risk communication, and on that basis it should remain as a candidate for the main glossary. In the next steps, these terms are further reviewed to determine if they should remain in the main glossary.

2.2.7 Step 1-7: Term Fundamental to Risk Communication

This step determines if any of the candidate terms from Steps 1-4 and 1-6, which would otherwise be excluded from the main glossary, should be retained because they are essential to a basic understanding of risk-informed activities. Terms that are potential candidates for exclusion from the main glossary include terms that have a well-documented definition source and that may or may not have a unique risk-context meaning.

Similar to Step 1-2, this step somewhat depends on the intended audience for the glossary. The audience may range from senior executives to subject experts to lay personnel. Nonetheless, some terms are fundamental to the basic understanding of risk-informed activities, and for the main glossary to be most effective, these terms should be included. However, while no special risk knowledge is presumed by the audience, a basic understanding of nuclear safety is assumed.

The guidance developed to identify fundamental terms include:

1. terms frequently used to communicate results of risk analyses
2. terms used in decisionmaking and communicating risk-informed decisions
3. terms sometimes misused or used in confusing ways

Guideline 1-7.1 – Terms Frequently Used to Communicate Risk Results

This guideline is used to identify terms that are frequently used and have a well-documented and consensual definition. Because of their frequent usage, these terms are useful to almost any audience for communicating the results of a risk analysis. A correct understanding of these terms is essential for accurately presenting results, as well as an accurate appreciation of what is being presented. Examples of terms used to communicate results include:

- health effects
- core damage frequency
- large early release frequency
- latent fatality

Also included are terms used to describe the analysis that produced the results. Examples of terms used to describe the analysis include:

- minimal cutset
- dose response model

Guideline 1-7.2 – Terms Used in Decisionmaking

This guideline is similar to Guideline 1-7.1, but it addresses terms used more often in decisionmaking and communication of decisions than in PRA results. A correct understanding of these terms is essential for accurately communicating and understanding a risk-informed decision.

Examples of terms useful for communicating and understanding decisions include:

- deterministic acceptance criteria
- high-level requirement
- consequences
- acute health effects

Guideline 1-7.3 – Terms Misused

This guideline is used to identify terms that are sometimes misused or used in confusing ways. Some terms may have established definitions in the published literature that are consistent with their risk-specific definition and are also fundamental to risk communication. However, their definitions may be complex or ambiguous (e.g., their usage is inconsistent with the intent of their meaning).

Examples of terms sometimes misused or used in confusing ways include:

- probability
- frequency
- model uncertainty
- internal event

OUTPUT

Once the guidance in this step has been applied, the final list of terms for the glossary has been identified.

Because a term has been identified as fundamental to a basic understanding of risk-informed activities does not mean that it should be included in the main glossary. The term may have policy implications and require a policy decision by the Commission. In the next step, these terms are reviewed for policy implications.

2.2.8 Step 1-8: Policy Implications

This step identifies if the terms from Steps 1-6 and 1-7 (which comprise the final list of terms) have policy implications. These terms may have risk-context definitions, definitions that are not consensually established, or may be terms fundamental to a basic understanding of risk communication. However, some of the terms may have policy implications and, therefore, its definitions could require Commission approval. These terms are identified and the policy implications discussed.

A definition for a term is considered to have policy implications if it:

- sets a precedent with broad ramifications
- states new Commission expectations
- deviates from current policy
- is fundamental to other decisions

Examples of terms that have potential policy issues include:

- defense-in-depth
- large release frequency

OUTPUT

Once the guidance in this step has been applied, terms that have a definition with policy implications have been identified, along with those that do not.

Because a term's definition has policy implications does not mean it should be excluded from the glossary. These terms remain in the glossary; however, a formal definition of the term is not developed. At this step in the process, a final list of terms has been identified and the next task is developing the definitions. As definitions are developed, however, new terms may be added or others may be deleted.

2.3 Task 2: Development of Definitions

Guidance was established for developing definitions. For example, for each term the glossary could provide a single definition or multiple definitions or the glossary could just document the various definitions found in the different sources. To meet the objective established for the glossary, such as reducing ambiguity and to be helpful to the user (regardless of the individual's level of risk experience or expertise), the glossary should provide more than just a simple definition. Understanding the meaning of a term may require some explanation (e.g., the bases for different definitions, the relationship to a related term). As such, guidance was developed to optimize the usability of the glossary. This guidance involves three major steps. These steps are not necessarily performed in a sequential manner, but more in an iterative and integrated fashion.

2-1 ***Develop Initial Definitions*** – For the terms identified from Task 1, the various definitions from the relevant sources are documented.

2-2 ***Identify Related Terms*** – For each term, the definitions are reviewed and terms are cross-referenced (e.g., related definitions) or grouped (similar definitions), where appropriate.

2-3 ***Finalize Definitions*** – For each term, although there may be published definitions, there may not be agreement; the bases for these disagreements are discussed. Furthermore, the definitions may not be easily understandable; therefore, a definition in plain English without the use of technical jargon is developed.

This process is illustrated in Figure 2-2 and discussed in detail below.

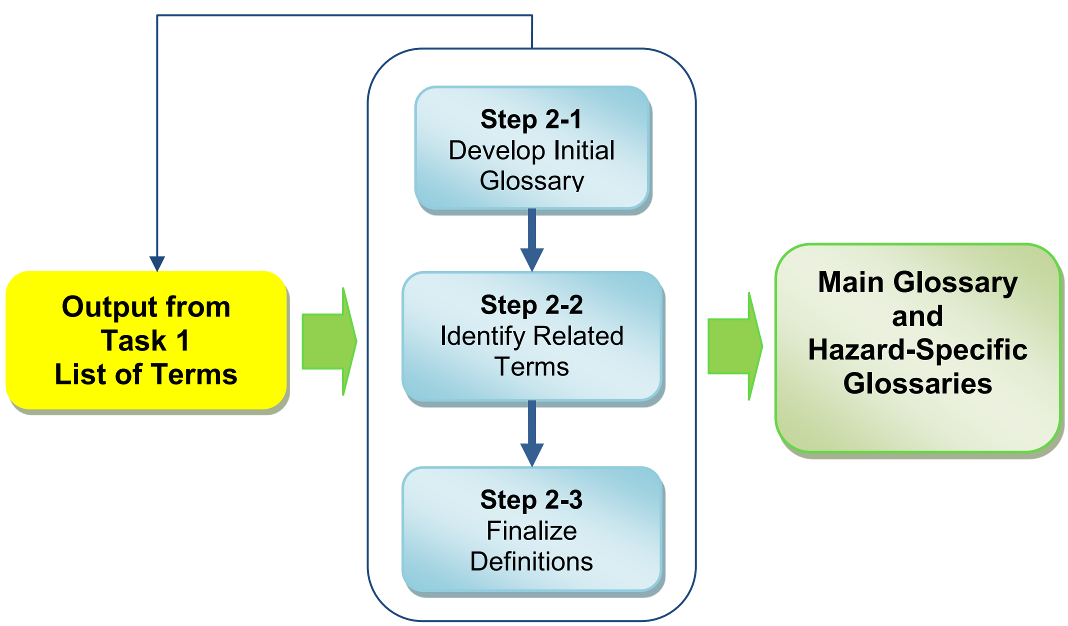


Figure 2-2 Process to develop definitions

2.3.1 Step 2-1: Develop Initial Glossary

This step collects the different definitions that exist for the final list of terms and documents both the definitions and their sources. For some terms, multiple or different definitions may exist; these are included in the glossary as potential definitions.

Most of the effort performed for this step was completed in Task 1. The main focus of this step is to document the definition (or definitions) associated with each term and its associated sources.

OUTPUT

Once the guidance in this step has been applied, the reference sources and definitions for the final list of terms have been collected and documented. The result is an initial glossary; however, there may be discrepancies in definitions. There also may be terms with similar definitions that are used interchangeably. Furthermore, some terms may be closely related, and understanding both terms is necessary. Identifying these situations and including explanations, where appropriate, helps in understanding the terms.

2.3.2 Step 2-2: Identify Related Terms

This step identifies terms related in some manner. For some related terms, an understanding of the relationship between the terms is needed to fully understand each one. These terms need to be cross-referenced to compare their differences or similarities with regard to their meanings and applications. In addition, there are terms that convey the same or similar meaning. These terms should be grouped together to avoid redundancy in definitions.

The guidance developed to identify related terms includes:

1. cross-referenced terms
2. grouped terms

Guideline 2-2.1 – Cross-Reference of Terms

This guideline is used to identify those terms that are related and should be cross-referenced in the glossary. These terms are:

- similar, but they have different meanings, and there are instances in which they are used incorrectly (e.g., “probability” versus “frequency,” “core damage” versus “core melt”)
- related, in that one is the opposite of the other (e.g., “deterministic” and “probabilistic”)
- closely related in meaning, and may be a subset of the related term(e.g., “harzard,” “external hazard” and “internal hazard”) or an example of the related term (e.g., “risk analysis” and “probabilistic risk assessment”, “accident consequence” and “health effects”).
- closely related, in that understanding one term depends on understanding the related term (e.g., “risk analysis” and “probabilistic risk assessment”, “accident consequence” and “health effects”).

These terms are identified and cross-referenced. In some cases, a definition is provided for each term, and the reason for the cross-reference is discussed. In other cases, the term is cross-referenced to another term and the definition is provided by the cross-referenced term For example:

- “Risk-informed approach” is a term in the glossary. A definition is not provided for this term. However, in the Discussion Column, it is noted that this term “is related to the term risk-informed and is defined under “Risk-Informed”.”

- “Probability” is a term in the glossary. A definition is provided and there is a cross-reference to frequency.

Guideline 2-2.2 – Grouping of Terms

This guideline is used to identify related terms that should be grouped in the glossary. This grouping assists the reader so that it is clear which terms should be defined together. These are terms that are:

- composed of multiple words with the same adjective (e.g., “significant,” “significant contributor,” “significant basic event”; “accident sequence class,” “accident sequence type,” “accident sequence group”)

- similar and convey the same meaning (e.g., “general transient” and “transient”)

- complements of each other (e.g., “availability” and “unavailability”)

These terms are identified and grouped. A single definition is provided in the glossary. When the group is a result of a common adjective, then the terms being modified are discussed where appropriate.

<u>OUTPUT</u>

Once the guidance in this step has been applied, terms that should be cross-referenced or grouped have been identified and the bases understood. With this accomplished, the actual definitions for each term need to be finalized and associated discussion included.

2.3.3 Step 2-3: Finalize Definitions in Glossary

This step finalizes the definitions and associated discussion to complete the glossary.

The guidance developed to complete the glossary includes:

1. modify definition
2. develop discussion

Guideline 2-3.1 – Modify Definition

This guideline is used to modify definition(s) of terms where necessary. As noted previously, the intent of the glossary is not to recreate definitions; consequently, if an appropriate definition exists, it is used as the starting point. However, it is the objective of the glossary that the definition provided is stated in plain English with little-to-no reliance on technical jargon.

The definitions collected from the various sources generally use technical jargon and are often written in a complex manner. For these terms, while the definition may be accurate, it can be difficult to understand and sometimes requires a specific expertise to understand. In these instances, the definition is modified and written in plain language (e.g., use of common, everyday words in short sentences) so that it can be clearly understood. The more complex definition (or explanation) is provided as part of the discussion.

Guideline 2-3.2 – Develop Discussion

This guideline is used to add the necessary discussion to each term to assist the user in understanding the meaning of the term as used in risk communication, where appropriate. Discussion might include:

- The definition provided is written in plain English, but does not necessarily provide a complete risk context. In these cases, the risk (in particular, PRA) context is explained in the discussion.

- There may be multiple legitimate definitions, each of which is included in the glossary. While the appropriate definition(s) are provided, a discussion is also included to explain why certain definitions are not appropriate.

- There may be discrepancies and inconsistencies among the definitions. An explanation is provided for the discrepancies or differences in use of the term.

- Where terms are cross-referenced, the reason for the cross-reference is provided where necessary (i.e., for some cross-references, the basis does not require an explanation, but for others, an explanation may be necessary to understand the relationship). Further, a discussion may be needed on the use of the terms. Examples include:

  — “Probability” and “frequency” are similar terms, but they have different meanings and it is not surprising that these terms should be cross-referenced. However, one of the main reasons for the cross-reference is to remind the reader that these terms are not the same and have been incorrectly used in the past.

  — “External events” and “external hazards” have the same meaning and should be cross-referenced. In current usage, external hazard has replaced the term external event, therefore a cross-reference was essential to emphasize this matter.

- There may be some terms whose definitions have policy implications(e.g., “defense-in-depth”). For these terms, the various definitions are provided; however, a single definition is not defined. Further, a discussion of why the definition is considered to have policy implications is provided.

OUTPUT

Once the guidance in this step has been applied, a complete glossary, given the scope and limitations of the document, has been developed. A definition(s) has been developed for each term. Discussion has been included to provide explanation and clarity to assist in understanding the meaning of each term. Terms whose definitions have policy implications also have been identified.

3. ABBREVIATIONS AND ACRONYMS

In risk communication, there are abbreviations and acronyms that have become common and in some cases, often are not defined. Table 3-1 is not meant to be all inclusive; it is meant to identify the more commonly used abbreviations or acronyms.

Table 3-1 Commonly Used Abbreviations and Acronyms

ACRS	Advisory Committee on Reactor Safeguards
ANS	American Nuclear Society
APET	accident progression event tree
ASME	(formerly) American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BE	basic event
BWR	boiling-water reactor
CCDF	complementary cumulative distribution function
CCDP	conditional core damage probability
CCF	common-cause failure
CD	core damage
CDF	core damage frequency
CDP	core damage probability
CET	containment event tree
CLERP	conditional large early release probability
CM	core melt
CMF	common-mode failure core-melt frequency
CRM	configuration risk management
CY	calendar year
DBA	design-basis accident
DBE	design-basis earthquake design-basis event
DCF	dose conversion factor
DCH	direct containment heating
EAB	exclusion area boundary
EP	emergency preparedness
EPRI	Electric Power Research Institute

3. ABBREVIATIONS AND ACRONYMS

Table 3-1 Commonly Used Abbreviations and Acronyms

ET	event tree
F&B	feed and bleed (bleed and feed)
FM	failure mode
FMEA	failure modes and effects analysis
FT	fault tree
HCLPF	high confidence in low probability of failure
HEP	human error probability
HFE	human failure event
HLR	high-level requirement
HPME	high-pressure melt ejection
HRA	human reliability analysis
IE	initiating event
IM	importance measure
ISLOCA	interfacing-systems loss-of-coolant accident
LBE	licensing-basis event
LERF	large early release frequency
LOCA	loss-of coolant accident
LOOP, LOSP	loss of offsite power; loss of station power
LP/SD	low power/shut down
LWR	light-water reactor
NEI	Nuclear Energy Institute
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
OG	owners group
PDS	plant damage state
POS	plant operational state
PRA	probabilistic risk assessment (base, baseline)
PWR	pressurized-water reactor
QA	quality assurance
QHO	quantitative health objective

3. ABBREVIATIONS AND ACRONYMS

Table 3-1 Commonly Used Abbreviations and Acronyms

RAW	risk achievement worth
RG	regulatory guide
RIDM	risk-informed decisionmaking
RY	reactor-year
SA	systems analysis
SB, SBO	station blackout
SGTR	steam generator tube rupture
SM	seismic margin
SOKC	state-of-knowledge correlation
SR	supporting requirement
ST	source term

4. GLOSSARY

4.1 Understanding the Format and Structure

This section describes the format and structure used in listing and defining the terms in the glossary. The understanding of this process is essential to being able to use the glossary effectively and efficiently.

For each term, a definition is provided in plain English with little-to-no technical jargon so that understanding does not depend on an individual's level of risk experience or expertise. In addition, a discussion is provided to include explanations to help the user understand the meaning of the term. This discussion, where appropriate, generally includes:

- the definition of the term in a risk context
- the different definitions of the term
- how the term has been and should be used
- how the term relates to other terms for a complete understanding of the definition

The table contains two columns. The first column provides the term and its definition and any appropriate cross-references; the second column contains the discussion of the term.

Where appropriate, terms are grouped, related, and cross-referenced as follows:

- When a term is related to another term(s), these related terms are referenced at the end of the definition of the main term. These related terms are not subsets of the main term or synonyms. The main term and related term each has a distinctly different definition; consequently, each term is listed separately in the glossary with its own definition. Furthermore, the reason for referencing any related term(s) is discussed in the discussion column.

  For example, common-cause failure and common-mode failure are related terms, but they have very different meanings:

Main term

Main term definition

Related terms

Discussion on relationship for the two terms

TERM AND DEFINITION	DISCUSSION
Common-Cause Failure	
A failure of two or more structure, system or components as a result of a single shared cause. *(see Common-Mode Failure, Failure Mode)*	In a PRA, common-cause failure (CCF) is a special form of dependent failure in which the failure of the structure, system, or component (SSC) has occurred from the same fault. CCF faults generally reflect errors occurring as a result of a common manufacturer, environment, maintenance, etc. The CCF term is often incorrectly used interchangeably with common-mode failure (CMF). CCF only accounts for the SSCs failing because of the same, single cause, not if they ultimately fail in the same manner (or in the same mode), which is CMF. In data provided to quantify CCF events, the failure mode is usually presented (i.e., failure to start, fail to run), and the cause is not always provided about why the failure mode occurs. There could be multiple causes lumped into the data presentation for a given failure mode. Thus, the available failure data dictate whether the PRA model is modeling CCF or CMF.

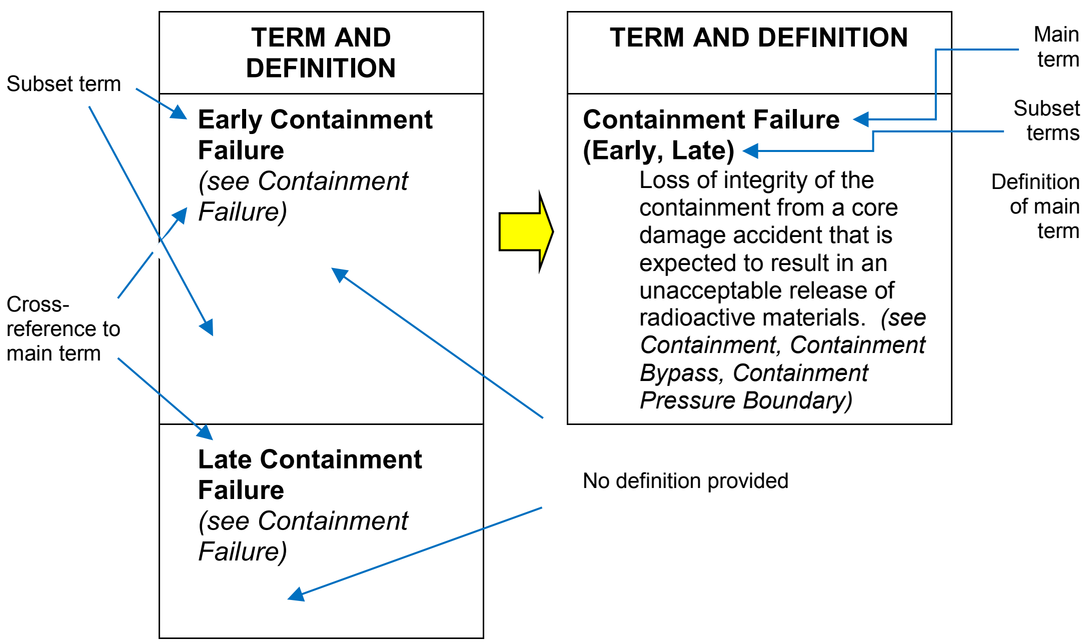
Moreover, for all the related terms, there is a cross-reference for each related term. For example:

TERM AND DEFINITION
Common-Cause Failure A failure of two or more structure, system or components as a result of a single shared cause. *(see Common-Mode Failure, Failure Mode)*

TERM AND DEFINITION
Common-Mode Failure A failure of two or more structures, systems, or components in the same manner or mode as the result of a single shared cause. *(see Common-Cause Failure, Failure Mode)*

- When a term is a subset of another term (i.e., main term), this subset term is listed with the main term. It is also listed separately in the glossary and cross-referenced to the main term. In these cases, no definition appears with the subset term; the subset term is defined with the main term. The definition provided is broad enough to encompass the main term and all the subset terms.

For example, early containment failure (ECF) is a subset of containment failure, which is the main term; therefore, ECF is cross-referenced to the main term, containment failure, where it is defined.



For situations in which terms are a subset of another term, the specific words causing the term to be a subset are enclosed in parentheses after the main term. The definitions for the subset terms are provided in the discussion column. For example, *early* containment failure is defined under containment failure. However, the definition provided is for containment failure, and the distinction and definition of early containment failure is provided in the discussion column.

Main term

Subset terms

Definition of main term

Discussion and definition of subset terms

TERM AND DEFINITION	DISCUSSION
Containment Failure (Early, Late)	
Loss of integrity of the containment from a core damage accident that is expected to result in an unacceptable release of radioactive materials. *(see Containment, Containment Bypass, Containment Pressure Boundary)*	In a PRA, determining when and if the containment fails or is bypassed during a severe accident is very important from a risk perspective. If the containment pressure boundary remains leak-tight, the offsite consequence will be low. Conversely, if the containment fails or is bypassed, then the consequence to the surrounding population can be potentially high. For specific containments there can be selected severe accident scenarios in which the containment fails before fission products have penetrated the primary system. If the accident is successfully arrested at this point, no release will occur. However, usually containment failure represents the failure of the final barrier preventing a radioactive material release. Containment failure is often categorized as early or late. Early containment failure occurs in a timeframe before the surrounding population within 1 mile of the site boundary can be evacuated. Late containment failure occurs in a timeframe that allows the surrounding population from 1 to 10 miles to be evacuated.

- When a term has the same meaning as another term, the terms are grouped. The terms generally are listed in the group alphabetically, unless one of the terms is prevalently used over the other terms. The prevalently used term is then listed first in the group. The first term listed in the group is the main term. The latter terms of the group are separately listed in the glossary where they are cross-referenced to the main term. No definition is provided for these latter similar terms; they are defined with the main term. For example, "accident sequence class," "accident sequence group," and "accident sequence type" are similar terms and are grouped together, with "accident class" identified as the main term. The terms are grouped alphabetically, and the first term is the main term.

Similar term

Cross-reference to main term

TERM AND DEFINITION
Accident Sequence Group *See Accident Sequence Class*
Accident Sequence Type *See Accident Sequence Class*

No definition provided

TERM AND DEFINITION
Accident Sequence Class, Accident Sequence Group, Accident Sequence Type, Event Sequence Class, Event Sequence Group, Event Sequence Type A grouping of accident sequences with similar characteristics or end states. (*see Accident Sequence*)

Main term

Similar terms

Definition of main term

For situations in which terms are similar or synonymous and are grouped, the terms are separated by commas, which indicates that they are terms with the same meaning and are not related terms. Moreover, the discussion will explain the reason for the grouping.

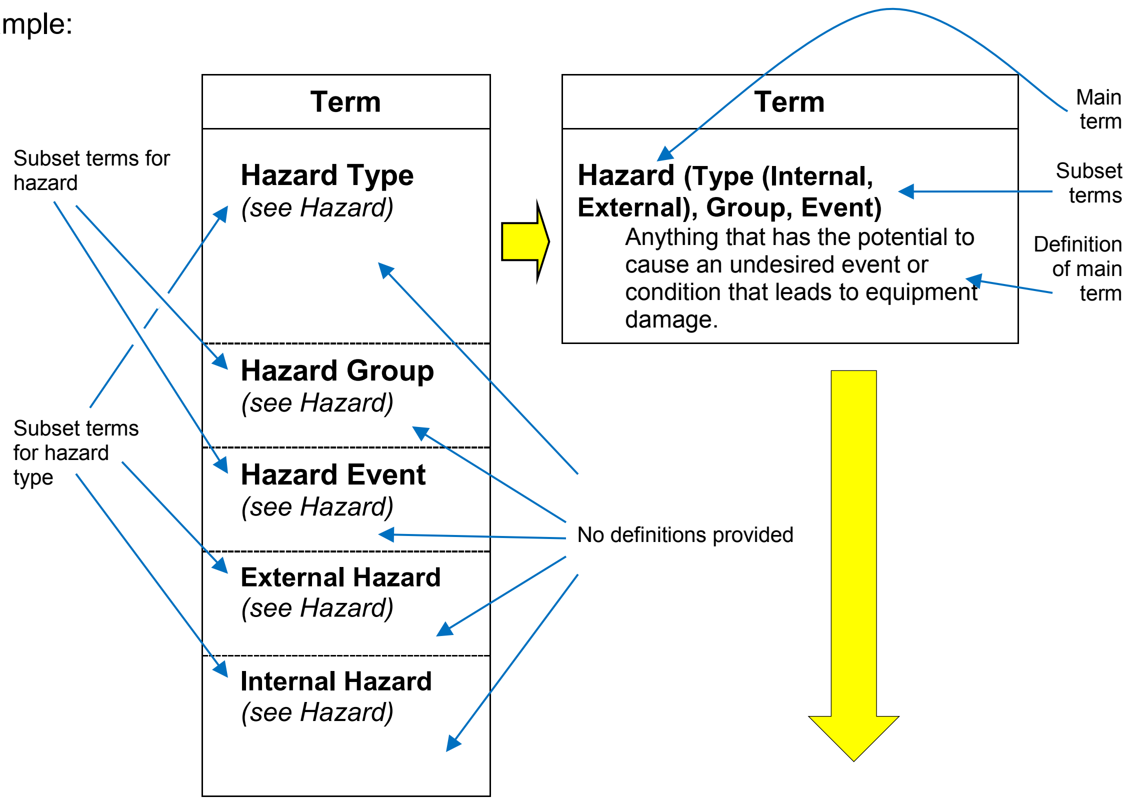
Term Group

Definition for group

TERM AND DEFINITION	DISCUSSION
Accident Sequence Class, Accident Sequence Group, Accident Sequence Type, Event Sequence Class, Event Sequence Group, Event Sequence Type	
A grouping of accident sequences with similar characteristics or end states. *(see Accident Sequence)*	In a PRA, the accident sequences typically are combined into accident sequence classes (groups or types). For example, an accident sequence class might represent a set of accident sequences with similar initiating events (e.g., loss-of-coolant accidents (LOCAs), loss of offsite power (LOOP), loss of heat removal or similar safety function responses. The purpose for combining like sequences is generally done to understand the type of sequences contributing to the risk. The terms accident sequence class, accident sequence group, and accident sequence type are similar in meaning and often correctly used interchangeably. Moreover, accident sequence is also used interchangeably with event sequence. Consequently, the terms event sequence class, event sequence group, and event sequence type also are similar in meaning and used interchangeably.

- There also may be instances of multiple subsets. For example, hazard type, hazard group, and hazard event are all subsets of hazard; internal and external hazards are both subsets of hazard type and hazard and are each listed in the glossary. There is still a single main term and the definition is provided for the main term. Each subset term is still separately listed, cross-referenced to the main term, and no definition is provided. The specific words causing the term to be a subset are still enclosed in parentheses after the main term. The definitions for the subset terms are provided in the discussion.

For example:



Discussion
In a PRA, there are three different uses of the term hazard as an adjective (the terms hazard and plant hazard tend to be correctly used interchangeably): types, groups, and events. The first, hazard type, classifies hazards as either internal or external to the plant. Within each hazard type, internal and external, there are subcategories, which are referred to as hazard groups. For internal hazards, this hazard group includes internal events, internal floods, and internal fires. For external hazards, this includes seismic events, high winds, external floods, and other external hazards. Finally, a hazard event represents the events brought about by the occurrence of the specified hazard. For example, those of interest in a PRA are ones that directly or indirectly cause an initiating event and may further cause safety system failures or operator errors that may lead to core damage or radioactive material release. As defined in Regulatory Guide 1.200 (Ref.91), a hazard group "is a group of similar causes of initiating events that are assessed in a PRA using a common approach, methods, and likelihood data for characterizing the effect on the plant." A hazard event is described in terms of the specific levels of severity of impact that a hazard can have on the plant. The hazard event is an occurrence of the phenomenon that can result in a plant trip and possibly other damage when the plant is at-power or result in the loss of a key safety function during non-power operations. The ASME/ANS PRA Standard (Ref. 2) states that there "is a range of hazard events associated with any given hazard, and, for analysis purposes, the range can be divided into bins characterized by their severity." An example of the overall concept of hazard, hazard event, and initiating event is as follows: • Earthquakes are a hazard; • 0.1g, 0.3g, 0.5g earthquakes and their associated spectral shapes and time histories may be defined as hazard events; • A manual plant trip is typically the initiating event for the 0.1g earthquake, and a loss of offsite power is typically assumed as the initiating event for the 0.3g and 0.5g earthquakes. The ASME/ANS PRA Standard (Ref. 2) defines a hazard as "an event or a natural phenomenon that poses some risk to a facility. Internal hazards include events such as equipment failures, human failures, and flooding and fires internal to the plant. External hazards include events such as flooding and fires external to the plant, tornadoes, earthquakes, and aircraft crashes."

4.2 Terms and Definitions

Table 4-1 provides the terms and their definitions with the associated discussion. The terms are listed alphabetically. Hazard-specific terms are listed, but their definitions are provided in the noted appendix.

Table 4-1 Term and Definition(s)

TERM AND DEFINITION	DISCUSSION
Accident Consequence	
The health effects or the economic costs resulting from a nuclear power plant accident. *(see Health Effects, Accident Consequence Analysis)*	In a Level 3 PRA, the consequences can be measured by health effects and economic costs resulting from a nuclear accident. The accident consequences analyzed in a risk analysis generally involve evaluating the extent to which the health of the surrounding population or the condition of the surrounding environment is affected. The health effects and economic costs of a nuclear accident can be incurred both on the plant site as well as in the surrounding community. In most cases, the focus is on offsite consequences (i.e., (1) radiation doses from various exposure pathways and consequent health effects to the public, and (2) the economic costs associated with protective measures, such as evacuation and relocation of the public, destruction of contaminated foodstuffs, and decontamination or interdiction of contaminated land and property).
Accident Consequence Analysis	
The calculation of the extent of health effects or the economic costs resulting from a nuclear power plant accident. *(see Accident Consequence)*	In a PRA, the accident consequence analysis is the actual quantification of the potential magnitude of health effects and/or economic costs that can result from a nuclear accident. Accident consequence analysis attempts to answer the third of the three questions used to define risk: (1) What can go wrong? (2) How likely is it? (3) What might be its consequences?
Accident Event Sequence	
(see Accident Sequence)	The term accident event sequence has the same meaning as accident sequence and is defined under “Accident Sequence.”
Accident Mitigation	
Actions taken to reduce the severity of an accident. *(see Accident Prevention, Emergency Preparedness, Emergency Response)*	In a PRA, accident mitigation typically refers to actions taken to reduce the severity of an accident once core damage has started, as opposed to actions to prevent a core damage event from occurring. Successful accident mitigation implies that a core damage event occurred, but its consequences were minimized. Some strategies used for accident mitigation include preventing fission product releases by maintaining barrier integrity, or reducing fission product releases by filtration. Also, accident mitigation measures typically refer to plans or actions taken on the plant site, while emergency preparedness measures and emergency response (e.g., evacuation, sheltering) refer to plans or actions taken to reduce exposure of onsite workers, as well as the surrounding population offsite.
Accident Precursor, Precursor Event	
A change in plant status that could lead to core damage accidents.	A PRA is used to evaluate an event to determine if it will be considered an accident precursor. A conditional core damage probability (CCDP) is calculated for the event. The event is considered a precursor event, according to the NRC’s Performance and Accountability Report (Ref. 55), if the event “has a probability of greater than 1 in 1 million of leading to substantial

<table>
<tr><th>TERM AND DEFINITION</th><th>DISCUSSION</th></tr>
<tr><td></td><td>damage to the reactor fuel.” An event is considered to be a “significant precursor” when the event “has a probability of 1 in 1,000 (or greater) of leading to substantial damage to the reactor fuel.”

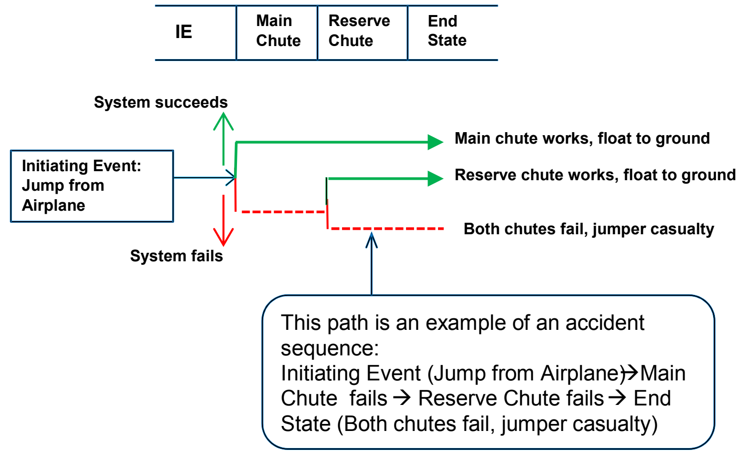
The terms accident precursor and precursor event generally have the same meaning. In some documents, the definition of accident precursor or precursor event includes quantitative criteria (e.g., as in the definition above), whereas some other definitions do not include quantitative criteria.</td></tr>
<tr><td colspan="2">Accident Prevention</td></tr>
<tr><td>Actions taken to reduce the likelihood of an accident. (see Accident Mitigation)</td><td>In a PRA, accident prevention typically refers to actions taken to prevent a core damage event from occurring, as opposed to reducing the severity once core damage has started. Successful accident prevention implies that a core damage event does not occur.

Some strategies used for accident prevention include: physical protection, maintaining plant stable operation, reactor protective systems, and maintaining barrier integrity.</td></tr>
<tr><td colspan="2">Accident Progression Event Tree</td></tr>
<tr><td>A logic diagram that begins with the onset of core damage and identifies the potential responses of the containment and associated equipment, as well as operator actions, to the severe accident loads. (see Bridge Tree, Containment Event Tree, Event Tree)</td><td>In the PRAs documented in the NUREG-1150 (Ref. 51) series of reports, an accident progression event tree (APET) was used to analyze containment response to severe accident loads. An APET is a detailed representation of the containment response to severe accident loads, including the interaction of phenomena, the availability of equipment, and the performance of operators. For most modern PRAs, a containment event tree (CET), which is a less complex representation, is used to emphasize the status of the containment and containment equipment during a severe accident. The end states of both the APET and the CET are no containment failure, various containment failure modes, or containment bypass.</td></tr>
<tr><td colspan="2">Accident Scenario</td></tr>
<tr><td>(see Accident Sequence)</td><td>The term accident scenario has the same meaning as accident sequence and is defined under “Accident Sequence.”</td></tr>
<tr><td colspan="2">Accident Sequence Analysis, Event Sequence Analysis</td></tr>
<tr><td>The process used to determine the series of events that can lead to undesired consequences. (see Accident Sequence)</td><td>In a PRA, accident sequence analysis is the process used to determine the combination of events that can lead to the undesired end state (e.g., core damage or large early release). The results of the accident sequence analysis are expressed in terms of individual accident sequences, each of which includes an initiating event followed by the necessary set of failures or successes of additional events (such as system, function, or operator performance) that will cause the undesired event.

The terms accident sequence analysis and event sequence analysis are similar in meaning and often correctly used interchangeably. However, generally the terminology “accident” refers to leading to core damage, and the terminology “event” does not necessarily reflect a negative outcome such as core damage.

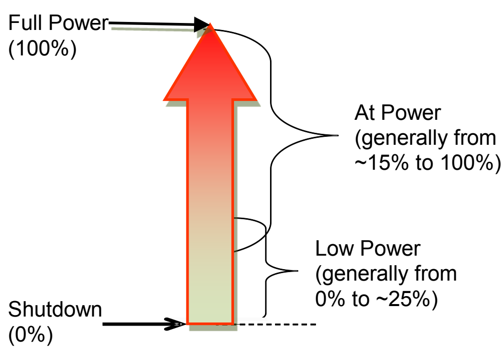
The ASME/ANS PRA Standard (Ref. 2) defines accident sequence analysis as “the process to determine the combinations of initiating events, safety functions, and system failures and successes that may lead to core damage or large early release.”</td></tr>
</table>

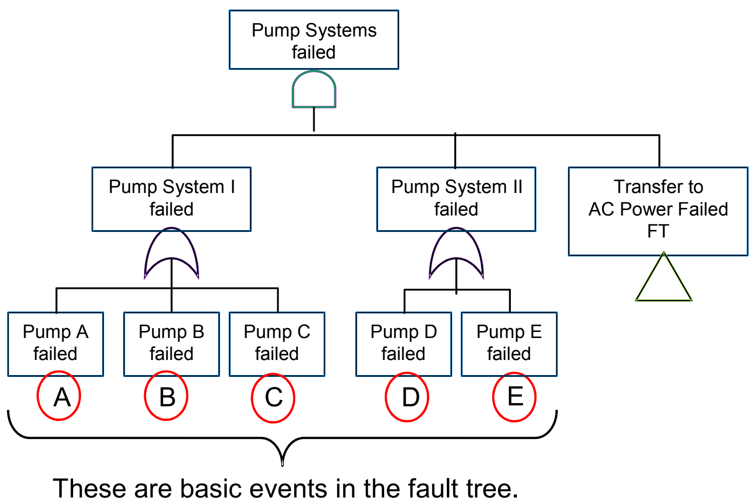
TERM AND DEFINITION	DISCUSSION
Accident Sequence Class, Accident Sequence Group, Accident Sequence Type , Event Sequence Class, Event Sequence Group, Event Sequence Type	
A grouping of accident sequences with similar characteristics or end states. *(see Accident Sequence)*	In a PRA, the accident sequences typically are combined into accident sequence classes (groups or types). For example, an accident sequence class might represent a set of accident sequences with similar initiating events (e.g., loss-of-coolant accidents (LOCAs), loss of offsite power (LOOP), loss of heat removal or similar safety function responses. The purpose for combining like sequences is generally done to understand the type of sequences contributing to the risk. The terms accident sequence class, accident sequence group, and accident sequence type are similar in meaning and often correctly used interchangeably. Moreover, accident sequence is also used interchangeably with event sequence. Consequently, the terms event sequence class, event sequence group, and event sequence type also are similar in meaning and used interchangeably.
Accident Sequence Frequency	
(see Frequency)	Accident sequence frequency is a type of frequency used in PRA and is defined in the discussion under "Frequency."
Accident Sequence Group	
(see Accident Sequence Class)	The term accident sequence group has the same meaning as accident sequence class and is defined under "Accident Sequence Class."
Accident Sequence Type	
(see Accident Sequence Class)	The term accident sequence type has the same meaning as accident sequence class and is defined under "Accident Sequence Class."
Accident Sequence, Accident Event Sequence, Accident Scenario, Event Sequence, Event Scenario, Event Tree Sequence	
A series of events that can lead to undesired consequences. *(see Accident Sequence Analysis, Severe Accident, End State, Event Tree)*	In a PRA, this series of events (e.g., an accident sequence, scenario, or event sequence) refers to an event tree pathway that follows from a particular initiating event, through system and operator responses, and ultimately to a well-defined end state, such as core damage. If the end state involves extensive core damage and radioactive material release into the reactor vessel and containment, with potential release to the environment, the accident sequence would represent a severe accident sequence. The system and operator responses may involve success, failure, or both. The terms accident sequence, accident event sequence, accident scenario, event scenario, event sequence, and event tree sequence are similar in meaning and are often correctly used interchangeably. The ASME/ANS PRA Standard (Ref. 2) defines an accident sequence as "a representation in terms of an initiating event followed by a sequence of failures or successes, of events (such as system, function or operator performance) that can lead to undesired consequences with a specified end state (e.g., core damage or large early release)." The following figure is an example of an accident sequence:

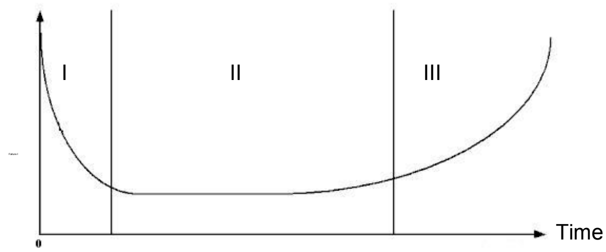
TERM AND DEFINITION	DISCUSSION
	
Active Component	
A component whose operation or function depends on an external source of power (e.g., air, electrical, hydraulic). *(see Passive Component)*	In a PRA, important elements of the model include both active and passive components. NUREG/CR-5695 (Ref. 74) defines active component as: "A component which normally is operating or can and should change state under normal operating conditions or in response to accident conditions (e.g., pumps, valves, switches)." Some examples of active components include pumps, fans, relays, and transistors. These are identified as active components because they rely on an external driving mechanism to perform their function. The IAEA Safety Glossary (Ref. 7) mentions "certain components, such as rupture discs, check valves, safety valves, injectors, and some solid state electronic devices, have characteristics that require special consideration before designation as an active or passive component." This special consideration implies that some components are not easily labeled as either active or passive because they may have characteristics of both. The ability to change state is sometimes considered as the defining characteristic of whether a component is active or passive. For example, a check valve normally has a passive function, but in a safety injection system it could be considered active since it needs to open and then reclose to prevent backflow.
Acute Exposure	
(see Exposure)	The term acute exposure is a type of exposure and is defined in the discussion under "Exposure."
Acute Health Effects	
(see Health Effects)	The term acute health effect refers to a type of health effect and is defined in the discussion under "Health Effects."
Aging	
General process in which characteristics of a structure or component gradually change (e.g., degrade) with time or use. *(see Bathtub Curve)*	In a PRA, the aging of a component is generally not explicitly modeled but is sometimes assumed to be reflected in the failure probability used to represent the performance of the component. The performance of structures or components may degrade with time (e.g., increasing failure rates, new failure modes) because of wearout and exposure to environmental conditions. Aging can lead to increasing failure rates in the later stages of life of a component. During the early life (burn-in) of a component, failure rates can decrease until a plateau is reached, as seen in the bathtub curve. The definition provided is based on the definition in the IAEA Safety Glossary (Ref. 7).

TERM AND DEFINITION	DISCUSSION
Air Submersion	
(see Cloudshine)	Air submersion has the same meaning as cloudshine and is defined under "Cloudshine."
Aleatory Uncertainty	
(see Uncertainty)	The term aleatory uncertainty is a specific type of uncertainty and is defined under the term "Uncertainty."
Anticipated Transient Without Scram	
An event that requires a plant trip and challenges safety systems but is followed by failure of control rod insertion to terminate the fission process. *(see Transient)*	In a PRA, anticipated transient without scram (ATWS) is referred to as both the initiating event and an accident sequence class. When referring to ATWS as the initiating event, this includes the initiating event (e.g., failure of the feedwater system) and failure of the reactor protection system (RPS). When referring to ATWS as an accident sequence, this includes the initiating event, failure of the RPS, and failure of other methods for terminating the fission process (e.g., emergency boron injection for a boiling-water reactor). A few examples of definitions for ATWS include: • "An ATWS is one of the "worst-case" accidents, consideration of which frequently motivates the NRC to take regulatory action. Such an accident could happen if the scram system (which provides a highly reliable means of shutting down the reactor) fails to work during a reactor event (anticipated transient). The types of events considered are those used for designing the plant." (NRC Web site Glossary, Ref. 36) • The *Code of Federal Regulations* formally defines ATWS as "an anticipated operational occurrence followed by the failure of the reactor trip portion of the protection system specified in General Design Criterion 20." (10 CFR Part 50, Appendix A, Ref. 22) • "The event is a perturbation in the state of some system or component at full reactor power that initiates a deviation from the full-power, steady-state operating conditions that have been previously considered and analyzed, which would normally result in a reactor scram. However, in this case, the reactor does not scram, either automatically or manually." (NUREG-1742, Ref. 59)
As-Built As-Operated (As-Designed)	
The accurate and current design and operation of the plant. *(see PRA Configuration Control, Living PRA, Plant Configuration Control)*	When applied to a PRA, as-built as-operated refers to the fidelity of the PRA model matching the current plant design, configuration, procedures, and performance data (e.g., component failure rates). Similarly, as-designed refers to the PRA matching the plant configuration in the design certification or combined operating license stage, in which the plant is not yet built or operated. Because the plant's configuration and operating procedures are continuously upgraded and modified and operating experience is accrued, the PRA model needs to be updated from time to time to reflect the as-built, as-operated plant. In that case, the model is said to be up-to-date (i.e., current). A PRA that is continuously updated to incorporate plant changes is called a living PRA. In the ASME/ANS PRA Standard (Ref. 2), as-built as-operated is defined as "a conceptual term that reflects the degree to which the PRA matches the current plant design, plant procedures, and plant performance data, relative to a specific point in time."

