



NUREG-2122

Glossary of Risk Related Terms in Support of Risk-Informed Decision Making

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Outline

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 - Objective
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Introduction

- Policy statement on the “Use of Probabilistic Risk Assessment in Nuclear Regulatory Activities”
 - PRA-Implementation Plan- used until 1999
 - Risk Informed Regulation and Implementation Plan- 2000
 - Risk Informed and Performance Based Plan- 2006
- Risk-information used in: regulation and guidance, licensing and certification, plant oversight and operational experience.
- A glossary of risk terms is needed for appropriate interpretation of terminology to support a risk-informed regulatory structure.
 - Example: “internal events.”



Objectives

- To identify and define terms that are used in risk-informed activities related to commercial NPPs.
- To reduce ambiguity in the definition of terms in order to facilitate risk informed communication.
- To explain individual definitions along with the context, to assure proper context-specific use, whenever more than one definition would be appropriate.
- Intent is not to re-invent definitions, but to acknowledge /address differences in definitions



References

- Over 75 references were used
 - To both identify terms and use as sources for definitions
- Sources were primarily NRC sources except for the use of the national consensus standards and IAEA documents
 - ASME/ANS PRA standard and NFPA 805 were a major source for the definitions
- Types of references
 - National consensus standards
 - NUREG reports
 - Regulatory guides
 - Code of Federal Regulations
 - Standard Review Plan
 - NRC Glossary and website
 - Commission Policy Statements
 - Staff SECY papers

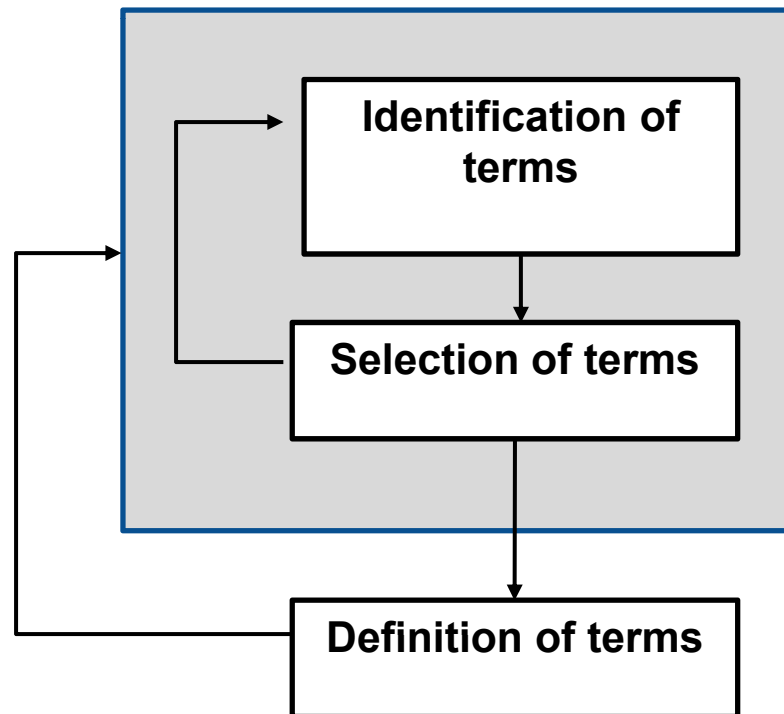


Scope and Limitations

- Sources primarily used limited to internal NRC sources except for national consensus standards
- Glossary contains terms:
 - Used in the three Levels of PRA
 - Common across all hazard groups
 - Used for at-power, low power and shutdown PRA
 - Terms related to risk-informed activities for commercial NPPs
 - Does not include scientific terms that do not take on additional meaning for risk-related activities

Approach

Steps in glossary development



Identification and Selection of Terms

- Step 1: Initial potential candidate
 - Initial list
 - High level screening

Example of initial terms:

- Accident consequences
- At-power
- ATHEANA
- Authority having jurisdiction
- Auxiliary feed water
- Basic event
- Becquerel
- Certified seismic design
- Deterministic
- Diagnosis
- IAEA
- Living PRA
- Mean
- Recovery action
- Technical adequacy



Identification and Selection of Terms

- Step 2: Important for risk communication
 - Related to risk analysis (e.g. consequence, probability, core damage frequency)
 - Role in risk-informed decisionmaking (e.g. safety margins, severe accidents, public health effects)
- Step 3: Risk context specific definition
 - Example: internal hazard
- Step 4: Availability of definition
 - Examples: seismic, aleatory/epistemic, probability



Identification and Selection of Terms

- Step 5: Multiple terms definition
 - Definition review and peer review
- Step 6: Consensually established definition
 - Definition review and peer review
- Step 7: Fundamental to risk communication
 - Frequently used to communicate risk results
 - Examples: large early release frequency, core damage frequency, health effects
 - Used in decisionmaking
 - Examples: deterministic acceptance criteria, high level requirements, consequences



Identification and Selection of Terms

- Step 7 (Cont'd): Fundamental to risk communication
 - Misused terms
 - Examples: Internal events, frequency, probability
- Step 8: Policy implications
 - Examples: Defense in depth, large release frequency