TERM AND DEFINITION	DISCUSSION
As-Designed	
(see As-Built As-Operated)	The term as-designed is defined in the discussion of the term "As-Built As-Operated."
Assumption (Key)	
A decision or judgment that is made in the development of a model or analysis. *(see Model Uncertainty)*	In a PRA, an assumption is either related to a source of model uncertainty or to scope or level of detail. An assumption related to a model uncertainty is made about the choice of the data, approach, or model used to address an issue because there is no consensus. A credible assumption is one that has a sound technical basis, such that the basis would receive broad acceptance within the relevant technical community. An assumption related to scope or level of detail is one that is made for modeling convenience. An assumption is considered to be key to a risk-informed decision when it could affect the PRA results that are being used in a decision and, consequently, may influence the decision being made. An effect on the PRA results could include the introduction of a new functional accident sequence or other changes to the risk profile (e.g., overall core damage frequency or large early release frequency, event importance measures). Key sources of model uncertainty are identified in the context of an application. The definition provided is based on the definition in the ASME/ANS PRA Standard (Ref. 2). The NRC Web site Glossary (Ref. 36) states, "in the context of individual plant examinations (IPEs), individual plant examinations for external events (IPEEE), and probabilistic risk assessments (PRAs), assumptions are those parts of the mathematical models that the analyst expects will hold true for the range of solutions used for making decisions."
Atmospheric Transport and Diffusion	
The movement and variation in concentration of a radioactive plume after release to the environment. *(see Atmospheric Transport and Diffusion Analysis, Level 1,2,3 PRA)*	In a PRA, assumptions about atmospheric transport and diffusion of the radioactive plume are used in the calculation of the health effects or economic consequences of a severe accident. A Level 3 PRA takes the result of a Level 2 PRA (frequencies, amounts, timing durations, and energies of radioactivity releases) and produces offsite consequences (health effects, economic consequences) as output. To calculate the offsite consequences, the movement and concentration of the radioactive plume under various weather conditions (e.g., high winds, rain) has to be determined. The plume characteristics can then be combined with the population information to calculate the health effects. The plume characteristics also can be used to determine land contamination and economic consequences of a severe accident.
Atmospheric Transport and Diffusion Analysis	
An analysis to determine the movement and concentration of a radioactive plume. *(see Atmospheric Transport and Diffusion)*	In a Level 3 PRA, atmospheric transport and diffusion (ATD) models are used in the consequence calculations. ATD models range from simple straight-line, steady-state Gaussian dispersion models, which calculate ground-level instantaneous and time-integrated airborne concentrations in the plume, to more sophisticated models that allow terrain-dependent effects and temporal variations in wind speed and atmospheric stability. Probabilistic consequence modeling codes typically include sampling of meteorological data from a site-specific annual database of hourly weather data to determine appropriately weighted scenarios of plume transport under different weather conditions to provide probabilistic results.
At-Power	
The state of operation in which the reactor is critical and	A PRA models the different plant operating states (POSs), generally defined as at-power, low-power, and shutdown. These POSs are distinguished in the PRA model because the plant responses (e.g., accident sequences) are different.

TERM AND DEFINITION	DISCUSSION
producing power from a range of states between full and low power. *(see Full Power, Low Power/ Shutdown, Plant Operational State)*	At-power plant status includes all power levels above low-power. In this instance, the reactor is producing a significant amount of power from fission in the core fuel, above and beyond the decay heat levels. The safety systems are on automatic actuation and not blocked or defeated (as they might be in low-power and shutdown states). The support systems are aligned in their normal configuration (e.g., electric power is being drawn from the grid). These are all important initial conditions for PRA modeling. The borderline between at-power and low-power and shutdown depends on plant evolution (the changes in configuration used to bring the plant down from full power or up from low-power and shutdown) and is typically on the order of 15%-25% of full power. Historically, the term "full power" was used for all power levels between low-power and 100% power. This has been modified such that at-power now refers to intermediate power levels ranging from low-power and up to and including 100% power, while "full power" is reserved for just 100% reactor power. The figure below is a pictorial representation of the different plant operating states.  Note: The overlap shows that PRAs have used different denominations for At-power and Low-power. The ASME/ANS PRA Standard (Ref. 2) defines at-power as "those plant operating states characterized by the reactor being critical and producing power, with automatic actuation of critical safety systems not blocked and with essential support systems aligned in their normal power operation configuration."
Availability (Unavailability)	
The probability that a system, structure, or component of interest is functional at a given point in time. (*see Reliability)*	In a PRA, unavailability is one of the attributes of a system, structure, or component that may affect the plant's response to an initiating event. Unavailability is the complement of availability (i.e., shortfall between availability and unity). In the ASME/ANS PRA Standard (Ref. 2), unavailability is defined as "the probability that a system or component is not capable of supporting its function including, but not limited to, the time it is disabled for test or maintenance." The definition provided is based on the definition in National Fire Protection Association (NFPA)-805 (Ref.11).
Base PRA, Baseline PRA	
(see PRA)	The terms base PRA and baseline PRA represent a specific type of PRA and are defined under "PRA."

TERM AND DEFINITION	DISCUSSION
Basic Event	
An element of the PRA model for which no further decomposition is performed because it is at the limit of resolution consistent with available data. *(see Component, Fault Tree)*	In a PRA, in developing the fault trees, the basic events represent those failures for which there is available data, and as such, represent the termination of a branch of the fault tree. There are typically two types of failures (or basic events): equipment unavailability and human errors. The term basic event can have other (more specific) definitions, as stated below: • "An event in a fault tree model that requires no further development, because the appropriate limit of resolution has been reached." (Ref.2) • The individual events that collectively form a cutset, which is a combination of failures needed to result in the occurrence of a condition of interest (e.g., accident sequence, system failure). In the quantification process of the PRA, the model uses or manipulates the basic events to model the core damage frequency. At this point, the initiating event is part of the quantification process; consequently, an initiating event is sometimes referred to as a basic event. The following figure is an example of a basic event: 
Basic Event Failure Probability	
(see Probability)	The term basic event failure probability is a specific type of failure probability and is defined under "Probability."
Bathtub Curve	
Graphical representation of failure rate time dependency in the life of a typical component. *(see Aging)*	In a PRA, the mid-life or constant failure rate stage in the life of a component is the one typically modeled. However, the life of certain types of components is often considered to have three stages of failure rate behavior: I) burn-in (or infant mortality) stage, characterized by failure rates decreasing with time, II) mid-life or constant failure rate stage, and III) wearout stage in which failure rates increase with time. These three stages together form a curve that looks like the cross-section of a bathtub. The following figure represents a bathtub curve:

TERM AND DEFINITION	DISCUSSION
	 • Region I – The failure rate is usually high at the beginning of a component's life because of defects. It decreases if the component survives. • Region II – The failure rate becomes stable and remains constant in the middle of the component's life. • Region III – The failure rate increases toward the end of the component's life.
Bayesian Analysis, Bayesian Estimation, Bayesian Statistics	
Type of data analysis in which an initial estimate about a parameter value is combined with evidence to arrive at a more informed estimate. *(see Frequentist, Bayesian Update)*	In a PRA, Bayesian analysis is commonly used in the computation of the frequencies and failure probabilities in which an initial estimation about a parameter value (e.g., event probability) is modified based on actual occurrences of the event. The initial parameter value may have a probability distribution associated with it. Thus, the event probability to be determined is based on a belief, rather than on occurrence ratios. Any actual occurrence or lack of occurrence of the event is used to measure consistency with the original hypothesis, which is then modified to reflect this evidence. The modified or updated hypothesis is the most meaningful estimate of the parameter. The initial hypothesis is called the "prior". The prior should be as relevant as possible to the parameter value in question. The final parameter estimate will depend on the prior chosen to a certain extent. For example, industry average (generic) data may be used as the prior. Noninformative priors can be used if no basis for making an educated guess exists. The prior is modified by actual observations of the event occurrences (e.g., plant-specific data) to calculate the "posterior" or best estimate of the parameter. The process is called "Bayesian update." Bayesian analysis is used when occurrences of an event are sparse or nonexistent, such that probability estimates using the proportion of actual event occurrences (frequentist approach) are not reliable. It also can be used to produce a probability distribution for the parameter in question. In risk analysis, both frequentist and Bayesian analysis may be used. Frequentist analysis is used when the occurrence data is sufficiently abundant, Bayesian analysis is used otherwise. The terms Bayesian analysis, Bayesian estimation, and Bayesian statistics are used interchangeably.
Bayesian Estimation	
(see Bayesian Analysis)	The term Bayesian estimation has the same meaning as Bayesian analysis and is defined the same as the term "Bayesian Analysis."
Bayesian Statistics	
(see Bayesian Analysis)	The term Bayesian statistics has the same meaning as Bayesian analysis and is defined the same as the term "Bayesian Analysis."

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Bayesian Update	
Modification of a probability (frequency) of an event by incorporating additional observations of event occurrence. *(see Bayesian Analysis)*	In a PRA, Bayesian update is the process of using the Bayesian approach to incorporate new information and combine it with existing information to come up with a new characterization of the state-of-knowledge about a parameter. It is used to incorporate new information as it becomes available or to account for plant-specific information when primarily relying on generic data (or some other initial guess) to generate event failure probabilities or frequencies. For example, an initial guess of a pump failure rate is based on industry generic data. Observations of a certain number of failures (or no failures) of that type of pump over a certain time period in the plant are used in the Bayesian update to obtain a better estimate of the pump failure rate in that particular plant. Industry generic failure rates might be used as the starting estimate (called the prior). These would be combined with the observed occurrences of failure of such components to calculate the updated failure rates. A similar process may be used to obtain plant-specific initiating event frequencies, by starting from generic data and updating with plant-experienced occurrences to arrive at the updated initiating event frequencies.
Best Estimate	
Approximation of a quantity based on the best available information. *(see Mean, Point Estimate)*	In a PRA, the term best estimate is not generally used. The term is sometimes mistakenly used in place of point estimate or mean value to characterize a parameter value estimate used in a PRA. The term is used for deterministic calculations, in which best estimate designates inputs or results obtained by using the most realistic assumptions available to the analyst (i.e., not biased by conservatism or optimism). For example, best estimate codes may be used to deterministically predict the pressure rise in containment from a hydrogen burn.
Beyond-Design-Basis Accident	
A postulated accident that is more severe than those accidents used to establish the design of a nuclear facility. *(see Design-Basis Accident, Severe Accident)*	In a PRA, beyond-design-basis accidents (BDBAs) are a major focus of the analysis. For example, PRAs for currently operating light-water reactors (LWRs) have focused almost exclusively on BDBAs. Recent PRAs for proposed high-temperature graphite reactors have included design-basis accidents and anticipated occurrences in the analysis. A nuclear facility must be designed and built to withstand a design-basis accident (DBA) without threatening public health and safety. However, the nuclear facility is not necessarily designed to withstand BDBAs. Therefore, an important role of PRA is to determine how a nuclear facility will behave in a BDBA and analyze the adequacy of the systems, structures, and components that are included to ensure public health and safety are maintained. Although BDBAs might exceed the design envelope, they do not necessarily result in significant core damage. Those BDBAs that do result in significant core damage are termed severe accidents. All severe accidents are by definition BDBAs since their challenges exceed the design envelope of the plant. The NRC Web site Glossary (Ref. 36) defines the term beyond-design-basis accident as “a technical way to discuss accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely. (In that sense, they are considered beyond the scope of design-basis accidents that a nuclear facility must be designed and built to withstand.) As the regulatory process strives to be as thorough as possible, beyond-design-basis accident sequences are analyzed to fully understand the capability of a design.”
Beyond-Design-Basis Event	
An event more severe than the	In a PRA, beyond-design-basis events (BDBEs) represent conditions beyond the plant design envelope and, therefore, exceed the already considered anticipated transients (e.g., tripping of

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events for which the nuclear power plant was designed to withstand and specified in the safety analysis. *(see Design-Basis Event, Severe Accident)*	turbine generator), anticipated operational occurrences (AOOs), design-basis accidents (DBAs), and design-basis natural phenomena. A BDBE challenges the systems, structures, and components that are included in the design to ensure public health and safety. Generally, BDBEs have been excluded from the design-basis because they were considered to have a low probability of occurrence. Extremely unlikely earthquakes or aircraft impacts would be considered beyond-design-basis events which, while not considered in the nuclear plant design, can be analyzed in the PRA to determine how the plant would respond given such an event.
Bin, Binning	
A group of initiating events or accident sequences with similar characteristics.	In a PRA, binning is a process used to group similar types of initiating events, accident scenarios, or sequences together to simplify the analysis. The term bin generally is associated with binning event tree sequences into groups that have similar characteristics and lead to similar end states called plant damage states. Initiating events also are grouped by similar characteristics (e.g., failure of a main steam isolation valve and failure of a feedwater pump are generally grouped (or binned) into a loss of feedwater initiator group). Bin is the actual group and binning is the process.
Birnbaum Importance	
(see Importance Measure)	The term Birnbaum importance is one type of importance measure and is defined under “Importance Measure.”
Bounding Analysis	
An analysis that uses assumptions such that the assessed outcome will meet or exceed the maximum severity of all credible outcomes, both in magnitude as well as frequency. (*see Conservative Analysis)*	In a PRA, a bounding analysis of a contributor or parameter may be performed to bound the risk or to screen the PRA item as a potential contributor to risk. When used for screening, the bounding analysis demonstrates that the item can be omitted from the PRA model because, even in the worst case, the impact on calculated risk is insignificant. As discussed in NUREG-1855 (Ref. 62), in the context of a specific PRA scope or level of detail item, a bounding analysis includes the worst credible outcome of all known possible outcomes that result from the risk assessment of that item. The worst credible outcome is the one that has the greatest impact on the defined risk metric(s). Thus, a bounding probabilistic analysis must be bounding both in terms of the potential outcome and the likelihood of that outcome. Consequently, a bounding analysis considers both the frequency of the event and the outcome of the event. NUREG-1855 states that if a bounding analysis is being used to bound the risk (i.e., determine the magnitude of the risk impact from an event), then both its frequency and outcome must be considered. However, if a bounding analysis is being used to screen the event (i.e., demonstrate that the risk from the event does not contribute to the defined risk metric(s)), then the event can be screened based on frequency, outcome, or both, depending on the specific event. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Bridge Event Tree	
(see Bridge Tree)	The term bridge event tree has the same meaning as bridge tree and is defined under “Bridge Tree.”

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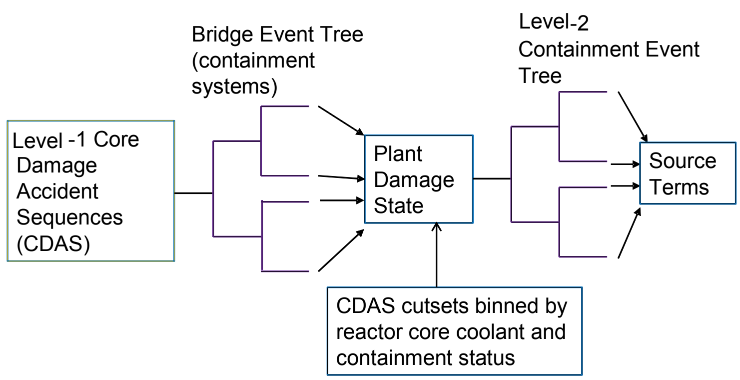
Bridge Tree, Bridge Event Tree	
An event tree used to transfer information from one analysis stage to another in a manner that ensures the critical information is preserved. *(see Containment Event Tree, Event Tree, Accident Progression Event Tree)*	In a PRA, the most common use of bridge trees is in linking the core damage states, which are the end points of the Level 1 PRA analysis, with the plant damage states. The plant damage states often are used as the starting point of the accident progression event tree or the containment event tree (i.e., Level 2 analysis). In this case, the bridge trees provide the information on the status of systems that were not relevant for determining core damage, but that can influence further accident progression. The terms bridge tree and bridge event tree are similar in meaning and often correctly used interchangeably. The figure below is an example of a bridge tree: 
Capability Categories	
Categories used to indicate different levels of detail, plant-specificity, and realism in defining technical requirements for an acceptable PRA.	For a PRA used with a risk-informed application, the level of detail, plant specificity, and realism needs to be commensurate with the scope of the specific application under consideration, as recognized in NRC Regulatory Guide 1.200 (Ref. 91). Capability categories are used in the ASME/ANS PRA Standard (Ref. 2) to recognize that the various elements in the PRA model can be constructed to different levels of detail, levels of plant-specificity, and levels of realism. The PRA standard defines three categories of the acceptable level of detail, plant-specificity and realism, starting at the minimal for capability Category I, and increasing through Category II, and Category III. The use of capability categories supports the concept that a PRA needs only to have the scope and level of detail necessary to support the application for which it is being used, but it always needs to be technically acceptable. As stated in the ASME/ANS PRA Standard (Ref. 2), “as the capability category increases, the depth of the analysis required also increases.” As further stated in the ASME/ANS PRA Standard, “the level of conservatism may decrease as the capability category increases and more detail and more realism are introduced into the analysis. However, this is not true for all requirements and should not be assumed.”
Chemical Element Group	
A group of radioactive materials with similar physical and chemical properties used to simplify the estimate for offsite health effects. *(see Source Term)*	In a PRA, the source term used to characterize the radioactive material release is based on the defined chemical element groups. During a core damage accident, the number of different radioactive materials released from the fuel, reactor vessel, and containment to the environment can be quite large. The number of radioactive materials considered can be reduced to a manageable size by grouping those with similar physical and chemical properties. For example, in NUREG-1150 (Ref. 51) the 60 radionuclides considered in the consequence calculation were not dealt with individually in the source term calculation. Since some different elements behave similarly enough both

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	chemically and physically that they can be considered together, the 60 isotopes were placed in nine radionuclide groups. These nine groups were treated individually in the source term analysis.
Chronic Exposure	
(see Exposure)	The term chronic exposure is a type of exposure and is defined in the discussion under "Exposure."
Cloudshine	
Direct external exposure from radioactive material in the atmosphere. *(see Exposure Pathways, Water Immersion, Groundshine, Inhalation, Ingestion, Skin Deposition)*	In a Level 3 PRA, cloudshine, also referred to as air immersion, is one of the assumed pathways by which an individual can receive doses in the consequence calculation. The pathways of exposure include: (1) direct external exposure from radioactive material in a plume, principally due to gamma radiation (air immersion or cloudshine), (2) direct exposure from radioactive material in contaminated water given to an individual immersed in the water, (3) exposure from inhalation of radioactive materials in the cloud and resuspended material deposited on the ground, (4) exposure to radioactive material deposited on the ground (groundshine), (5) radioactive material deposited onto the body surfaces (skin deposition), and (6) ingestion from deposited radioactive materials that make their way into the food and water pathway.
Cohort	
A group of individuals that is defined by some statistical or demographic factor. (*see Emergency Response)*	In the emergency response modeling of a Level 3 PRA, a cohort is a subset of the offsite population that mobilizes or moves differently from others. The planning and analysis of the offsite response to a severe accident is driven by the demographics of the surrounding population (i.e., the attributes (e.g., age, location) of the various cohorts (e.g., school children, hospital patients, prisoners) and their potential for being exposed to severe health effects).
Collective Dose	
(see Dose)	The collective dose is a summation of dose that is defined under "Dose."
Committed Dose Equivalent	
(see Dose Equivalent)	The committed dose equivalent is one measure of dose that can be used to calculate the effect of radiation received by an individual and is defined under "Dose Equivalent."
Committed Effective Dose Equivalent	
(see Dose Equivalent)	The committed effective dose equivalent is one measure of dose that can be used to calculate the effect of radiation received by an individual and is defined under "Dose Equivalent."
Common Cause Component Group	
Similar components that are modeled as a group because they are subject to failure by a common cause. *(see Common-Cause Failure)*	In a PRA, one failure mechanism of a component may be from a common cause that also fails other components. A common cause component group is a collection of like components considered to have the potential to fail by the same cause. For example, redundant diesel generators in a nuclear power plant are modeled as having the potential to fail by common cause (as well as independently) and form a common cause component group. Turbine-driven and motor-driven pumps in a secondary cooling system may form a common cause component group (failures because of a common environment), while at the same time the motor-driven pumps may form a separate common cause group because of separate common cause failures.

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	Common cause failure among like components usually is not modeled to occur across system boundaries. This is because the operating regime may be different and thus failure rates may be different. An exception may be in external events, such as seismic events, in which components may be subject to similar stresses.
Common-Cause Failure	
A failure of two or more structures, systems, or components as a result of a single shared cause. *(see Common-Mode Failure, Failure Mode)*	In a PRA, common-cause failure (CCF) is a special form of dependent failure in which the failure of the structure, system, or component (SSC) has occurred from the same fault. CCF faults generally reflect errors occurring as a result of a common manufacturer, environment, maintenance, etc. The CCF term is often incorrectly used interchangeably with common-mode failure (CMF). CCF only accounts for the SSCs failing because of the same, single cause, not if they ultimately fail in the same manner (or in the same mode), which is CMF. In data provided to quantify CCF events, the failure mode is usually presented (i.e., failure to start, fail to run), and the cause is not always provided about why the failure mode occurs. There could be multiple causes lumped into the data presentation for a given failure mode. Thus, the available failure data dictate whether the PRA model is modeling CCF or CMF. To illustrate the relationship between CCF and CMF, consider potential causes of failure for emergency diesel generators (EDGs) as shown in the figure below. Potential failure causes include a plugged radiator, a failed load sequencer, bad fuel oil, or faulty bearings. As indicated in the figure below, each of these causes can result in failure of multiple diesel generators in either the same failure mode or in different failure modes. Diesel failure modes included in this example are fails to start (FTS) and fails to run (FTR).

Failure Cause	Failure Mode		Basic Event	Comments	CCF Types
	EDG A	EDG B			
Plugged radiator	FTS	FTR	CCF-DG-AB-FTS/R-1	Same cause results in a different failure mode of each DG	CCF without CMF
Failed load sequencer	FTR	FRT	CCF-DG-AB-FTR	Same cause results in the same failure mode of both EDGs	CCF with CMF
Bad fuel oil	FTS	FTS	CCF-DG-AB-FTS	Same cause results in the same failure mode of both EDGs	CCF with CMF
Faulty Bearings	FTS	FTR	CCF-DG-AB-FTS-R2	Same cause results in a different failure mode of each DG	CCF without CMF

The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).

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Common-Mode Failure	
A failure of two or more structures, systems, or components in the same manner or mode as the result of a single shared cause. *(see Common-Cause Failure, Failure Mode)*	In a PRA, common-mode failure (CMF) is a special form of dependent failure that reflects (1) a common manner of failure (e.g., failure to start, failure to run) and (2) failure from a common cause. Consequently, CMF is actually a type of common-cause failure (CCF) in which the SSCs fail in the same way and from the same cause. CMF and CCF are often incorrectly used interchangeably. However, CCF only addresses the cause of the failure, while CMF addresses both the cause and the manner. In data provided to quantify CCF or CMF events, the failure mode is usually presented (i.e., fails to start (FTS), fails to run (FTR)), and the cause is not always provided about why the failure mode occurs. There could be multiple causes lumped into the data presentation for a given failure mode. Thus, the available failure data dictate if the PRA model is modeling CCF or CMF.

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	Consider the figure displayed in the discussion section for CCF. Potential failure modes for emergency diesel generators are FTS and FTR. Potential failure causes include a plugged radiator, a failed load sequencer, bad fuel oil, or faulty bearings. As indicated in the figure for CCF, each of these causes can result in failure of multiple diesel generators in either the same failure mode or in different failure modes. Examples of CMF are shown in the comment column under the term "Common-Cause Failure." The definition provided was based on the definition in the IAEA Safety Glossary (Ref. 7).
Complementary Cumulative Distribution Function	
(see Cumulative Distribution Function)	The term complementary cumulative distribution function is a type of cumulative distribution function and is defined under "Cumulative Distribution Function."
Completeness Uncertainty	
(see Uncertainty)	The term completeness uncertainty is related to epistemic uncertainty and defined under "Uncertainty."
Component	
A part of a system in a nuclear power plant. *(see Basic Event)*	In a PRA, the plant is usually modeled at the component level. The ASME/ANS PRA Standard (Ref. 2) defines a component as "an item in a nuclear power plant, such as a vessel, pump, valve, or circuit breaker." Basic events are associated with individual components, such that different basic events will be associated with different failure modes of a particular component.
Conditional Containment Failure Probability	
(see Conditional Probability)	The term conditional containment failure probability is a type of conditional probability and is defined under "Conditional Probability."
Conditional Core Damage Probability	
(see Conditional Probability)	The term conditional core damage probability is a type of conditional probability and is defined under "conditional probability."
Conditional Large Early Release Probability	
(see Conditional Probability)	The term conditional large early release probability is a type of conditional probability and is defined under "Conditional Probability."
Conditional Probability (Containment Failure, Core Damage, Large Early Release)	
Probability of occurrence of an event, given that a prior event has occurred. *(see Probability)*	In a PRA, a conditional probability can be calculated for containment failure, core damage, and large early release given the knowledge of a variety of prior events have occurred. Examples include: • Conditional containment failure probability can be calculated given that a particular accident type (large loss-of-coolant accident, transient) has occurred. • Conditional core damage probability can be calculated given an initiating event (a plant upset causing a demand for shutdown) has occurred, or given that a certain plant system has been taken out of service.

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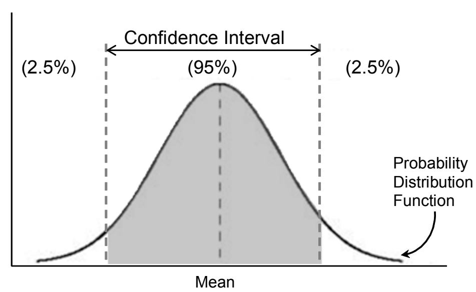
	• Conditional large early release probability can be calculated given that a core damage event has occurred, or given that a bypass sequence has occurred. Conditional probability exists in other contexts. For example, seismic fragility is the conditional probability of a component, structure, or system failure given a seismic motion of a certain magnitude.
Confidence Interval	
A range of values that has a specified likelihood of including the true value of a random variable. *(see Uncertainty Interval)*	In a PRA, a confidence interval is sometimes used to describe the uncertainty of a parameter input. However, confidence intervals cannot be propagated through the PRA model. A confidence interval with a confidence level *p* is defined such that the probability that the true value of a random variable contained within that interval *p* can be stated with a specified likelihood. The confidence level can take a specified value, with the most common being 95% or 99%. The following figure shows a 95% confidence interval. In this case, 2.5% of the probability distribution is greater than the 95% confidence interval (shaded area under the probability distribution function curve), while 2.5% of the probability distribution is less than the 95% confidence interval.



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Configuration Risk Profile	
(see PRA Configuration Control)	The configuration risk profile is related to configuration control and is defined under "PRA Configuration Control."
Consequence	
(see Accident Consequence)	In the context of a PRA, the term consequence has the same meaning as accident consequence, which is defined under "Accident Consequence."
Consequence Analysis	
(see Accident Consequence Analysis)	In the context of a PRA, the term consequence analysis has the same meaning as accident consequence analysis, which is defined under "Accident Consequence Analysis."
Consequential Steam Generator Tube Rupture, Induced Steam Generator Tube Rupture	
A break or breach in a steam generator tube caused by the consequences of an accident. *(see Steam Generator Tube Rupture, Containment Bypass)*	In a PRA for a pressurized-water reactor, steam generator tube ruptures (SGTRs) are modeled either as an initiating event or a subsequent failure as part of an accident sequence. If the SGTR occurs randomly while the plant is operating, it is an initiating event modeled in the PRA. However, if the SGTR occurs because of excessive conditions produced as a result of the accident, it is considered to be a consequential or induced SGTR and is modeled in the PRA as an event in an accident sequence. These excessive conditions generally involve high pressures or high temperatures that could rupture a steam generator tube. For example, this might occur if the steam generator were to boil dry (steam generator dryout).

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	Accidents involving SGTRs are modeled in PRAs because it allows reactor coolant to flow from the reactor vessel to the secondary side of the steam generator. As such, an SGTR can become a significant contributor to risk because it can serve as a possible mechanism for radioactive material transport to the environment. There is the potential that if a tube bursts while a plant is operating, radioactivity from the primary coolant system could escape directly to the atmosphere through the safety valves on the secondary side. This scenario is referred to as containment bypass.
Conservative Analysis (Demonstrably)	
An analysis that uses assumptions such that the assessed outcome is meant to be less favorable than the expected outcome. *(see Bounding Analysis)*	In a PRA, conservative analysis may be performed to show that a certain contributor is not significant to risk, and thus, resources do not need to be spent on more accurate modeling. A conservative analysis provides a result that may not be the worst result of a set of outcomes, but produces a quantified estimate of a risk metric that is significantly greater than the risk metric estimate obtained by using the most realistic information obtainable (i.e., a realistic analysis). Therefore, in a PRA, if there is not much change in risk with the contributor in question set at an unfavorable value (as opposed to its most favorable value), then the contributor can be omitted from the analysis. For example, a licensee's request for change in technical specifications may show that the requested change will result in acceptable risk increases, even with pessimistic assumptions associated with the proposed change. If that is the case, then it may be acceptable not to perform a realistic assessment of the proposed change since it may involve detailed and time-consuming modeling. Conservative analysis also may be used to demonstrate that an item that is not modeled in the PRA has negligible impact on risk and therefore can be justifiably neglected. A conservative analysis provides a result that may not be the worst result of a set of outcomes, but produces a quantified estimate of a risk metric that is significantly greater than the risk metric estimate obtained by using a best-estimate evaluation. A conservative analysis should be distinguished from a bounding analysis in which assumptions and parameters are chosen such that the impact on risk is as detrimental as possible; therefore, bounding analysis is a special case of conservative analysis. For example, for a conservative analysis a human error probability event can be set to a value that is unlikely to be exceeded, whereas for a bounding analysis, the error probability would be set to 1.0. Conservative analyses, then, include a spectrum of assessments with results less favorable than those of realistic analysis all the way to bounding assessments with the most unfavorable results. Examples of areas in which conservative analyses can be used in Level 1 risk assessments are initiating events, success criteria, thermal-hydraulics, and human error probabilities. The terms conservative and demonstrably conservative are used interchangeably. The definition is based on the ASME/ANS PRA Standard (Ref. 2), which defines demonstrably conservative analysis as one "that uses assumptions such that the assessed outcome will be conservative relative to the expected outcome."
Containment Building	
(see Containment)	The term containment building has the same meaning as containment and is defined under "Containment."
Containment Bypass	
A flow path that allows the unintended release of radioactive material directly to	In a PRA, the potential for containment bypass is modeled and such a bypass often is determined to be a significant risk contributor. A containment bypass circumvents the containment's design function, which is to confine and reduce a release of radioactive material. Therefore, a containment bypass can lead to a significant release of fission products in the event of a core damage accident. A containment bypass can result from the failure of various containment components so that a direct path to the environment is opened. For example, a

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the environment, bypassing the containment. *(see Containment Failure, Containment Isolation Failure, Interfacing Systems Loss-of-Coolant Accident)*	containment bypass can result from an interfacing-system loss-of-coolant accident (i.e., an accident in which a high-pressure system containing fission products leaks into a lower-pressure system, part of which is outside of containment). For example, a steam generator tube rupture in a core damage accident provides a pathway for the fission products in the high-pressure primary system to enter the low-pressure side of the steam generator, which has relief valves outside of containment. Containment bypass is distinct from containment isolation failure in which the containment is not acceptably leak-tight. The definition provided is based on the definition found in the ASME/ANS PRA Standard (Ref. 2).
Containment Capacity	
The ability of the containment to withstand the challenges that result from accidents. *(see Containment, Containment Capacity Analysis, Containment Pressure Boundary)*	In a Level 2 PRA, the containment capacity is evaluated so that it can be compared against the postulated challenges to the containment that could result from a severe accident, both pre- and post-core damage. As such, the containment performance in response to severe accident conditions can be assessed. The containment capacity is the ability of the structures, systems, and components that make up the containment pressure boundary to withstand postulated loads and challenges.
Containment Capacity Analysis	
A calculation that estimates the ability of the containment to withstand the challenges that result from accidents. *(see Containment Capacity)*	In a Level 2 PRA, the containment capacity analysis involves selecting a method or methods to evaluate the structural capacity to withstand challenges (e.g, high pressure, temperature, etc.) of the structures, systems, and components (SSCs) that make up the containment pressure boundary. A plant-specific containment capacity analysis usually involves developing and solving a computer model of the relevant SSCs using finite element analysis or similar techniques. In the simplest case, the containment capacity can be inferred from that of a previously analyzed similar containment of a reference plant.
Containment Event Tree	
A logic diagram that graphically represents the status of the containment and containment equipment when subjected to severe accident loads. *(see Accident Progression Event Tree, Event Tree)*	In a PRA, a containment event tree (CET) begins with the onset of core damage and progresses through a limited number of branches that depict the various scenarios of the containment and containment equipment performance when subjected to severe accident loads (e.g., high temperatures, pressures). As noted in NUREG-1150 (Ref.51), an accident progression event tree (APET) is a more detailed representation of the containment response to severe accident loads. The APET includes the interaction of phenomena, the availability of equipment, and the performance of operators. The end states of both the CET and the APET are: no containment failure, various containment failure modes, or containment bypass.

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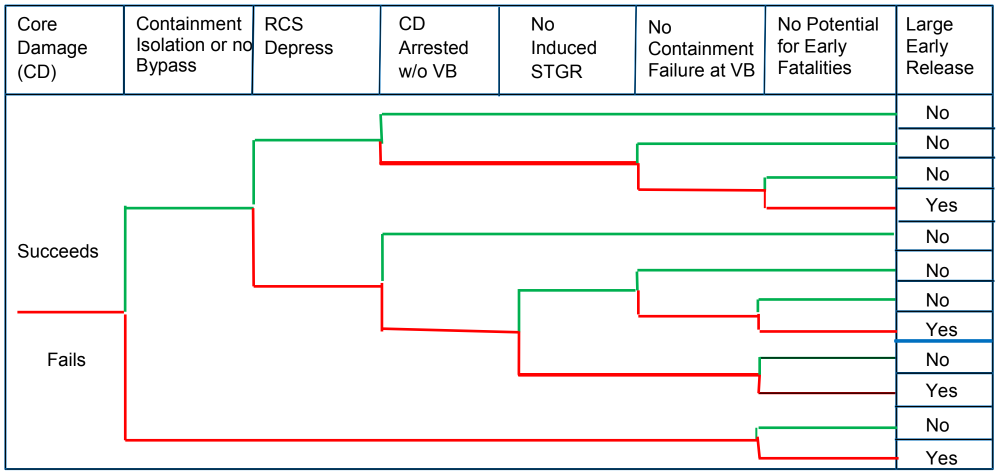
	The figure below represents a containment event tree with the following acronyms: Core Damage (CD), Reactor Coolant System depressurization (RCS Depress), Vessel Breach (VB), Steam Generator Tube Rupture (SGTR). 
Containment Failure Mode	
The various ways in which the ability of the containment to prevent radioactive material release is compromised. *(see Containment Failure, Containment Bypass, Containment Isolation Failure)*	In a PRA, the modes of containment failure define the manner in which containment integrity is lost (i.e., the way a radioactive material release pathway from inside the containment to the environment is created). Containment failure mode encompasses both structural failures of containment induced by containment challenges when they exceed containment capability, as well as the failure modes of containment induced by human failure events, isolation failures, or bypass events such as interfacing-systems loss-of-coolant accidents. The definition provided is based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Containment Failure Probability	
(see Probability)	The term containment failure probability is a type of failure probability that is computed based on the likelihood of containment failure and is discussed under the discussion for the term "Probability."
Containment Failure (Early, Late)	
Loss of integrity of the containment from a core damage accident that is expected to result in an unacceptable release of radioactive materials. *(see Containment, Containment Bypass,*	In a PRA, determining when and if the containment fails or is bypassed during a severe accident is very important from a risk perspective. If the containment pressure boundary remains leak-tight, the offsite consequence will be low. Conversely, if the containment fails or is bypassed, then the consequence to the surrounding population can be potentially high. For specific containments there can be selected severe accident scenarios in which the containment fails before fission products have penetrated the primary system. If the accident is successfully arrested at this point, no release will occur. However, usually containment failure represents the failure of the final barrier preventing a radioactive material release. Containment failure is often categorized as early or late. Early containment failure occurs in a timeframe before the surrounding population within 1 mile of the site boundary can be evacuated. Late containment failure occurs in a timeframe that allows the surrounding population from 1 to 10 miles to be evacuated.

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Containment Pressure Boundary)	Containment bypass failures (e.g., interfacing-system loss-of-coolant accidents) occur in the early timeframe but usually are categorized separately from early structural failures of the containment. The definition is derived from the ASME/ANS PRA Standard (Ref. 2).
Containment Integrity	
The ability of the containment to function as a barrier to prevent release of radioactive materials as a result of an accident. *(see Containment Failure Mode)*	In a Level 2 PRA, an important concern is the potential loss of containment integrity. Containment integrity depends on the structures, systems, and components of the reactor containment pressure boundary that perform the containment function. Maintaining containment integrity largely depends on the individual containment design and the particular phenomena or load that challenges the integrity of the containment. Examples of particular severe accident challenges to the containment integrity include overpressure, internal missiles, external missiles, melt-through, and bypass.
Containment Isolation Failure	
A failure in the piping, valves, or actuators that isolate the containment. *(see Containment Bypass, Containment Failure Mode)*	In a PRA, containment isolation failures are one of the containment failure modes considered in a Level 2 analysis. Containment isolation is provided to prevent or limit the escape of fission products that may result from postulated accidents. In a containment isolation failure, fission products can pass to the environment through the containment because the containment is not properly isolated (i.e., not acceptably leak-tight). In some severe accident scenarios, an accident management strategy, referred to as containment venting, may be used. Containment venting involves a deliberate breach of containment isolation by the plant operators who open a controlled, filtered or unfiltered, pathway from the containment to the environment to prevent an uncontrolled overpressure failure of the containment. The containment isolation system consists of the piping, valves, and actuators that are designed so that fluid lines penetrating the containment boundary are isolated in the event of an accident.
Containment Pressure Boundary	
Those parts of the reactor containment that sustain loading and provide a pressure boundary in the performance of the containment function. *(see Containment)*	In a Level 2 PRA, the evaluation of containment integrity is an evaluation of the structures, systems, and components of the reactor containment pressure boundary that perform the containment function (i.e., that form the containment system). As stated in NUREG-0800 (Ref. 44), the reactor containment system design must include the functional capability of enclosing the reactor system and of providing a final barrier (boundary) against the release of radioactive fission products in case of postulated accidents. Leak-tightness of the containment is ensured by a continuous pressure boundary consisting of nonmetallic seals and gaskets and metallic components that are either welded or bolted together. Each containment also includes numerous access and process penetrations that complete the pressure boundary. The definition provided is derived from Chapter 6 of NUREG-0800 (Ref. 46).
Containment Structure	
(see Containment)	The term containment structure has the same meaning as containment and is defined under “Containment.”

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Containment, Containment Building, Containment Structure	
A physical structure surrounding a reactor that is designed to prevent or control the release of radioactive material. *(see Containment Capacity, Containment Failure, Containment Failure Mode, Containment Integrity, Containment Pressure Boundary)*	In a Level 2 PRA, the ability of the containment (containment building or containment structure) to contain fission products that have escaped from the reactor is analyzed to estimate the limits of the containment's capacity. A containment, containment building, or containment structure, in its most common usage, is a steel or reinforced concrete structure enclosing a nuclear reactor designed to contain the escape of radiation to the environment. The containment is the final barrier to radioactive material release. Containments are designed to remain intact when subject to the pressure and temperature loads from design-basis accidents (DBAs). Moreover, because of safety factors built into containment designs, they are predicted to fail at pressures and temperatures (from core melt accidents) that are significantly higher than those of DBAs. The NRC Web site Glossary (Ref. 36) defines the term containment building as an "air-tight building, which houses a nuclear reactor and its pressurizer, reactor coolant pumps, steam generator, and other equipment or piping that might otherwise release fission products to the atmosphere in the event of an accident. Such buildings usually are made of steel-reinforced concrete." The NRC Web site Glossary (Ref. 36) also defines the term containment structure as "a gas-tight shell or other enclosure around a nuclear reactor to confine fission products that otherwise might be released to the atmosphere in the event of an accident. Such enclosures are usually dome-shaped and made of steel-reinforced concrete."
Core Damage	
Sufficient damage that could lead to a release of radioactive material from the core that could affect public health. *(see Core Melt, Core Damage Frequency, Core Damage Probability)*	In a PRA, the potential for core damage is evaluated in the Level 1 part of the analysis. Specifically, a Level 1 PRA calculates the core damage frequency given the design and operation of the plant. In this context, core damage in a Level 1 PRA is actually the onset of core damage; that is, being the onset of sufficient damage to the core that (1) if not immediately arrested could potentially result in a release of radioactive material from the core, and (2) if released from the vessel and containment, could result in offsite public health effects. In deterministic analyses, quantitative criteria often are used to define the onset of core damage (e.g. a peak clad temperature of 2,200 degrees Fahrenheit). The ASME/ANS PRA Standard (Ref. 2) defines core damage as "uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects." The terms core damage and core melt are sometimes incorrectly used as synonyms. However, core melt occurs after the onset of core damage. Core damage does not necessarily indicate that the reactor fuel has melted, only that radioactive material could be released from the core into the reactor vessel. An illustration differentiating the concepts of core damage, core melt, and their timing is provided below.

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	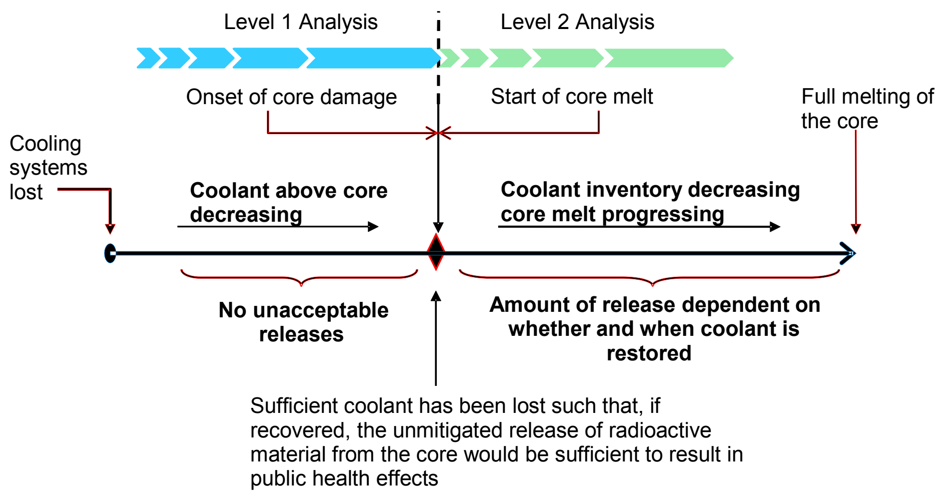
Core Damage Frequency	
(see Frequency)	The term core damage frequency is a type of frequency used in PRA and is defined under "Frequency."
Core Damage Probability	
(see Probability)	The term core damage probability is a type of probability used in PRA and is defined under "Probability."
Core Melt	
Damage beyond the onset of core damage that could progress to a complete melting of the core. *(see Core Damage, High- Pressure Melt Ejection, Reactor Core)*	In a PRA, the potential for core melt is evaluated in the Level 2 part of the analysis. A Level 1 PRA calculates the onset of core damage, while the Level 2 evaluates the effects starting with the onset of core damage and then progressing to a complete melting of the core. The evaluation considers the different releases that can occur during the core melt progression and the frequencies of the associated accident progressions. The terms core melt and core damage are sometimes incorrectly used as synonyms. However, core damage entails only the potential release of radioactive material from the core into the reactor vessel, not necessarily the melting of any portion of the reactor core. An illustration differentiating the concepts of core damage, core melt, and their timing is provided under the discussion for the term "Core Damage." The NRC Web site Glossary (Ref. 36) defines a core melt accident as "an event or sequence of events that result in the melting of part of the fuel in the reactor core."
Cumulative Distribution Function (Complementary)	
A function that provides the probability that a parameter is less than or equal to a given value. *(see Probability Distribution)*	In a PRA, the cumulative distribution function is often used to present the results of the analysis. The cumulative distribution function gives the probability that the random variable does not exceed a specified value. The cumulative distribution function is the integral of the probability distribution functions. The cumulative distribution function adds up the probabilities of occurrence of all possible parameter values less than the specified value, as represented by the probability distribution function of the parameter. The following graphs illustrate the cumulative distribution function and the probability distribution function.

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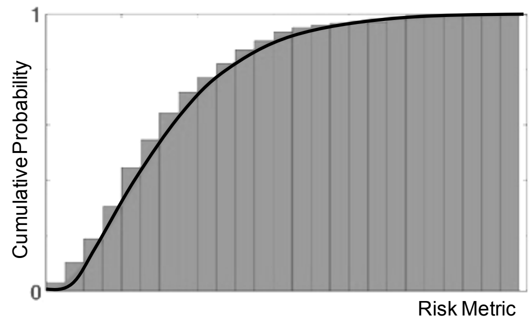
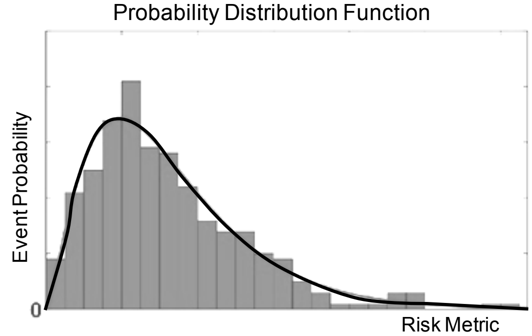
Cumulative Distribution Function



Probability Distribution Function



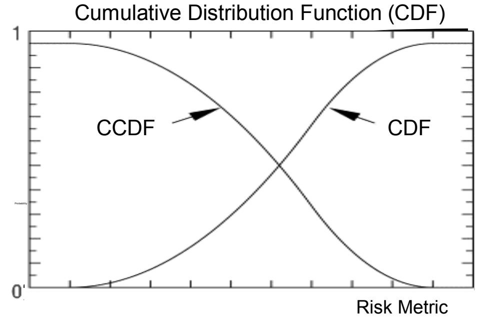
The cumulative distribution function may be used to calculate the quantiles or the probability of not exceeding the mean of a risk metric.

Other examples of using the cumulative distribution function are calculation of the seismic fragility of a component, or the calculation of probability of recovery of offsite power within a certain time period.

NUREG/CR-6823 (Ref. 78) defines cumulative distribution function as one that “gives the probability that the random variable does not exceed a given value.”

The complementary cumulative distribution function is the complement of the cumulative distribution function (i.e., the result of subtracting the cumulative distribution function from unity). Therefore, the complementary cumulative distribution function can be defined as a function that provides the probability that a parameter value is greater than a given value. The following graphs illustrate the complementary cumulative distribution function and its corresponding cumulative distribution function.

Complementary Cumulative Distribution Function (CCDF)
Vs.
Cumulative Distribution Function (CDF)



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	Some examples of using the complementary cumulative distribution function are calculating the probability of exceeding a certain release fraction of radioactive material in core melt accidents, calculating the frequency of exceeding a certain intensity of external hazard occurrence, calculating the frequency of loss of offsite power events exceeding a certain duration, or calculating the probability of emergency diesel generator repair lasting longer than a certain time period. The definition provided was based on the definition in NUREG/CR-6823 (Ref. 78).
Cumulative Dose	
(see Dose)	The cumulative dose is a total dose that is defined under "Dose."
Cutset (Minimal Cutset)	
A combination of failures that result in a particular outcome. *(see Truncation Limit)*	In a PRA, a cutset (sometimes also written as "cut set") is the product (i.e., result) of the analysis and identifies a combination of failures that would result in core damage or containment failure. However, the cutsets produced by the PRA are minimal cutsets in which each minimal cutset is the smallest combination of failures needed to cause core damage or containment failure. Cutsets are expressed in the form of combinations of basic events. Basic events represent elements of the PRA model for which no further decomposition is performed because they are at the limit of resolution consistent with available failure data. Basic events can represent equipment unavailability, human errors, and initiating events. NUREG-1560 (Ref. 56) defines cutset as a "combination of a set of events (e.g., initiating event and component failures) that, if they occur, will result in an undesirable condition (such as the onset of core damage or containment failure)." In addition, NUREG-1560 defines the term "minimal cutset" as "the minimum combination of the set of events that would result in the undesirable condition." The Fault Tree Handbook (Ref. 49) defines minimal cutset in the context of a fault tree as "a smallest combination of component failures which, if they all occur, will cause the top event to occur." To illustrate the concept of a minimal cutset, consider an accident involving the combination of loss of offsite power, emergency diesel generator (EDG) failure, and electrically-driven emergency cooling pump failure: • For this postulated accident, a "cutset" may include separate events that represent (1) failure of offsite power, (2) failure of all EDGs, and (3) independent failure of the electrically-driven emergency cooling pumps; however, this would represent a nonminimal cutset because the electrically-driven emergency cooling pumps rely on the EDGs. If the EDGs fail, the electrically-driven emergency cooling pumps will not function, regardless if they independently fail. • For this accident, a "minimal cutset" would represent (1) failure of offsite power and (2) failure of all EDGs. These are the minimal failures required to cause failure of emergency cooling regardless if the electrically-driven emergency cooling pumps fail.

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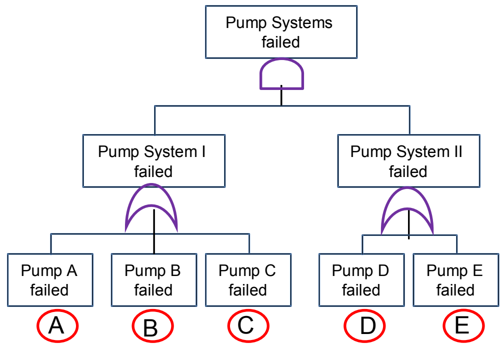
Cutset Example forPump Systems:



Possible Cutsets:
A*D
A*E
A*B*D
A*B*E
A*C*D
A*C*E
A*B*C*D
A*B*C*E
A*B*C*D*E
A*C*D*E
A*B*D*E
B*D
B*E
B*C*D
B*C*E
B*D*E
B*C*D*E
C*D
C*E
C*D*E

Minimal Cutsets:
A*D
A*E
B*D
B*E
C*D
C*E

Deep Dose Equivalent

(see Dose Equivalent)

The deep dose equivalent is one measure of dose that can be used to calculate the effect of radiation received by an individual and is defined under "Dose Equivalent."

Defense-in-Depth

Formal definition requires Commission approval. *(see Safety Margin, Uncertainty, Rationalist, Structuralist)*

In a PRA, defense-in-depth is not an explicitly modeled element. Rather, the results of the PRA provide insights into defense-in-depth.

Over time, various definitions have been used for defense-in-depth, including:

- three barriers to contain radioactive material: fuel cladding, primary system boundary, and the containment
- the use of successive measures to prevent an accident or to mitigate the consequences of an accident
- the use of redundancy and diversity
- implementation of the single failure criterion

Regardless of its definition, defense-in-depth is an integral part of the NRC's safety philosophy.

The NRC Web site Glossary (Ref. 36) defines defense-in-depth as: "An approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures."

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	The NRC Commission has referred to defense-in-depth as a concept that: Has always been and will continue to be a fundamental tenet of regulatory practice in the nuclear field, particularly regarding nuclear facilities. Risk insights can make the elements of defense-in-depth clearer by quantifying them to the extent practicable. Although the uncertainties associated with the importance of some elements of defense may be substantial, the fact that these elements and uncertainties have been quantified can aid in determining how much defense makes regulatory sense. Decisions on the adequacy of, or the necessity for, elements of defense should reflect risk insights gained through identification of the individual performance of each defense system in relation to overall performance. The Commission further states: Defense-in-depth is an element of the NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance, and operation is that the facility or system in question tends to be more tolerant of failures and external challenges.
Demonstrably Conservative Analysis	
(see Conservative Analysis)	A demonstrably conservative analysis has the same meaning as a conservative analysis and is defined under "Conservative Analysis."
Dependency	
Reliance of a function, system, component, or human action on another part of the system or another human action to accomplish its function.	Dependency is significant to the fidelity of a PRA model to capture the interrelationship between the modeled systems and human actions. As an example of systems dependency, many core cooling systems depend on electric power or cooling water systems. Also, operator actions closely spaced in time may have dependency in that a failure to perform a certain action may negatively affect successful performance of a subsequent action. Dependency has also been defined as: • "Requirement external to an item and upon which its function depends and is associated with dependent events that are determined by, influenced by, or correlated to other events or occurrences." (Ref. 2) • "Requirement external to a structure, system, or component (SSC), and upon which the SSC's function depends." (Ref. 59)
Design-Basis Accident	
A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and	In a PRA, the accidents traditionally modeled are not design-basis accidents (DBAs). Instead, the PRA typically models accidents that are more severe than DBAs, which are referred to as beyond-design-basis accidents (BDBAs) or severe accidents. It is important, though, to distinguish that the term "severe accident" indicates that core damage occurred; however, the term "beyond-design-basis accident" merely indicates that the accident exceeded the design limits of the plant. When developing a nuclear power plant, DBAs are selected to bound credible accident conditions and to ensure that the nuclear power plant can withstand and recover from these

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components necessary to ensure public health and safety. (*see Beyond-Design-Basis Accident, Severe Accident, Design-Basis Event*)	accidents. An example of a DBA is a major rupture of a pipe containing reactor coolant up to and including the double-ended rupture of the largest pipe containing reactor coolant. Another term, design-basis event (DBE), is used to broadly describe any event, internal or external to the plant, which could challenge safety functions. Therefore, DBAs are a subset of DBEs, and other examples of DBEs are anticipated transients (e.g., tripping of turbine generator), external events, and natural phenomena. NUREG-0800, Standard Review Plan 15.0 (Ref. 47), defines design-basis accidents as "postulated accidents that are used to set design criteria and limits for the design and sizing of safety-related systems and components." The definition provided was based on the definition in the NRC Web site Glossary (Ref. 36).
Design-Basis Event	
Any of the events specified in the nuclear power plant's safety analysis that are used to establish acceptable performance for safety-related functions. (*see Design-Basis Accident, Severe Accident*)	In a PRA, the outcome of concern is whether or not a particular accident leads to core damage. Therefore, beyond-design-basis accidents (BDBAs) that exceed the design envelope and lead to core damage are typically modeled. In this instance, these BDBAs that lead to core damage are referred to as severe accidents. Because a plant is designed and engineered to contend with design-basis accidents (DBAs), they typically are not the focus of current PRAs. However, DBAs represent only a portion of a broader category, design-basis events (DBEs). DBEs represent conditions within the plant design envelope and include anticipated transients (e.g., tripping of turbine generator), anticipated operational occurrences (AOOs), DBAs, external events, and natural phenomena. AOOs, an example of a DBE mentioned above, are a type of DBE described in NUREG-0800, Standard Review Plan 15.0 (Ref. 47), as "conditions of normal operation that are expected to occur one or more times during the life of the nuclear plant unit," (e.g., example loss of all offsite power). DBAs are a subset of DBEs, as noted above. An example of a DBA is a major rupture of a pipe containing reactor coolant up to and including the double-ended rupture of the largest pipe containing reactor coolant. The definition provided was based on the definition in NUREG-1560 (Ref. 56).
Deterministic (Analysis, Approach, Regulation)	
A characteristic of decisionmaking in which results from engineering analyses, not involving probabilistic considerations, are used to support a decision. *(see Risk-Informed, Probabilistic)*	A PRA represents an approach for assessing the likelihood of accidents and their potential consequences. However, the PRA model cannot be separated from and depends on deterministic analyses. For example, success criteria for various systems used in PRA to prevent and mitigate core damage are based on deterministic analyses. Another example of a deterministic analysis would be the calculation of peak cladding temperatures after emergency core cooling system actuation in a loss-of-coolant accident, or the timing of vessel breach in a core melt accident. As discussed in SECY-98-144 (Ref. 96), a deterministic regulation assumes that adverse conditions can exist and establishes a specific set of design-basis events (i.e., what can go wrong?). The deterministic approach involves implied, but unquantified, elements of probability in the selection of the specific accidents to be analyzed as design-basis events. It then requires that the design include safety systems capable of preventing or mitigating the consequences (i.e., what are the consequences?) of those design-basis events to protect public health and safety. The NRC Web site Glossary (Ref. 36) defines the term deterministic as "consistent with the principles of 'determinism,' which hold that specific causes completely and certainly determine effects of all sorts. As applied in nuclear technology, it generally deals with evaluating the safety of a nuclear power plant in terms of the consequences of a predetermined bounding subset of accident sequences." A deterministic approach or regulation is the opposite of a risk-

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	informed approach or regulation in which the likelihood of potential accidents is integrated. Deterministic approaches or regulations do not account for likelihood, and thus do not incorporate risk results obtained from a PRA.
Deterministic Analysis	
(see Deterministic)	The term deterministic analysis is defined under "Deterministic."
Deterministic Approach	
(see Deterministic)	The term deterministic approach is defined under "Deterministic."
Deterministic Regulation	
(see Deterministic)	The term deterministic regulation is defined under "Deterministic."
Direct Containment Heating	
(see High-Pressure Melt Ejection)	The term direct containment heating is a mechanism for challenging containment integrity and is defined under "High-Pressure Melt Ejection."
Dose	
A measure of the amount of radiation absorbed by a person. *(see Dose Equivalent)*	In a Level 3 PRA, dose is calculated to assess offsite health effects. The NRC Web site Glossary (Ref. 36) defines dose as "a general term, which may be used to refer to the amount of energy absorbed by an object or person per unit mass. Known as the 'absorbed dose,' this reflects the amount of energy that ionizing radiation sources deposit in materials through which they pass, and is measured in units of radiation-absorbed dose (rad). The related international system unit is the gray (Gy), where 1 Gy is equivalent to 100 rad. By contrast, the biological dose or dose equivalent, given in rems or sieverts (Sv), is a measure of the biological damage to living tissue as a result of radiation exposure." The collective dose (i.e., total dose obtained by summing over individual exposures of the affected population) is also used as a risk measure in value-impact analyses carried out in conjunction with PRAs. NUREG-0713, Vol. 28 (Ref. 45), states that the concept of collective dose is used by the NRC to denote the summation of the total effective dose equivalent received by all monitored workers at a nuclear facility, usually over the course of a year, and is reported in units of person-rem per year. The cumulative dose is the total dose that an individual receives as a result of repeated exposures to ionizing radiation to the same portion of the body, or to the whole body, over time. Cumulative dose usually is used for measuring occupational exposures of workers in the nuclear industry. When defining dose and the way it is used in PRAs to estimate health effects the following considerations are relevant: Under 'radiation dose' two concepts commonly used are: deterministic or non-stochastic dose and stochastic dose. The former implies that a health effect will occur within a short period following exposure with near certainty; the latter that a health effect may occur at some later time with some probability. In a PRA, the former is used with a threshold (depending on organ) to estimate early health effects. The latter is used, usually with a linear no-threshold model, to estimate latent cancers.

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Dose Coefficient	
Dose coefficients relate the dose to organs and tissues of the body from concentrations of radionuclides. *(see Dose, Dose Conversion Factor)*	In a Level 3 PRA, dose coefficients are incorporated into the consequence model. Dose coefficients relate the dose to organs and tissues of the body from concentrations of radionuclides. Dose coefficients for external exposure relate the organ and tissue doses to the concentrations of radionuclides in environmental media. Since the radiation arises outside the body, this is referred to as external exposure, while dose coefficients for internal exposure relate the organ and tissue doses to the intake of radionuclides by inhalation or ingestion, where the radiation is emitted inside the body.
Dose Conversion Factor	
A factor used to determine the biological effect of different types of radiation on an individual's organs. *(see Dose)*	In a Level 3 PRA, dose conversion factors are incorporated into the consequence model and used to calculate the effect of radiation received by an individual on different organs. As discussed in WASH-1400 (Ref. 44), dose conversion factors for the incorporation of radioactive material in the body give the dose received by individual organs over a time interval per curie intake by inhalation or ingestion. For external exposure, the dose conversion factors give the dose received by each organ per curie of radioactive material in a cubic meter of air or per curie of radioactive material deposited uniformly on a square meter of horizontal surface. The calculation of these dose conversion factors requires elaborate computer models with appropriate physiological parameters for a human body. These calculations need only be performed once for each type of radioactive material, organ, exposure mode, and time interval. From these calculations, a table can be prepared for use in the consequence model.
Dose Equivalent	
A measure of the biological damage to living tissue as a result of radiation exposure. *(see Dose)*	In a Level 3 PRA, a measure of biological damage because of radiation exposure is needed to estimate health effects. The dose equivalent is calculated as the product of absorbed dose in tissue multiplied by a quality factor and then sometimes multiplied by other necessary modifying factors at the location of interest. The dose equivalent is expressed numerically in units of rems or sieverts. The NRC Web site Glossary (Ref. 36) states that as defined in Title 10 of the *Code of Federal Regulations* (10 CFR) 20.1003, "Definitions" (Ref. 13), the committed dose equivalent (CDE) is the dose to some specific organ or tissue of reference that will be received from an intake of radioactive material by an individual during the 50-year period following the intake. In the event that an individual inhales or ingests radioactive material, the individual will continue to receive a dose from this event for the rest of his or her life. The NRC Web site Glossary (Ref. 36) also states that as defined in 10 CFR 20.1003 (Ref. 12), the committed effective dose equivalent (CEDE) is the sum of the products of the committed dose equivalents for each of the body organs or tissues that are irradiated, multiplied by the weighting factors applicable to each of those organs or tissues. The CEDE reflects the fact that different organs in the body are affected differently by radiation. The total effective dose equivalent (TEDE) is the sum of the external and the internal doses to an individual exposed to radiation. In a PRA, the total effective dose equivalent is needed to calculate offsite health effects. According to the NRC Web site Glossary (Ref. 36), the TEDE is the sum of the deep-dose equivalent (for external exposures) and the CEDE (for internal exposures). The deep-dose equivalent is the external whole-body exposure dose equivalent at a tissue depth of 1 cm. Whole body exposure includes at least the external exposure, head, trunk, arms above the elbow, or legs above the knee. Where a radioisotope is uniformly distributed throughout the body tissues, rather than being concentrated in certain parts, the irradiation can be considered as whole-body exposure.

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Dose Rate	
The amount of absorbed dose delivered per unit time. *(see Dose, Exposure, Exposure Time)*	In a Level 3 PRA a dose rate is needed to calculate the health effects. The units in which the dose rate is expressed are usually rems or sieverts per hour. Dose rate is the same as exposure rate. A PRA considers two types of exposures: acute and chronic. An acute exposure involves a large exposure received over a short period of time, i.e., a high exposure rate. Chronic exposures involve exposure at a low rate received over a long period of time, such as during a lifetime.
Dose Response Model	
A model that reflects the relationship between low doses of ionizing radiation and the potential for cancer. *(see Dose, Linear No-Threshold Model)*	In a Level 3 PRA, a dose response model is used to calculate frequency of latent cancers in the affected population, based on the dose received from the postulated accidents. There is some debate about the appropriate dose-response relationship for cancer risk following exposure to ionizing radiation. For example, in most PRAs, a linear relationship is assumed in which the cancer risk increases in direct proportion to the dose and there is no lower dose limit below which there is no risk. Others believe there is a nonlinear relationship, in which cancer risk increases in a more complex manner relative to dose.
Dosimetry	
The measurement and calculation of the absorbed dose in matter and tissue resulting from the exposure to ionizing radiation. *(see Dose)*	In a Level 3 PRA, dose is calculated to estimate health effects on the population affected by a severe accident. Dosimetry is the process of determining dose from exposure to radiation. To determine the dose received by exposed individuals, dosimetry attempts to estimate the dose received directly or indirectly via the various dose pathways, including cloudshine, water immersion, groundshine, skin deposition, inhalation, and ingestion.
Dynamic PRA	
A PRA that accounts for time-dependent effects by integrating them directly into the computer model. *(see PRA, Living PRA)*	In a traditional PRA, the coupling of deterministic analyses into the PRA model is achieved by manually constructing the linkage between the probabilistic and deterministic models. Thus, the manner in which an accident evolves with time (i.e., time-dependent effects) is based on a set of system and operator response characteristics that are manually entered into the PRA model. This is done by constructing event sequences in a discrete way such that they bound the contribution from all the scenarios that differ in the timing of the contributing events. In contrast, a dynamic PRA models accident sequences by automatically constructing the linkage between the probabilistic and deterministic models such that system and operator response characteristics are automatically accounted for in the PRA model. A dynamic PRA is not the same as a living PRA. In a living PRA, the PRA is updated as necessary to reflect changes in plant characteristics (e.g., design, operations) so that it represents the as-built as-operated plant.
Early Containment Failure	
(see Containment Failure)	The term early containment failure is discussed under the discussion for the term “Containment Failure.”

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Early Fatality	
(see Fatality)	The term early fatality is discussed under the discussion for the term “Fatality.”
Early Fatality Risk	
(see Fatality)	The term early fatality risk is a type of risk-involved fatality caused by exposure to radioactive materials and is defined under “Fatality.”
Economic Factors	
The considerations taken into account when assessing costs related to a release of radioactive material to the environment. *(see Economic Impact)*	The Level 3 portion of a PRA assesses the injuries and economic losses that might result if radioactivity escaped from containment. The economic factors in assessing risk include the costs of various actions taken to protect the public from short-term and long-term exposure through different exposure pathways (e.g., evacuation, relocation, decontamination), the costs of health effects and health care following exposure, and secondary economic effects. An illustrative list of required cost inputs from NUREG/CR-2300 (Ref. 69) includes: • evacuation cost per person • value of residential, business, and public areas per person • relocation cost per person • decontamination cost per acre for farm areas • decontamination cost per person for residential, business, and public areas • compensation rate per year for residential, business, and public areas (i.e., fraction of value) • average value of farmland per acre for state, county, or smaller areas • average annual value of farm sales per acre for state, county, or smaller areas • miscellaneous information, such as seeding and harvesting month, fraction of land devoted to farming, and fraction of farm sales due to dairy production.
Economic Impact	
The incurred costs of evacuation and relocation of the population, the costs of land condemnation, and the cost of condemned crops and other farm products as a result of an accident. *(see Economic Factors)*	In a Level 3 PRA, in addition to the health effects on the surrounding population, the impact of the severe accident on the surrounding economy is often estimated. Therefore, the economic impact risk is one of the risk categories calculated in a Level 3 PRA. The economic model in a Level 3 PRA includes the direct costs associated with protective actions taken after the accident, such as evacuation and relocation of the population, temporary or permanent interdiction of contaminated land and property, destruction of crops and foodstuffs. The model also may include other direct costs of actions, such as decontamination. Therefore, costs are a function of the stringency of post-accident radiation protection measures. Other direct costs may include costs of treatment of individuals exposed to radiation. Some models may include indirect economic impacts (e.g., litigation costs, government spending for disaster relief, regional economic activity impacts).
Economic Impact Risk	
(see Economic Impact)	The economic impact risk is the risk resulting from the economic impact of the accident and is defined in the discussion under “Economic Impact.”

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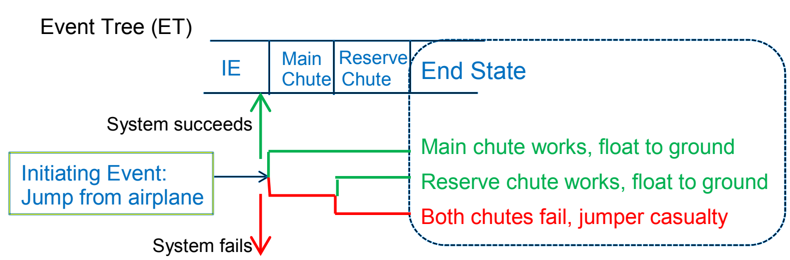
Emergency Preparedness	
The actions put into place to prepare personnel to rapidly identify, evaluate, and react to emergencies. *(see Emergency Response, Accident Mitigation)*	In a Level 3 PRA, to credit an effective emergency response when calculating the consequences of postulated accidents, adequate emergency preparedness (EP) is assumed. EP includes the programs, plans, training, exercises, and resources necessary to prepare emergency personnel to respond to emergencies, including those arising from terrorism or natural events such as hurricanes. EP strives to ensure that nuclear power plant operators can implement measures to protect public health and safety in the event of a radiological emergency. The definition provided is based on the definition in the NRC Web site Glossary (Ref. 36).
Emergency Response	
The actions initiated by the plant to mitigate the consequences of an accident that could potentially result in radioactive material release. *(see Emergency Preparedness, Accident Mitigation, Cohort)*	In a Level 3 PRA, the emergency response is taken into account when calculating the consequences of the postulated accidents. The emergency response encompasses the actions used to mitigate the consequences of an emergency, such as a severe nuclear accident, to human health and safety, quality of life, property, and the environment. The feasibility of some emergency actions may be limited by the hazard type (e.g., seismic events). The definition provided is based on the definition in the IAEA Safety Glossary (Ref. 7).
End State	
A set of conditions selected to characterize the plant states at the end of a chain of events. *(see Accident Sequence)*	In most PRAs, end states associated with Level 1 accident sequences typically include: success states (i.e., those states with negligible impact), and core damage or plant damage states. End states associated with Level 2 sequences usually are containment failure modes or release categories. The following figure illustrates different end states of an event tree:  The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Environmental Qualification	
A process for demonstrating that equipment will be capable of withstanding the accident ambient	In most PRAs, the focus is on severe accidents. The environment during a severe accident can be quite harsh and affect equipment performance. Safety equipment may experience high temperatures, pressures, humidity, radiation levels, and aerosol and particulate levels. The equipment may or may not be credited in the PRA as continuing to function under these conditions for many hours. One issue is that the environmental qualification carried out for equipment in currently operating reactors is carried out for the ambient conditions expected for

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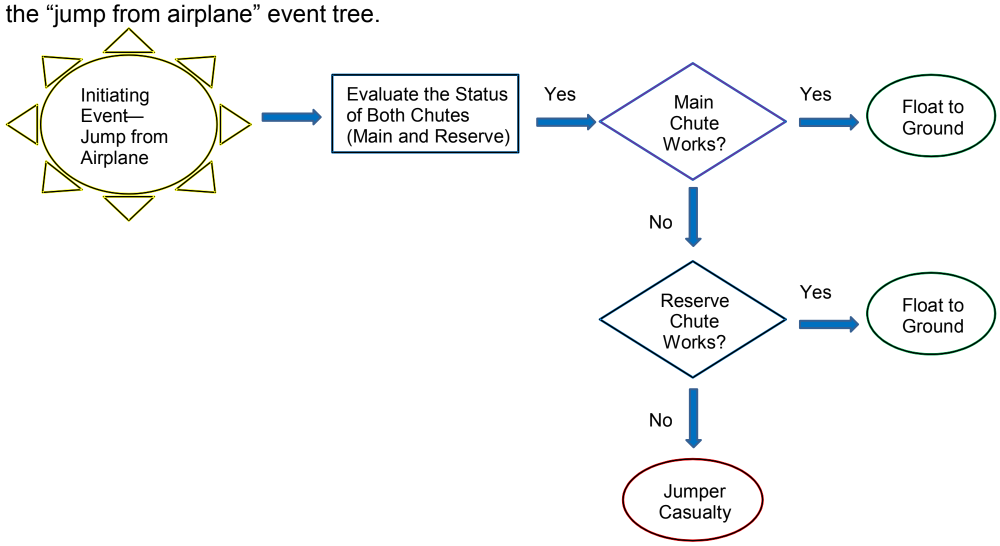
conditions that could exist when functionality is required.	design-basis accidents, and these conditions are likely to differ from those encountered in a severe accident. 10 CFR 50.49 (Ref.18) establishes requirements for environmental qualification for safety electric equipment important to safety for nuclear power plants. The definition provided was based on the definition in the NRC Web site Glossary (Ref. 36).
Epistemic Uncertainty	
(see Uncertainty)	Epistemic uncertainty is a type of uncertainty and is defined under "Uncertainty."
Error Factor (Human)	
A measure of uncertainty associated with probability estimates.	In a PRA, error factors are used to account for the uncertainty of the various parameters in the PRA model, such as the probability associated with a component failure or human error event. The error factor is a measure of the spread of the distribution of a parameter in the calculation of these types of failure. The term human error factor refers to the uncertainty in the probability of a human error. The probability of a human error event is often referred to as the human error probability. From a mathematical perspective, when the uncertainty distribution for an event failure probability is characterized by the log-normal distribution, uncertainties on these probability estimates are expressed as error factors. The lognormal error factor is defined as the 95th percentile divided by the median (i.e., the 50th percentile).
Event Scenario	
(see Accident Sequence)	The term event scenario has the same meaning as accident sequence and is defined under "Accident Sequence."
Event Sequence	
(see Accident Sequence)	The term event sequence has the same meaning as accident sequence and is defined under "Accident Sequence."
Event Sequence Analysis	
(see Accident Sequence Analysis)	The term event sequence analysis is another way of describing an accident sequence and is defined under "Accident Sequence Analysis."
Event Sequence Class	
(see Accident Sequence Class)	The term event sequence class has the same meaning as accident sequence class and is defined under "Accident Sequence Class."
Event Sequence Diagram	
A flowchart that represents various accident scenarios that can occur as a result of a plant upset condition. *(see Event Tree, Top Event)*	In a PRA, event sequence diagrams (ESDs) sometimes have been used to represent the progression of an initiating event by asking questions about successes and failures of plant responses to that initiating event. Each leg of the ESD ends with a successful or undesired end state for individual sequences. Once an ESD is developed, it can be mapped into an event tree, which relates more directly to a practical quantification of accident scenarios in a PRA. However, in comparison to event trees, ESDs tend to include additional supporting details on plant design and operational information that illustrates why a branch in the event tree proceeds down a particular success path. In this regard, ESDs are related to event trees in

<table>
<tr><th>TERM AND DEFINITION</th><th>DISCUSSION</th></tr>
<tr><td></td><td>
that they can help document the assumptions used in constructing an event tree.

The following figure illustrates a simple ESD. The oval to the left corresponds to top events in the “jump from airplane” event tree.



</td></tr>
<tr><td colspan="2">Event Sequence Group</td></tr>
<tr><td>(see Accident Sequence Class)</td><td>The term event sequence group has the same meaning as accident sequence group and is defined under “Accident Sequence Class.”</td></tr>
<tr><td colspan="2">Event Sequence Type</td></tr>
<tr><td>(see Accident Sequence Class)</td><td>The term event sequence type has the same meaning as accident sequence type and is defined under “Accident Sequence Class.”</td></tr>
<tr><td colspan="2">Event Tree</td></tr>
<tr><td>A logic diagram that graphically represents the various scenarios that can occur as a result of an upset condition. (see Accident Sequence, Containment Event Tree, Top Event, Accident Progression Event Tree, Bridge Tree)</td><td>
In a PRA, event trees are used in various parts of the analysis:

- Level 1 event trees provide the plant response logic from the initiating event to the successful prevention of core damage or core damage end states.
- Bridge event trees often are used as the interface between the Level 1 event trees and Level 2 event trees, in that they define the initial conditions for the Level 2 analysis (i.e., plant damage states), based on the plant conditions when core damage occurs.
- Level 2 event trees provide the plant response logic from the plant damage states to the successful prevention of containment failure or containment failure and release end states. In Level 2, these event trees are referred to as a containment event tree or accident progression event tree.

Event trees start with an initiating event and progress through questions about successes and failures of plant responses to that initiating event, ending with a successful or undesired end state for individual sequences. Individual sequences are pathways through the event tree.

An example of a simple event tree is shown below:
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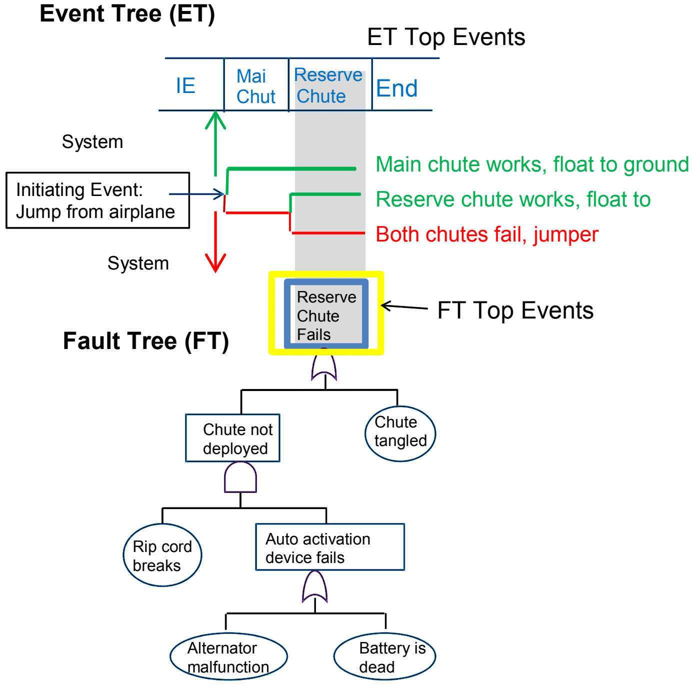
	Event Tree (ET)  An event tree has also been defined as: • “A logic diagram that begins with an initiating event or condition and progresses through a series of branches that represent expected system or operator performance that either succeeds or fails. The progression arrives at either a successful or failed end state.” (Ref. 2) • “An event tree graphically represents the various accident scenarios that can occur as a result of an initiating event (i.e., a challenge to plant operation). Toward that end, an event tree starts with an initiating event and develops scenarios, or sequences, based on whether a plant system succeeds or fails in performing its function. The event tree then considers all of the related systems that could respond to an initiating event, until the sequence ends in either a safe recovery or reactor core damage.” (Ref. 36)
Event Tree Sequence	
(see Accident Sequence)	The term event tree sequence is a specific description of an accident sequence and is defined under “Accident Sequence.”
Event Tree Top Event	
(see Top Event)	The term event tree top event is discussed under the discussion for the term “Top Event.” An illustration of an event tree top event is shown under the discussion for the term “Event Tree.”
Exclusion Area Boundary	
The boundary of the area surrounding the plant where the plant owner has the authority to determine all	PRA consequence calculations usually are concerned with the consequences outside of the exclusion area boundary. The exclusion area is that area around the plant where public residence is not normally permitted. The exclusion area boundary is the inner edge of the low population zone. The exclusion area and its boundary are important for reactor siting considerations as a location where acceptable dose limits following a release must be met. For example, Title 10

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activities, including exclusion or removal of personnel and property.	of the Code of Federal Regulations (10 CFR) 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance" (Ref. 26), states that the applicant (of a siting permit) should determine the following: an exclusion area of such size that an individual located at any point on its boundary for 2 hours immediately following onset of the postulated fission product release would not receive a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure. The definition provided is based on the definition in the NRC Web site Glossary (Ref. 36).
Expert Elicitation	
A formal, structured, and documented process in which judgments from expert(s) are obtained. *(see Expert Judgment)*	In a PRA, expert elicitation may be used to obtain information from technical experts on topics that are uncertain. An expert elicitation is a process in which experts are assembled and their judgment is sought and aggregated in a formal way. NUREG-1563 (Ref. 57) states, "Typically an elicitation is conducted to evaluate uncertainty. The uncertainty could be associated with: the value of a parameter to be used in a model; the likelihood and frequency of various future events; or the relative merits of alternative conceptual models. In each of these cases, the information regarding uncertainty would be represented by encoding the subjective probabilities from each subject-matter expert." An expert elicitation is a more formal process than expert judgment. Expert judgment may be the opinion of one or more experts, whereas expert elicitation is a highly structured process in which the opinions of several experts are sought, collected, and aggregated in a very formal way. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Expert Judgment	
Information (or opinion) provided by one or more technical experts that is based on their experience and knowledge. *(see Expert Elicitation)*	In a PRA, expert judgment is used when there is a lack of information. For example, if certain parameter values are unknown, or there are questions about phenomenology in accident progression, then expert judgment may be used. Expert judgment may be part of a structured approach, such as expert elicitation. Obtaining expert judgment is not necessarily as formal as invoking an expert elicitation process. Expert judgment may be the opinion of one or more experts, whereas expert elicitation is a highly structured process in which the opinions of several experts are sought, collected, and aggregated in a very formal way. NUREG-1563 (Ref. 57) states, "expert judgments may also be opinions that can be analyzed and interpreted, and used in subsequent technical assessments. Expert judgments can be either qualitative or quantitative. Expert judgments also can be judgments about uncertain quantities or judgments about value preferences." The ASME/ANS PRA Standard (Ref. 2) defines expert judgment as "information provided by a technical expert, in the expert's area of expertise, based on opinion, or on an interpretation based on reasoning that includes evaluations of theories, models, or experiments."
Exposure	
The state of being subjected to ionizing radiation. *(see Exposure Time, Cloudshine, Groundshine, Inhalation, Ingestion, Skin Deposition, Health Effects)*	In a Level 3 PRA, the offsite health effects resulting from exposure to ionizing radiation is considered. As stated in the NRC Web site Glossary (Ref.36), exposure occurs through absorption of ionizing radiation because of an external source or an internal exposure caused by inhalation or ingestion of a radioisotope. Acute exposure is a large exposure received over a short period of time. Chronic exposure is exposure received over a long period of time, such as during a lifetime.

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Exposure Pathways	
The various means by which exposure to radiation occurs and dose to recipients is delivered. *(See Exposure, Exposure Time, Cloudshine, Water Immersion, Groundshine, Inhalation, Ingestion, Skin Deposition, Health Effects)*	In a Level 3 PRA, exposure pathways to an individual are assumed for the consequence calculations. Cloudshine, sometimes referred to as air submersion, is the pathway by which external dose is given to an individual exposed to contaminated air; water immersion is a pathway by which external dose is given to an individual immersed in contaminated water (e.g., by bathing or swimming); inhalation is the pathway by which internal dose is given by breathing in contaminated air (resuspension inhalation is the pathway by which internal dose is given to an individual from breathing resuspended material previously deposited on the ground); ingestion is the pathway by which internal dose is given from consuming contaminated food or water; groundshine is the pathway by which external dose is given to an individual standing on contaminated ground; and skin deposition is exposure resulting from radioactive material deposited directly onto the surface of the body.
Exposure Rate	
(see Dose Rate)	The exposure has the same meaning as dose rate and is defined under "Dose Rate".
Exposure Time	
Duration of radiation exposure used to estimate the dose received by an individual. *(see Health Effects, Exposure)*	In a Level 3 PRA, the exposure time is needed to calculate the dose and subsequent health consequences to affected individuals. The PRA considers two types of exposures: acute and chronic. An acute exposure involves a large exposure received over a short period of time. Chronic exposures involve exposure received over a long period of time, such as during a lifetime.
External Event	
The term external event is no longer used and has been replaced by the term external hazard. *(see Hazard)*	A full scope PRA includes accidents resulting from both internal and external hazards. Internal hazards could include internal events, internal floods, and internal fires. External hazards could include seismic events, high winds, external floods, and other external hazards. The no-longer-used term, external event, is defined in the ASME/ANS PRA Standard (Ref. 2) as "an event originating outside a nuclear power plant that directly or indirectly causes an initiating event and may cause safety system failures or operator errors that may lead to core damage or large early release. Events such as earthquakes, tornadoes, and floods from sources outside the plant and fires from sources inside or outside the plant are considered external events. By historical convention, loss of offsite power not caused by another external event is considered to be an internal event." Historically, the difference between an internal event and an external event was the equipment boundary. The internal event represented something that occurred "internal" to the boundary of the piece of equipment. Conversely, occurrences external to the equipment boundary but within the plant boundary were classified as external events. With time, the definition for internal hazards has come to encompass all the hazards within the plant boundary, not just within the equipment. Thus, the external events have changed to currently represent events that occur outside the plant boundary but can cause undesired outcomes or conditions leading to plant equipment damage. Loss of offsite power is still considered an internal event. The term external event and external hazard have been used incorrectly interchangeably. The term external event is no longer used and has been subsumed by the term external hazard.

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External Flood	
A flood initiated outside the plant boundary that can affect the operability of the plant. *(see Hazard, External Flood Analysis, Internal Flood)*	In a PRA, external floods are a specific hazard group in which the flood occurs outside the plant boundary. The PRA considers floods because they have the potential to cause equipment failure by the intrusion of water into plant equipment through submergence, spray, dripping, or splashing. The definition provided was based on the definition in NUREG-1742 (Ref. 59).
External Flood Analysis	
A process used to assess potential risk from external floods. *(see Hazard Analysis, External Flood)*	In a PRA, an external flood analysis quantifies the risk contribution (e.g., core damage frequency and large release frequency) as a result of an external flood. The analysis models the potential failures of plant systems and components from external floods, as well as random failures. Floods have the potential to cause equipment failure by the intrusion of water into plant equipment through submergence, spray, dripping, or splashing. The likelihood of an external flood is determined through an external flood hazard analysis, which evaluates the frequency of occurrence of different external flood severities. The frequency of the external flood is used as input to the model used to assess external flood risk.
External Flood Fragility Analysis	
(see Fragility Analysis)	The term external flood fragility analysis is a type of fragility analysis and is included in the discussion to the term "Fragility Analysis."
External Flood Hazard Analysis	
(see Hazard Analysis)	The term external flood hazard analysis is a specific type of hazard analysis and is defined under "Hazard Analysis."
External Flood Plant Response Analysis/Model	
(see Plant Response Analysis/Model)	The term external flood plant response analysis is a type of plant response analysis and is included under "Plant Response Analysis/Model."
External Hazard	
(see Hazard)	The term external hazard is related to the term hazard and is defined under "Hazard."
External Hazard Analysis	
(see Hazard Analysis)	The term external hazard analysis is a type of hazard analysis and is defined under "Hazard Analysis."
Failure Mechanism	
The fault associated with a component that causes it to malfunction. *(see Failure Mode)*	In a PRA, the concept of failure mechanism is used to explain the immediate cause of component failure. The fault that causes failure could be electrical, mechanical, chemical, physical, thermal, or human error. An example of a failure mechanism would be an electrical short in the electric motor winding that causes failure of a pump to start. The ASME/ANS PRA Standard (Ref. 2) defines failure mechanism as "any of the processes that results in failure modes, including chemical, electrical, mechanical, physical, thermal, and human error." While failure mechanism is a cause of failure, failure mode is the functional manifestation of failure (e.g., failure to start, failure to run).

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Failure Mode	
The manner in which a component fails to perform its function. *(see Failure Mechanism, Failure Modes and Effects Analysis)*	In a PRA, the failure modes of a component are represented as basic events, and while it is a visible manifestation of failure, it is distinguished from failure mechanism, which is a cause of failure. Failure of a component is distinguished by its failure mode. Each failure mode is modeled separately, with its own failure probability. Failure mode is failure in a distinct functionality of a component that is necessary for it to successfully operate (e.g., failure modes of a valve might be failure to open, failure to close, or inadvertent opening). Failure of a pump may be distinguished into two separate failure modes, namely failure to run or failure to start. In a fire PRA, spurious (unintended) operation is also defined as a failure mode. The ASME/ANS PRA Standard (Ref. 2) defines failure mode as "a specific functional manifestation of a failure (i.e., the means by which an observer can determine that a failure has occurred) by precluding the successful operation of a piece of equipment, a component, or a system (e.g., fails to start, fails to run, leaks)." A failure modes and effects analysis can be used to identify component failure modes and evaluate their effects on other components, subsystems, and systems.
Failure Modes and Effects Analysis	
A process for identifying failure modes of specific components and evaluating their effects on other components, subsystems, and systems. *(see Failure Mode)*	In a PRA, a failure modes and effects analysis (FMEA) generally is not used except to identify initiating events for a new plant design with no operational history or failure data. A FMEA is aimed at analyzing the effects of a single component or function failure on other components, systems, and subsystems. A FMEA can be useful in identifying initiating events that involve support system failures and the expected effects on the plant (especially on mitigating systems). The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Failure Probability	
(see Probability)	The term failure probability is a specific type of probability and is defined under "Probability."
Fatality (Early, Latent, Prompt, Latent Cancer)	
Death occurring as a result of exposure to radioactive material. *(see Exposure, Quantitative Health Objectives)*	In a Level 3 PRA, one of the objectives is to calculate the dose received by the population surrounding the plant as a result of a potential release of radioactive material. Depending on the amount of dose and the duration over which it is received, early and latent fatalities can occur. The risk of incurring fatalities, both early and latent fatalities, is one of the most important outputs of a Level 3 PRA. Early fatalities, synonymous with prompt fatalities, are defined as deaths from the acute effects of radiation that may occur within a few months of the exposure. Latent cancer fatalities are defined as deaths from cancer caused by chronic effects of radiation exposure; latent cancer fatalities may occur years after the exposure. Prompt or early fatalities are usually the result of acute exposures (large exposure received over a short period of time). Latent fatalities resulting from cancer that became active after a latent period can result from exposure from early pathways (e.g., groundshine, cloudshine, and skin deposition), as well as long-term pathways (e.g., resuspension inhalation and ingestion).
Fatality Risk (Early, Latent, Prompt)	
(see Fatality)	The fatality risk (early or prompt fatality risk, latent fatality risk) is the risk involving fatalities caused by exposure to radioactive materials and is defined in the discussion under "Fatality."

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Fault Tree

A deductive logic diagram that graphically represents the various failures that can lead to a predefined undesired event. *(see Top Event, Event Tree)*

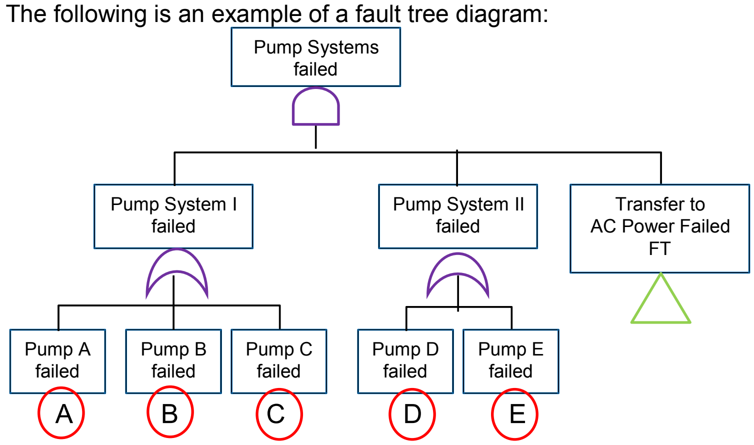
In a PRA, fault trees are used to depict the various pathways that lead to a system failure.

Fault trees describe how failures of top events occur because of various failure modes of components, human errors, initiator effects, and failures of support systems that combine to cause a failure of a top event in the event trees.

A fault tree also has been defined as:

- "A deductive logic diagram that depicts how a particular undesired event can occur as a logical combination of other undesired events." (Ref. 2)
- "A fault tree identifies all of the pathways that lead to a system failure. Toward that end, the fault tree starts with the top event, as defined by the event tree, and identifies …what equipment and operator actions, if failed, would prevent successful operation of the system. All components and operator actions that are necessary for system function are considered. Thus, the fault tree is developed to a point where data are available for the failure rate of the modeled component or operator action." (Ref. 36)

The following is an example of a fault tree diagram:



Fault Tree Top Event

(see Top Event)

The term fault tree top event is a type of top event in a PRA model and is defined under "Top Event." An illustration of a fault tree top event is shown under the discussion for the term "Event Tree."

Feed and Bleed, Bleed and Feed

A method of core cooling in a pressurized-water reactor by providing cooling water to the reactor while removing heated coolant through open reactor vessel relief valves.

In a PRA, feed and bleed is often included as a core heat removal option for pressurized-water reactors when secondary cooling (e.g., auxiliary feedwater) is unavailable. To remove the core (i.e., decay) heat from the reactor vessel, water from a storage tank or recirculated from the containment sump is injected into the reactor vessel through safety or nonsafety grade pumping systems (feed), and the pressurizer power-operated relief valves (PORVs) or safety valves are opened to discharge the heated coolant from the reactor vessel (bleed).

The terms feed and bleed and bleed and feed are similar in meaning and often used interchangeably. However, in certain instances, these terms may be used to distinguish the manner in which this decay heat removal option is accomplished. In some plants, the injection pumps may be capable of injecting coolant at full reactor coolant system pressure while discharging reactor coolant through the safety valves. In this design, the injection of water

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	(feed) can occur before opening the safety valves (bleed), such that this decay heat option may be referred to as feed and bleed. In other plants, the injection pumps are not capable of injecting coolant at full system pressure, but instead must rely upon operator actions to open one or more PORVs in a timely matter. In this situation, the reactor vessel pressure is first reduced by the release of coolant (bleed), with subsequent injection of coolant from the injection pumps (feed). This decay heat option may be referred to as bleed and feed.
Fire Probabilistic Risk Assessment Plant Response Model (Analysis)	
(see Plant Response Analysis)	The term fire probabilistic risk assessment plant response analysis is a type of plant response analysis and is defined under "Plant Response Analysis/Model." The term fire probabilistic risk assessment plant response model is also a technical element for internal fires in the ASME/ANS PRA Standard (Ref. 2) whose objective is to identify the initiating events that can be caused by a fire event and develop a related accident sequence model, and to depict the logical relationships among equipment failures (both random and fire induced) and human failure events for core damage frequency and large early release frequency assessment when combined with the initiating event frequencies.
Fission Product (Release)	
The byproduct of the nuclear fission process. *(See Radioactive Material, Radionuclide)*	In a PRA, the terms radionuclide, radioactive material, and fission product are used interchangeably. These terms are meant to refer to the substance that is the source of the risk being evaluated. A fission product release, therefore, refers to the release of the radioactive material from the reactor and from the containment that could adversely affect public health and safety. The NRC Web site Glossary (Ref. 36) defines fission product as, "The nuclei (fission fragments) formed by the fission of heavy elements, plus the nuclide formed by the fission fragment's radioactive decay."
Fission Product Release	
(see Radioactive Material Release)	For purposes of a Level 2 and Level 3 PRA, the term fission product release is used interchangeably with radioactive material release.
Fragility	
The likelihood that a component, system, or structure will cease to function given the occurrence of a hazard event of a certain intensity. *(see Fragility Analysis, High Confidence of Low Probability of Failure, Fragility Curve)*	In a PRA, fragility is a concept used in the evaluation of external hazards. The fragility of a component, system, or structure is generally calculated for seismic events, high wind events, and external flood events Since a given component may fail because of various mechanisms (e.g., seismic motion may cause anchor failure, structural failure, systems interactions), fragility can be calculated for each of these failure mechanisms, or the results can be presented for the dominant mechanism. The ASME/ANS PRA Standard (Ref. 2) states, "fragility of a structure, system or component (SSC) is the conditional probability of its failure at a given hazard input level. The input could be earthquake motion, wind speed, or flood level."
Fragility Analysis (External Flood, High Winds, Other External Hazards, Seismic)	
Estimation of the likelihood that a given component, system, or	In a PRA, fragility analysis identifies the components, systems, and structures susceptible to the effects of an external hazard and estimates their fragility parameters. Those parameters are then used to calculate fragility (conditional probability of failure) of the component, system, or structure at a certain intensity level of the hazard event. Fragility analysis considers all

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structure will cease to function given the occurrence of a hazard event of a certain intensity. *(see Fragility, Fragility Curve)*	failure mechanisms due to the occurrence of an external hazard event and calculates fragility parameters for each mechanism. This is true whether the fragility analysis is used for an external flood hazard, fire hazard, high wind hazard, seismic hazard, or other external hazards. For example, for seismic events, anchor failure, structural failure, and systems interactions are some of the failure mechanisms that would be considered.
Fragility Curve	
A graph that plots the likelihood that a structure, system or component will fail versus the increasing intensity of a hazard event. *(see Fragility, Fragility Analysis)*	In a PRA, fragility curves generally are used in seismic analyses and provide the conditional frequency of failure for structures, systems, or components as a function of an earthquake-intensity parameter, such as peak ground acceleration. Fragility curves also can be used in PRAs examining other hazards, such as high winds or external floods.
Frequency (Accident Sequence, Core Damage, Initiating Event, Large Early Release, Large Release, Radioactive Material Release)	
The expected number of occurrences of an event or accident condition expressed per unit of time. (*see Probability)*	In a PRA, a frequency is calculated for various events. For a Level 1 PRA, frequencies are calculated for the initiating events and for the core damage accident sequences; the latter frequencies are summed to provide an overall core damage frequency. For a Level 2 PRA, frequencies are calculated for the plant damage states and for the release of radioactive material (e.g., large early release frequency, large release frequency, and the overall radioactive material release frequency). For a Level 3 PRA, frequencies are calculated for accident consequences (i.e.; early and latent fatalities) and, sometimes, economic consequences. Frequency is normally expressed in events per plant (or reactor) operating year or events per plant (or reactor) calendar year. The subset terms of frequency can be defined as follows: • <u>Accident Sequence Frequency:</u> The frequency associated with a series of events that follow from a particular initiating event, through system and operator responses, and ultimately to a well-defined end state, such as core damage. *(see Accident Sequence)* • <u>Core Damage Frequency:</u> The sum of the accident sequence frequencies of those accident sequences whose end state is core damage. • <u>Initiating Event Frequency:</u> The frequency of an event originating from an internal or external hazard that both challenges normal plant operation and requires successful mitigation. • <u>Large Early Release Frequency:</u> The frequency of a rapid, unmitigated release of airborne fission products from the containment to the environment that occurs before effective implementation of offsite emergency response, and protective actions, such that there is a potential for early health effects. • <u>Large Release Frequency:</u> The Commission has not approved a formal definition of a large release or a large release frequency. One informal definition for large release frequency is the frequency of an unmitigated release of airborne fission products from the containment to the environment that is of sufficient magnitude to cause severe

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	health effects, regardless of its timing. The history of the use of the term "Large Release Frequency" is provided in SECY-13-0029 (Ref.100). *(see Large Release)* • Radioactive Material Release Frequency: The frequency of the release of radioactive material from the containment to the environment. This may refer to the total frequency of all releases regardless of size or timing. The radioactive material release frequency may also be subdivided depending on the size and timing of the release. Large early release frequency and large release frequency are defined above. A small early release frequency can be defined as the frequency of early releases of low enough magnitude to have minimum potential for early health effects. A small late release frequency can be defined as the frequency of late releases of low enough magnitude and with a long enough delay to have minimum potential for early health effects. A large late release frequency can be defined as the frequency of late releases that have sufficient magnitude to cause severe health effects, but which occur in a timeframe that allows effective emergency response and protective actions so that the offsite health effects will be significantly reduced compared to those of a large early release. *(see Radioactive Material Release)* In some instances, the terms frequency and probability are used interchangeably, but incorrectly. Unlike frequency, probability represents a unitless quantity.
Frequentist Analysis, Frequentist Estimation, Frequentist Statistics	
A type of data analysis that relies solely on actual occurrences of the event under consideration. *(see Bayesian Analysis)*	In a PRA, frequentist analysis is only used when occurrences of an event are sufficiently abundant such that a reliable estimate of event probability can be expressed as the ratio of number of event occurrences to total number of occurrences in which the event could occur. In frequentist statistics, error probability can be calculated as the number of errors experienced over some number of tries divided by the number of tries. In the frequentist approach, the probability of a random event is interpreted as the fraction of times that the event would occur, in a large number of trials. In risk analysis, both frequentist and Bayesian analysis may be used, depending on whether occurrence data is sufficiently abundant. The terms frequentist analysis, frequentist estimation, and frequentist statistics are used interchangeably.
Frequentist Estimation	
(see Frequentist Analysis)	The term frequentist estimation has the same meaning as frequentist analysis and is defined the same as the term "Frequentist Analysis."
Frequentist Statistics	
(see Frequentist Analysis)	The term frequentist statistics has the same meaning as frequentist analysis and is defined the same as the term "Frequentist Analysis."
Front-Line System	
A system used to directly provide a safety function. *(see Support System)*	In a PRA, front-line systems are modeled to help represent the ways in which a plant can prevent core damage or prevent containment failure. The ASME/ANS PRA Standard (Ref. 2) defines a front-line system as "a system (safety or non-safety) that is capable of directly performing one of the accident mitigating functions (e.g., core or containment cooling, coolant makeup, reactivity control, or reactor vessel pressure control) modeled in the PRA." In some references, the definition of a front-line system only includes safety-related systems. However, other definitions are more generalized to include the possibility that a front-line system can be a nonsafety system, such as the ASME/PRA Standard definition cited above.

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Full Power	
The state of operation in which the reactor is critical and producing 100-percent power. *(see At-Power, Low Power and Shutdown)*	A PRA models the different plant operating states (POSs) of the plant. Operation at full power is one POS, while several POSs are needed to characterize the plant during the various stages of low-power and shutdown. These POSs are distinguished in the PRA model because the plant response (e.g., accident sequences) differs during different POSs. Historically, the term full power was used to denote any power level between low power and 100-percent power. This definition has been recently modified so that full power currently refers just to 100-percent power of the reactor core, while at-power covers the range of powers from low power up to and including 100-percent power.
Full-Scope PRA	
A PRA that considers all the various challenges that could contribute to the risk posed by the plant to the health and safety of the public. (*see PRA, Risk Metric*)	A full-scope PRA generally only considers the reactor and associated systems and is comprised of three distinct parts, referred to as Levels. The full-scope PRA includes a Level 1 (core damage), Level 2 (radioactive material release) and Level 3 (consequences) PRA that addresses both internal and external hazards at all power modes (at-power, low-power, and shutdown). These power modes commonly are referred to as plant operating states (POSs). A full-scope site PRA may also consider risks from the spent fuel pool and any other fuel storage facility on site. Offsite risk metrics in the Level 3 portion may include both health effects and economic considerations brought about by the release of radioactive material.
Fussell-Vesely Importance	
(see Importance Measure)	The term Fussell-Vesely importance is one type of importance measure and is defined under "Importance Measure."
General Transient	
(see Transient)	The term general transient has the same meaning as transient and is defined under "Transient."
Groundshine	
Exposure from radioactive material deposited on the ground. *(see Exposure Pathways, Cloudshine, Water Immersion, Inhalation, Ingestion, Skin Deposition)*	In a Level 3 PRA, for the consequence calculation groundshine is one of the assumed pathways by which an individual can receive doses. The pathways of exposure include: (1) direct external exposure from radioactive material in a plume, principally due to gamma radiation (air immersion or cloudshine), (2) direct exposure from radioactive material in contaminated water given to an individual immersed in the water, (3) exposure from inhalation of radioactive materials in the cloud and resuspended material deposited on the ground, (4) exposure to radioactive material deposited on the ground (groundshine), (5) radioactive material deposited onto the body surfaces (skin deposition), and (6) ingestion from deposited radioactive materials that make their way into the food and water pathway.
Hazard (Type (Internal, External), Group, Event)	
Anything that has the potential to cause an undesired event or condition that leads to equipment damage. *(see Hazard Analysis, Initiating Event)*	In a PRA, there are three different uses of the term hazard as an adjective (the terms hazard and plant hazard tend to be correctly used interchangeably): types, groups, and events. The first, hazard type, classifies hazards as either internal or external to the plant. Within each hazard type, internal and external, there are subcategories, which are referred to as hazard groups. For internal hazards, this hazard group includes internal events, internal floods, and internal fires. For external hazards, this includes seismic events, high winds, external floods, and other external hazards. Finally, a hazard event represents the events brought about by the occurrence of the specified hazard. For example, those of interest in a PRA are ones that directly or indirectly cause an initiating event and may further cause safety system failures or operator errors that may lead to core damage or radioactive material release.

<table>
<tr><th>TERM AND DEFINITION</th><th>DISCUSSION</th></tr>
<tr><td></td><td>As defined in Regulatory Guide 1.200 (Ref.91), a hazard group “is a group of similar causes of initiating events that are assessed in a PRA using a common approach, methods, and likelihood data for characterizing the effect on the plant.”

A hazard event is described in terms of the specific levels of severity of impact that a hazard can have on the plant. The hazard event is an occurrence of the phenomenon that can result in a plant trip and possibly other damage when the plant is at-power or result in the loss of a key safety function during non-power operations. The ASME/ANS PRA Standard (Ref. 2) states that there “is a range of hazard events associated with any given hazard, and, for analysis purposes, the range can be divided into bins characterized by their severity.” An example of the overall concept of hazard, hazard event, and initiating event is as follows:

• Earthquakes are a hazard;

• 0.1g, 0.3g, 0.5g earthquakes and their associated spectral shapes and time histories may be defined as hazard events;

• A manual plant trip is typically the initiating event for the 0.1g earthquake, and a loss of offsite power is typically assumed as the initiating event for the 0.3g and 0.5g earthquakes.

The ASME/ANS PRA Standard (Ref. 2) defines a hazard as “an event or a natural phenomenon that poses some risk to a facility. Internal hazards include events such as equipment failures, human failures, and flooding and fires internal to the plant. External hazards include events such as flooding and fires external to the plant, tornadoes, earthquakes, and aircraft crashes.”</td></tr>
<tr><td colspan="2">Hazard Analysis (External, External Flood, High Wind, (Probabilistic) Seismic, Other Hazards)</td></tr>
<tr><td>A process used to assess potential plant challenges, including natural phenomena, and to assess their likelihood, typically as a function of severity.</td><td>In a PRA, it is important to identify and characterize the nature and causes of specific types of hazards. A hazard represents an event or a natural phenomenon that poses some challenge to a facility. Examples of external hazards typically evaluated in a PRA include external floods, high winds, seismic events, and external fires. A hazard analysis is used to evaluate the frequency of occurrence of different severities for the hazard being analyzed. Results from the hazard analysis are used as input to the PRA, which subsequently examines the hazards with respect to risk.

Listed below are specific types of hazard analyses:

• <u>External hazard analysis</u>: The objective is to evaluate the frequency of occurrence of different severities or intensities of external events or natural phenomena (e.g., external floods or high winds).

• <u>External flood hazard analysis</u>: The objective is to evaluate the frequency of occurrence of different external flood severities.

• <u>High wind hazard analysis</u>: The objective is to evaluate the frequency of occurrence of different intensities of high winds.

• <u>(Probabilistic) seismic hazard analysis:</u> A seismic hazard analysis expresses “the seismic hazard in terms of the frequency of exceedance for selected ground motion parameters during a specified time interval. The analysis involves identification of earthquake sources, evaluation of the regional earthquake history, and an estimate of the intensity of the earthquake-induced ground motion at the site. As stated in Regulatory Guide 1.200 (Ref. 86): “at most sites, the objective is to estimate the probability or frequency of exceeding different levels of vibratory ground motion” The term probabilistic seismic hazard analysis is similar in meaning to the definition of seismic hazard analysis as stated above.</td></tr>
</table>

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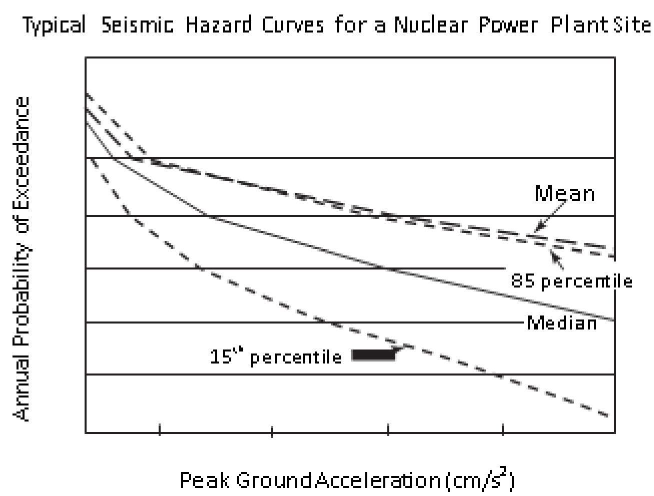
	• <u>Other hazards analysis</u>: Evaluates the frequency of occurrence of different intensities of other internal or external hazards (e.g., external fires). The ASME/ANS PRA Standard (Ref. 2) defines hazard analysis as “the process to determine an estimate of the expected frequency of exceedance (over some specified time interval) of various levels of some characteristic measure of the intensity of a hazard (e.g., peak ground acceleration to characterize ground shaking from an earthquake). The time period of interest is often taken as 1 year, in which case the estimate is called the annual frequency of exceedance.” An example of a hazard curve is shown below. 
Hazard Event *(see Hazard)*	The term hazard event is related to the term hazard and is defined under “Hazard.”
Hazard Group *(see Hazard)*	The term hazard group is related to the term hazard and is defined under “Hazard.”
Hazard Type *(see Hazard)*	The term hazard type is related to the term hazard and is defined under “Hazard.”
Health Effects The effects of radioactive material on the health and safety of exposed individuals. *(see Quantitative Health Objectives, Accident Consequence, Exposure Time, Land Contamination)*	In a Level 3 PRA, the health effects represent the main component of the calculated risk. Health effects from radioactive material (i.e., ionizing radiation) usually are distinguished as acute or latent. Acute health effects are adverse health symptoms (e.g., fatalities) occurring within a short time (days or months rather than years) of an exposure to large radiation doses. Acute fatalities and injuries are expected to occur within 1 year of an accident or sooner. Latent health effects refer to cancer deaths that may occur with a considerable latency period, from approximately 2 to 25 years, depending on the type of cancer involved. Public health effects refer to illnesses or fatalities to the population beyond the site boundary resulting from the release of radiation.

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High Confidence of Low Probability of Failure	
A measure of seismic capacity of a structure, system, or component, expressed in terms of a threshold earthquake intensity, below which failure of the structure, system, or component is highly unlikely. *(see Seismic Margin, Fragility)*	In a seismic PRA, the high confidence in low probability of failure (HCLPF) measure is generally not used, but it is a key parameter primarily in a seismic margin analysis. The HCLPF capacity is a measure of the seismic capacity of a structure, system, or component (SSC) or of the whole plant. It indicates an earthquake intensity level at which there is high (95%) confidence the conditional probability of failure of the SSC is low (5% or less). At the plant level, HCLPF can refer to the peak ground acceleration level at which there is a high (95%) confidence of low (5%) conditional probability of core damage. It is used extensively in a seismic margin analysis. The ASME/ANS PRA Standard (Ref. 2) states that "HCLPF capacity: refers to the High Confidence of Low Probability of Failure capacity, which is a measure of seismic margin."
High-Level Requirements	
The minimum requirements for a technically acceptable baseline PRA, independent of application. *(see Supporting Requirements)*	For a base PRA, NRC Regulatory Guide (RG) 1.200 (Ref. 91) defines a set of technical characteristics and associated attributes that make it technically acceptable. One approach to demonstrate a PRA is acceptable is to use a national consensus PRA standard, supplemented to account for the NRC staff's regulatory positions. The ASME/ANS PRA Standard (Ref. 2) is one example of a national consensus PRA standard. The ASME/ANS PRA Standard uses high-level requirements and supporting requirements. RG 1.200 states, "Technical requirements may be defined at two different levels: (1) high-level requirements and (2) supporting requirements. High-level requirements are defined for each technical element and capture the objective of the technical element. These high-level requirements are defined in general terms, need to be met regardless of the level of analysis resolution and specificity (capability category), and accommodate different approaches. Supporting requirements are defined for each high-level requirement. These supporting requirements are those minimal requirements needed to satisfy the high-level requirement." The ASME/ANS PRA Standard (Ref. 2) states, "The high level requirements are defined in general terms and present the top level logic for the derivation of more detailed supporting requirements. The high level requirements reflect not only the diversity of approaches that have been used to develop the existing PRAs, but also the need to accommodate future technological innovations." The definition provided was based on the definition in the introduction section of ASME/ANS PRA Standard (Ref. 2).
High-Pressure Melt Ejection	
A phenomenon in which molten core material penetrates the reactor vessel and is forcibly ejected under high pressure. *(see Core Melt)*	In a PRA, high-pressure melt ejection (HPME) is a phenomenon that could lead to containment failure and release of radioactive material to the environment before evacuation of the surrounding population. If the core melts and penetrates the reactor pressure vessel while the reactor coolant system is at high pressure (>400psi), the core debris would be ejected into the reactor cavity. This phenomenon is called HPME. A phenomenon often associated with HPME is direct containment heating (DCH). DCH can occur in the following manner: As the core debris is being ejected from the reactor vessel (depending on the configuration of the reactor cavity), it is possible that it will be transported into the containment atmosphere and directly heat the atmosphere. This heating can substantially increase the pressures in containment. It is also possible that combustible gases in the containment atmosphere could ignite and burn as a result of the transported core debris, adding to the containment heating and therefore the pressure in containment.

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High-Wind Fragility Analysis	
(see Fragility Analysis)	High-wind fragility analysis is a type of fragility analysis and is included in the discussion under "Fragility Analysis."
High-Wind Hazard Analysis	
(see Hazard Analysis)	The term high-wind hazard analysis is a specific type of hazard analysis and is defined under "Hazard Analysis."
High-Wind Plant Response Analysis/Model	
(see Plant Response Analysis/Model)	The high-wind plant response analysis is a type of plant response analysis and is included in the discussion under "Plant Response Analysis/Model."
High Winds	
Winds of a certain size that could potentially damage or affect the operability of a nuclear power plant. *(see Hazard)*	In a PRA, the typical high winds analyzed as a hazard include the following: tornadoes, hurricanes (or cyclones or typhoons as they are known outside of the United States), extratropical (thunderstorm) winds, and other wind phenomena depending on the site location (Ref. 2). High winds are a hazard group and, more specifically, a type of external hazard.
Human Action (Operator Action)	
An action performed by plant personnel. *(see Human Failure Event, Human Reliability Analysis)*	In a PRA, the human actions that are modeled include those actions that plant personnel might fail to perform or might fail to perform correctly. Plant personnel interact with the plant in a number of ways. For example, maintenance personnel perform surveillance tests, calibrate equipment, and repair failed equipment. Control room operators control the plant and, after an initiating event, bring the plant to a safe stable state using as guidance written or memorized procedures. These actions are of concern for the PRA because failure to perform any of the actions correctly can lead to a reduced capability of responding to a transient or accident. For example, failure to restore a system following maintenance can lead to its unavailability to perform its function when called upon. Failure of the control room crew to correctly follow their procedures might lead to a loss of a critical safety function. A human action and an operator action do not necessarily mean the same thing. A human action can be performed by different types of nuclear power plant personnel, while an operator action is an action performed by a licensed individual in the control room. Human actions are an important component in conducting a human reliability analysis (HRA). HRA is used to support the development of a PRA by identifying relevant human actions and the associated human errors that might occur. Human errors modeled in the PRA are referred to as human failure events.
Human Error (Operator Error)	
Any human action, including inaction, which exceeds some limit of acceptability, excluding	In a PRA, human (operator) errors are modeled in the PRA as human failure events if they are unrecovered and lead to the failure or unavailability of a component, system, or function. Human errors of interest are those that result in the unavailability of a component, system, or function, or a failure to initiate, terminate, or control a system or function that can affect an accident sequence.

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malevolent behavior. *(see Human Failure Event, Human Reliability Analysis)*	A human error and an operator error do not necessarily mean the same thing. A human error can be attributed to different types of nuclear power plant personnel, while an operator error is specifically attributed to a licensed individual (i.e., operator) in the control room. Human reliability analysis (HRA) is used to identify the possible human errors that might occur. The term human failure event is synonymous with and has replaced the term human error in the PRA lexicon. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Human Error Event	
(see Human Failure Event)	A human error event is a type of human error modeled in a PRA and is defined under “Human Failure Event.”
Human Error Factor	
(see Error Factor)	A human error factor is a specific type of error factor applicable to human reliability analysis and is defined under “Error Factor.”
Human Error Probability	
(see Probability)	A human error probability is a specific type of probability applicable to human reliability analysis and is defined under “Probability.”
Human Failure Event, Human Error Event	
A basic event that represents a failure or unavailability of a component, system, or function that is caused by human inaction, or inappropriate action. *(see Human Action, Human Error)*	In a PRA, potential human errors (i.e., human actions or inappropriate human actions) are modeled as basic events. The term human failure event is synonymous with and has replaced the term human error in the PRA lexicon. Human failure events can be classified as either errors of omission or errors of commission. An error of omission would be failure to perform a system-required task or action. An error of commission would be incorrectly performing a system-required task or action, or performing an extraneous task that is not required and could contribute to component, system, or function failure or unavailability. In the PRA, failures to restore a function, referred to as recovery, are also modeled as human failure events. The terms human failure event and human error event have the same meaning in a PRA context and it is correct and appropriate to use them interchangeably. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Human Reliability Analysis	
A structured approach used to identify potential human failure events and to systematically estimate the probability of those events using data, models, or expert judgment. *(see Human Action, Human Error)*	In a PRA, a human reliability analysis is used to identify relevant human actions and possible human errors that might occur. Human actions considered in the human reliability analysis include those actions that plant personnel might fail to perform or might fail to perform correctly. Failure to correctly perform certain human actions can lead to a reduced capability of responding to a transient or accident, including the loss of one or more critical safety functions. The failure to correctly perform a human action is referred to as a human error. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).

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Importance Measure (Risk Reduction Worth, Risk Achievement Worth, Fussell-Vesely, Birnbaum Importance, Uncertainty Importance)	
A metric that provides either the absolute or relative contribution of a component, system, structure, or human action to the defined risk.	In a PRA, importance measures are used to determine the contribution of the basic events to a number of risk metrics, such as core damage frequency. By using importance measures, the PRA analyst can determine the risk-significance of structures, systems, and components (SSCs) or human actions. Different importance measures provide different perspectives. For example, importance measures can evaluate the risk-reduction potential of improving SSC performance or human action, or they can show the significance of an SSC or human failure event for maintaining the current risk level. There are five importance measures typically used in a PRA: • Risk Reduction Worth : As defined in NUREG/CR-3385 (Ref.71), risk reduction worth is: "The decrease in risk if a plant feature (e.g., system or component) were assumed to be optimized or were assumed to be made perfectly reliable. Depending on how the decrease in risk is measured, the risk reduction worth can either be defined as a ratio or an interval." • Risk Achievement Worth: The increase in risk if a plant feature (e.g., system or component) was assumed to be failed or was assumed to be always unavailable. Depending on how the increase in risk is measured, the risk achievement worth can either be defined as a ratio or an interval. Sometimes risk achievement worth is referred to as "risk increase." • Fussell-Vesely: For a specified basic event, Fussell-Vesely importance is the relative contribution of a basic event to the calculated risk. This relative or fractional contribution is obtained by determining the reduction of the risk if the probability of the basic event to zero. • Birnbaum Importance (Bi): NUREG-1489 (Ref.54) defines birnbaum importance as: "An indication of the sensitivity of the accident sequence frequency to a particular basic event." Bi measures the change in total risk as a result of changes to the probability of an individual basic event. • Uncertainty Importance: The uncertainty in each input parameter, as expressed through its probability distribution, contributes to the uncertainty in the output parameter of interest (e.g., core damage frequency). The uncertainty importance measure attempts to quantify the contribution of each individual basic event's uncertainty to this total output uncertainty. The uncertainty importance is the Birnbaum importance multiplied by the standard deviation of the input probability distribution (Ref.83).
Important to Safety	
(see Safety Significant)	The term important to safety has a safety connotation and is defined under "Safety Significant."
Incremental Conditional Probability (Core Damage, Large Early Release)	
A measure of the impact of a temporary plant modification on the probability of an undesired end state. *(see Conditional Probability,*	As applied to PRA and plant risk evaluations, the term incremental conditional probability refers to the change in the probability of an undesired plant end state attributable to (conditional on) a temporary modification in plant configuration or operations, over the time that the modification is in place. Usually, this incremental change in conditional probability is reflected as an increase in the probability of an undesired end state such as core damage when compared to the baseline core damage probability. Because the probability of core damage depends on the temporary modification or change at the plant, it is therefore a conditional probability. Incremental conditional probability also is calculated in a PRA for large early release.

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Instantaneous Conditional Probability).	Incremental conditional probability differs from instantaneous conditional probability in that instantaneous conditional probability represents the probability that an undesired plant end state is reached given an initiating event and the actual (instantaneous) plant configuration. The incremental conditional probability is integrated over the duration of the temporary condition, while the instantaneous conditional probability represents a point-in-time measure.
Induced Steam Generator Tube Rupture	
(see Consequential Steam Generator Tube Rupture)	The term induced steam generator tube rupture is similar in definition to, and is grouped with, the term "Consequential Steam Generator Tube Rupture."
Ingestion	
Exposure from intake of food and water contaminated with radioactive material. *(see Exposure Pathways, Exposure, Exposure Time, Cloudshine, Water Immersion, Groundshine, Inhalation, Skin Deposition, Health Effects)*	In a Level 3 PRA, for the consequence calculation ingestion is one of the assumed pathways by which an individual can receive doses. The pathways of exposure include: (1) direct external exposure from radioactive material in a plume, principally due to gamma radiation (air immersion or cloudshine), (2) direct exposure from radioactive material in contaminated water given to an individual immersed in the water, (3) exposure from inhalation of radioactive materials in the cloud and resuspended material deposited on the ground, (4) exposure to radioactive material deposited on the ground (groundshine), (5) radioactive material deposited onto the body surfaces (skin deposition), and (6) ingestion from deposited radioactive materials that make their way into the food and water pathway.
Inhalation	
Exposure from breathing radioactive material. *(see Exposure Pathways, Cloudshine, Water Immersion, Groundshine, Ingestion, Skin Deposition)*	In a Level 3 PRA, for the consequence calculation inhalation is one of the assumed pathways by which an individual can receive doses. The pathways of exposure include: (1) direct external exposure from radioactive material in a plume, principally due to gamma radiation (air immersion or cloudshine), (2) direct exposure from radioactive material in contaminated water given to an individual immersed in the water, (3) exposure from inhalation of radioactive materials in the cloud and resuspended material deposited on the ground, (4) exposure to radioactive material deposited on the ground (groundshine), (5) radioactive material deposited onto the body surfaces (skin deposition), and (6) ingestion from deposited radioactive materials that make their way into the food and water pathway.
Initiating Event, Initiator	
An event that perturbs the steady-state operation of the plant and could lead to an undesired plant condition.	In a PRA, an initiating event is an event originating from an internal or external hazard that both challenges normal plant operation and requires successful mitigation. As such, these events represent the beginning of accident sequences modeled in the PRA. Having a reasonably complete set of initiating events is crucial in determining what events could propagate to core damage. Initiating events can arise from the following: • <u>Internal Hazards, which include:</u> — Internal event (*see Internal Event*)

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	— Floods (*see Internal Flood*) — Fires (*see Appendix A for fire terms*) • <u>External Hazards, which include:</u> — Floods (*see External Flood*) — High winds (*see High Winds*) — Seismic events (*see Hazard Analysis*) — Other external hazards These hazards result in different types of initiating events. Examples of initiating events are transients (*see Transient*) and loss-of-coolant accidents (*see Loss-of-Coolant Accident*). The terms initiating event and initiator are both used in a PRA context and generally have the same meaning. In some cases, the term initiator may refer to a class of initiators (e.g., transient), while the term initiating event may refer to the actual event (e.g., loss of a feedwater pump resulting in a transient). The ASME/ANS PRA Standard (Ref. 2) defines an initiating event as "an event either internal or external to that which perturbs the steady state operation of the plant by challenging plant control and safety systems whose failure could potentially lead to core damage or release of airborne fission products. These events include human-caused perturbations and failure of equipment from either internal plant causes (such as hardware faults, floods, or fires) or external plant causes (such as earthquakes or high winds)."
Initiating Event Analysis	
The process used to identify events that perturb the steady- state operation of the plant and could lead to an undesired plant condition. *(see Initiating Event, Master Logic Diagram)*	In a PRA, the initiating event analysis considers how accidents can start by identifying and quantifying those events that challenge plant operation and require successful mitigation to prevent core damage from occurring. To facilitate the efficient modeling of potential accidents, initiating events typically are identified using a systematic process (e.g., master logic diagram) and grouped according to their mitigation requirements. The frequencies of these initiating event groups are then quantified. NRC Regulatory Guide 1.200 (Ref. 91) states that initiating event analysis "identifies and characterizes the events that both challenge normal plant operation during power or shutdown conditions and require successful mitigation by plant equipment and personnel to prevent core damage from occurring. Events that have occurred at the plant and those that have a reasonable probability of occurring are identified and characterized. An understanding of the nature of the events is performed such that a grouping of the events, with the groups defined by similarity of system and plant responses (based on the success criteria), may be performed to manage the large number of potential events that can challenge the plant."
Initiating Event Frequency	
(see Frequency)	The term initiating event frequency is a type of frequency that is defined under "Frequency."
Initiator	
(see Initiating Event)	The term initiator is similar in meaning to initiating event and is defined under "Initiating Event."
Instantaneous Conditional Probability (Core Damage, Large Early Release)	
Event probability at the specific time the plant is analyzed, given that a prior event	Using a PRA, instantaneous conditional probability can be calculated for core damage and large early release. The probability of either of those undesired outcomes occurring depends on the occurrence of an initiating event while the plant is in a given configuration. Thus, core damage or large early release is "conditional" on the probability of a prior event occurring.

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has occurred. *(see Conditional Probability, Incremental Conditional Probability)*	The following are other definitions that could describe instantaneous conditional probability: • The probability that an undesired plant end state is reached given an initiating event and the actual (instantaneous) plant configuration. • The average probability that an undesired plant end state is reached, weighted over all credible initiating events, for the actual (instantaneous) plant configuration. Instantaneous conditional probability differs from incremental conditional probability in that incremental conditional probability represents the impact of a temporary plant modification on the probability of an undesired end state. The incremental conditional probability is integrated over the duration of the temporary condition, while the instantaneous conditional probability represents a point-in-time measure.
Interfacing-Systems Loss-of-Coolant Accident	
A loss-of-coolant accident characterized by high-pressure reactor coolant being released into a low-pressure system. *(see Loss-of-Coolant Accident)*	In a PRA, accidents involving an interfacing-systems loss-of-coolant accident (ISLOCA) are modeled because they represent a loss of isolation between an ancillary system and the reactor coolant system, which contains radioactive material. This type of accident is important in the PRA because it may lead to radioactive material bypassing containment and loss of reactor coolant inventory. The ASME/ANS PRA Standard (Ref. 2) defines ISLOCA as “a loss of coolant accident (LOCA) when a breach occurs in a system that interfaces with the reactor coolant system, where isolation between the breached system and the reactor coolant system fails. An ISLOCA is usually characterized by the over-pressurization of a low-pressure system when subjected to reactor coolant system pressure and can result in containment bypass.” ISLOCAs of most concern are those accidents during which the break flow is discharged outside the reactor containment building. The two main reasons for this concern are: (1) potential high offsite radiological consequences caused by radioactive material bypassing the containment and (2) potential loss of long-term core cooling resulting from loss of reactor coolant system inventory that would otherwise be available for recirculation from the containment sumps.
Internal Event	
Failure of equipment as a result of either an internal random cause or a human event which perturbs the steady-state operation of the plant and could lead to an undesired plant condition. *(see Hazard)*	In a PRA, internal events result from or involve random mechanical, electrical, structural, or human failures within the plant boundary and are a specific hazard group. An example of an internal event modeled in a PRA would be the random structural failure of a reactor coolant system pipe resulting in a loss-of-coolant accident (LOCA) initiating event. Until the 2009 ASME/ANS PRA Standard revision (Ref. 2), this term did not have a consistent definition. In some cases, a fire or flood or both occurring within the plant were considered an internal event. The ASME/ANS PRA Standard has been revised and internal flood and internal fire are not considered internal events. The ASME/ANS PRA Standard (Ref.2) defines an internal event as “an event resulting from or involving random mechanical, electrical, structural, or human failures from causes originating within a nuclear power plant that directly or indirectly causes an initiating event and may cause safety system failures or operator errors that may lead to core damage or large early release. By historical convention, loss of offsite power is considered to be an internal event, and internal fire is considered to be an external event, except when the loss is caused by an external hazard that is treated separately (e.g., seismic-induced loss of offsite power). Internal floods sometimes have been included with internal events and sometimes considered as external events. For this standard, internal floods are considered to be internal hazards separate from internal events.”

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Internal Fire	
A fire initiated within the plant that can affect the operability of the plant. *(see Hazard and Appendix A)*	In a PRA, internal fires are a specific hazard group in which the fire occurs within the plant boundary. The PRA considers fires because they have the potential to cause equipment failure by direct flame impact or high thermal radiation.
Internal Flood, Internal Flooding Event	
A flood initiated within the plant that can affect the operability of the plant. *(see Hazard, External Flood)*	In a PRA, internal floods are a specific hazard group in which the flood occurs within the plant boundary. The PRA considers floods because they have the potential to cause equipment failure by the intrusion of water into plant equipment through submergence, spray, dripping, or splashing. The term internal flooding event represents the occurrence of an internal flood.
Internal Flooding Event	
(see Internal Flood)	The term internal flooding event is the occurrence of an internal flood and is defined under "Internal Flood."
Internal Hazard	
(see Hazard)	The term internal hazard is a specific type of hazard and is defined under "Hazard."
Key Assumption	
(see Assumption)	The term key assumption is a specific type of assumption and is defined under "Assumption."
Key Model Uncertainty	
(see Uncertainty)	The term key model uncertainty is a type of uncertainty and is defined under "Uncertainty."
Key Source of Model Uncertainty	
(see Uncertainty)	The term key source of model uncertainty is defined under "Uncertainty."
Key Source of Uncertainty	
(see Uncertainty)	The term key source of uncertainty is defined under "Uncertainty."
Land Contamination	
Contamination of land outside of the nuclear power plant site boundary with radioactive material released in an accident. *(see Health Effects)*	In a Level 3 PRA, land contamination often is evaluated along with health effects. Land contamination refers to the radioactive material deposited on the ground by gravitational settling or the impact during plume passage. Land contamination depends on the characteristics of the radioactivity release and how the land surrounding the plant is used. Land contamination risk involves the frequency and amount of land contamination and its associated cost.

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Land Contamination Risk	
(see Land Contamination)	Land contamination risk is sometimes calculated in a Level 3 PRA and is defined in the discussion under "Land Contamination."
Large Early Release	
(see Radioactive Material Release)	The term large early release is a type of radioactive material release and is defined in the discussion under "Radioactive Material Release."
Large Early Release Frequency	
(see Frequency)	The term large early release frequency is a type of frequency used in PRA calculation and is defined in the discussion under "Frequency."
Large Early Release Frequency Analysis	
(see Radioactive Material Release Frequency Analysis)	The term large early release frequency analysis is a type of radioactive material release frequency analysis and is defined under "Radioactive Material Release Frequency Analysis."
Large Late Release	
(see Radioactive Material Release)	The term large late release is a type of radioactive material release and is defined in the discussion under "Radioactive Material Release."
Large Late Release Frequency	
(see Frequency)	The term large late release frequency is a type of frequency used in PRA calculation and is defined in the discussion under "Frequency."
Large Late Release Frequency Analysis	
(see Radioactive Material Release Frequency Analysis)	The term large late release frequency analysis is a type of radioactive material release frequency analysis and is defined under "Radioactive Material Release Frequency Analysis."
Large Release	
Formal definition requires Commission approval. *(see Radioactive Material Release)*	The notion of a large release implies that in the range of possible releases there exists a threshold value that distinguishes large releases from not large releases. Many PRAs include their own specific definitions of a large release, but no universally accepted definition has been established. Attempts have been made to define a large release magnitude based on offsite health effects. There is an inherent arbitrariness in definitions since offsite health effects depend not only on release magnitude but also on site-specific parameters, such as population. Therefore, what would be a large release at one site would not necessarily be one at another site. Weather and wind variability are other site-specific factors. In the past, the NRC staff has considered several alternate definitions of a large release. These include: • A release that would result in one or more early fatalities;

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	• A release that has the potential to result in one early offsite fatality within 1 mile of the plant boundary; • A definition of a large release source term in the traditional form of a fractional release of the core inventory of various radionuclide groups to the environment, the timing of the release, etc. • Any release from an event that involves severe core damage, primary system pressure boundary failure, and early containment failure. The Commission has not approved a formal definition for the term large release.
Large Release Frequency	
(see Frequency)	The term large release frequency is a type of frequency used in PRA calculation and is defined in the discussion under “Frequency.”
Late Containment Failure	
(see Containment Failure)	The term late containment failure is a type of containment failure and is defined under “Containment Failure.”
Latent Cancer Fatality	
(see Fatality)	The term latent cancer fatality is a type of fatality caused by exposure to radioactive materials and is defined under “Fatality.”
Latent Fatality	
(see Fatality)	The term latent fatality is a type of fatality caused by exposure to radioactive materials and is defined under “Fatality.”
Latent Fatality Risk	
(see Fatality)	The term latent fatality risk is a type of risk-involved fatality caused by exposure to radioactive materials and is defined under “Fatality.”
Latent Health Effects	
(see Health Effects)	The term latent health effect refers to a type of health effect and is defined in the discussion under “Health Effects.”
Level 1, 2, 3 PRA	
A characterization of the scope of a PRA in terms of increasing specification of consequences. *(see PRA)*	The three types of PRA are distinguished by the risk metric calculated, and when all three are calculated for a particular plant, it is referred to as a full-scope PRA. Level 1 refers to core damage frequency as the risk measure, Level 2 refers to radioactivity releases as the risk measure, and Level 3 refers to offsite consequences as the risk measure. A Level 2 PRA takes the results of the Level 1 PRA (accident sequences resulting in core damage) as input and produces frequencies of radioactivity releases as output. A Level 3 PRA takes the results of the Level 2 PRA as input and produces offsite consequences (health effects, economic consequences) as output. In some usages, a Level 2 PRA includes the Level 1 analysis, and the Level 3 PRA includes both the Level 1 and Level 2 analyses. The figure below illustrates the different PRA “Levels” and what each calculates.

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	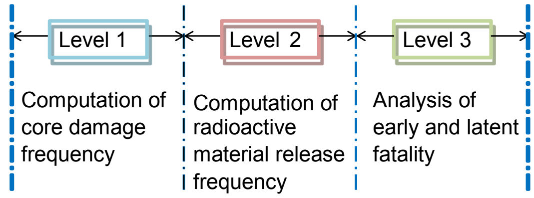
Level of Detail	
The degree of resolution or specificity in the analyses performed in the PRA. *(see Model, Capability Categories)*	In a PRA, the level of detail generally refers to the level to which a system is modeled (e.g., function level, train level, component level), the extent to which systems are included in the success criteria (e.g., safety systems and nonsafety systems), the extent to which phenomena are included in the challenges to the plant in the Level 2 analysis, and the extent to which operator actions are considered (e.g., accident management strategies). Level of detail generally is dictated by four factors: (1) the level of detail to which information is available, (2) the level of detail required so that dependencies are included, (3) the level of detail so that the risk contributors are included, and (4) the level of detail sufficient to support the application. In the ASME/ANS PRA Standard (Ref. 2), the degree to which the level of detail (and scope) of the plant design, operation, and maintenance are modeled forms one of the bases for the capability categories defined in the Standard.
Licensing Basis	
The collection of documents or technical criteria that provides the basis upon which the NRC issues a license to construct or operate a nuclear facility.	A PRA is part of the licensing basis for plants licensed under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” A PRA also is used to support changes to the licensing basis carried out using regulatory guidance documents such as Regulatory Guide (RG) 1.174 (Ref. 84), RG 1.175 (Ref. 85), or RG 1.177 (Ref. 86). The NRC Web site Glossary (Ref. 36) defines licensing basis as “the collection of documents or technical criteria that provides the basis upon which the NRC issues a license to construct or operate a nuclear facility; to conduct operations involving the emission of radiation; or to receive, possess, use, transfer, or dispose of source material, byproduct material, or special nuclear material.” 10 CFR Part 54 (Ref. 27) defines current licensing basis (CLB) as “the set of NRC requirements applicable to a specific plant and a licensee’s written commitments for ensuring compliance with and operation within applicable NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect.” The CLB includes NRC regulations, orders, license conditions, exemptions, technical specifications, final safety analysis reports, and licensee commitments to NRC bulletins, generic letters, enforcement actions, and licensee event reports. The definition provided was based on the definition in the NRC Web site Glossary (Ref. 36).
Licensing-Basis Event	
A postulated accident that a nuclear facility must be designed and built to withstand.	The term licensing-basis event (LBE) is not used in current PRAs or the current nuclear power plant regulatory licensing structure. It is a term being used for a potentially new regulatory process. Further information on this regulatory framework can be found in NUREG-1860 (Ref. 63). This potential future licensing structure is a process that uses both deterministic and probabilistic criteria for selecting the postulated accidents, called LBEs, which a nuclear facility

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	must demonstrate it can withstand (i.e., the facility design and operation must be able to withstand the impact of LBEs without loss to the structures, systems, and components (SSCs) needed to ensure public health and safety).
Linear No-Threshold Model	
A dose response model that assumes cancer risk is proportional to the dose received no matter how small the dose. *(see Dose, Dose Response Model)*	In a Level 3 PRA, a dose response model is used to calculate the cancer risk for given levels of a dose to individuals after a severe accident. There is some debate on the appropriate dose response relationship for cancer risk following exposure to ionizing radiation. A linear relationship in which the cancer risk increases in direct proportion to the dose is one view. Another view advocates a nonlinear relationship, in which cancer risk increases in a more complex manner relative to dose. There is also a question about whether a minimum dose exists, below which no increased risk of cancer is found (threshold model), or whether any dose, no matter how small, increases cancer risk (no-threshold model).
Living PRA	
A probabilistic risk assessment that is maintained so that it reflects the current plant design and operational features. *(see Dynamic PRA, PRA Configuration Control, As-Built As-Operated)*	The term living PRA designates a PRA that is updated as necessary to reflect any changes in the plant (e.g., design, operating procedures, data) to continue to represent the as-built as-operated plant. Therefore, the living PRA can be used in risk-informed decisionmaking processes, such as plant-specific changes to the licensing basis discussed in NRC Regulatory Guide 1.174 (Ref. 84). PRA configuration control is part of the process used to support a living PRA. A living PRA is not the same as a dynamic PRA. A dynamic PRA refers to a PRA that accounts for time-dependent effects by integrating these effects directly into the computer model.
Loss-of-Coolant Accident (Small, Medium, Large)	
An accident that results in a loss of coolant from the reactor. *(see Interfacing-Systems Loss-of-Coolant Accident)*	In a PRA, two major categories of initiating events are evaluated; namely, transients and loss-of-coolant accidents (LOCAs). LOCAs represent a particularly challenging accident because reactor coolant, usually water, cannot be replaced at a sufficient rate to prevent uncovering the reactor core leading to core damage and potential fueling melting. Once considered to be the most severe design-basis accident, PRAs have revealed that other accident initiators, such as long-term station blackout, are far more frequent and can lead to equally undesired consequences. LOCA initiating event frequencies used in the PRA are dependent on the size of LOCA. These sizes are typically referred to as small, medium, or large LOCAs. The break sizes which define small, medium, and large LOCAs are also dependent on the type of reactor, either PWR or BWR, and whether the lost coolant is liquid or steam. NUREG/CR-6928 (Ref. 82) provides the following description for BWRs: • Small LOCA (SLOCA): A break size less than 0.004 square feet (1-inch inside diameter pipe equivalent) for liquid and less than 0.05 square feet (approximately 4-inch inside diameter pipe equivalent) for steam in a primary system pipe with leakage rate greater than100 gallons per minute. • Medium LOCA (MLOCA): A break size between 0.004 to 0.1 square feet (approximately 1- to 5-inch inside diameter pipe equivalent) for liquid and between 0.05 to 0.1 square feet (approximately 4- to 5-inch inside diameter pipe equivalent) for steam in a primary system pipe.

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	• <u>Large LOCA (LLOCA)</u>: A break size greater than 0.1 square feet (approximately 5-inch inside diameter pipe equivalent) for liquid or steam in a primary system pipe. NUREG/CR-6928 (Ref. 82) also provides the following description for PWRs: • <u>Small LOCA (SLOCA)</u>: A pipe break in the primary system boundary with an inside diameter between 0.5- and 2-inches • <u>Medium LOCA (MLOCA)</u>: A pipe break in the primary system boundary with an inside diameter between 2- and 6- inches. • <u>Large LOCA (LLOCA)</u>: A pipe break in the primary system boundary with an equivalent inside diameter greater than 6-inches. Historically, NUREG-1150 (Ref. 51) defines SLOCA as < 1 inch, MLOCA as 1 to 5 inches, and LLOCA as > 5 inches for BWRs and SLOCA as 0.5 to 2 inches, MLOCA as 2 to 6 inches, and LLOCA as > 6 inches for PWRS. Appendix A to 10 CFR Part 50 (Ref. 22) and the NRC Web site Glossary (Ref. 34) define the term LOCAs as “those postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.”
Loss of Offsite Power	
The loss of all AC power from the electrical grid to the plant. *(see Transient)*	In a PRA, loss of offsite power (LOOP) is referred to as both an initiating event and an accident sequence class. As an initiating event, LOOP to the plant can be a result of a weather-related fault, a grid-centered fault, or a plant-centered fault. During an accident sequence, LOOP can be a random failure. Generally, LOOP is considered to be a transient initiating event. NUREG/CR-6890 (Ref.80) defines a LOOP as “the simultaneous loss of electrical power to all plant safety buses, requiring all emergency power generators to start and supply power to the safety buses.”
Low-Power and Shutdown	
The states of nuclear power plant operation when the reactor is producing power in a range below a specified level or is shutdown. (see *Full Power, At-Power*)	A PRA models the different plant operating states (POSs) of the plant. Operation at-power is one POS, while several POSs are needed to characterize the plant during the various stages of low-power and shutdown. These POSs are distinguished in the PRA model because the plant response (e.g., accident sequences) differs during different POSs. Low power and shutdown is the term applicable for other than at-power conditions (i.e., the reactor is typically producing less than 15-25% of its rated power). Low-power and shutdown analysis is further separated into consideration of low power and shutdown states. In a low-power initial condition, the core is producing power from fissioning of fuel, over and above the decay heat levels, although at lower amounts than at-power. Most safety systems are on automatic actuation but some may be disabled or blocked (e.g., main feedwater trip in boiling-water reactors). The support systems are aligned in their normal configuration (e.g., electrical power is being drawn from the grid). In these POSs, the power level may be changing as the reactor is shutting down or starting up, or the power level may be constant at a reduced level. The power level that distinguishes nominal full power from low power is the power level below which major plant evolutions are required to reduce or increase power that significantly increase the likelihood of a plant trip (e.g., taking manual control of feedwater level).

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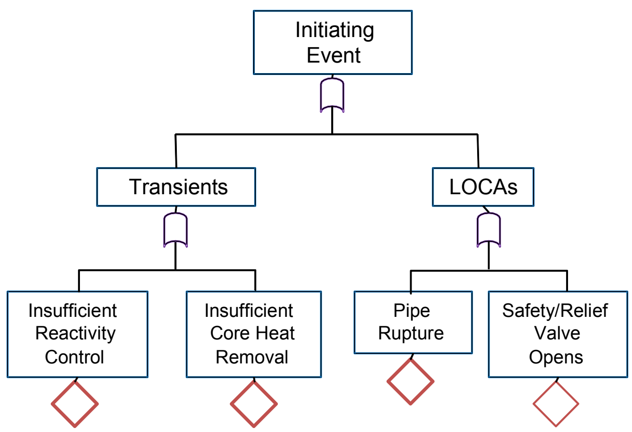
	In shutdown conditions, the core is not producing power (i.e., the reactor is subcritical). The reactor temperature and pressure are lower than at-power, coolant inventory may be lower or higher, the reactor may be relying on alternate cooling systems, some safety systems may be defeated, or containment may be open. A representation of the different plant operating states (i.e., low power and shutdown) is shown under the discussion for the term At-Power.
Master Logic Diagram	
A graphical model that can be constructed to guide the selection of initiating events. *(see Fault Tree)*	In a PRA, a master logic diagram (MLD) is often used to identify the specific events that are potential initiating events and to group them according to the challenges they pose to plant safety. An MLD is developed using fault tree logic to show general categories of initiating events proceeding to increasingly detailed information at lower levels, with specific initiating events presented at the bottom level. In a more general sense, an MLD is a fault tree identifying all the hazards that affect a system or mission. An MLD generally uses a fault tree logic approach to identify the logic or relationship between events. However, the difference between an MLD and a fault tree is that a fault tree focuses on accounting for the specific causes leading to failure of a system or group of systems, whereas the MLD focuses on listing the hazards that can affect a top event. An example of an MLD is provided below.  The ASME/ANS PRA Standard (Ref. 2) defines an MLD as a "summary fault tree constructed to guide the identification and grouping of initiating events and their associated sequences to ensure completeness."
Mean	
The expected value of a random variable. *(see Median, Best Estimate, Point Estimate, Probability Distribution)*	In a PRA, the metrics (e.g., core damage frequency, large early release frequency) generally are evaluated and presented as mean values to reflect the uncertainties in the parameter values used as input to the evaluation of the metrics. The mean values and the distributions from which they are calculated can be used to address the parameter uncertainties. The mean is the average value from a probability distribution. It is the expected value one would get from many samples taken of the random variable. The random variable in question could be a risk parameter, such as a component failure probability, or a risk measure, such as core damage frequency. The mean and median provide different information and cannot be used interchangeably. An illustration of the difference between mean and median is shown below.

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	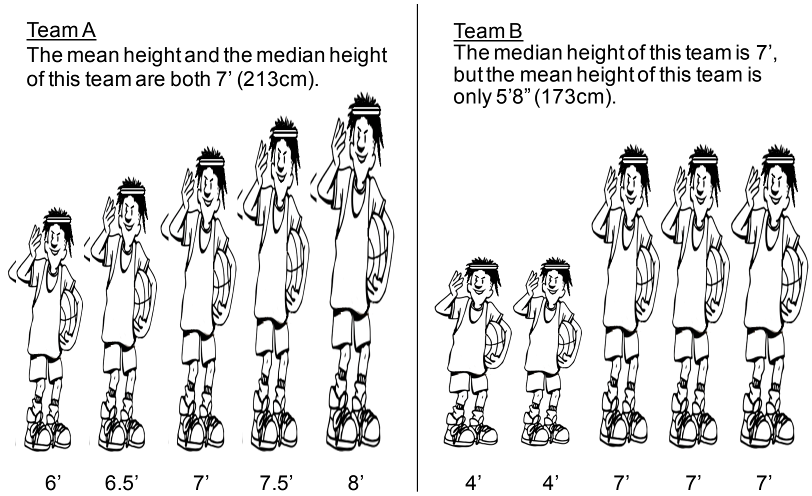
Mechanistic Source Term	
A source term that is calculated considering the characteristics of specific accidents. (*see Source Term)*	In a Level 2 PRA, the source term calculated is usually a mechanistic source term. A mechanistic source term is calculated using validated models and supporting scientific data that simulate the physical and chemical processes that describe the radioactive material inventories and the time-dependent radioactive material transport mechanisms necessary and sufficient to predict the source term. For licensing calculations not involving a PRA, current light-water reactors (LWRs) use a deterministic predetermined source term into containment for different accidents, instead of a mechanistic source term, to analyze the effectiveness of the containment and site suitability for licensing purposes.
Median	
That value of a random variable for which the occurrence of larger values is just as likely as occurrence of smaller values. *(see Mean, Probability Distribution)*	In a PRA, median values are not usually calculated. In some cases, median values of the risk metric are calculated in addition to the mean to provide a perspective on the distribution of the risk metric. Conclusions can be made about the spread and shape of a probability distribution of a risk metric or a parameter by comparing the median to the mean and to the other quantiles. The median is the middle value in a probability distribution. It is a reference point in which half the data values in a probability distribution (e.g., uncertainty distribution) lie below it and half lie above it. For example, if the median of a failure rate of a particular type of electric motor is $2x10^{-4}$/hr then half of all electric motors of that type would have failure rates below $2x10^{-4}$/hr and half would have failure rates above $2x10^{-4}$/hr. An illustration of the difference between mean and median is under the discussion of the term "Mean."
Minimal Cutset	
(see Cutset)	The term minimal cutset is a type of cutset used in PRA and is defined under "Cutset."
Mission Time	
The time period that a system or component is required to operate	In a PRA, the failure probability of a component to operate is directly related to its mission time. By convention, in a Level 1 internal events PRA, mission time usually is specified as 24 hours. After that initial time period, multiple options for dealing with the accident would become available so that the residual risk results, beyond the 24-hour timeframe, would be negligibly

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to successfully perform its function.	small. For Level 1 PRAs that examine external hazards, the mission times usually are longer (e.g., 72 hours) because of area wide effects of such events. The definition provided is based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Mitigating System	
A plant system designed to minimize the effects of initiating events. *(see Initiating Event, Front-Line System, Support System)*	In a PRA, the accident mitigating functions and mitigating systems modeled are based on the initiating event(s) being analyzed. Mitigating systems can prevent an accident or reduce the consequences of a potential accident by directly performing or supporting one or more accident mitigating functions (e.g., core or containment cooling, coolant makeup, reactivity control, or reactor vessel pressure control). Front-line systems are mitigating systems that directly perform an accident mitigating function. Typically, support systems (e.g., electric power, control power, or cooling) are required to enable the operation of systems that directly perform an accident mitigating function. In this regard, support systems also may be considered mitigating systems.
Model (PRA)	
A representation of a physical process or system that allows one to predict the system's behavior. *(see Uncertainty)*	The term "model" is used in a variety of ways in a PRA: • The entire PRA is sometimes referred to as a PRA model or risk model. • Different submodels are used inside the PRA in the performance of the various technical elements (system model, human reliability analysis model). • Other submodels may be phenomenological models (e.g., direct containment heating or core-concrete interaction). All of these types of models may be sources of model uncertainty in the PRA.
Model Uncertainty	
(see Uncertainty)	The term model uncertainty is related to epistemic uncertainty and is defined under "Uncertainty."
Nonsafety Related	
(see Safety Significant)	The term nonsafety related indicates the safety category of a structure, system, or component and is defined under "Safety Significant."
Operating-Basis Earthquake	
An earthquake that could be expected to affect the site of a nuclear reactor, but for which the plant's power production equipment is designed to remain functional without undue risk to public health and safety. *(see Safe-Shutdown Earthquake)*	In a seismic PRA, the operating-basis earthquake (OBE) is sometimes used in the initiating event selection process to develop a hierarchy to ensure that every earthquake greater than a certain defined size produces a plant shutdown within the systems model. As noted in the ASME/ANS PRA Standard (Ref. 2), it is generally a requirement at all nuclear power plants that any earthquake larger than a certain size—usually defined as the OBE—will require the plant to shut down to reduce energies that may cause loss-of-coolant accidents and to enable inspection for possible earthquake-caused damage. The ASME/ANS PRA Standard (Ref. 2) defines an OBE as "that earthquake for which those features of the nuclear power plant necessary for continued operation without undue risk to health and safety are designed to remain functional. In the past, the OBE was commonly chosen to be one-half of the safe shutdown earthquake (SSE)." The definition provided is based on the definition in the NRC Web site Glossary (Ref. 36).

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Operator Action	
(see Human Action)	The term operator action is a specific type of human action that is defined under the term "Human Action."
Operator Error	
(see Human Error)	The term operator error is a specific type of human error that is defined under the term "Human Error."
Other External Hazard Fragility Evaluation/ Analysis	
(see Fragility Analysis)	The term other external hazard fragility analysis is a type of fragility analysis and is included in the discussion under "Fragility Analysis."
Other External Hazard Plant Response Analysis/Model	
(see Plant Response Analysis)	The term other external hazard plant response analysis is a type of plant response analysis and is included the discussion under "Plant Response Analysis/Model."
Other Hazards Analysis	
(see Hazard Analysis)	The term other hazards analysis is a specific type of hazard analysis and is defined under the term "Hazard Analysis."
Parameter	
The variables used to calculate and describe frequencies and probabilities. (*see Uncertainty, Point Estimate)*	In a PRA, parameters are used directly in supporting PRA models. Initiating event frequencies, component failure rates and probabilities, and human error probabilities are several parameters used in quantifying the accident sequence frequencies. Generally accepted probability models exist for many of the basic events modeled in the PRA model. These "basic event" models typically are simple mathematical models with only one or two parameters. An example is the simple constant failure rate reliability model, which assumes that the failures of components in a standby state occur at a constant rate. The parameter(s) of such models may be estimated using appropriate data, which, in the example above, may come from the number of failures observed in a population of like components in a given period of time. Statistical uncertainties are associated with the estimates of the model's parameters. Because most of the events that constitute the building blocks of the risk model (e.g., some initiating events, operator errors, and equipment failures) are relatively rare, the data are scarce and the uncertainties can be relatively significant.
Parameter Uncertainty	
(see Uncertainty)	The term parameter uncertainty is related to epistemic uncertainty and is defined under "Uncertainty."
Passive Component	
A component whose operation or function does not depend on an external source of motive power. (*see Active Component)*	In a PRA, both passive and active components are modeled. A passive component has no moving parts, and it can experience changes in pressure, temperature, or fluid flow in performing its functions. Some examples of passive components include heat exchangers, pipes, vessels, and electrical cables and structures. The IAEA Safety Glossary (Ref. 7) defines passive components as "a component whose functioning does not depend on an external input such as actuation, mechanical movement, or supply of power."

<table>
<tr><th>TERM AND DEFINITION</th><th>DISCUSSION</th></tr>
<tr><td colspan="2">Performance-Based (Approach, Regulation, Regulatory Action)</td></tr>
<tr><td>Focusing on measurable outcomes, rather than prescriptive processes, techniques, or procedures. (see Risk-Based)</td><td>In a PRA, a quantitative evaluation is made about the performance of the plant in response to potential accident conditions. The results of this evaluation can be used to support a performance-based approach to plant operations in which measureable outcomes are used to show compliance with regulation.

NUREG/BR-0318 (Ref. 66) defines the term performance-based as “an approach to regulatory practice that establishes performance and results as the primary bases for decision-making. Performance-based regulations have four common attributes: (1) Measurable, calculable, or objectively observable parameters exist or can be developed to monitor performance. (2) Objective criteria exist or can be developed to assess performance. (3) Licensees have flexibility to determine how to meet the established performance criteria in ways that encourage and reward improved outcomes. (4) A framework exists or can be developed in which the failure to meet a performance criterion, while undesirable, will not constitute or result in an immediate safety concern.”

The terms performance-based regulation and performance-based regulatory action are defined below based on the NRC Web site Glossary (Ref. 36):

• <u>Performance-Based Regulation</u>: “A regulatory approach that focuses on desired, measurable outcomes, rather than prescriptive processes, techniques, or procedures. Performance-based regulation leads to defined results without specific direction regarding how those results are to be obtained. At the NRC, performance-based regulatory actions focus on identifying performance measures that ensure an adequate safety margin and offer incentives for licensees to improve safety without formal regulatory intervention by the agency.”

• <u>Performance-Based Regulatory Action</u>: “Licensee attainment of defined objectives and results without detailed direction from the NRC on how these results are to be obtained.”</td></tr>
<tr><td colspan="2">Performance-Based Approach</td></tr>
<tr><td>(see Performance-Based)</td><td>The term performance-based approach indicates an evaluation that is based on measureable outcomes and is defined under “Performance-Based.”</td></tr>
<tr><td colspan="2">Plant Configuration Control</td></tr>
<tr><td>The process of maintaining consistency between the physical condition of a nuclear plant and its associated design and engineering records.</td><td>A PRA relies on plant configuration control to ensure that the as-built as-operated plant is accurately modeled. Without plant configuration control, uncertainty can be introduced about the extent to which the PRA accurately reflects important characteristics of the plant (e.g., the design of plant structures, systems, and components (SSCs)).

Plant configuration control represents the process of identifying and documenting the characteristics (e.g., design or operating conditions) of plant SSCs, and of ensuring that changes to these characteristics are properly developed, assessed, approved, issued, implemented, verified, recorded, and incorporated into the facility documentation.</td></tr>
<tr><td colspan="2">Plant Damage State</td></tr>
<tr><td>A group of accident sequence end states that share similar characteristics with accident progression, and containment or engineered safety feature operability. (see Bin)</td><td>In a Level 2 PRA, the critical first step is developing a structured process for defining the specific accident conditions to be examined. Attributes have to be determined for binning the large number of accident sequences developed for Level 1 PRA analysis into a practical number for detailed Level 2 analysis. Combinations of attributes of similar accident conditions define the plant damage states.

The definition provided is based on the definition in the ASME/ANS PRA Standard (Ref. 2).</td></tr>
</table>

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Plant Hazard	
(see Hazard)	The term plant hazard has the same meaning as hazard and is defined under "Hazard."
Plant Operational Mode	
(see Plant Operational State)	The term plant operational mode has the same meaning as plant operational state and is defined with "Plant Operational State."
Plant Operational State, Plant Operational Mode	
A particular plant configuration with specified operational characteristics.	The scope of the PRA determines the various individual plant operational states (POSs) that the PRA model must include for the risk estimation results (i.e., if a PRA is being conducted for at-power operations, the plant configuration in that state or mode will be considered to obtain the risk results). The term plant operational state has the same meaning as plant operational mode. The plant conditions that define a POS usually include core decay heat level, primary water level, primary temperature, primary vent status, containment status, and decay heat removal mechanisms. A POS can be a steady state or represent a transition between steady-state POSs. For example, full power and cold shutdown while on residual heat removal cooling may be two steady-state POSs. To transition from full power to cold shutdown, there may be one or more transition POSs to cover the range of temperatures and pressures the plant goes through in shutting down to cold shutdown. Note that the impacts of unavailability of individual systems or components because of test or maintenance typically are not included as part of the definition of a POS. The complete set of POSs for a specific outage type shows a discrete representation of the outage from a risk perspective.
Plant Partitioning	
The defining of the plant physical boundary affected by the flood and fire hazard and the segmenting of the physical boundary into smaller spatial units.	In a PRA, plant partitioning is used in flood and fire evaluations to define the physical analysis units in terms of flood or fire areas and flood or fire compartments. In the ASME/ANS PRA Standard (Ref. 2), the objective of plant partitioning for internal floods (referred to as internal flood plant partitioning) is to account for plant-specific physical layouts and separations in such a way as to identify in the PRA plant areas where internal floods could lead to core damage.
Plant Response Analysis, Plant Response Model (External Floods, Internal Fire, High Winds, Other External Hazard, Seismic)	
The logic framework for identification and analysis of accident scenarios resulting from the effects of a hazard on the plant.	In a PRA conducted to evaluate the effect of an external hazard group on the plant, or the effect of internal fires on the plant, plant response analysis usually involves modification of the internal events PRA model. This modification includes the event trees and fault trees and the initiating event set. It involves identifying and selecting important initiating events, deleting unlikely events from event trees, deleting unimportant internal failures and human errors (from fault trees or event trees), modifying event tree logic to conform to event-specific procedures, and adding hazard event induced failure events and human errors (to fault trees and event trees). These modifications are performed when the plant response model is used in conducting an external flood, internal fire, high wind, seismic, or other external hazards analysis.

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	For example, in a seismic analysis, the initiating event is assumed to be a loss of offsite power. Recovery of offsite power is trimmed from the event trees. Seismic failures of structures and equipment are added and comparatively unimportant internal failures are trimmed. Human errors and their probabilities are adjusted. Mission time is extended, usually to 72 hours. A simplified plant response model also can be constructed "from scratch" (ad hoc model), without starting with the internal events model. Note that in an internal flood PRA the plant response also is determined in a manner similar to that described above. The ASME/ANS PRA Standard (Ref. 2) states that the expected plant response(s) to the selected set of flood scenarios is determined, and an accident sequence, from the internal events at power PRA that is reasonably representative of this response is selected for each scenario.
Plant Response Model	
(see Plant Response Analysis)	The term plant response model has the same meaning as plant response analysis and is defined under "Plant Response Analysis."
Plant Risk Profile	
(see Risk Profile)	The term plant risk profile has the same meaning as risk profile and is defined under "Risk Profile."
Point Estimate	
An estimate of a parameter in the form of a single value. *(see Mean)*	In a PRA, the preferred parameter point estimate is the mean of the value obtained from a probability distribution for the parameter. NUREG-1855 (Ref. 62) states, "a point estimate is a single value estimate for a parameter population. For example, the mean of a sample of values of a random variable X (i.e., expected value) is a commonly used point estimate of the mean of the distribution. When parameter distributions are not available, a maximum likelihood estimate or a value obtained from expert elicitation can serve as a point estimate." For a point estimate of a risk metric (e.g., core damage frequency) mean values of various parameters are used. The mean value of the risk metric usually is very close to this point estimate. The definition provided was based on the definition in NUREG/CR-6823 (Ref.78).
PRA, Probabilistic Safety Assessment (Base, Baseline)	
A systematic method for assessing the likelihood of accidents and their potential consequences. *(see Probability, Dynamic PRA, Full-Scope PRA, Level 1, 2, 3 PRA)*	The term probabilistic risk assessment (PRA) has numerous, similar definitions. Some of the formal definitions used are presented below: • "A qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public (also referred to as a probabilistic safety assessment (PSA))." (Ref. 2) • "For a method or approach to be considered a PRA, the method or approach provides (1) a quantitative assessment of the identified risk in terms of scenarios that result in undesired consequence (e.g., core damage or large early release) and their frequencies, and (2) is comprised of specific technical elements in performing the quantification." (Ref. 91)

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	• “A systematic method for assessing three questions used to define “risk.” These questions consider (1) what can go wrong, (2) how likely it is, and (3) what its consequences might be. These questions allow understanding of likely outcomes, sensitivities, areas of importance, system interactions, and areas of uncertainty, which can identify risk-significant scenarios. The PRA determines a numeric estimate of risk to provide insights into the strengths and weaknesses of the design and operation of a nuclear power plant.” (Ref. 36) A specific type of PRA is the base or baseline PRA, which represents the as-built as-operated plant to the extent needed to support the application. For a nuclear power plant at the design certification or combined operating license stage, where the plant is not built or operated, the base(line) PRA model reflects the as-designed plant. This type of PRA is also used as a benchmark to estimate the change in risk from a proposed design change. A dynamic PRA is a special type of PRA that automatically accounts for time-dependent effects by integrating these effects directly into the computer model. In a traditional PRA, time-dependent effects are accounted for manually. A full-scope PRA addresses three specific levels of analysis; namely, Level 1 (core damage), Level 2 (radioactive material release), and Level 3 (consequences). The term probabilistic safety assessment is another term that is sometimes used interchangeably with PRA. Typically, the term probabilistic safety assessment is used outside of the U.S.
PRA Configuration Control (Maintenance, Upgrade)	
A process that maintains and updates the probabilistic risk assessment so that it reflects the as-built as-operated facility. *(see Living PRA, Risk Management)*	In a PRA, updates to the model may be needed to ensure that the PRA reflects the as-built as-operated facility. As described in the ASME/ANS PRA Standard (Ref. 2), a “PRA configuration control program shall include a process to monitor changes in the design, operation, maintenance, and industry-wide operational history that could affect the PRA. These changes shall include inputs that impact operating procedures, design configuration, initiating event frequencies, system or subsystem unavailability, and component failure rates. The program should include monitoring of changes to the PRA technology and industry experience that could change the results of the PRA model.” As further described in the ASME/ANS PRA Standard (Ref. 2), PRA maintenance involves “update of the PRA models to reflect plant changes such as modifications, procedure changes, or plant performance (data).” Additionally, the ASME/ANS PRA Standard (Ref. 2) states that a PRA upgrade involves “the incorporation into a PRA model of a new methodology or changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. This could include items such as new human error analysis methodology, new data update methods, new approaches to quantification or truncation, or new treatment of common cause failure.” PRA configuration control is part of the process used to support a living PRA (i.e., a PRA that is continuously updated to reflect current plant design, configuration, operating procedures, and plant-specific data). Listed below are definitions of related terms: • <u>Configuration risk management</u>: The term configuration risk management is the same as risk management and is defined under “Risk Management.” • <u>Configuration risk profile</u>: A change in the overall nuclear power plant risk metric value as a result of a change from the initial plant configuration. Results from a PRA can be used as the basis for developing configuration risk profiles using various risk metrics (e.g., core damage frequency, large early release frequency). The configuration risk profile can depend on the plant operational status. For example, during certain shutdown operations when the containment function is not maintained,

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	the risk metric represented by large early release fraction is not applicable; therefore, licensees may use more stringent baseline core damage frequency guidelines to maintain an equivalent risk profile.
PRA Maintenance	
(see PRA Configuration Control)	The term PRA maintenance is part of PRA configuration control and is defined under "PRA Configuration Control."
PRA Model	
(see Model)	The term PRA model is a specific type of model and is defined under the term "Model."
PRA Technical Acceptability	
(see Technical Acceptability)	The term PRA technical acceptability is discussed in the discussion for the term "Technical Acceptability."
PRA Technical Adequacy	
(see Technical Adequacy)	The term PRA technical adequacy is discussed in the discussion for the term "Technical Adequacy."
PRA Technical Elements	
The basic pieces (or analyses) required to produce the PRA model. (*see Appendix B*)	The individual analyses used in the development of a PRA model are organized according to a set of PRA technical elements. As described in the ASME/ANS PRA Standard (Ref. 2), a number of specific PRA technical elements are used to support the evaluation of contributors to risk (e.g., the evaluation of hazard groups). Examples of PRA technical elements include the following: initiating events analysis, accident sequence analysis, and high wind hazard analysis.
PRA Upgrade	
(see PRA Configuration Control)	The term PRA upgrade is part of PRA configuration control and is defined under "PRA Configuration Control."
Precursor Event	
(see Accident Precursor)	The term precursor event is the same as accident precursor and is defined under "Accident Precursor."
Probabilistic (Analysis, Approach)	
A characteristic of an evaluation that includes consideration of events with regard to their likelihood. *(see Deterministic, PRA, Risk-Based, Risk-Informed)*	A PRA is an example of a probabilistic analysis, which can be defined as a mathematical evaluation of random (stochastic) events or processes and their consequences. While a PRA uses probabilistic analysis, a PRA also depends on deterministic analyses. For example, success criteria for various systems modeled in a PRA to prevent and mitigate core damage are based on deterministic analyses. A probabilistic approach can be defined as a method that accounts for the likelihood of possible states that a physical entity or system can assume and predictions of models describing the entity or system. Both risk-based and risk-informed approaches to decisionmaking and regulation rely upon probabilistic analysis. A risk-based approach to decisionmaking or regulation means that the decision or regulation is based only on risk information generated from a probabilistic analysis

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	(e.g., from a PRA), whereas a risk-informed approach combines risk information generated from a probabilistic analysis with other factors to arrive at a decision or develop regulations. The NRC Web site Glossary (Ref. 36) states the following: "The term 'probabilistic' is associated with an evaluation that explicitly accounts for the likelihood and consequences of possible accident sequences in an integrated fashion." Therefore, a probabilistic analysis or approach is unlike a deterministic analysis or approach, which does not include consideration of events with regard to their likelihood.
Probabilistic Analysis	
(see Probabilistic)	The term probabilistic analysis is defined under "Probabilistic."
Probabilistic Approach	
(see Probabilistic)	The term probabilistic approach is defined under "Probabilistic."
Probabilistic Safety Assessment	
(see PRA)	The term probabilistic safety assessment is another term for PRA and is defined under "PRA."
Probabilistic Seismic Hazard Analysis	
(see Hazard Analysis)	The term probabilistic seismic hazard analysis is a specific type of hazard analysis and is defined under "Hazard Analysis."
Probability (Basic Event Failure, Containment Failure, Core Damage, Failure, Human Error)	
The likelihood that an event will occur as expressed by the ratio of the number of actual occurrences to the total number of possible occurrences. *(see Frequency)*	In a PRA, probability is calculated for various types of PRA input and output parameters (e.g., failures of equipment associated with basic events, core damage, and containment failure). The probability assigned to a basic event is often referred to as the basic event failure probability. A basic event is an element of the PRA model for which no further decomposition is performed because it is at the limit of resolution consistent with available data. A failure probability is calculated for each failure mode of a component (e.g., failure to start and failure to run for a pump). In addition, a failure probability may be calculated for a system failing to perform its function or a structure failing (e.g., given a seismic event). For example, containment failure probability is the likelihood that the containment structure fails to perform its function of retaining fission products. The ASME/ANS PRA Standard (Ref. 2) defines failure probability as "the likelihood that a system or component will fail to operate upon demand or fail to operate for a specific mission time." Failure probability is also calculated for human actions and is then called human error probability. The ASME/ANS Standard (Ref. 2) defines human error probability as a measure of the likelihood that plant personnel will fail to initiate the correct, required, or specified action or response in a given situation, or by commission performs the wrong action. Some PRA studies also calculate the probability of core damage, also referred to as core damage probability, given a particular initiating event or set of initiating events. There is a tendency in risk communication to use frequency and probability synonymously, but incorrectly. Probability only conveys the likelihood of an event; frequency conveys that likelihood per unit time. The definition provided was based on the definition in NUREG/CR-6823 (Ref. 78).

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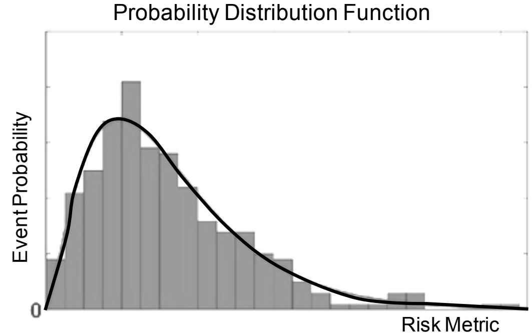
Probability Density Function	
(see Probability Distribution)	The term probability density function is an equivalent term for probability distribution and is defined under "Probability Distribution."
Probability Distribution (Probability Density Function)	
A curve that shows all the values that a random variable can take and the likelihood that each will occur. *(see Cumulative Distribution Function, Mean, Median, Uncertainty Interval)*	In a PRA, probability distributions are used to express uncertainties associated with the state-of-knowledge about the parameter values and models used in constructing the PRA. A probability distribution can represent either a discrete or continuous set of values for a random variable. It is usually represented as a probability density function. The probability density function is a function of a continuous random variable whose integral over an interval gives the probability that its value will fall within the interval. In comparison, the cumulative distribution function adds up the probabilities of occurrence of all possible parameter values in a probability distribution function that are less than a specified value. An illustration of a probability distribution function and its corresponding cumulative distribution function is shown under the discussion for the term "Cumulative Distribution." 
Prompt Fatality	
(see Fatality)	The term prompt fatality is a type of fatality caused by exposure to radioactive materials and is defined under "Fatality."
Prompt Fatality Risk	
(see Fatality)	The term prompt fatality risk is a type of fatality caused by exposure to radioactive materials and is defined under "Fatality."
Public Health Effects	
(see Health Effects)	The term public health effect refers to a type of health effect and is defined in the discussion under "Health Effects."
Qualitative Risk Assessment	
(see Risk)	A qualitative risk assessment is one type of risk assessment and is defined under "Risk."
Qualitative Screening	
(see Screening)	A qualitative screening is one type of screening performed and is defined under "Screening."
Quantitative Health Objectives	
Numerical criteria for the acceptable levels of risk to	In some risk-informed decisions, the results of a PRA are used to compare the risk from the plant with the quantitative health objectives (QHOs) that support the NRC's reactor safety goals.

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public health and safety in the population surrounding a nuclear power plant that satisfy the NRC's reactor safety goals. *(see Fatality, Risk to Average individual)*	The NRC safety goals (Ref. 30) are expressed by two QHOs: (1) the annual average individual probability of prompt fatality in the population within 1 mile of the site boundary of a nuclear power plant should not exceed one-tenth of 1 percent of the risk of prompt fatality due to all other risks (non-nuclear) that the U.S. population is generally exposed to, and (2) the annual average individual probability of latent cancer fatality in the population within 10 miles of the site boundary of a nuclear power plant should not exceed one-tenth of 1 percent of the U.S. cancer fatality rate due to all other (non-nuclear) causes.
Quantitative Screening	
(see Screening)	A quantitative screening is one type of screening and is defined under "Screening."
Radioactive Material	
A substance that emits ionizing radiation. *(see Radionuclide, Fission Product)*	In a PRA, the terms radionuclide, radioactive material, and fission product are used interchangeably. These terms are meant to refer to the substance that is the source of the risk being evaluated. However, a release of this substance (i.e., radioactive material) from the reactor and from the containment, or from another source such as the spent fuel pool, could have an adverse impact on public health and safety is generally not referred to as radioactive material release. Generally, either radionuclide release or fission product release, or just 'release' is used.
Radioactive Material Release (Large Early, Small Early, Large Late, Small Late)	
The release of radioactive material to the environment. *(see Radioactive Material, Radioactive Material Release Frequency Analysis, Health Effects)*	In a Level 2 PRA, the release of radioactive material from the reactor core to the environment is calculated. Usually this is referred to as the 'release,' 'radionuclide release,' or 'fission product release.' This release may occur early or late and may be large or small. In the ASME/ANS PRA Standard (Ref. 2), a large early release is defined as a rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions so there is a potential for early health effects. A small early release is of low enough magnitude to have minimal potential for early health effects. A large late release can be defined as a release of airborne fission products from the containment to the environment of sufficient magnitude to cause severe health effects. However, the release occurs in a timeframe that allows the effective implementation of offsite emergency response and protective actions such that the offsite health effects can be significantly reduced compared to those of a large early release. A small late release is of low enough magnitude and is delayed long enough to have minimal potential for health effects. For both early and late large releases, significant land contamination and property damage is to be expected. The term large release is discussed as its own entry in this glossary. The Commission has not approved a formal definition for the term large release.
Radioactive Material Release Frequency (Large Early, Small Early, Large Late, Small Late)	
(see Frequency)	The term radioactive material release frequency (large early, small early, large late, small late) is a type of frequency used in PRA and is defined in the discussion under "Frequency."

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Radioactive Material Release Frequency Analysis (Large Early, Small Early, Large Late, Small Late)	
An estimation of the frequency of radioactive material releases by evaluating the core and containment behavior under severe accident conditions. *(see Radioactive Material Release, Health Effects)*	In a Level 2 PRA, the frequency of release of radioactive material from the reactor core to the environment is calculated. This release may occur early or late and may be large or small. For operating reactors, a large early release frequency is one of the risk metrics used for risk-informed decisions. For new reactors, a large release frequency is one of the risk metrics used for risk-informed decisions.
Radiological Source Term	
(see Source Term)	The term radiological source term has the same meaning as source term and is defined under "Source Term."
Radiological Source Term Analysis	
(see Source Term Analysis)	The term radiological source term analysis has the same meaning as source term analysis and is defined under "Source Term Analysis."
Radionuclide	
An atom with an unstable nucleus that emits radiation *(see Radioactive Material, Fission Product)*	In a PRA, the terms radionuclide, radioactive material, and fission product are used interchangeably. These terms are meant to refer to the substance that is the source of the risk being evaluated. A radionuclide release, therefore, refers to the release of the substance (i.e., radionuclides) from the reactor and from the containment that could have an adverse impact on public health and safety. The NRC Web site Glossary (Ref. 38) defines radionuclide as "an unstable isotope of an element that decays or disintegrates spontaneously, thereby emitting radiation. Approximately 5,000 natural and artificial radioisotopes have been identified."
Random Failure	
A failure not anticipated to occur at a certain time (i.e., occurring with no specific pattern).	In a PRA, potential failures of the modeled structures, systems, and components (SSCs) are treated as random events. This treatment is necessary because it is not possible to predict when an SSC will possibly fail. Instead, it is only possible to predict the likelihood that an SSC will fail. The likelihood that an SSC will fail is based on failure rate data, which represents the expected number of failures of the SSC per unit time. Failure rate data are developed for each SSC modeled in a PRA.
Random Uncertainty	
(see Uncertainty)	The term random uncertainty is related to aleatory uncertainty and defined under "Uncertainty."
Rare Initiator	
An initiating event that is extremely unlikely and not expected to occur	In a PRA, rare initiators generally are screened because of their low frequencies. Examples of rare initiators include aircraft impact, meteor strikes, and very large earthquakes. These occurrences are also correctly referred to as rare events.

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in nuclear power plants. *(see Initiating Event)*	The ASME/ANS PRA Standard (Ref. 2) defines the term rare event as “one that might be expected to occur only a few times throughout the world nuclear industry over many years (e.g., <1E-4/r-yr).” However, the ASME/ANS PRA Standard only allows screening of initiating events if the frequency is much lower than 1E-4/yr (e.g., if the frequency <1E-7/yr and the event does not involve either an ISLOCA, containment bypass, or reactor, or reactor pressure vessel rupture).
Rationalist	
An approach to defense-in-depth that uses probabilistic information to evaluate the uncertainties and to determine what steps should be taken to compensate for those uncertainties. *(see Structuralist, Defense-in- Depth)*	When used in a PRA context, the term rationalist is a relatively new term associated with defense-in-depth. The rationalist approach is made practical by the ability to quantify risk and estimate uncertainties using PRA techniques. In this approach, results from a PRA or other probabilistic analysis are used to assess the strengths and weaknesses of defense-in-depth, while accounting for analysis uncertainties. Ultimately, the rationalist approach provides a way to increase the degree of confidence in the conclusion that the defense-in-depth is sufficiently robust to achieve adequate safety. In contrast, the fundamental principle of the structuralist approach is that if a system is designed to withstand all the worst-case credible accidents, then it is by definition protected against any credible accident. It is a deterministic method of establishing how precautions can be placed into a system, just in case an existing barrier or system fails. The Advisory Committee on Reactor Safeguards (Ref. 39) describes that the rationalist will “(1) establish quantitative acceptance criteria, such as the quantitative health objectives, core damage frequency and large early release frequency, (2) analyze the system using PRA methods to establish that the acceptance criteria are met, and (3) evaluate the uncertainties in the analysis, especially those due to model incompleteness, and determine what steps should be taken to compensate for those uncertainties.”
Reactor Core	
The location within a nuclear reactor where the fission process occurs.	In a PRA, the source of risk generally evaluated is the reactor core with an understanding that the actual risk is from the fuel. The reactor core includes the fuel assemblies, moderator, neutron poisons, control rods, and support structures. The other sources of risk at the plant site (e.g., spent fuel) generally are not included in the reactor core PRA; however, there are several PRAs, separate from the reactor core PRAs, which evaluate the risk of the spent fuel. The NRC Web site Glossary (Ref. 36) defines reactor core as “the central portion of a nuclear reactor, which contains the fuel assemblies, moderator, neutron poisons, control rods, and support structures. The reactor core is where fission takes place.”
Reactor-Operating-State-Year	
(see Reactor-Year)	The term reactor-operating-state-year is related to the term reactor-year and is defined under “Reactor-Year.”
Reactor-Year (Reactor-Operating-State-Year)	
A unit of time by which risk parameters are measured in a PRA. *(see Plant Operational State)*	In a PRA, the terms reactor-year and reactor-operating-state-year refer to units of time by which risk parameters (e.g., core damage frequency, large early release frequency) are measured. The ASME/ANS PRA Standard (Ref. 2) defines the term reactor-year as “a calendar year in the operating life of one reactor, regardless of power level.” The term reactor-year assumes that more than one reactor can operate during a year (e.g., a calendar year during which five reactors operated would be the experience equivalent of 5 reactor-years). For some applications, such as configuration risk management or analyses that compare specific risks during different modes of operation, it may be appropriate to develop risk metrics that consider the time period associated with a given plant operational state. For at-power

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	operation, this basis is sometimes referred to as per reactor critical year (i.e., assuming that the reactor operated continuously for a year). On a more general basis, it could be considered to be per reactor-operating-state-year. The ASME/ANS PRA Standard (Ref. 2) defines the term reactor-operating state-year as "an equivalent calendar year of operation in a particular plant operating state."
Realistic Analysis	
(see Conservative Analysis)	The term realistic analysis is discussed in the discussion for "Conservative Analysis" and is defined there.
Recovery	
Restoration of a failed function. *(see Repair)*	In a PRA, the term recovery usually refers to an action or series of actions performed by an operator or other plant personnel to restore a function in response to a failed component or system. This term is sometimes used incorrectly as a synonym for repair. However, repair is restoring a failed function by fixing the actual cause of the failure while recovery is restoring the function in some other way. The ASME/ANS PRA Standard (Ref. 2) defines the term recovery as "restoration of a function lost as a result of a failed structure, system or component (SSC) by overcoming or compensating for its failure. Generally modeled by using human reliability analysis (HRA) techniques."
Release	
(see Radioactive Material Release)	For purposes of a Level 2 and Level 3 PRA, the term release is used interchangeably with "Radioactive Material Release."
Release Category	
A group of radioactive material releases expected to result in similar consequences. *(see Source Term)*	In a Level 2 PRA, a release category is a grouping of accident sequences into an accident sequence class or family based on a common potential for release of radioactive material. The release categories are characterized by a bounding mechanistic source term. This grouping is based on the following common attributes: common initiating events, combination of successful and failed safety functions, release magnitude, release timing and location, and radioactive material species released from the plant as a result of an accident.
Release Fraction	
The amount of radioactive material released from the reactor core expressed as a fraction of the original inventory of the radioactive material. *(see Source Term)*	In a Level 2 PRA, the release fraction specifies the amount of radioactive materials released to the environment and provides the basis for the subsequent dose calculations to the affected population. NUREG-1489 (Ref. 54) states that the release fraction is expressed as the amount of radioactive material released from the containment as a function of time given as a fraction of the fission product inventory in the core at the time of the start of the accident.
Release Timing and Duration	
The time of release and the timeframe over	In a Level 3 PRA, the time of release and its duration are used to calculate the health consequences to the affected population. Both the timing and duration of the release also form the basis for potential offsite protective action strategies.

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which the radioactive materials are released to the environment during an accident. *(see Source Term)*	
Reliability (Unreliability)	
The likelihood that a system, structure, or component performs its required function(s) for a specified period of time. *(see Availability)*	In a PRA, the unreliability of systems, structures and components, as well as human actions, are used as input to the PRA model, as opposed to the reliability. Unreliability is the complement of reliability and is the likelihood that a structure, system, and component (SSC) does not operate for its mission time when required. The term reliability is often inappropriately used interchangeably with the term availability. Availability only represents the degree to which a SSC is operational and accessible when required for use, with no reference to a mission time. Availability is the likelihood that the SSC is in a state to perform its required function at a given moment in time. In the ASME/ANS PRA Standard (Ref. 2), unreliability is defined as “the probability that a system or component will not perform its specified function under given conditions upon demand or for a prescribed time.”
Repair	
The restoration of a failed function by correcting the cause of failure. *(see Recovery)*	In a PRA, the term repair usually refers to an action or series of actions performed by an operator or other plant personnel to restore the function of a failed structure, system, or component (SSC) by correcting the cause of failure and returning the failed SSC to service so that it can perform its intended function(s). This term is sometimes used incorrectly as a synonym for the term recovery. However, repair is restoring a failed function by fixing the actual cause of the failure while recovery is restoring the function in some other way. The ASME/ANS PRA Standard (Ref. 2) defines the term repair as “restoration of a failed SSC by correcting the cause of failure and returning the failed SSC to its modeled functionality. Generally modeled by using actuarial data.”
Response Time	
The period of time something takes to react to a given input.	In a PRA, the term response time has different connotations, depending on the situation. Some of these connotations are as follows: • When referring to plant components, response time is “the period of time necessary for a component to achieve a specified output state from the time that it receives a signal requiring it to assume that output state.” (Ref. 7) • When referring to human reliability analysis, response time is the time required for “the actions carried out after the operator has received and processed information related to his tasks. These responses constitute the human outputs in a man-machine system and serve as inputs to the man-machine interfaces.” (Ref. 68) • When referring to a Level 3 PRA emergency response, response time is the time required for offsite responders to arrive at a plant site during an emergency (as related to accident response and accident preparedness).

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Risk (Assessment, Analysis)	
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. *(see PRA, Level 1, 2, 3 PRA, Risk Metric)*	Risk assessment or risk analysis and PRA are often incorrectly used as synonyms. A PRA is one type of risk assessment or risk analysis. The PRA has a structured format and quantifies the ultimate consequences. A risk assessment or risk analysis does not necessarily reflect all the technical elements. For example, a seismic margin risk analysis is not a PRA. A qualitative risk assessment or analysis is a risk evaluation that uses descriptions or distinctions based on some characteristic rather than on some quantity or measured value. In comparison to a risk assessment or analysis, a PRA generates different ways to measure risk, called risk metrics, which satisfy specified safety objectives or goals. The consequences are manifested in the onset of core damage and each level of the PRA uses different risk metrics, which can be found in the discussion of Level 1, 2, 3 PRA. The ASME/ANS PRA Standard (Ref. 2) defines the term risk as the "probability and consequences of an event, as expressed by the "risk triplet" that is the answer to the following three questions: *(a)* What can go wrong? *(b)* How likely is it? *(c)* What are the consequences if it occurs?" The definition provided was based on the definition in the NRC Web site Glossary (Ref. 36).
Risk Achievement Worth	
(see Importance Measure)	The term risk achievement worth is one type of importance measure and is defined under "Importance Measure."
Risk Characterization	
(see Risk Metric)	The term risk characterization is a process that uses risk metrics to determine risk and is defined under "Risk Metric."
Risk Insights	
The understanding about a facility's response to postulated accidents. *(see Risk, Risk-Based, Risk-Informed)*	One of the main objectives of a PRA is to gain insights about a facility's response to initiating events and accident progression, including the expected interactions among facility structures, systems, and components (SSCs), and between the facility and its operating staff. Risk insights are derived by investigating in a systematic manner: (1) what can go wrong, (2) how likely it is, and (3) what the consequences are. A risk assessment is a systematic method for addressing these questions as they relate to understanding issues like: important hazards and initiators, important accident sequences and their associated SSC failures and human errors, system interactions, vulnerable plant areas, likely outcomes, sensitivities, and areas of uncertainty. Risk insights can be obtained via both quantitative and qualitative investigations. As noted in RG 1.174 (Ref.84), quantitative risk results from PRA calculations are typically the most useful and complete characterization of risk, but they are generally supplemented by qualitative risk insights and traditional engineering analysis. Qualitative risk insights include generic results, i.e., results that have been learned from numerous PRAs that have been performed in the past, and from operational experience, and that are applicable to a group of similar plants. Risk insights are an important part of risk-informed regulation, in which regulatory decisions are made by integrating risk insights with considerations of defense-in-depth and safety margins.
Risk Management	
A process used at a nuclear power plant to keep the risk at acceptable levels.	A PRA is a tool used to evaluate a nuclear plant from a risk management perspective. The PRA quantifies the plant risk and also quantifies changes in plant risk because of modifications of the plant design or operation. Examples of risk management activities that are supported by PRA are listed below:

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	• A PRA represents an important risk management tool that, as stated in Regulatory Guide (RG) 1.177 (Ref. 86), "ensures that other potentially lower probability, but nonetheless risk-significant, configurations resulting from plant maintenance and other operational activities are identified and compensated for." • Regarding the use of PRA findings and risk insights to support licensee requests for changes to a plant's licensing basis, RG 1.174 (Ref. 84) states the following: "All safety impacts of the proposed change are evaluated in an integrated manner as part of an overall risk management approach in which the licensee is using risk analysis to improve operational and engineering decisions broadly by identifying and taking advantage of opportunities to reduce risk and not just to eliminate requirements the licensee sees as undesirable. For those cases in which risk increases are proposed, the benefits should be described and should be commensurate with the proposed risk increases. The approach used to identify changes in requirements should be used to identify areas in which requirements should be increased as well as those in which they can be reduced." • In reference to the Maintenance Rule, 10 CFR 50.65 (Ref. 20) states, "the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities. The scope of the assessment may be limited to structures, systems, and components that a risk-informed evaluation process has shown to be significant to public health and safety." Risk Management is used in a broader context in NUREG-2150 (Ref.67), "A Proposed Risk Management Regulatory Framework," to refer to an approach for achieving a more comprehensive, holistic, risk-informed, performance-based regulation for reactors, materials, waste, fuel cycle, and transportation that would continue to ensure the safe and secure use of nuclear material. The objective of such an approach is described NUREG-2150 as managing the risks from the use of byproduct, source and special nuclear materials through appropriate performance based regulatory controls and oversight.
Risk Metric	
A measure that is used to express the risk quantity of interest. *(see Risk, Level 1, 2, 3 PRA, Risk Profile, Full-Scope PRA)*	In a PRA, several risk metrics are evaluated. Examples of risk metrics are core damage frequency, developed as part of a Level 1 PRA and large early release frequency, developed as part of a Level 2 PRA. Health effects developed in a Level 3 PRA also can be used as a risk metric. In this instance, limiting to a threshold value the annual average individual probability of death due to acute radiation syndrome within 1 mile of the site boundary would be an example of a risk metric. A full-scope PRA develops risk metrics associated with Levels 1, 2, and 3. Risk metrics are used among other things, to illustrate compliance with safety goals. Risk metrics focus attention on those areas where risk is most likely (such as events that cause core damage) and how the risk metric value for that area is achieving the desired safety objective. Risk metrics can be used in performing risk characterization. Risk characterization combines the major components of risk (hazards, consequences, frequency, and probability), along with quantitative estimates of risk, to give a combined and integrated risk perspective (i.e., a risk profile). Additionally, it shows the key assumptions and rationale, expert elicitation, uncertainties associated with the analysis, and sensitivity analysis.
Risk Monitor	
A plant-specific analysis tool used to determine the risk in real-time based on the current plant configuration. *(see Living PRA)*	The model the risk monitor uses is based on, and is consistent with, the living PRA for the facility. At any given time, the risk monitor reflects the current plant configuration in terms of the known status of the various systems or components (e.g., if any components are out of service for maintenance or tests). The risk monitor assists plant personnel in making decisions about plant configuration changes.

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Risk Profile (Plant)	
The major results generated by a PRA that characterize plant risk.	A plant risk profile presents a concise synopsis of the major PRA results. This synopsis may consist of numerous characterizations of risk, including: • Core damage frequency and large early release frequency for internally and externally initiated events during various modes of operation. • Percentage contributions to core damage frequency and large early release frequency by initiating event and accident sequence type. • Ranking of the contribution of individual basic events and cutsets to core damage frequency and large early release frequency, based on various importance measures. • Comparison of PRA results to PRAs for other plants. • Qualitative risk insights on plant design features.
Risk Reduction Worth	
(see Importance Measure)	The term risk reduction worth is one type of importance measure and is defined under "Importance Measure."
Risk Significant	
A level of risk associated with a facility's system, structure, component, human action or modeled accident sequence that exceeds a predetermined level. *(see Safety Significant, Significant)*	A principal focus of a PRA is to determine the risk significance of the various 'features,' i.e., the systems, structures, and components (SSCs), human actions or the accident sequences involving those SSCs, of the facility being analyzed. Usually, an item is considered risk significant when the risk associated with it exceeds a predetermined limit for contributing to the risk associated with the facility. Since the overall risk of a nuclear facility can be calculated in terms of core damage frequency (CDF) (Level 1 PRA), or releases (Level 2 PRA), or health effects (Level 3 PRA), risk significance can also be determined as related to these various risk measures. Note that risk significant does not have the same meaning as safety significant (defined elsewhere in this glossary) and safety significance is not evaluated in a PRA. The term also describes a level of risk exceeding a predetermined 'significance' level. (Ref. 36)
Risk Significant Equipment	
(see Significant)	The term risk significant equipment is related to the term significant and is defined under "Significant."
Risk to Average Individual	
A measure of the risk to an individual that represents an average over the parameters characterizing the population at risk *(see Fatality, Quantitative Health Objectives)*	In a Level 3 PRA, the risk to an average individual is calculated as the total fatalities in the surrounding population as a result of an accident divided by the total population. For example, the risk of prompt fatality to an average individual within 1 mile of the plant boundary can be calculated as the number of prompt fatalities per year to the total population within 1 mile of the plant boundary because of each accident sequence, summed over all accident sequences weighted by their frequency of concurrence, divided by the population within 1 mile. The average individual in the vicinity of the plant is defined as the average individual biologically (in terms of age and other risk factors) and who resides within 1 mile of the plant site boundary.

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Risk-Based Approach	
(see Risk-Based)	The term risk-based approach is related to the term risk-based and is defined under "Risk-Based."
Risk-Based (Approach, Decisionmaking, Regulation)	
A characteristic of decisionmaking in which a decision is solely based on the results of a risk assessment. *(see Risk-Informed)*	The modifying term "risk-based" is applied to decisionmaking and regulation activities that rely solely on the use of risk information from PRA results. The terms risk-based approach, risk-based decisionmaking, and risk-based regulation are often used interchangeably and somewhat correctly to describe the same concept; therefore, these terms are grouped under the same definition. However, as indicated below, each of these terms has its own distinct meaning: • <u>Risk-Based Approach</u>: A philosophy on decisionmaking "in which a safety decision is solely based on the numerical results of a risk assessment." (Ref. 96) • <u>Risk-Based Decisionmaking</u>: "An approach to regulatory decisionmaking that considers only the results of a probabilistic risk assessment." (Ref. 36) • <u>Risk-Based Regulation</u>: An approach to regulation that uses the results of a risk assessment to develop applicable regulations. Risk-informed is a term that is often used incorrectly in place of risk-based. These terms are not synonyms. Unlike a risk-based approach, a risk-informed approach to decisionmaking or regulation combines risk information with other factors (e.g., engineering design features) to arrive at a decision or develop regulations. Since risk-based approaches, decisionmaking, and regulation put a greater emphasis on risk assessment results than is currently practical because of uncertainties in PRA, such as completeness, the Commission does not endorse a solely "risk-based" approach.
Risk-Based Decisionmaking	
(see Risk-Based)	The term risk-based decisionmaking is related to the term risk-based and is defined under "Risk-Based."
Risk-Based Regulation	
(see Risk-Based)	The term risk-based regulation is related to the term risk-based and is defined under "Risk-Based."
Risk-Informed (Approach, Decisionmaking, Regulation)	
A characteristic of decisionmaking in which risk results or insights are used together with other factors to support a decision. *(see Risk-Based, Deterministic, Probabilistic)*	The modifying term "risk-informed" is applied to decisionmaking and regulation activities that combine risk information (e.g., PRA results) with other factors (e.g., engineering design features) to arrive at a decision. The terms risk-informed approach, risk-informed decisionmaking, and risk-informed regulation are often used interchangeably and somewhat correctly to describe the same concept; therefore, these terms are grouped under the same definition. However, as indicated below, each of these terms has its own distinct meaning: • <u>Risk-Informed Approach</u>: "A 'risk-informed' approach to regulatory decision-making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety. A 'risk-informed' approach enhances the traditional approach by: (a) allowing explicit consideration of a broader set of potential challenges to safety, (b) providing a logical means for prioritizing these challenges based on risk significance, operating experience, and/or engineering judgment, (c) facilitating consideration of a broader

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	set of resources to defend against these challenges, (d) explicitly identifying and quantifying sources of uncertainty in the analysis, and (e) leading to better decision-making by providing a means to test the sensitivity of the results to key assumptions. Where appropriate, a risk-informed regulatory approach can also be used to reduce unnecessary conservatism in deterministic approaches, or can be used to identify areas with insufficient conservatism and provide the bases for additional requirements or regulatory actions." (Ref. 96) • Risk-Informed Decisionmaking: "An approach to regulatory decision making, in which insights from probabilistic risk assessment are considered with other engineering insights." (Ref. 36) • Risk-Informed Regulation: "An approach to regulation taken by the NRC, which incorporates an assessment of safety significance or relative risk. This approach ensures that the regulatory burden imposed by an individual regulation or process is appropriate to its importance in protecting the health and safety of the public and the environment." (Ref. 36) A term often used incorrectly in place of risk-informed is risk-based; these terms are not synonyms. A risk-based approach to decisionmaking or regulation means that the decision or regulation is based only on risk information (e.g., risk results obtained from a PRA), whereas a risk-informed approach combines risk information with other factors to arrive at a decision or develop regulations.
Risk-Informed Approach	
(see Risk-Informed)	The term risk-informed approach is related to the term risk-informed and is defined under "Risk-Informed."
Risk-Informed Decisionmaking	
(see Risk-Informed)	The term risk-informed decisionmaking is related to the term risk-informed and is defined under "Risk-Informed."
Risk-Informed Regulation	
(see Risk-Informed)	The term risk-informed regulation is related to the term risk-informed and is defined under "Risk-Informed."
Safe-Shutdown Earthquake	
The maximum earthquake for which certain structures, systems, and components are designed to remain functional to shut down the reactor. *(see Seismic Margin Analysis)*	In a seismic PRA, the plant's response to earthquakes of all magnitudes appropriate for the site are evaluated. In a seismic margin analysis, the capability of the plant to withstand an earthquake larger that the safe-shutdown earthquake (SSE) is often assessed. The ASME/ANS PRA Standard (Ref. 2) defines the SSE as "that earthquake for which certain structures, systems and components (SSCs) are designed to remain functional. In the past, the SSE has been commonly characterized by a standardized spectral shape anchored to a peak ground acceleration value." Appendix S to 10 CFR 50 (Ref.25) states that the "safe-shutdown earthquake ground motion (SSE) is the vibratory ground motion for which certain structures, systems, and components must be designed to remain functional." The SSCs required to withstand the effects of the safe-shutdown earthquake ground motion are those necessary to ensure: (1) The integrity of the reactor coolant pressure boundary; (2) The capability to shut down the reactor and maintain it in a safe-shutdown condition; or

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	(3) The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1) (Ref. 14). The definition provided is based on the definition in the NRC Web site Glossary (Ref. 36).
Safe Stable State	
Condition of the reactor in which the necessary safety functions are achieved.	In a PRA, safe stable states are represented by success paths in modeling of accident sequences. A safe stable state implies that the plant conditions are controllable within the success criteria for maintenance of safety functions. The ASME/ANS PRA Standard (Ref. 2) defines the term safe stable state as “a plant condition, following an initiating event, in which reactor coolant system conditions are controllable at or near desired values.”
Safety Function	
Those functions needed to shut down the reactor, remove the residual heat, and contain any radioactive material release.	A PRA involves the analysis of the performance of the plant safety functions in response to accidents. The common general safety functions for a nuclear power plant as stated in the IAEA Safety Glossary (Ref. 7) are: • The capability to safely shut down the reactor and maintain it in a safe shutdown condition during and after appropriate operational states and accident conditions. • The capability to remove residual heat from the reactor core after shutdown, and during and after appropriate operational states and accident conditions. • The capability to reduce the potential for the release of radioactive material and to ensure that any releases are within prescribed limits during and after operational states and within acceptable limits during and after design-basis accidents. The ASME/ANS PRA Standard (Ref. 2) defines safety function as “function that must be performed to control the sources of energy in the plant and radiation hazards.”
Safety Margin	
The extra capacity factored into the design of a structure, system, or component so that it can cope with conditions beyond the expected to compensate for uncertainty. *(see Defense-in-Depth, Uncertainty)*	In a PRA, the extra capacity of systems, structures, and components (SSC) provided by the safety margin is used in calculating the plant response to an accident. A safety margin is used to provide capacity for emergency situations, unexpected loads, misuse, or attrition. Many engineering codes and standards provide quantitative guidance on appropriate safety margin for a particular design application. However, the term safety margin also is often found in regulatory documents that contain phrases such as “maintain adequate safety margin,” or “provide sufficient safety margin,” without specification of a particular quantitative margin. Safety margins can be considered a part of, or complementary to, defense-in-depth in that they provide extra (redundant) capacity. Incorporation of safety margins is one of the ways designers deal with the uncertainty of the challenges that the designed SSCs face. The figure below illustrates several concepts on safety margins. A regulator may impose the requirement that a margin is maintained between a component’s allowable limit of operation, the regulatory limit, and the component’s ultimate capacity. The component designer may want to design or select the component so that during normal operation it operates below, rather than right at, the regulatory limit (i.e., he or she may want to add an additional margin). The total safety margin then encompasses both the designer and regulatory margins.

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	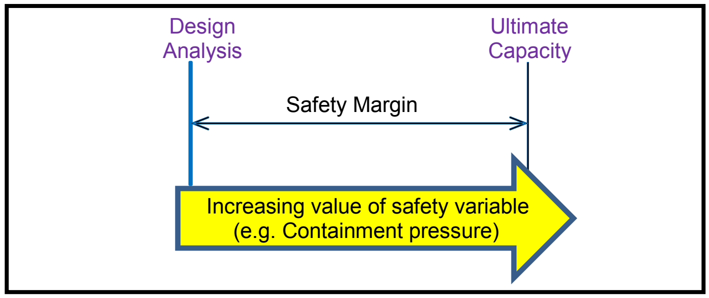
Safety-Related	
(see Safety Significant)	The term safety-related indicates the safety significance of a structure, system, or component and is defined under "Safety Significant."
Safety Significant (Important to Safety, Safety-Related, Nonsafety-Related)	
A qualifying term that indicates if something does not meet some predetermined criterion, it has the potential to affect safety.	In a PRA, the risk significance of nuclear power plant structures, systems, and components (SSCs) are determined, not the safety significance. This risk significance is then used in a risk-informed regulatory framework to determine the safety significance of SSCs. The term safety significant is generally used to categorize nuclear power plant SSCs using the process outlined in 10 CFR 50.69 (Ref. 21). In this application, a plant-specific PRA is used to delineate and quantify severe accident scenarios resulting from internal initiating events at full-power operation. In 10 CFR 50.36, Technical Specifications, (Ref. 15) Criterion 4 requires that "a structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety" must have a technical specification limiting condition for operation established for it. The term important to safety refers to both safety related and non-safety related SSCs that have been deemed important. In Regulatory Guide 1.201 (Ref. 92), the NRC has stated that it does not endorse the Nuclear Energy Institute (NEI) usage of important to safety as having the same connotation as safety significant. Another term, safety related, has a specific meaning in the regulatory arena. Part 50 of the Code of Federal Regulations (Ref. 13), as well as the NRC Web site Glossary (Ref. 36) state that the term "safety-related applies to systems, structures, components, procedures, and controls (of a facility or process) that are relied upon to remain functional during and following design basis events. Their functionality ensures that key regulatory criteria, such as levels of radioactivity released, are met. Examples of safety related functions include shutting down the nuclear reactor and maintaining it in a safe-shutdown condition." Conversely, nonsafety-related indicates that the SSCs, procedures, and controls are *not* relied upon to remain functional during a design-basis event. The NRC Web site Glossary (Ref. 36) makes the following statement about the term safety significant: "When used to qualify an object, such as a system, structure, component, or accident sequence, this term identifies that object as having an impact on safety, whether determined through risk analysis or other means, which exceeds a predetermined significance criterion." Safety significance is not evaluated in a PRA.
Screening (Analysis, Criteria, Qualitative, Quantitative)	
A process that distinguishes items that should be included or excluded from an analysis based on defined criteria.	In a PRA, screening may be applied in a variety of ways (e.g., screening out (eliminating) component failure events from the PRA based on a low probability or frequency). Another form of screening is to identify the more significant events that should be analyzed in a detailed manner. Insignificant events may be addressed using less detailed and usually conservative methods. Screening is an integral step in most PRAs to reduce the complexity of the PRA model using sound judgment. The terms screening and screening analysis are similar in meaning and often used interchangeably.

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	The definitions of the grouped terms are presented below as they apply to screening: • Screening criteria: "The values and conditions used to determine whether an item is a negligible contributor to the probability of an accident sequence or its consequences." (Ref. 2) • Qualitative screening: The objective is to identify portions of the analysis whose potential risk contribution can be judged negligible without quantitative analysis. • Quantitative screening: The objective is to eliminate portions of the analysis from further consideration based on preliminary estimates of risk contribution through the use of established quantitative screening criteria. The ASME/ANS PRA Standard (Ref. 2) defines screening as "a process that eliminates items from further consideration based on their negligible contribution to the probability of an accident or its consequences."
Screening Analysis	
(see Screening)	The term screening analysis is similar in meaning to screening and is discussed under "Screening."
Screening Criteria	
(see Screening)	The term screening criteria is defined under "Screening."
Seismic Fragility Analysis	
(see Fragility Analysis)	Seismic fragility analysis is a type of fragility analysis and is included in the discussion under "Fragility Analysis."
Seismic Hazard Analysis	
(see Hazard Analysis)	The term seismic hazard analysis is a type of hazard analysis and is defined under "Hazard Analysis."
Seismic Margin	
A measure of the capacity of the plant to withstand an earthquake more severe than the design-basis earthquake. *(see High Confidence of Low Probability of Failure, Safe Shutdown Earthquake, Seismic Margin Analysis)*	For some applications, seismic margin, rather than a PRA risk metric, has been used to estimate the ability of a plant to safely withstand seismic events. The ASME/ANS PRA Standard (Ref. 2) states that "seismic margin is expressed in terms of the earthquake motion level that compromises plant safety, specifically leading to severe core damage. The margin concept also can be extended to any particular structure, function, system, equipment item, or component for which 'compromising safety' means sufficient loss of safety function to contribute to core damage either independently or in combination with other failures." NUREG-1742 (Ref. 59) defines seismic margin as "the ability of a plant, system, component or structure to safely withstand seismic demands or input ground-motion levels beyond those imposed by the design basis earthquake."
Seismic Margin Analysis	
The process to estimate the seismic margin of the plant and to	For some applications, seismic margin analysis is an alternative to a seismic PRA for identifying seismic vulnerabilities at a plant. The earthquake specified for assessing the seismic margin can depend on a number of factors, usually the plant's location. In the individual plant examination for external events (IPEEE), plants were assessed against a

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identify any seismic vulnerabilities in the plant. *(see High Confidence of Low Probability of Failure, Seismic Margin, Safe-Shutdown Earthquake)*	review-level earthquake whose intensity was higher than the design-basis earthquake and varied according to the plant location. Seismic margin analysis is performed to show high confidence of low probability of failure (HCLPF) at a certain earthquake level (peak ground acceleration) above the design-basis (safe-shutdown) earthquake. A number of methods can be used to calculate seismic margin: • In the IPEEEs, most licensees that carried out a seismic margin analysis used a method developed by the Electric Power Research Institute (EPRI). In the EPRI method, two success paths, addressing transients, are developed based on a group of safety functions capable of bringing the plant to a safe-shutdown condition after an earthquake. Each success path has to rely on different equipment and each path assumes a loss of offsite power. One path also has to be capable of mitigating a small loss-of-coolant accident (LOCA). HCLPFs are developed for the two success paths. The NRC also developed a seismic margin method for the IPEEEs, used by a few licensees. In the NRC IPEEE method, accident sequence models are developed for transients and small LOCAs and HCLPF values are evaluated for the accident sequences developed from these two initiators. Neither the EPRI nor the NRC method requires fragility curves to be developed and allow HCLPFs to be based on the conservative deterministic failure margin method. • More recently, the NRC has endorsed a seismic margin method in which fragility curves are developed. In this PRA-based method, accident sequence models are developed for all the initiators and HCLPF values are evaluated for the accident sequences developed from all the initiators. The definition provided is based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Seismic Plant Response Analysis/Model	
(see Plant Response Analysis/Model)	The term seismic plant response analysis is a type of plant response analysis and is included in the discussion under "Plant Response Analysis/Model."
Sensitivity Analysis	
An analysis in which one or more input parameters to a model are varied in order to observe their effects on the model results.	In a PRA, sensitivity analyses often are performed to help assess the results. Sensitivity analyses often involve variations of quantitative parameters (e.g., component failure probabilities, initiating event frequencies, human error rates). The definition provided was based on the definition in NUREG-1560 (Ref. 56).
Severe Accident (Sequence, Progression Sequence)	
A type of accident that involves core damage. *(see Accident Sequence, Beyond-Design-Basis Accident, Design-Basis Accident)*	In a PRA, beyond-design-basis accidents (BDBAs) are analyzed to determine which ones could lead to core damage. The BDBAs that have an end state resulting in core damage are termed severe accidents. All severe accidents are by definition beyond-design-basis accidents since their challenges exceed the design envelope of the plant. However, not all beyond-design-basis accidents are severe accidents, since the design envelope can be exceeded without core damage occurring. The ASME/ANS PRA Standard (Ref. 2) defines a severe accident as "an accident that involves extensive core damage and fission product release into the reactor vessel and containment, with potential release to the environment."

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	In a Level 1 PRA, severe accident sequences are a subset of the accident sequences (i.e., many of the accident sequences in a Level 1 PRA do not result in core damage). In a Level 2 PRA, severe accident sequences are the only sequences considered because they involve core damage. The term severe accident progression sequence usually is used correctly as a synonym for the term severe accident sequence.
Severe Accident Progression Sequence	
(see Severe Accident)	Severe accident progression sequence has the same meaning as severe accident sequence and is defined under "Severe Accident."
Severe Accident Sequence	
(see Severe Accident)	A severe accident sequence is an accident sequence that results in a severe accident and is defined under "Severe Accident."
Shutdown	
(see Low-Power and Shutdown)	The term shutdown is part of low power and shutdown operation and is defined under "Low-Power and Shutdown."
Significant (Accident Sequence, Accident Progression Sequence, Basic Event, Containment Challenge, Contributor, Cutset, Equipment)	
A factor that can have a major or notable influence on the results of a risk analysis.	In a PRA, the modifying term significant is applied to factors that have an important influence on causing a measurement of risk to exceed a predetermined level or limit. The terms significant and risk significant have the same meaning in a PRA context and are often used interchangeably, which is correct and appropriate in this context. As discussed in NRC Regulatory Guide 1.200 (Ref. 91), the determination of significance is a function of how the PRA is being, or is intended to be, used. When a PRA is being used to support an application, the significance of an accident sequence or contributor is measured with respect to whether its consideration has an effect on the decision being made. Quantitative thresholds (criteria) often are used to determine if a basic event, cutset, accident sequence, or accident progression sequence is considered significant from a risk perspective (e.g., based on importance measures, percentage contribution). The previously mentioned items (e.g., basic event, cutset) represent the different types of significant risk contributors that could influence the results of a risk analysis. These quantitative criteria may vary, depending on the source of the guidance. The following terms (excluding risk significant) and the subsequent definitions are based on the ASME/ANS PRA Standard (Ref. 2): • <u>Significant Accident Sequence</u>: "One of the sets of accident sequences resulting from the analysis of a specific hazard group, defined at the functional or systematic level, which, when rank-ordered by decreasing frequency, sum to a specified percentage of the core damage frequency for that hazard group, or that individually contribute more than a specified percentage of core damage frequency. For this version of the Standard [RA-Sa-2009], the summed percentage is 95% and the individual percentage is 1% of the applicable hazard." (Ref. 2) • <u>Significant Accident Progression Sequence</u>: "One of the sets of accident sequences contributing to large early release frequency resulting from the analysis of a specific hazard group that, when rank-ordered by decreasing frequency, sum to a specified percentage of the large early release frequency, or that individually contribute more than a specified percentage of large early release frequency for that hazard group. For this version of the Standard [RA-Sa-2009], the summed percentage is 95% and the individual percentage is 1% of the applicable hazard." (Ref. 2)

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	• Significant Basic Event: "A basic event that contributes significantly to the computed risks for a specific hazard group. For internal events, this includes any basic event that has an FV importance greater than 0.005 or a RAW importance greater than 2." (Ref. 2) • Significant Containment Challenge: "A containment challenge that results in a containment failure mode that is represented in a significant accident progression sequence." (Ref. 2) • Significant Cutset: "One of the sets of cutsets resulting from the analysis of a specific hazard group that, when rank-ordered by decreasing frequency, sum to a specified percentage of the core damage frequency (or large early release frequency) for that hazard group, or that individually contribute more than a specified percentage of core damage frequency (or large early release frequency). For this version of the Standard [RA-Sa-2009], the summed percentage is 95% and the individual percentage is 1% of the applicable hazard." (Ref. 2) • Risk Significant Equipment: "Equipment associated with a significant basic event." (Ref. 2) A significant contributor can refer to an important factor associated with a significant accident sequence, such as a particular accident sequence cutset, a significant basic event, or an initiating event. As stated in the ASME/ANS PRA Standard (Ref. 2), a significant contributor also can be "an essential characteristic (e.g., containment failure mode, physical phenomena) of a significant accident progression sequence, and if not modeled would lead to the omission of the sequence."
Significant Accident Progression Sequence	
(see Significant)	The term significant accident progression sequence is related to the term significant and is defined under "Significant."
Significant Accident Sequence	
(see Significant)	The term significant accident sequence is related to the term significant and is defined under "Significant."
Significant Basic Event	
(see Significant)	The term significant basic event is related to the term significant and is defined under "Significant."
Significant Containment Challenge	
(see Significant)	The term significant containment challenge is related to the term significant and is defined under "Significant."
Significant Contributor	
(see Significant)	The term significant contributor is related to the term significant and is defined under "Significant."
Significant Cutset	
(see Significant)	The term significant cutset is related to the term significant and is defined under "Significant."
Skin Deposition	
Exposure resulting from radioactive	In a Level 3 PRA, for the consequence calculation skin deposition is one of the assumed pathways by which an individual can receive doses. The pathways of exposure include: (1)

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material deposited directly onto the surface of the body. *(see Exposure Pathways, Exposure, Exposure Time, Cloudshine, Water Immersion, Groundshine, Inhalation, Ingestion, Health Effects)*	direct external exposure from radioactive material in a plume, principally due to gamma radiation (air immersion or cloudshine), (2) direct exposure from radioactive material in contaminated water given to an individual immersed in the water, (3) exposure from inhalation of radioactive materials in the cloud and resuspended material deposited on the ground, (4) exposure to radioactive material deposited on the ground (groundshine), (5) radioactive material deposited onto the body surfaces (skin deposition), and (6) ingestion from deposited radioactive materials that make their way into the food and water pathway.
Small Early Release	
(see Radioactive Material Release)	The term small early release is a type of radioactive material release and is defined in the discussion under “Radioactive Material Release.”
Small Early Release Frequency	
(see Frequency)	The term small early release frequency is a type of frequency used in PRA calculation and is defined in the discussion under “Frequency.”
Small Early Release Frequency Analysis	
(see Radioactive Material Release Frequency Analysis)	The term small early release frequency analysis is a type of radioactive material release frequency analysis and is defined under “Radioactive Material Release Frequency Analysis.”
Small Late Release	
(see Radioactive Material Release)	The term small late release is a type of radioactive material release and is defined in the discussion under “Radioactive Material Release.”
Small Late Release Frequency	
(see Frequency)	The term small late release frequency is a type of frequency used in PRA calculation and is defined in the discussion under “Frequency.”
Small Late Release Frequency Analysis	
(see Radioactive Material Release Frequency Analysis)	The term large late release frequency analysis is a type of radioactive material release frequency analysis and is defined under “Radioactive Material Release Frequency Analysis.”
Source of Risk	
A substance that can pose danger or threat to public health. *(see Hazard, Initiating Event)*	In a PRA, sources of risk at nuclear power plants include, for example, the nuclear fuel contained within the reactor core and the spent fuel pool. These sources of risk could be affected by hazards which directly or indirectly cause initiating events and may further cause safety system failures or operator errors leading to core damage or radioactive material release. For instance, in a non-nuclear application, a leak in a pool may not cause a negative consequence other than having an empty pool. However, because the pool at a nuclear power

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	plant contains nuclear fuel, there could be a negative consequence if that pool drained and radioactive material (the source of risk) was released. The terms source of risk and hazard are sometimes incorrectly used as synonyms. A hazard is anything that has the potential to cause an undesired event. Intrinsically, a source of risk does not cause an event, but a hazard can cause an initiating event leading to core damage. For example, an earthquake (hazard) with particular frequency could cause a loss-of-coolant accident (initiating event) which may result in core damage of the nuclear fuel (source of risk).
Source Term	
Types and amounts of radioactive or hazardous material released to the environment following an accident. *(see Release Category, Mechanistic Source Term, Chemical Element Group, Release Fraction, Release Timing and Duration, Source Term Analysis)*	In a Level 2 PRA, the source term is one of the end products of the analysis and involves the characterization of the release from containment to the environment. This characterization involves a description of the radionuclide release at a particular location, including the physical and chemical properties of released material, release magnitude, heat content (or energy) of the carrier fluid, location relative to local obstacles that would affect transport away from the release point, and the temporal variations in these parameters (e.g., time of release duration). The information used to define a source term can vary, depending on the objective and intended application of the PRA. For instance, if the Level 2 PRA results will be used in a Level 3 consequence assessment, it may be necessary to provide more detailed source term information than if no Level 3 assessment will be performed. For a Level 3 assessment, the source term information needs to be sufficient to estimate offsite radiation doses and, in some cases, other radiological consequences such as land contamination.
Source Term Analysis	
An analysis to determine the characteristics of the radioactive material released to the environment following an accident. *(see Source Term)*	In a Level 2 PRA, the source term analysis determines the release of radioactive material from the fuel or core debris and the transport of this material through the primary system and containment to the environment. (The scope of the PRA source term analysis usually does not include releases from the spent fuel pool.) NUREG-1489 (Ref. 54) states that there are three parts to a source term analysis: (1) the estimation of the release of radioactive material from the fuel and core debris, (2) the transport of this material through the primary system and the containment, and (3) the characterization of the release from containment to the environment.
Split Fraction	
The likelihood that one specific outcome from a set of possible outcomes will be observed. *(see Event Tree, Probability)*	A split fraction is a unitless parameter (i.e., probability). This term typically is used with regard to the quantification of an event tree of a PRA model. It represents the fraction with which each possible outcome, or branch, of a particular top event in an event tree may be expected to occur. Split fractions are, in general, conditional on prior events. At any event tree branch point, the sum of all the split fractions representing the possible outcomes should be unity. The ASME/ANS PRA Standard (Ref. 2) defines the term split fraction as “a unitless quantity that represents the conditional (on preceding events) probability of choosing one direction rather than the other through a branch point of an event tree.”
State-of-Knowledge Correlation	
A type of dependency that arises when the same data is used	In a PRA, when the basic event mean values and uncertainty distributions are propagated without accounting for the state-of-knowledge correlation (SOKC), the calculated mean value of the relevant risk metric and the uncertainty about this mean value will be underestimated.

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to quantify the individual probabilities of two or more basic events. *(see Uncertainty)*	When the same data is used to quantify the individual probabilities of two or more basic events, the uncertainty associated with such basic event probabilities must be correlated to correctly propagate the parameter uncertainty through the risk calculation. The SOKC arises because, for identical or similar components, the state-of-knowledge about their failure parameters is the same. In other words, the data used to obtain mean values and uncertainties of the parameters in the basic event models of these components may come from a common source and, therefore, are not independent, but are correlated. The ASME/ANS PRA Standard (Ref. 2) defines the term SOKC as “the correlation that arises between sample values when performing uncertainty analysis for cut sets consisting of basic events using a sampling approach (such as the Monte Carlo method); when taken into account, this results, for each sample, in the same value being used for all basic event probabilities to which the same data applies.”
State-of-Knowledge Uncertainty	
(see Uncertainty)	The term state-of-knowledge uncertainty is related to epistemic uncertainty and defined under “Uncertainty.”
Station Blackout	
The complete loss of alternating current electric power in a nuclear plant. *(see Transient)*	In a PRA, station blackout (SBO) accidents are analyzed because alternating current (AC) power is an important support system for numerous plant systems and components. A plant subjected to an SBO condition must achieve safe-shutdown by relying on mitigating systems and components that do not require AC power (e.g., steam-driven pumps and battery-powered valves and instrumentation). However, for operating plants, core cooling may not be indefinitely maintained without AC power. Important factors that influence the risk associated with SBO include the potential for recovery of AC power, battery depletion times, and the reliability of the mitigating systems and components that do not require AC power. 10 CFR 50.2 (Ref. 13) defines the term station blackout as “the complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). SBO does not include the loss of available AC power to buses fed by station batteries through inverters or by alternate AC sources, nor does it assume a concurrent single failure or design basis accident.” The ASME/ANS PRA Standard (Ref. 2) defines the term SBO as “complete loss of AC electric power to the essential and nonessential switchgear buses in a nuclear power plant.”
Steam Generator Tube Rupture	
A break or breach of a steam generator tube. *(see Consequential (Induced) Steam Generator Tube Rupture)*	In a PRA for a pressurized-water reactor, steam generator tube ruptures (SGTRs) are modeled either as an initiating event or a subsequent failure as part of an accident sequence. If the SGTR occurs randomly while the plant is operating, it is an initiating event modeled in the PRA. However, if the SGTR occurs because of excessive conditions produced as a result of an accident, it is considered to be a consequential or induced SGTR. An SGTR allows reactor coolant to flow from the reactor vessel to the secondary side of the steam generator. As such, it can become a significant contributor to risk because an SGTR can serve as a possible mechanism for radioactive material transport to the environment because it can be a containment bypass mechanism. There is the potential that if a tube bursts or leaks while a plant is operating, radioactivity from the primary coolant system could escape directly to the atmosphere through the safety valves on the secondary side.
Stochastic Uncertainty	
(see Uncertainty)	The term stochastic uncertainty is related to aleatory uncertainty and defined under “Uncertainty.”

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Structuralist	
An approach to defense-in-depth that relies on multiple strategies in the design and operation of a facility to compensate for both known and unknown uncertainties. *(see Rationalist, Deterministic, Defense-in-Depth)*	A PRA is not used in the structuralist approach to defense-in-depth, unlike the rationalist approach. Instead, the structuralist approach asserts that safety margins associated with defense-in-depth are embodied within the regulations and in the design of a facility built to comply with those regulations. The fundamental principle of the structuralist approach is that if a system is designed to withstand all the worst-case credible accidents, then it is by definition protected against any credible accident. It is a method that is solely based on deterministic analyses and principles to establish how precautions can be placed into a system, just in case an existing barrier or protective system fails. By comparison, a rationalist approach uses PRA methods to quantify and reduce system uncertainties, as opposed to relying on potentially overly conservative safety margins.
Success Criteria	
The minimum combination of systems and components needed to carry out the safety functions given an initiating event.	In a PRA, success criteria are used at different places or levels in the analysis. At a high level, the success criteria define the safety functions that must be performed following an initiating event. Success criteria are then defined for each safety function, which are expressed in terms of requirements for the systems needed to support that function. Success criteria also are developed for the components within these systems. The success criteria specify how the systems and components must function, when they must begin to function, and how long they must function. Success criteria for PRA studies typically are developed through the use of deterministic analyses that represent the design and operation of the plant being evaluated. Success criteria may be defined in a number of ways, including the following: • In terms of the equipment required (e.g., one out of two service water pumps). • In terms of equipment performance (e.g., at least 50 percent of the maximum system flow rate). • In terms of the timing (e.g., system must be initiated within 30 minutes and operate for 24 hours). The ASME/ANS PRA Standard (Ref. 2) defines the term success criteria as “criteria for establishing the minimum number or combinations of systems or components required to operate, or minimum levels of performance per component during a specific period of time, to ensure that the safety functions are satisfied.”
Success Path	
A sequence of events (responding to an upset condition) that result in a successful state of a system, the reactor, or the containment. *(see Event Tree, Safe Stable State)*	In a PRA, the term success path often is used in the context of describing an event tree path that leads to a safe stable state of the reactor. Alternatively, a fault tree model can be transformed into its logical complement, a success tree that shows the specific ways (success paths) in which an undesired event (e.g., system failure) can be prevented from occurring. A successful state of a system occurs when the system is able to perform its intended function (e.g., provide injection water at a sufficient flow rate and pressure). A successful state of a reactor is achieved if adequate core cooling is maintained throughout the sequence of events following an upset condition. For the containment, a successful state is achieved if the containment pressure boundary remains intact throughout the sequence of events following an upset condition. The ASME/ANS PRA Standard (Ref. 2) defines a success path as “a set of systems and associated components that can be used to bring the plant to a stable hot or cold condition and maintain this condition for at least 72 hrs.”

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Supplementary Analysis	
Any evaluation that is performed to support another study or evaluation.	In a PRA context, the term supplementary analysis often is used to denote an evaluation made to facilitate the development or review of a PRA consistent with the ASME/ANS PRA Standard (Ref. 2). An example of a supplementary analysis would be an evaluation of plant-specific component failure data to support derivation of plant-specific component failure rates for use in a PRA. Sometimes the supplementary analysis is performed instead of following the specific requirements in the ASME/ANS PRA Standard. In this situation, the supplementary analysis is performed to meet the Standard's intent, but it is outside the scope of the Standard. Therefore, performing a supplementary analysis does not meet all the Standard's criteria.
Support System	
A system that enables the operation of one or more systems. *(see Front-Line System, Support System Initiating Event)*	In a PRA, support system failures are evaluated to determine the effect of these failures on the operability of other plant systems and components. Often one support system, such as component cooling water, provides functionality to multiple systems or components, and therefore, needs to be considered in PRA modeling to assess what happens if that capability is lost to multiple systems. Examples of support systems include electrical power, cooling water, instrument air, and heating, ventilation, and air conditioning. Support systems (e.g., cooling water) can require other support systems for operation (e.g., electric power may be needed to operate the cooling water pumps). Front-line systems typically require one or more support systems. In some instances, a failed support system can lead to an undesired plant condition that requires successful mitigation by plant equipment and personnel to prevent core damage from occurring. In this situation, the support system failure would be characterized as a support system initiating event. The ASME/ANS PRA Standard (Ref. 2) defines the term support system as "a system that provides a support function (e.g., electric power, control power, or cooling) for one or more other systems."
Support System Initiating Event	
A support system failure that perturbs the steady-state operation of the plant and could lead to an undesired plant condition. *(see Initiating Event, Support System)*	In a PRA, the failures of support systems are evaluated to determine if they could potentially cause an undesired plant condition (i.e., a manual trip or a reactor shutdown). At the same time, this failed support system also may have the potential to disable one or more systems that could be used to mitigate the undesired plant condition. An example of a support system initiating event would be the loss of the component cooling water (CCW) system at a pressurized-water reactor. The failure of this system would, in turn, lead to the consequential failure of a number of other important systems that depend on CCW, which might include the reactor coolant pumps (RCPs) and emergency core cooling system (ECCS) equipment. Loss of the RCPs would result in a plant trip, and loss of ECCS functionality would reduce the number of plant mitigating systems that could be used to maintain core cooling following the plant trip.
Supporting Requirements	
Requirements that support the high-level requirements in defining the minimum needed for a technically	For a base PRA, NRC Regulatory Guide 1.200 (Ref. 91) defines a set of technical characteristics and associated attributes that make it technically acceptable. One approach to demonstrate a PRA is acceptable is to use a national consensus PRA standard, supplemented to account for the NRC staff's regulatory positions. The ASME/ANS PRA Standard (Ref. 2) is one example of such a national consensus PRA standard. The ASME/ANS PRA Standard uses high-level requirements and supporting requirements.

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acceptable baseline PRA. *(see High-Level Requirements, Capability Categories)*	Regulatory Guide 1.200 (Ref.91) states, "Technical requirements may be defined at two different levels: (1) high-level requirements and (2) supporting requirements. High-level requirements are defined for each technical element and capture the objective of the technical element. These high-level requirements are defined in general terms, need to be met regardless of the level of analysis resolution and specificity (capability category), and accommodate different approaches. Supporting requirements are defined for each high-level requirement. These supporting requirements are those minimal requirements needed to satisfy the high-level requirement." To use a PRA for a risk-informed application, it is recognized that not every PRA item will be, or needs to be, developed to the same level of detail, same degree of plant-specificity, or the same degree of realism. The ASME/ANS PRA Standard (Ref. 2) uses three capability categories to distinguish levels of detail, plant specificity, and realism. Furthermore, the supporting requirements are developed commensurate with each capability category. Therefore, while the high-level requirements are the same across all three capability categories, their supporting requirements reflect the differences in levels of detail, plant specificity, and realism across the three categories.
Systems Analysis	
The evaluation of the reliability and availability of a system. *(see Availability, Reliability)*	In a PRA, the term systems analysis can refer to a qualitative or quantitative evaluation of the failure modes of an individual system or group of systems (e.g., a fault tree analysis of a cooling water system or an electrical distribution system).
Technical Acceptability, Technical Quality (PRA)	
Refers to a set of characteristics and related attributes that provide the minimum qualities a base PRA must satisfy to be used in risk-informed decisionmaking. *(see Technical Adequacy)*	For a PRA to be technically acceptable, it must satisfy a set of technical characteristics and associated attributes. Regulatory Guide (RG) 1.200 (Ref. 91) defines such a set of characteristics and accompanying attributes that need to be addressed in a technically acceptable base PRA (i.e., independent of the application for which the PRA is used). RG 1.200 guidance is for operating reactors and contains cautions for new advanced light-water reactors. Technical acceptability and technical quality mean the same thing and are used interchangeably.
Technical Adequacy (PRA)	
Refers to the fact that the PRA has the scope and level of detail necessary to support the application for which it is being used and is also technically acceptable. *(see Technical Acceptability)*	The scope of a PRA (i.e., risk characterization, level of detail, plant specificity and realism) needs to be commensurate with the scope of the specific risk-informed application that it is supporting. Some applications (e.g., extension of diesel generator allowed outage time) may only use a portion of the base PRA, whereas other applications (e.g., safety significance categorization of structures, systems, and components) may require the complete model. Regulatory Guide 1.200 (Ref. 91) provides guidance on an acceptable approach for demonstrating the technical adequacy of a PRA used to support a regulatory application. Central to this approach is the concept that the PRA needs to only have the scope and level of detail necessary to support the application for which it is being used, but it always needs to be technically acceptable.

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Technical Elements	
(see PRA Technical Elements)	The term technical elements has the same meaning as PRA technical elements in the context of PRA and is defined under “PRA Technical Elements.”
Technical Quality	
(see Technical Acceptability)	The term technical quality has the same meaning as technical acceptability and is defined the same as the term “Technical Acceptability.”
Top Event (Event Tree Top Event)	
The events across the top of an event tree needed to mitigate an accident. *(see Event Tree, Fault Tree)*	The NRC Web site Glossary (Ref. 36) defines top events as “the events across the top of the event tree, which graphically represent the systems needed to keep the plant in a safe state following an initiating event (i.e., a challenge to plant operation). A top event is the starting point of the fault tree, which identifies all of the pathways that lead to a system failure.” The fault tree starts with the top event, as defined by the event tree, and identifies what equipment and operator actions, if failed, would prevent successful operation of the system. The ASME/ANS PRA Standard (Ref. 2) includes two terms: event tree top event and top event. Event tree top event is defined as “the conditions (i.e., system behavior or operability, human actions, or phenomenological events) that are considered at each branch point in an event tree.” Top event is defined as the “undesired state of a system in the fault tree model (e.g., the failure of the system to accomplish its function) that is the starting point (at the top) of the fault tree.” An illustration of a top event is shown under the discussion for the term “Event Tree.”
Total Effective Dose Equivalent	
(see Dose Equivalent)	The total effective dose equivalent is one measure of dose that can be used to calculate the effect of radiation received by an individual and is defined under “Dose Equivalent.”
Transient, General Transient	
An event that could require a plant trip that might challenge safety systems but does not lead to a loss of significant quantities of reactor coolant. *(see Initiating Event, Station Blackout)*	In a PRA, two major categories of initiating events are evaluated; namely, transients and loss-of-coolant accidents. Transients can represent a variety of initiating events (e.g., manual reactor trip, loss of main feedwater, turbine trip, loss of offsite power, and loss of primary flow). Each of these initiating events subsequently leads to changes in reactor temperature or pressure that could demand functioning of safety systems. Transients are modeled in the PRA if they lead to a plant trip, thus challenging safety systems leading to positive or negative outcomes. The terms transient and general transient often are used interchangeably, which is appropriate and correct in a PRA context. NUREG/CR-6572 (Ref. 76) defines the term general transient as “events in which high pressure can be maintained in the primary system, active core cooling is required, and high pressure makeup may be needed.” The NRC Web site Glossary (Ref. 36) defines the term transient as “a change in the reactor coolant system temperature, pressure, or both, attributed to a change in the reactor’s power output. Transients can be caused by (1) adding or removing neutron poisons, (2) increasing or decreasing electrical load on the turbine generator, or (3) accident conditions.”
Truncation Limit	
The minimum value of	In a PRA, a truncation limit is a numerical criterion that defines the boundaries, in terms of frequencies or probabilities, of what is retained and what is screened out. The truncation limit

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contributors retained in the PRA quantification process. *(see Accident Sequence, Cutset)*	determines what accident sequences or cutsets are retained for or excluded from further analysis. Since truncation limit affects PRA quantification, Regulatory Guide 1.200 (Ref. 91) notes that truncation values should be set relative to the total plant core damage frequency (CDF) such that the CDF is stable with respect to further reduction in the truncation value. The ASME/ANS PRA Standard (Ref. 2) defines truncation limit as "the numerical cutoff value of probability or frequency below which results are not retained in the quantitative PRA model or used in subsequent calculations (such limits can apply to accident sequences-cutsets, system level cutsets, and sequence-cutset database retention)."
Unavailability	
(see Availability)	The term unavailability is the opposite of availability and is defined under "availability."
Uncertainty (Aleatory, Random, Stochastic, Epistemic, State-of-Knowledge, Model, Source of Model, Key Source of Model, Parameter, Completeness)	
Variability in an estimate because of the randomness of the data or the lack of knowledge.	When used in the context of a PRA, the term uncertainty is associated with the lack of information or knowledge, or the random behavior of a system or model that is taken into account in the PRA in different ways. In defining uncertainty, there are two types: aleatory and epistemic. Aleatory uncertainty is based on the randomness of the nature of the events or phenomena and cannot be reduced by increasing the analyst's knowledge of the systems being modeled. Therefore, it is also known as random uncertainty or stochastic uncertainty. Epistemic uncertainty is the uncertainty related to the lack of knowledge or confidence about the system or model and is also known as state-of-knowledge uncertainty. The PRA model itself reflects aleatory uncertainty. The PRA model contains epistemic uncertainty that includes model uncertainty, parameter uncertainty, or completeness uncertainty. In the ASME/ANS PRA Standard (Ref. 2), uncertainty is defined as "a representation of the confidence in the state-of-knowledge about the parameter values and models used in constructing the PRA." In the ASME/ANS PRA Standard (Ref. 2), aleatory uncertainty is defined as "the uncertainty inherent in a nondeterministic (stochastic, random) phenomenon. Aleatory uncertainty is reflected by modeling the phenomenon in terms of a probabilistic model. In principle, aleatory uncertainty cannot be reduced by the accumulation of more data or additional information. (Aleatory uncertainty is sometimes called 'randomness.')" In the ASME/ANS PRA Standard (Ref. 2), epistemic uncertainty is defined as "the uncertainty attributable to incomplete knowledge about a phenomenon that affects our ability to model it. Epistemic uncertainty is reflected in ranges of values for parameters, a range of viable models, the level of model detail, multiple expert interpretations, and statistical confidence. In principle, epistemic uncertainty can be reduced by the accumulation of additional information. (Epistemic uncertainty is sometimes also called 'modeling uncertainty.')" Model uncertainty is discussed in NUREG-1855 (Ref. 60) as follows: "Model uncertainty is related to an issue for which no consensus approach or model exists and where the choice of approach or model is known to have an effect on the PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, and introduction of a new initiating event). A model uncertainty results from a lack of knowledge of how structures, systems and components (SSCs) behave under the conditions arising during the development of an accident. A model uncertainty can arise for the following reasons:

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	• The phenomenon being modeled is itself not completely understood (e.g., behavior of gravity-driven passive systems in new reactors, or crack growth resulting from previously unknown mechanisms). For some phenomena, some data or other information may exist, but it needs to be interpreted to infer behavior under conditions different from those in which the data were collected (e.g., RCP seal LOCA information). • The nature of the failure modes is not completely understood or is unknown (e.g., digital instrumentation and controls)." In the ASME/ANS PRA Standard (Ref. 2), source of model uncertainty is defined as: "a source that is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event). A source of model uncertainty is labeled "key" when it could impact the PRA results that are being used in a decision, and consequently, may influence the decision being made. Therefore, a key source of model uncertainty is identified in the context of an application. This impact would need to be significant enough that it changes the degree to which the risk acceptance criteria are met, and therefore, could potentially influence the decision." NUREG-1855 (Ref. 62) has additional discussion on key sources of model uncertainty. The terms key model uncertainty and key sources of model uncertainty have the same meaning. Parameter uncertainty is the uncertainty in the values of the parameters of a model represented by a probabilistic distribution. Examples of parameters that could be uncertain include initiating event frequencies, component failure rates and probabilities, and human error probabilities that are used in the quantification of the accident sequence frequencies. Completeness uncertainty is caused by the limitations in the scope of the model, such as whether all applicable physical phenomena have been adequately represented, and all accident scenarios that could significantly affect the determination of risk have been identified. Completeness uncertainty also can be thought of as a type of model uncertainty. However, completeness uncertainty is separated from model uncertainty because it represents a type of uncertainty that cannot be quantified. It also represents those aspects of the system that are, either knowingly or unknowingly, not addressed in the model. (Ref. 62)
Uncertainty Analysis	
A process for determining the level of imprecision in the results of the PRA and its parameters.	In a PRA, the ways in which the uncertainty in the results is presented includes the following: • A continuous probability distribution on numerical results. • A discrete probability distribution representing the impact of different models or assumptions. • Sensitivity studies that provide a discrete set of results that represent the results of making different assumptions or using different models, or that represent the impact of varying key parameters in the model that have significant uncertainty, without providing weights or probabilities to the members of the set. • Bounds or ranges of results that represent the results of the extreme assumptions. • An identification of limitations in the scope of the model (e.g., incompleteness) and how they might influence the applicability of the PRA. The ASME/ANS PRA Standard (Ref. 2) defines uncertainty analysis as "the process of identifying and characterizing the sources of uncertainty in the analysis, and evaluating their impact on the PRA results and developing a quantitative measure to the extent practical."

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Uncertainty Distribution	
(see Probability Distribution)	The term uncertainty distribution is related to the term probability distribution and is defined under "Probability Distribution."
Uncertainty Interval, Uncertainty Range	
A range that bounds the uncertainty value(s) of a parameter or analysis result by establishing upper and lower limits. *(see Confidence Interval, Probability Distribution)*	In a PRA, uncertainty intervals can provide the range of the frequency or probability of the various inputs (e.g., initiating event frequencies, component failure probabilities, human error probabilities), as well as outputs of the analysis (e.g., core damage frequency, conditional containment failure probability). However, in most cases, a probability distribution of the uncertainty around a mean value is preferred. NUREG 1855 (Ref. 62) defines uncertainty interval as "a characterization of the uncertainty. This characterization could, in the simplest approach, take the form of an interval (i.e., a range of values within which the value lies). However, it is more usual to characterize the uncertainty in terms of a probability distribution on the value of the quantity of concern, whether it is a parameter, accident sequence frequency, or a core damage frequency." The NRC Web site Glossary (Ref. 34) defines uncertainty range as "an interval within which a numerical result is expected to lie within a specified level of confidence. The interval often used is the 5–95 percentile of the distribution reporting the uncertainty." The definition provided was based on definitions in the NRC Web site Glossary (Ref. 36) and in NUREG-1855 (Ref. 62).
Uncertainty Range	
(see Uncertainty Interval)	The term uncertainty range has the same meaning as uncertainty interval and is defined under "Uncertainty Interval."
Unreliability	
(see Reliability)	The term unreliability is the opposite of reliability and is defined under "Reliability."
Up-to-Date	
(see PRA Configuration Control, As-Built As-Operated)	The term up-to-date is related to PRA configuration control and is defined under "PRA Configuration Control" or "As-Built As-Operated."
Vulnerability	
Weakness in the design or operation of a system, component, or structure that could disable its function.	Results from a PRA of a nuclear power plant (NPP) model can be used to identify plant vulnerabilities (e.g., vulnerabilities related to system design or plant operations). The term vulnerability was used in Generic Letter (GL) 88-20, "Individual Plant Examination For Severe Accident Vulnerabilities" (Ref. 40). As part of GL 88-20, each licensee was asked to perform a systematic examination of its NPP to identify any plant-specific vulnerabilities to severe accidents. The NRC, however, did not define vulnerability; it was the licensee's responsibility to define vulnerability. The method all licensees used to identify vulnerabilities was a PRA. For some licensees, vulnerabilities were based on the contribution of accident sequence types or individual failure events (e.g., fault tree basic events) to overall plant core damage frequency (CDF) or a percent contribution to CDF (e.g., a functional accident sequence with a CDF that exceeds 1E-04/yr, or one that contributes more than 50% to the total plant CDF).

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Water Immersion	
Direct exposure from radioactive material in contaminated water given to an individual immersed in the water. *(see Exposure Pathways, Cloudshine, Groundshine, Inhalation, Ingestion, Skin Deposition)*	In a Level 3 PRA, for the consequence calculation, water immersion, is one of the assumed pathways by which an individual can receive doses. The pathways of exposure include: (1) direct external exposure from radioactive material in a plume, principally due to gamma radiation (air immersion or cloudshine), (2) direct exposure from radioactive material in contaminated water given to an individual immersed in the water, (3) exposure from inhalation of radioactive materials in the cloud and resuspended material deposited on the ground, (4) exposure to radioactive material deposited on the ground (groundshine), (5) radioactive material deposited onto the body surfaces (skin deposition), and (6) ingestion from deposited radioactive materials that make their way into the food and water pathway.

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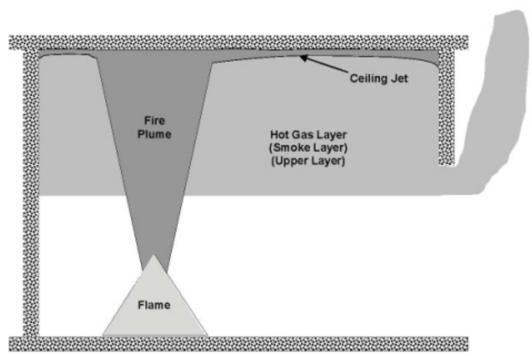
APPENDIX A

INTERNAL FIRE GLOSSARY

Table A-1 provides internal fire terms and their definitions with the associated discussion. The terms are listed alphabetically.

Table A-1 Internal Fire Terms and Definition

TERM AND DEFINITION (S)	DISCUSSION
Active Fire Barriers	
A fire barrier that must be physically repositioned from its normal configuration to an alternate configuration in order to provide its protective function.	In a fire PRA, fire barriers impede the spread of fires and limit potential damage to safety equipment, thus reducing probabilities of fire spread to additional components and the probability of accident sequences. Ventilation system fire dampers, normally open fire doors, and water curtains are examples of passive fire barriers. The definition provided was based on the definition in NUREG-1805 (Ref. 60).
Algebraic Fire Models	
A type of fire model that provides a method for calculating simple fire phenomena based on a closed-form algebraic formulation.	In a fire PRA, fire models predict fire damage of components, and thus contribute to the failure of those components, given failure of suppression. Algebraic models may be standalone equations found in the literature or may be contained within spreadsheets, such as the NRC's fire dynamics tools (FDTs). These equations are typically closed-form algebraic expressions, many of which were developed as correlations from empirical data. In some cases, they may take the form of a first-order ordinary differential equation and can provide an estimate of fire variables, such as hot gas layer (HGL) temperature, heat flux from flames or the HGL, smoke production rate, depth of the hot gas layer, and the actuation time for detectors. Algebraic models are helpful because they require minimal computational time and a limited number of input variables. Other than for very simple situations, algebraic models are useful primarily as screening tools. The definition provided was based on the definition in NUREG-1934 (Ref. 65).
Authority Having Jurisdiction	
The organization, office, or individual responsible for approving equipment, materials, an installation, or a procedure.	The NRC is the authority having jurisdiction for NFPA 805 as it is applied under 10 CFR 50.48 (Ref.17). The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).
Cable and Raceway (Database) System	
Cross-reference of power, control, or instrument cables associated with certain components or systems and their location throughout the plant, as it relates to specific cable raceways, tracks, or conduits where they may be situated.	The Cable and Raceway System (CRS) generally correlates cables to raceways, raceways to locations within the plant, and tracks basic cable and raceway attributes. Newer CRSs typically contain sophisticated database sort and query features. The information in the CRS may be used to determine how a fire in a certain location may affect the cables nearby and thus determine which components and systems may be affected. The location of cables is then used for the development of fire scenarios that are quantified in the fire PRA. This is then used in a PRA as input in constructing and calculating accident sequences. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).

TERM AND DEFINITION (S)	DISCUSSION
Cable Failure Mode	
The behavior of an electrical cable upon fire-induced failure. (*see Intercable Shorting, Intracable Shorting)*	In a fire PRA, component failure modes can be attributed to cable failure modes resulting from fire. The ASME/ANS PRA Standard (Ref. 2) indicates that "failure modes for electrical cables include intractable shorting, intercable shorting, open circuit (loss of conductor continuity), and/or shorts between a conductor and an external ground."
Ceiling Jet	
The relatively rapid gas flow in a shallow layer beneath the ceiling surface that is driven by the buoyancy of hot combustion products.	Typically, a fire plume will form above a burning object. The fire plume will rise until obstructed by a horizontal surface, such as a ceiling. Upon hitting the ceiling, the hot gases in the fire plume will turn and flow along the ceiling in the form of a ceiling jet. When the ceiling jet gases are blocked by vertical surfaces, such as walls, they will accumulate into a hot gas layer or smoke layer. As more hot gas accumulates in the layer, the interface between the hot gas layer and cooler layer below will continue to drop toward the floor of the enclosure. As stated in NUREG/CR-6850 (Ref. 79), "ceiling jets form when a fire plume impinges under a ceiling and hot gases spread away."  The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Circuit Failure Analysis	
The evaluation of electrical circuits to determine both the potential failure modes and their impact on the systems and equipment supported by the circuit.	Circuit failure analysis can include the assignment of probabilities to the likelihood of the cable failure modes of concern. Circuit failure analysis would include consideration of the impact of cable failures on circuit function. The equipment failures associated with those circuit failure modes would be input to the PRA and contribute to accident sequence quantification.
Circuit Failure Mode	
The manner in which conductor failures from an electrical cable are manifested in the circuit. *(see Cable Failure Mode)*	In a fire PRA, equipment failures associated with circuit failure modes are analyzed and contribute to accident sequence quantification. Examples of circuit failure modes include loss of motive power, loss of control, loss of or false indication, open circuit conditions, and spurious operation. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).

TERM AND DEFINITION (S)	DISCUSSION
Code of Record	
The edition of the code or standard in effect at the time the fire protection systems or feature was designed or specifically committed to the authority having jurisdiction. *(see Authority Having Jurisdiction)*	If the 1996 edition of NFPA 13 was in effect at the time a sprinkler system was designed, the code of record would be NFPA 13, *Standard for the Installation of Sprinkler Systems* – 1996 edition. The definition provided was based on the definition in the NFPA 805 Standard (Ref.11).
Compensatory Actions	
Actions taken to counteract or reduce an impairment to a required fire protection system, feature, or component.	In the NFPA 805 Standard (Ref.11), compensatory actions are described as **"**actions taken if an impairment to a required system, feature, or component prevents that system, feature, or component from performing its intended function. These actions are a temporary alternative means of providing reasonable assurance that the necessary function will be compensated for during the impairment, or an act to mitigate the consequence of a fire. Compensatory measures include, but are not limited to, actions such as fire watches, administrative controls, temporary systems, and features of components." The term compensatory measures may be used in place of compensatory actions (e.g., fire watch compensatory actions may improve detection in the affected vicinity). The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).
Concurrent Hot Shorts	
The occurrence of two or more hot shorts such that the shorts overlap in time. *(see Conductor-to-Conductor Short)*	In a fire PRA, concurrent hot shorts are important because they can cause multiple equipment failures, complicate operator response, and increase human error probabilities in a fire PRA. These challenges may be more difficult to overcome than would be the case given only a single spurious operation at a time. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Conductor-to-Conductor Short	
An abnormal connection (including an arc) of relatively low impedance between two conductors.	In a fire PRA, conductor-to-conductor shorts may be caused by fire and in turn may cause failure of equipment, thus contributing to accident sequences. As described in NUREG/CR-6850 (Ref. 76), a conductor-to-conductor short can occur in the following manner: "a conductor-to-conductor short between an energized conductor of a grounded circuit and a grounded conductor results in a ground fault. A conductor-to-conductor short between an energized conductor and a non-grounded conductor results in a hot short. A conductor-to-conductor short between an energized conductor of an ungrounded circuit and a neutral conductor has the same functional impact as a ground fault." The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).

TERM AND DEFINITION (S)	DISCUSSION
Damage Criteria	
Those characteristics of the fire-induced environment that are specified as indicating failure of a damage target or set of damage targets. *(see Damage Target, Damage Threshold)*	In a fire PRA, cables and their associated components are failed in the PRA model upon damage. Damage criteria commonly refer to certain temperatures or heat fluxes at target locations that when exceeded indicate failure of the targets. The damage target may be a cable, set of cables, or a component in a location near the fire. The damage criteria also may be based on any other environmental effect of the fire (e.g., smoke density). The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Damage Target	
Any cable, equipment, or structural element in the fire PRA whose function can be adversely affected by the modeled fire.	In a fire PRA, cables and their associated components are failed in the PRA model upon damage. The ASME/ANS PRA Standard (Ref. 2) defines the term damage target as "a cable or equipment item that belongs to the Fire PRA cable or equipment list and that is included in event trees and fault trees for fire risk estimation. Damage targets also may include structural elements (e.g., structural steel) in the case of certain high-hazard fire sources, such as very large oil spills." The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Damage Threshold	
The values corresponding to the damage criteria that will be taken as indicative of the onset of fire-induced failure of a damage target or set of damage targets. *(see Damage Criteria)*	An example of a damage threshold would be the temperature at a cable location that when exceeded would indicate failure of the cable. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Electrical Cable	
A construct consisting of one or more insulated conductors designed to carry signals or power between points in a circuit.	In a fire PRA, fire damage to a cable may result in disablement or spurious operation of safety-related equipment (affecting probability of failure of safety systems) and/or generation of an initiating event. Cables are used to connect points in a common electrical circuit and may be used to transmit power, control signals, indications, or instrument signals. Cables are important to risk because they connect equipment necessary for safe operation of the plant to sources of power and control over relatively long distances in the plant. This increases the possibility that an undesired event (e.g., a fire) at an intervening location will affect the cable and disrupt the continued operation of equipment.
Electrical Raceway Fire Barrier System	
Non-load-bearing partition type envelope system installed around electrical components and cabling that are rated by test laboratories in hours of fire resistance and	In a fire PRA, electrical raceway fire barrier systems (ERFBSs) are modeled because they provide protection for electrical cables and delay or prevent damage from fires. A fire rated ERFBS provides additional time before damage for those protected cables in a fire PRA. The definition provided was based on the definition in Regulatory Guide 1.189 (Ref. 90).

TERM AND DEFINITION (S)	DISCUSSION
used to maintain safe-shutdown functions free of fire damage. *(see Wrap)*	
External Hot Short	
A hot short in which the source conductor and target conductor are from separate cables. *(see Hot Short, Intercable Short Circuit)*	The term external hot short can be used interchangeably and correctly with intercable short circuit, which is also referred to as intercable conductor-to-conductor short circuit. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Field Models	
A type of fire model that provides a method for calculating fluid flow through a volume using numerical solutions of the governing equations for conservation of total mass, chemical species, momentum, and energy.	In a fire PRA, the results from a field model can be used as input in determining the probability of damage from a particular fire to targets nearby and to associated safety-related equipment. Field models are computational fluid dynamics models that can be used to predict fire-induced environmental conditions (e.g., temperature at different times). The equations used in field models are approximated using finite differences over discrete control volumes, and the solution is obtained using the discretized equations. The calculations are performed over a period of time to obtain a transient (time-dependent) solution, or iterated over many times to provide a steady-state (time-independent) solution. The model typically is comprised of a large number of control volumes from thousands to millions. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Fire Analysis Tool	
A method used to estimate or calculate one or more physical fire effects. *(see Field Model, Zone Model, Algebraic Fire Model)*	Fire analysis tools include, but are not limited to, computerized compartment fire models, such as zone or field models, closed-form algebraic fire models, empirical correlations such as those provided in a handbook, and lookup tables that relate input parameters to a predicted output. The fire analysis tool used is based on the objectives of the specific analysis and a predefined set of input parameter values as defined by the fire scenario being analyzed. Examples of calculated physical fire effects are temperature, heat flux, time to failure of a damage target, rate of flame spread over a fuel package, heat release rate for a burning material, and smoke density. The ASME/ANS PRA Standard (Ref. 2) defines the term fire analysis tool as “any method used to estimate or calculate one or more physical fire effects (e.g., temperature, heat flux, time to failure of a damage target, rate of flame spread over a fuel package, heat release rate for a burning material, smoke density, etc.) based on a predefined set of input parameter values as defined by the fire scenario being analyzed. Fire analysis tools include, but are not limited to, computerized compartment fire models, closed-form analytical formulations, empirical correlations such as those provided in a handbook, and lookup tables that relate input parameters to a predicted output.” The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Area	
An area enclosed by rated fire barriers capable of preventing	In a fire PRA, the spread of fire and fire effects is limited (reduced probability of propagation) across fire areas. A multicompartment fire analysis is done across fire areas to evaluate the risk significance of these fire scenarios.

TERM AND DEFINITION (S)	DISCUSSION
or inhibiting spread of fires to and from the outside. *(see Fire Barrier)*	A fire area must be made up of rated fire barriers with openings in the barriers provided with fire doors, fire dampers, and fire penetration seal assemblies with a fire resistance rating at least equivalent to the barrier in which it exists (e.g., this term is defined in the analysis in Appendix R to 10 CFR Part 50 (Ref. 24)). Fire areas tend to confine most fires within the area. In a PRA, the fire area concept may simplify analysis, as each fire area generally may be treated independently from others. Fires may spread from one area to the next should a portion of the barrier be defeated (e.g., fire door left open). Regulatory Guide 1.189 (Ref. 90) defines the term fire area as “the portion of a building or plant that is separated from other areas by rated fire barriers adequate for the fire hazard.”
Fire Barrier	
A component intended to impede spreading of a fire and its effects. *(see Passive Fire Barrier, Active Fire Barrier)*	In a fire PRA, fire barriers are modeled to prevent or reduce the spread of fires between fire areas. Therefore, fire barriers reduce the probability of damage to safety-related equipment in adjacent areas, and thus reduce the frequency of undesired end states. Fire barriers can be active, indicating the barrier requires some physical repositioning to function, or passive, indicating the barrier provides protection in its normal orientation. Certification of a fire barrier’s fire resistance endurance rating typically is based on standardized tests, such as the American Society of Testing and Materials (ASTM) Standard E-119. Examples of solid construction made of fire-resistant material could be a wall or door. NUREG/CR-6850 (Ref. 79) defines the term fire barrier as “components of construction (walls, floors, and their supports), including beams, joists, columns, penetration seals or closures, fire doors, and fire dampers that are rated by approving laboratories in hours of resistance to fire, that are used to prevent the spread of fire and restrict spread of heat and smoke.”
Fire Compartment	
A subdivision of a building or plant that is a well-defined enclosed room, not necessarily bounded by rated fire barriers, which essentially confines the fire.	In a fire PRA, fire compartments are modeled because they reduce the probability of fire spread across boundaries. Boundaries of a fire compartment may have open equipment hatches, stairways, doorways, or unsealed penetrations. As discussed in the ASME/ANS PRA Standard (Ref. 2), “a fire compartment generally falls within a fire area and is bounded by noncombustible barriers where heat and products of combustion from a fire within the enclosure will be substantially confined.” The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Control	
The stage of firefighting in which a fire incident is controlled and not allowed to escalate in magnitude.	In current fire PRA practice, the concept of fire control generally is not used because there is large uncertainty associated with declaring when a fire has been brought under control as opposed to having been fully extinguished. Also, fire control is not modeled in fire models. Fire control can be achieved by water-based fixed systems or through the application of other fire suppression means (e.g., hose streams, portable extinguishers). Furthermore, gaseous fixed systems can prevent fire damage from extending beyond the locations damaged when the system is actuated. The concept of fire control may also include managed fire burnout whereby a fire is allowed to continue burning until the fuel source is exhausted (e.g., in the case of a leak of flammable compressed gases such as hydrogen). The definition provided was based on the definition in NUREG-1805 (Ref. 60).

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Fire Event	
A particular case where a fire has occurred in a nuclear power plant.	Fire events are characterized in the fire events database. A fire event is described by its initiation, the progression of the fire, detection and suppression, and the impact on plant systems.
Fire Events Database	
A collection of fire events that indicates characteristics of the fire and response by fire protection systems and plant personnel as well as the impact of the fire on plant equipment and operations.	In a fire PRA, the fire events database is used to provide raw data to calculate fire ignition frequencies and manual suppression reliability for different types of fires.
Fire Extinguishment	
The stage of a fire when combustible materials are no longer burning.	In a fire PRA, fire extinguishment concludes the duration of a fire and implies that all burning materials have been fully suppressed. Fire damage generally is modeled in fire PRA until fire extinguishment.
Fire Hazard Analysis	
An analysis to evaluate potential fire sources and combustibles, and appropriate fire protection systems, and features used to mitigate the effects.	Fire hazards analyses are generally of a qualitative or semi-quantitative nature as compared to a quantitative PRA. Regulatory Guide 1.189 (Ref. 90) defines fire hazard analysis as “an analysis used to evaluate the capability of a nuclear power plant to perform safe-shutdown functions and minimize radioactive releases to the environment in the event of a fire. The analysis includes the following features: identification of fixed and transient fire hazards; identification and evaluation of fire prevention and protection measures relative to the identified hazards; evaluation of the impact of fire in any plant area on the ability to safely shut down the reactor and maintain shutdown conditions, as well as to minimize and control the release of radioactive material.” The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).
Fire Human Reliability Analysis	
A structured approach used to identify potential human error events that may occur in a sequence of events following a fire and to systematically estimate the probability of those errors using data, models, or expert judgment as applied to a fire.	Fire human reliability analysis is used to quantify the potential impact of fire-generated environmental effects and stressors on human performance and the likelihood that errors might occur during execution of fire response procedures for specific areas of the plant, including control room evacuation. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).

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Fire Ignition Frequency	
Frequency of fire occurrence generally expressed as fire ignitions per reactor-year.	In a fire PRA, fire ignition frequency is normally calculated based on fires events that have the potential to cause damage to targets outside the ignition source. Fire ignition frequency is the factor that, in quantification, introduces the frequency element into the fire-induced core damage frequency. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire-Induced Initiating Event	
The initiating event assigned to occur in the fire PRA plant response model for a given fire scenario. *(see Fire Plant Response Model)*	The term initiating event is defined in the exact same context as is used in internal events PRA. That is, the initiating event is not the fire, it is induced by the fire. For example, a fire affects a pilot operated relief valve (PORV) control cable, causing spurious operation of a PORV, and thus an initiating event. Fire-induced initiating events trigger sequences of events that challenge plant control and safety systems whose failure potentially could lead to core damage or large early release. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Model	
A mathematical prediction of fire growth, environmental conditions, and potential effects on structures, systems, or components based on the conservation equations or empirical data.	The American Society of Testing and Materials (ASTM) Standard E176-10a, “Standard Terminology of Fire Standards” (Ref. 3), defines fire model as “a physical representation or set of mathematical equations that approximately simulate the dynamics of burning and associated processes.” The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).
Fire Plant Response Model	
A representation of a combination of equipment, cable, circuit, and system function, and operator failures or successes, of an accident that when combined with a fire-induced initiating event can lead to undesired consequences, with a specified end state (e.g., core damage or large early release).	In a fire PRA, the fire plant response model contains the event trees and fault trees that will be used to analyze fire-induced initiating events. Given a fire scenario leading to fire-induced failure of a fire damage target set, a plant damage state (fire-induced damage to plant systems and components including equipment failure modes) is defined and incorporated into the fire plant response model. The event tree/fault tree models are then manipulated to depict the logical relationships among equipment failures (both random and fire-induced) and human failure events. As in internal events, the fire plant response model estimates the conditional core damage probability given loss of a fire damage target set. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Plume	
Buoyant stream of hot gases rising above a localized area undergoing combustion	Typically, a fire plume will form above a burning object. The fire plume will rise until obstructed by a horizontal surface, such as a ceiling. Upon hitting the ceiling, the hot gases in the fire plume will turn and flow along the ceiling in the form of a ceiling jet. When the ceiling jet gases are blocked by vertical surfaces, such as walls, they will accumulate into a

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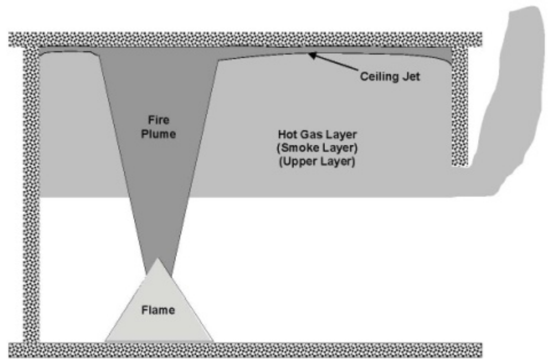
into surrounding space of essentially uncontaminated air.	hot gas layer or smoke layer. As more hot gas accumulates in the layer, the interface between the hot gas layer and cooler layer below will continue to drop toward the floor of the enclosure.  The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Fire PRA, Fire Probabilistic Safety Assessment	
An approach to quantitatively evaluate the risk from hazards associated with a fire. *(see Main Glossary: PRA)*	This quantitative approach consists of fire ignition frequencies, the associated initiating event produced by the ignition, the probability of fire damage from those ignition sources, and the resulting impact on the plant. The term probabilistic safety assessment is another term that can be used interchangeably and correctly with PRA. Typically, the term probabilistic safety assessment is used internationally.
Fire Prevention	
Measures directed toward reducing the likelihood of fire.	Fire prevention is not generally modeled in fire PRA, although it is reflected in fire ignition frequency. Lower fire frequencies could be due, at least in part, to an effective fire prevention program. The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).
Fire Probabilistic Safety Assessment	
(*see Fire PRA)*	The term fire probabilistic safety assessment has the same meaning as fire PRA and is defined under "Fire PRA."
Fire Protection Defense-In-Depth	
The principle of providing multiple and diverse fire protection systems and features.	Fire protection defense-in-depth is modeled explicitly in fire PRA. In particular, fire PRA will credit defense-in-depth fire protection measures and will predict the likelihood that those measures fail to prevent fire-induced damage to plant equipment and cables. The fire protection defense-in-depth objectives, as indicated in Appendix R to 10 CFR Part 50, (Ref. 24), are "(1) to prevent fires from starting; (2) to detect rapidly, control, and extinguish promptly those fires that do occur; and (3) to provide protection for structures, systems, and components important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant." Multiple and diverse fire protection systems and features attain these objectives.
Fire Protection Design Elements	
Any aspect of the fire protection program supported by specific	Fire protection design elements can include active fire protection systems such as sprinkler or smoke detector systems, passive systems such as electrical raceway fire barriers, and programmatic elements.

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design requirements and/or analyses.	The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Protection Feature	
Administrative controls, emergency lighting, fire barriers, fire detection and suppression systems, fire brigade personnel, and other features provided for fire protection purposes.	In a fire PRA, fire protection features would be credited in accident sequences in which a fire endangers stable operation of the plant. Fire protection features are important to risk because they reduce damage due to fire and thus the frequency of accidents with undesired consequences because of fires. The definition provided was based on the definition in Regulatory Guide 1.189 (Ref. 90).
Fire Protection Program	
The integrated effort involving equipment, procedures, and personnel used in carrying out all activities of fire protection.	The ASME/ANS PRA Standard (Ref. 2) states that the fire protection program includes “system and facility design, fire prevention, fire detection, annunciation, confinement, suppression, administrative controls, fire brigade organization, inspection and maintenance, training, quality assurance, and testing.” The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Protection Program Element	
Any specific aspect or provision included as a part of the fire protection program.	As described in the ASME/ANS Standard (Ref. 2), fire protection program elements include “system and facility design, fire prevention, fire detection, annunciation, confinement, suppression, administrative controls, fire brigade organization, inspection and maintenance, training, quality assurance, and testing.” The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Protection System	
Fire detection, notification, and fire suppression systems designed, installed, and maintained in accordance with the applicable National Fire Protection Association codes and standards.	Fire protection systems are systems installed to provide detection, warning, or suppression of fires. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Response Procedure	
A procedure established for operators to respond to a fire.	An example of a fire response procedure is to evacuate the control room when certain environmental conditions are reached due to a control room fire. Specific facilities may have alternate names for the fire response procedures such as fire emergency procedures, pre-fire plans, or emergency response procedures. The fire response procedures also may be embedded within a more general set of emergency operating procedures designed to deal with a range of potential off-normal plant operating states, including fires.

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Fire Risk Analysis	
(see Fire PRA)	The term fire risk analysis has the same meaning as fire PRA and is defined under "Fire PRA."
Fire Safe-Shutdown Analysis	
The deterministic process or method conducted to identify and evaluate the capability of structures, systems, and components necessary to accomplish and maintain safe shutdown conditions in the event of a fire.	Fire safe shutdown analysis is conducted based on a fire scenario in fire PRA and affects the plant response mode. For fire events, safe shutdown are those plant conditions specified in the plant technical specifications as hot standby, hot shutdown, or cold shutdown. The definition provided was based on the definition in Regulatory Guide 1.189 (Ref. 90).
Fire Scenario	
A set of elements that describe a fire event.	A fire scenario includes a description of the fire and any factors affecting it from ignition to suppression. As a result, the fire scenario describes the progression of the fire from ignition to damage in the fire PRA. The ASME/ANS Standard (Ref. 2) states that the elements of a fire scenario include "a physical analysis unit, a source fire location and characteristics, detection and suppression features to be considered, damage targets, and intervening combustibles." The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Suppression	
The process of controlling and ultimately extinguishing fires.	In fire PRA, fire suppression is a process, but successful completion of that process implies fire extinguishment, which represents the termination of the fire itself. An accident sequence caused by the fire may continue beyond extinguishment of the fire. Traditional fire protection definitions refer to fire suppression as controlling and extinguishing fires, which is consistent with the term as applied in fire PRA. Fire suppression can be either manual or automatic. Manual fire suppression is the use of hoses, portable extinguishers, or manually actuated fixed suppression systems by plant personnel. Automatic fire suppression is the use of automatic fixed systems, such as sprinkler, Halon, and CO_2 systems. Manual fire suppression is modeled as a time-dependent activity in fire PRA, occurring at potentially different times in the scenario, in which automatic fixed suppression is modeled as occurring early in the scenario and often can be treated as time-independent.
Fire Suppression System	
Typically, permanently installed fire protection systems provided for the express purpose of suppressing fires.	In a fire PRA, the effectiveness of the fire suppression system is an important consideration, in addition to the system availability and reliability. The ASME/ANS Standard (Ref. 2) states that a fire suppression system "may be either automatically or manually actuated. However, once activated, the system should perform its design function with little or no manual intervention." The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).

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Fire Wrap	
A localized protective covering designed to protect cables, cable raceways, or other equipment from fire-induced damage.	Fire wrap, used to protect against thermal damage, is the common term usually used to denote a type of electronic raceway fire barrier system. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Fire Zone	
1. Subdivisions of a fire area not necessarily bounded by fire rated assemblies. 2. Subdivisions of a fire detection or suppression systems, which provide alarm indications at the central alarm panel.	The term fire zone is not widely used in current fire PRA practice but, when used, can have different meanings. A fire zone may be a loosely defined spatial area such as a partially enclosed space within a larger fire compartment or fire area (per definition (1)). The term also may be used in the more traditional context of a zone of coverage for fixed fire protection features such as fire detection and fire suppression (per definition (2)). The term fire zone may also be encountered in older fire PRAs in which terminology was as yet unsettled. That is, some older fire PRAs may use the term fire zone in the same context that the ASME/ANS Standard uses the term physical analysis unit. The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).
Fire-Resistance Rating	
The time that materials or assemblies have withstood a fire exposure as established in accordance with an approved test procedure appropriate for the structure, building material, or component under consideration.	In a fire PRA, the greater the fire-resistance rating, the longer time to damage is modeled. American Society of Testing and Materials (ASTM) Standard E-119 is the test standard for determining fire resistance. The fire-resistance rating is provided in units of minutes or hours. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Flame Spread Rating	
A relative measurement of the surface burning characteristics of building materials.	The flame spread rating is tested in accordance with NFPA 255, “Standard Method of Test Surface Burning Characteristics of Building Materials” (Ref. 10). The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).
Free of Fire Damage	
The structure, system, or component under consideration remains capable of performing its intended function during and after the postulated fire.	A component free of fire damage in the fire PRA model is given full credit to performing its function. The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).

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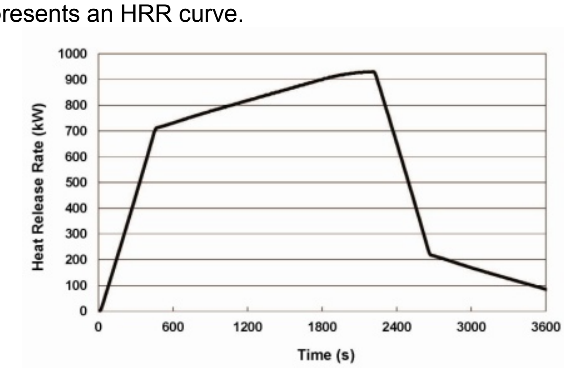
Ground Fault	
A type of short circuit involving an abnormal connection between a conductor and a grounded conducting medium.	NUREG/CR-6850 (Ref. 79) describes a ground fault as being characterized by "an abnormal current surge (fault current) attributable to the lack of any significant circuit burden (i.e., load). A ground fault should trigger over-current protective action for a properly designed circuit." As used in the definition, the grounded conducting medium refers to any conduction path associated with the reference ground of the circuit. This might include structural elements (e.g., tray, conduit, enclosures, metal beams) or intentionally grounded conductors of the circuit (neutral conductor). The term ground fault is used interchangeably and correctly with the term short-to-ground. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Heat Release Rate	
The amount of heat generated by a burning object per unit time.	The heat release rate (HRR) is the key driver in determining the extent of damage in a fire scenario and is usually expressed in units of kW. An example of an HRR can be found in an HRR profile. An HRR profile refers to the behavior of the HRR as a function of time (an HRR versus time plot). For example, a fire with a constant HRR has an intensity that does not change. The American Society of Testing and Materials (ASTM) Standard E176-10a, "Standard Terminology of Fire Standards" (Ref. 3), defines heat release rate as "the thermal energy released per unit time by an item during combustion under specified conditions." The following figure represents an HRR curve.  The definition provided was based on the definition in NUREG/CR-6850 (Ref.79).
High-Energy Arcing Fault	
A high-current, electrical fault that produces an energetic discharge of electrical and thermal energy and may be followed by a fire.	High-energy arcing faults are unique in fire PRA since damage is assumed to occur instantaneously to targets, regardless of the potential presence of a fixed suppression system. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
High-Hazard Fire Source	
A fire source that can lead to fires of a particularly severe and challenging nature.	In a fire PRA, high-hazard fire sources may cause extensive damage, potentially including the failure of structural elements such as steel, which is mapped into failures of equipment. Examples of high-hazard fire sources include catastrophic failure of an oil-filled transformer, an unconfined release of flammable or combustible liquid, leaks from a pressurized system

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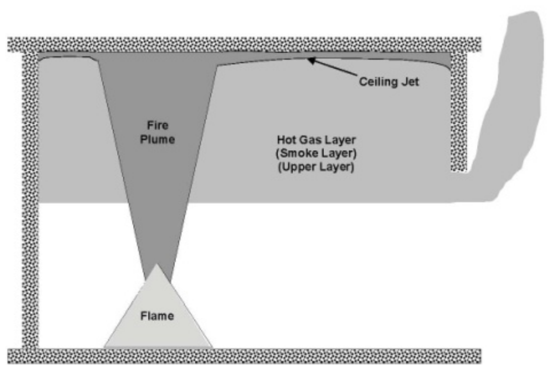
	containing flammable or combustible liquids, and significant releases or leakage of hydrogen or other flammable gases (Ref. 2). The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
High-Low Pressure Interface	
Interface between the reactor coolant system and lower-pressure systems.	In a fire PRA, regulations stipulate that at least one isolation valve at the interface of high- and low-pressure systems must remain closed despite any damage that may be caused by fire.
Hot Gas Layer	
The volume under the ceiling of a fire enclosure where smoke accumulates and high gas temperatures are observed.	Typically, a fire plume will form above a burning object. The fire plume will rise until obstructed by a horizontal surface, such as a ceiling. Upon hitting the ceiling, the hot gases in the fire plume will turn and flow along the ceiling in the form of a ceiling jet. When the ceiling jet gases are blocked by vertical surfaces, such as walls, they will accumulate into a hot gas layer or smoke layer. As more hot gas accumulates in the layer, the interface between the hot gas layer and cooler layer below will continue to drop toward the floor of the enclosure. Hot gas layer is the upper zone in a two-zone fire model formulation.  The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Hot Short	
The condition in which individual conductors of the same or different cables come in contact with each other. At least one of the conductors involved in the shorting is energized, resulting in an impressed voltage or current on the circuit being analyzed.	In a fire PRA, a hot short can cause a spurious operation, which is one possible failure mode considered in the accident sequence model. Hot shorts also can cause misleading instrumentation and indication signals. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Ignition Source	
A piece of equipment or activity that causes a fire.	Ignition source is the first link to an accident sequence caused by fire. A fire started by an ignition source may damage equipment, causing an initiating event, and possibly damaging safety systems required for response.

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	Fixed ignition sources are permanently installed, and transient ignition sources are temporarily located. Examples of transient ignition sources are a welder or grinder being used for hot work. Examples of fixed ignition sources are switchgear cabinets, transformers, pumps, and cables. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Intercable Short Circuit	
Electrical contact between individual conductors in two or more separate cables due to damaged insulation and cable wrapping. *(see Intracable Short Circuit)*	As analyzed in a PRA, an intercable short circuit may lead to any one of several possible conductor fault modes including hot shorts and ground faults. Such faults may disable safety-related systems, cause the spurious operation of plant components, and may lead to or contribute to an accident sequence. An intercable short circuit may be caused by fire-induced damage to grouped electrical cables. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Internal Fire	
A hazard group in which a fire occurs from within the plant that is evaluated in fire PRA.	For fire PRA, the phrase within the plant as used in this definition is any location that lies within the global analysis boundary as defined by the plant partitioning technical element under Part 4 of the ASME/ANS Standard (Ref. 2). Examples of internal fires are fires that occur in the confines of the plant, including any buildings associated with plant operations, the switchyard, transformer yard, and service water supply. Forest fires are classified as external fires.
Internal Hot Short	
A hot short in which both the source conductor and target conductor are in the same multi-conductor cable. *(see Hot Short, Intracable Short Circuit)*	Internal hot shorts have greater probabilities of occurrence than external hot shorts. The term internal hot short can be used interchangeably and correctly with intracable short circuit, which is also referred to as intracable conductor-to-conductor short circuit. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Intervening Combustibles	
Materials that may burn but are not ignition sources.	The fire scenario becomes more extensive in the presence of intervening combustibles. This is because intervening combustibles, located between the ignition source and target, contribute to fire propagation along this path. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Intracable Short Circuit	
Electrical contact between individual conductors in a cable due to damaged insulation between the conductors. (*see Intercable Short Circuit)*	As analyzed in a PRA, intractable short circuits may lead to any of the defined cable and circuit failure modes, including hot shorts and ground faults. Such faults may cause the spurious operation of plant components, disable safety-related systems, and lead to or contribute to an accident sequence. Intracable short circuits may occur because of a fire damaging insulation between the conductors of any multi-conductor cable, or they may occur because of insulation faults. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).

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Limiting Fire Scenario	
Fire scenario(s) in which one or more of the inputs to the fire modeling calculation are varied to the point that particular equipment is failed.	The intent of the limiting fire scenario is to determine that there is a reasonable margin between the expected fire scenario conditions and the point of this failure. Examples of fire modeling inputs that could be varied include heat, release rate, initiation location, or ventilation rate. The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).
Maximum Expected Fire Scenario	
Scenarios that represent the most challenging fire that could be reasonably anticipated for the occupancy type and conditions in the space.	Maximum expected fire scenario is a term for an analysis in the fire modeling track of NFPA 805 and is not specifically related to fire PRA. Maximum expected fire scenarios can be based on industry experience using plant-specific conditions and fire experience (Ref. 11). The definition provided was based on the definition in the NFPA 805 Standard (Ref. 11).
Multiple Spurious Operations	
Concurrent spurious operations of two or more equipment items. *(see Concurrent Hot Shorts)*	Multiple spurious operations may cause multiple equipment failures and complicate operator actions in a fire accident sequence in comparison to single spurious operations. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Natural Ventilation	
The condition in which gas flows into or out of the room because of density differences between the fluids.	Ventilation (supplying fresh air) may cause the fire to burn more intensely, while at the same time potentially removing part of the hot gas layer. Therefore, ventilation may affect the probability of damage to equipment, given a fire in a certain location. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Open Circuit	
A loss of electrical continuity in an electrical circuit, either intentional or unintentional.	In a fire PRA, open circuits will cause the associated electrical equipment to be inoperable. This may increase the probability of system failures and probabilities of relevant accident sequences. Open circuits could result from a loss of conductor continuity or from the triggering of circuit protection devices such as a blown fuse or open circuit breaker, or because of a loss of physical continuity in one or more cable conductors (Ref. 79). The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Passive Fire Barriers	
A fire barrier that provides its protective function while in its normal orientation, without any need to be repositioned.	In a fire PRA, fire barriers impede the spread of fires and limit potential damage to safety equipment, thereby reducing probabilities of fire spread to additional components and the probability of accident sequences. Walls and normally closed fire doors are examples of passive fire barriers. The definition provided was based on the definition in NUREG-1805 (Ref. 60).
Physical Analysis Unit	
A spatial subdivision of the plant on which the fire PRA is based.	In a fire PRA, the physical analysis units are the fundamental spatial element considered as being affected by fires. While the fire PRA will include consideration of fires affecting more than one physical analysis unit at a time (the multicompartment analysis), most fire scenarios

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	are assumed to remain confined to one physical analysis unit. Physical analysis units usually are based on fire areas or fire compartments, but they also may be based on factors such as spatial separation (as opposed to physical barriers), nonrated partitioning elements, and active fire barrier systems (e.g., a water curtain). Since a physical analysis unit substantially contains the effects of a fire, it generally reduces the probability of additional component damage. This term was coined in relation to the fire portion of the ASME/ANS PRA Standard to refer generally to fire compartments, fire zones, and fire areas. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Probability of Nonsuppression	
Probability of failing to suppress a fire before target damage occurs.	In a fire PRA, probability of nonsuppression is used to calculate the probability of target damage (and, consequently, probability of component or system failure), given a fire of a certain intensity in a certain location. Probability of nonsuppression depends on the characteristics of the fire, fire suppression method, and the time available until target damage. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Qualified Cable	
A cable that has been tested and certified as meeting all aspects of IEEE-383 standard including both the equipment qualification and flame spread elements.	The IEEE-383 standard primarily deals with the equipment qualification issues of cable aging and severe accident environmental exposures. The standard also includes a vertical flame spread test. In practice, cables that have been only tested against the flame spread portion of the standard, but have not been subjected to the equipment qualification elements, may be referred to as low flame spread cables, but they would not be considered fully qualified. A cable that does not meet this criterion is referred to as unqualified or nonqualified.
Raceway	
An enclosed channel of metallic or nonmetallic materials designed expressly for holding wires, cables, or bus bars, with additional functions as permitted by code.	In a fire PRA, generally all cables in a raceway are affected equally by the modeled fire. Open cable trays (e.g., ladder style trays) also are referred to as raceways. The ASME/ANS PRA Standard (Ref. 2) states that raceways include, but are not limited to, “rigid metal conduit, rigid nonmetallic conduit, intermediate metal conduit, liquid-tight flexible conduit, flexible metallic tubing, flexible metal conduit, electrical nonmetallic tubing, electrical metallic tubing, underfloor raceways, cellular concrete floor raceways, cellular metal floor raceways, surface raceways, wireways, and busways.” The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Short Circuit	
An abnormal connection (including an arc) of relatively low impedance between two conductors or points of different potential.	With regard to control circuit failures, short circuits could involve a ground fault or hot short. Either may cause disablement or undesired operation of safety-related equipment and contribute to initiation or propagation of an accident sequence. Short circuits also can cause the failure or maloperation of the indication elements of a control circuit, instrument circuits, and power circuits. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).

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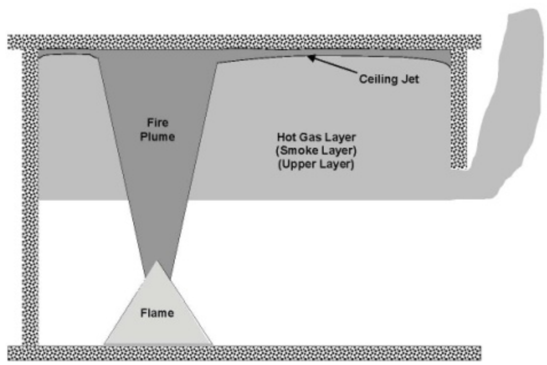
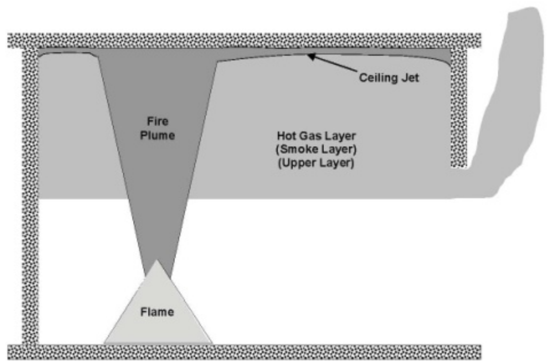
Short-to-Ground	
A type of short circuit involving an abnormal connection between a conductor and a grounded conducting medium.	NUREG/CR-6850 (Ref. 79) describes a ground fault as being characterized by "an abnormal current surge (fault current) attributable to the lack of any significant circuit burden (i.e., load). A ground fault should trigger over-current protective action for a properly designed circuit." As used in the definition, the grounded conducting medium refers to any conduction path associated with the reference ground of the circuit. This might include structural elements (e.g., tray, conduit, enclosures, metal beams) or intentionally grounded conductors of the circuit (neutral conductor). The term short-to-ground is used interchangeably and correctly with the term ground fault. The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Smoke Layer	
The volume under the ceiling of a fire enclosure where smoke accumulates and high gas temperatures are observed. *(see Upper Layer, Hot Gas Layer)*	Typically, a fire plume will form above a burning object. The fire plume will rise until obstructed by a horizontal surface, such as a ceiling. Upon hitting the ceiling, the hot gases in the fire plume will turn and flow along the ceiling in the form of a ceiling jet. When the ceiling jet gases are blocked by vertical surfaces, such as walls, they will accumulate into a hot gas layer or smoke layer. As more hot gas accumulates in the layer, the interface between the hot gas layer and cooler layer below will continue to drop toward the floor of the enclosure. The smoke layer is the upper zone in a two-zone model formulation.  The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Spurious Operation	
The undesired operation of equipment resulting from a fire that could affect the capability to achieve and maintain safe shutdown.	Spurious operation results from a hot short and may result in undesired change of state or disablement of safety-related equipment, thereby resulting in initiation of an accident sequence or damage to a component within the accident sequence. In some cases, ground faults or open circuits also may cause spurious operation, depending on the specific circuit design. The definition provided was based on the definition in the ASME/ANS PRA Standard (Ref. 2).
Transient Combustible	
Combustible materials placed in a temporary location.	In a fire PRA, a transient combustible is one of many potential ignition sources. As discussed in NUREG/CR-6850 (Ref. 79), transient combustibles "are usually associated with (but not limited to) maintenance or modifications involving combustible and flammable liquids, wood and plastic products, waste, scrap, rags, or other combustibles resulting from the work activity." The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).

TERM AND DEFINITION (S)	DISCUSSION
Upper Layer	
The volume under the ceiling of a fire enclosure where smoke accumulates and high gas temperatures are observed. *(see Smoke Layer, Hot Gas Layer)*	Typically, a fire plume will form above a burning object. The fire plume will rise until obstructed by a horizontal surface, such as a ceiling. Upon hitting the ceiling, the hot gases in the fire plume will turn and flow along the ceiling in the form of a ceiling jet. When the ceiling jet gases are blocked by vertical surfaces, such as walls, they will accumulate into a hot gas layer or smoke layer. As more hot gas accumulates in the layer, the interface between the hot gas layer and cooler layer below will continue to drop toward the floor of the enclosure. The smoke layer is the upper zone in a two-zone model formulation.  The definition provided was based on the definition in NUREG/CR-6850 (Ref. 79).
Ventilation Rate	
Amount of air injected or extracted by a mechanical ventilation system into or from a location, respectively.	The ventilation rate is usually measured in cubic meters per second (m^3/sec).
Zone Model	
A type of fire model that provides a method for calculating fire environment conditions in control volumes, or zones, within a space by applying conservation equations and the ideal gas law.	The fundamental idea behind a zone model is that each zone is well-mixed and that all fire environment variables (e.g., temperature, smoke concentration), therefore, are uniform throughout the zone. The variables in each zone change as a function of time and rely on the initial conditions that the user specifies. It is assumed that there is a well-defined boundary separating the two zones, though this boundary may move up or down throughout the simulation. Zone models can easily analyze conditions resulting from fires involving single compartments or compartments with adjacent spaces, and they are often used to compute the hot gas layer temperature, hot gas layer composition, and target heat fluxes. Zone models also are capable of modeling some effects of natural and mechanical ventilation in both horizontal and vertical directions. Smoke production, fire plume dynamics, ceiling jet characteristics, heat transfer, and ventilation flows are all algebraic models embedded within zone models. The definition provided was based on the definition in NUREG-1934 (Ref. 65).
Zone of Influence	
That vicinity of the fire in which fire damage or fire spread to secondary combustibles is possible.	Fire damage or spread may require some time to occur. The zone of influence is associated with the potential for fire damage or fire spread, regardless of the time available. Zone of influence generally does not encompass hot gas layer effects; instead, it focuses on direct radiant heating, plume, and ceiling jet effects. Typically a component is not damaged initially in the fire scenario if it is outside the zone of influence for an ignition source.

PRA TECHNICAL ELEMENTS

Table B-1 provides the technical elements as defined in the ASME PRA Standard for Level 1, Level 2 and Level 3 PRA with the associated discussion. The technical elements are listed alphabetically by level of the PRA and hazard groups.

Table B-1 Technical Elements and Discussion

TECHNICAL ELEMENT	DISCUSSION
Level 1 Internal Events	
Accident Sequence Analysis	The term accident sequence analysis is a technical element in the ASME/ANS PRA Standard whose objectives are to ensure that the response of the plant's systems and operators to an initiating event is reflected in the assessment of CDF and LERF.
Data Analysis	The term data analysis is a Level 1 technical element in the ASME/ANS PRA Standard (Ref. 2) whose objectives are to provide estimates of the parameters used to determine the probabilities of the basic events representing equipment failures and unavailabilities modeled in the PRA.
Human Reliability Analysis	The term human reliability analysis is a Level 1 technical element in the ASME/ANS PRA Standard whose objective is to ensure that the impacts of plant personnel actions are reflected in the risk assessment.
Initiating Event Analysis	The term initiating event analysis is a technical element in the ASME/ANS PRA Standard (Ref.2) whose objective is to identify and quantify events that could lead to core damage.
Large Early Release Frequency Analysis	The term large early release frequency (LERF) analysis is a technical element of Part 2 of the ASME/ANS "Combined Standard: Requirements for Internal Events At-Power PRA." The objectives of the LERF analysis element are to identify and quantify the contributors to large early releases based on the plant-specific core damage scenarios.
Quantification	The term quantification is a technical element in the ASME/ANS Level 1 PRA Standard (Ref. 2) whose objective is to provide an estimate of core damage frequency (and support the quantification of large early release frequency) based on the plant-specific core damage scenarios.
Success Criteria	The term accident success criteria is a technical element in the ASME/ANS PRA Standard whose objectives are to define the plant-specific measures of success and failure that support the other technical elements of the PRA.
Systems Analysis	The term systems analysis is also a technical element in the ASME/ANS PRA Standard (Ref.2) whose objectives are to identify and quantify the causes of failure for each plant system represented in the initiating event analysis and accident sequence analysis.
Level 1 Internal Flood At-Power	
Internal Flood Accident Sequences and Quantification	The term internal flood accident sequences and quantification is a technical element in the ASME/ANS Level 1 PRA Standard (Ref. 2) whose objective is to quantify the core damage frequency and large early release frequency for the internal flood plant response sequences.
Internal Flood Plant Partitioning	The term internal flood plant partitioning is a technical element in the ASME/ANS Level 1 PRA Standard whose objectives are to identify plant areas where internal floods could lead to core damage in such a way that plant-specific physical layouts and separations are accounted for.

Internal Flood Scenarios	The term internal flood scenarios is a technical element in the ASME/ANS Level 1 PRA Standard (Ref. 2) whose objective is to develop a set of internal flood scenarios relating flood source, propagation path(s), and affected equipment.
Internal Flood Source Identification and Characterization	The term internal flood source identification and characterization is a technical element in the ASME/ANS Level 1 PRA Standard (Ref. 2) whose objective is to identify the various sources of floods and equipment spray within the plant, along with the mechanisms resulting in flood or spray from the sources, and a characterization of the flood/spray sources is made.
Internal Flood-Induced Initiating Events	The term internal flood-induced initiating events is a technical element in the ASME/ANS Level 1 PRA Standard (Ref. 2) whose objective is to determine the expected plant response to the selected set of flood scenarios, and an accident sequence from the internal event at power PRA that is reasonably representative of this response is selected for each scenario.
Internal Fire At-Power	
Circuit Failure Analysis	The term circuit failure analysis is a technical element in the ASME/ANS Level 1 PRA Standard (Ref. 2) whose objectives are to treat fire-induced cable failures and their impact on the plant equipment, systems, and functions, and estimate the relative likelihood of various circuit failure modes.
Fire Ignition Frequency	The term fire ignition frequency is a technical element in the ASME/ANS Level 1 Internal Fire PRA Standard (Ref. 2) whose objective is to estimate the frequency of fires (expressed as fire ignitions per reactor-year).
Fire PRA Cable Selection	The term fire probabilistic risk assessment cable selection is a technical element in the ASME/ANS Level 1 Internal Fire PRA Standard (Ref. 2) whose objectives are to identify and locate cables required to support the operation of fire PRA equipment selected and cables whose failure could adversely affect credited systems and functions.
Fire PRA Equipment Selection	The term fire probabilistic risk assessment equipment selection is a technical element in the ASME/ANS Level 1 Internal Fire PRA Standard (Ref. 2) whose objective is to identify the set of plant equipment that will be included in the fire PRA.
Fire PRA Plant Response Model	The term fire probabilistic risk assessment plant response model is a technical element for internal fires in the ASME/ANS PRA Standard (Ref. 2) whose objective is to identify the initiating events that can be caused by a fire event and develop a related accident sequence model; and to depict the logical relationships among equipment failures (both random and fire-induced) and human failure events for core damage frequency and large early release frequency assessment when combined with the initiating event frequencies.
Fire Risk Quantification	The term fire risk quantification is a technical element in the ASME/ANS Level 1 Internal Fire PRA Standard (Ref. 2) whose objective is to quantify and present fire risk results.
Fire Scenario Selection and Analysis	The term fire scenario selection and analysis is a technical element in the ASME/ANS Level 1 Internal Fire PRA Standard (Ref. 2) whose objectives are to select a set of fire scenarios for each unscreened physical analysis unit upon which fire risk estimates will be based, characterize the selected fire scenarios, determine the likelihood and extent of risk-relevant fire damage for each select fire scenario, and examine multicompartment fire scenarios.
Plant Boundary Definition and Partitioning	The term plant boundary definition and partitioning is a technical element in the ASME/ANS PRA Standard (Ref. 2) for internal fire whose objective is to define the physical boundaries of the analysis and divide the various volumes within that boundary into physical analysis units.
Post-Fire Human Reliability Analysis	The term post-fire human reliability analysis is a technical element in the ASME/ANS PRA Standard (Ref. 2) whose objective is to consider the operator actions as needed for safe shutdown, including those called out in the relevant plant fire response procedures.
Qualitative Screening	The term fire probabilistic risk assessment cable selection is a technical element in the ASME/ANS Level 1 Internal PRA Standard whose objective is to identify physical analysis

	units whose potential fire risk contribution can be judged negligible without quantitative analysis
Quantitative Screening	The term fire ignition frequency is a technical element in the ASME/ANS Level 1 Internal Fire PRA Standard (Ref. 2) whose objective is to screen physical analysis units from further consideration based on preliminary estimates of fire risk contribution and using established quantitative screening criteria.
Seismic/Fire Interactions	The term seismic/fire interactions is a technical element in the ASME/ANS Level 1 PRA Standard (Ref. 2) whose objective is to provide a qualitative review of potential interactions between an earthquake and fire that might contribute to plant risk.
Uncertainty and Sensitivity Analyses	The term uncertainty and sensitivity analysis is a technical element in the ASME/ANS Level 1 Internal Fire PRA Standard (Ref. 2) whose objectives are the identification and treatment of uncertainties throughout the Fire PRA process.
Seismic Events	
Probabilistic Seismic Hazard Analysis	The term probabilistic seismic hazard analysis is a technical element for seismic PRA in the ASME/ANS PRA Standard (Ref. 2) whose objective is to estimate the probability or frequency of exceeding different levels of vibratory ground motion.
Seismic Fragility Analysis	The term seismic fragility analysis is a technical element for seismic PRA in the ASME/ANS PRA Standard (Ref. 2) whose objective is to determine the plant-specific failure probabilities of structures, systems, and components as a function of the seismic event intensity level, usually given in peak ground acceleration.
Seismic Plant Response Analysis	The term seismic plant response analysis is a technical element in seismic PRA in the ASME/ANS PRA Standard (Ref. 2) whose objective is to develop a plant response model that addresses the initiating events and other failures resulting from the effects of the seismic hazard that can lead to core damage or large early release. The model usually is based on the internal events, at-power PRA model to incorporate those aspects that are different, because of the seismic hazard's effects, from the corresponding aspects of the at-power, internal events model.
High Winds	
High Wind Fragility Analysis	The term high wind fragility analysis is a technical element for high wind hazards in the ASME/ANS PRA Standard (Ref. 2) whose objective is to identify those structures, systems, and components susceptible to the effects of high winds and to determine their plant-specific failure probabilities as a function of the wind intensity.
High Wind Plant Response Analysis	The term high wind plant response analysis is a technical element for high winds PRA in the ASME/ANS PRA Standard (Ref. 2). The objective is: (1) to modify the internal events of the at-power PRA model to include the effects of high wind events in terms of the initiating events and failures induced, and (2) to exercise the resulting model to obtain quantitative results in terms of core damage frequency and large early release frequency.
High Winds Hazard Analysis	The term high winds hazard analysis is a technical element for high wind hazards in the ASME/ANS PRA Standard (Ref. 2) whose objective is to assess the frequency of occurrence of high wind as a function of intensity on a site-specific basis.
External Floods	
External Flood Fragility Analysis	The term external flood fragility analysis is a technical element for external floods in the ASME/ANS PRA Standard (Ref. 2) whose objective is to identify those structures, systems, and components susceptible to the effects of external floods and to determine their plant-specific failure probabilities as a function of the severity of the external flood.
External Flood Hazard Analysis	The term external flood hazard analysis is a technical element for external floods in the ASME/ANS PRA Standard (Ref. 2) whose objective is to assess the frequency of occurrence of external floods as a function of severity on a site-specific basis.

External Flood Plant Response Model and Quantification	The term external flood plant response model and quantification is a technical element for external floods in the ASME/ANS PRA Standard (Ref. 2) whose objectives are to: • develop an external flood plant response model by modifying the internal events at-power PRA model to include the effects of the external flood in terms of initiating events and failures caused; • quantify this model to provide the conditional core damage probability (CCDP) and conditional large early release probability (CLERP) for each defined external flood plant damage state; • evaluate the unconditional CDF and LERF by integrating the CCDP/CLERP with the frequencies of the plant damage states obtained by combining the external flood hazard analysis and external flood fragility analysis.
Other External Hazards	
External Hazard Analysis	The term external hazard analysis is also a technical element for other external hazards in the ASME/ANS PRA Standard (Ref. 2) whose objective is to assess the frequency of occurrence of the external hazard as a function of intensity on a site-specific basis.
External Hazard Fragility Evaluation/ Analysis	The term external hazard fragility evaluation is also a technical element for other external hazards in the ASME/ANS PRA Standard (Ref. 2) whose objective is to identify those structures, systems, and components susceptible to the effects of the other external hazard and to determine their plant-specific failure probabilities as a function of the intensity of the hazard.
External Hazard Plant Response Model/Analysis	The term external hazard plant response model is a technical element for other external hazards in the ASME/ANS PRA Standard (Ref. 2) whose objective is to develop a plant response model that addresses the initiating events and other failures resulting from the effects of the external hazard that can lead to core damage or large early release. The model is based on the internal events, at-power PRA model to incorporate those aspects that are different, because of the external hazard's effects, from the corresponding aspects of the at-power, internal events model.
Level 2	
Containment Capacity Analysis	The term containment capacity analysis is a technical element of a Level 2 PRA whose objective is to select an analysis method and calculate the ability of the containment to withstand challenges.
Interface Between a Level 2 and Level 3 PRA	The term interface between Level 2 and Level 3 PRA is a technical element of a Level 2 PRA whose objectives are to provide clear traceability of the release category quantification back to the Level 2 analysis, to assure that initiating event information that could affect the Level 3 analysis is communicated, and to assure that all information required for the Level 3 analysis is provided in suitable form.
Level 1-2 Interface	The term level 1-2 interface is a technical element of a Level 2 PRA whose objective is to consolidate or group accident sequences (or individual cutsets) from the Level 1 PRA in a way that reduces the number of unique scenarios for evaluation, but preserves initial and boundary conditions to the analysis of plant response (i.e., plant damage states or equivalent).
Probabilistic Treatment of Event Progression and Source Terms	The term probabilistic treatment of event progression and source terms is a technical element of a Level 2 PRA whose objective is to establish a framework to support the systematic quantification of the potential severe accident sequences evolving from each Level 2 core damage sequence in sufficient detail.
Radiological Source Term Analysis	The term radiological source term analysis is a technical element in the draft Level 2 PRA whose objective is to develop a quantitative basis for associating a unique radiological source term to the environment for each accident progression sequence and release category. The metrics used to define a source term can vary, depending on the objective and intended application of the PRA.

Severe Accident Progression Analysis	The term severe accident progression analysis is a technical element of a Level 2 PRA whose objective is to generate a technical basis, rooted in realistic deterministic analysis for describing the chronology of postulated accident involving significant damage to reactor fuel, quantitatively characterizing thermal and mechanical challenges to engineered barriers to fission product release to the environment, and generating quantitative estimates of radioactive material release to the environment for accident sequences identified as contributors to the frequency of release.
Level 3 PRA	
Atmospheric Transport and Diffusion	The term atmospheric transport and diffusion (ATD) is a technical element of a Level 3 PRA that refers to the process by which material that has been released from containment, moves through and spreads upon release to the atmosphere. The objective of ATD is to model the transport of radioactive material as it travels for many hours in the atmosphere under the meteorological conditions prevailing at and beyond the site that can change in both space and time. ATD models range from simple straight-line, steady-state Gaussian dispersion models that calculate ground-level instantaneous and time-integrated airborne concentrations in the plume, to more sophisticated models that allow terrain-dependent effects and temporal variations in wind speed and atmospheric stability. Probabilistic consequence modeling codes typically include sampling of meteorological data from a site-specific annual data base of hourly weather data to determine appropriately weighted scenarios of plume transport under different weather conditions to provide probabilistic results, model ATD for accident- and site-specific input parameters, accommodate temporal and spatial changes in meteorological conditions, calculate wet and dry deposition of particulate and halogen radionuclides, and document algorithms, assumptions, limitations, and uncertainties.
Dosimetry	The term dosimetry is a technical element in a Level 3 PRA whose objectives are to determine dose by including all applicable dose pathways such as cloudshine, groundshine, skin deposition, inhalation and ingestion; apply the effect of mitigation actions such as shielding; apply recognized dose conversion factors; and document assumptions, limitation and uncertainties associated with dosimetry.
Economic Factors	The term economic factor is a technical element in a Level 3 PRA whose objective is to determine the economic impacts of the release on the surrounding land and the population.
Meteorological Data	The term meteorological data is a technical element of a Level 3 PRA whose objective is to provide valid and representative meteorological data that are input into the atmospheric transport and dispersion codes, which provide the basis for consequences analysis calculations.
Protective Action Parameters and Other Site Data	The term protective action parameters and other site data is a technical element in a Level 3 PRA whose objectives are to model appropriate emergency response actions and protective actions; use appropriate site, local, and regional data; and document site-specific data, emergency response planning modeling, assumptions, limitations, and uncertainties.
Quantification and Reporting	The term quantification and reporting is a technical element of a Level 3 PRA whose objectives are to ensure that the Level 3 model executes properly, proves appropriate results, and is documented in a manner that facilitates risk assessments, PRA applications, upgrades and peer reviews.
Risk Integration	The term risk integration is a technical element of a Level 3 PRA whose objective is to combine the Level 3 analyses with the results from the Level 1-2 analyses to obtain a characterization of the overall risk, including uncertainty.
Transition from the Radionuclide (Radioactive Material) Release to Level 3	The term transition from radioactive material release to Level 3 is a technical element of a Level 3 PRA whose objectives are to provide clear traceability of the release category quantification back to the radioactive material release analysis, to ensure that initiating event information that could affect the Level 3 analysis is communicated, and to ensure that all information required for the Level 3 analysis is provided in suitable form.

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NRC: M. Drouin, M. Gonzalez, J.S Hyslop, S. Lai, D.Stroup
Brokeheaven National Laboratories: J.Lehner, T.Pratt
Sandia National Laboratories: M.Dennis, J.LaChance, T.Wheeler
Energy Math Solutions: Z. Musicki
OMICRON: P.Carrara, W.Thomas

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The policy statement on the "Use of Probabilistic Risk Assessment (PRA) Methods in Nuclear Regulatory Activities" expressed the Commission's belief that the use of PRA technology in U.S. Nuclear Regulatory Commission (NRC) regulatory activities should be increased. Consequently, the NRC carried out numerous risk-informed activities in all areas of NRC regulation. With increased risk-informed activities came the recognition that the agency could enhance regulatory stability and efficiency if it implemented the many potential applications of risk information in a consistent and predictable manner. An essential part of consistent and predictable implementation is the use of consistent terminology to ensure accurate communication and transfer of information. Further, the NRC recognizes that some risk related terms have been used in ambiguous ways by practitioners. The increased development of guidance documents, regulations, and procedures related to risk-informed activities makes the fundamental understanding of these risk-related terms more imperative. Consistent terminology is essential to the appropriate implementation of risk-informed activities and the communication between the NRC and its stakeholders. It allows practitioners to eliminate communication issues and avoid unnecessary discussions that may have been erroneously perceived as technical issues. Therefore, a glossary with agreed-upon definitions of risk-informed related terms is an essential tool for future risk-informed activities.

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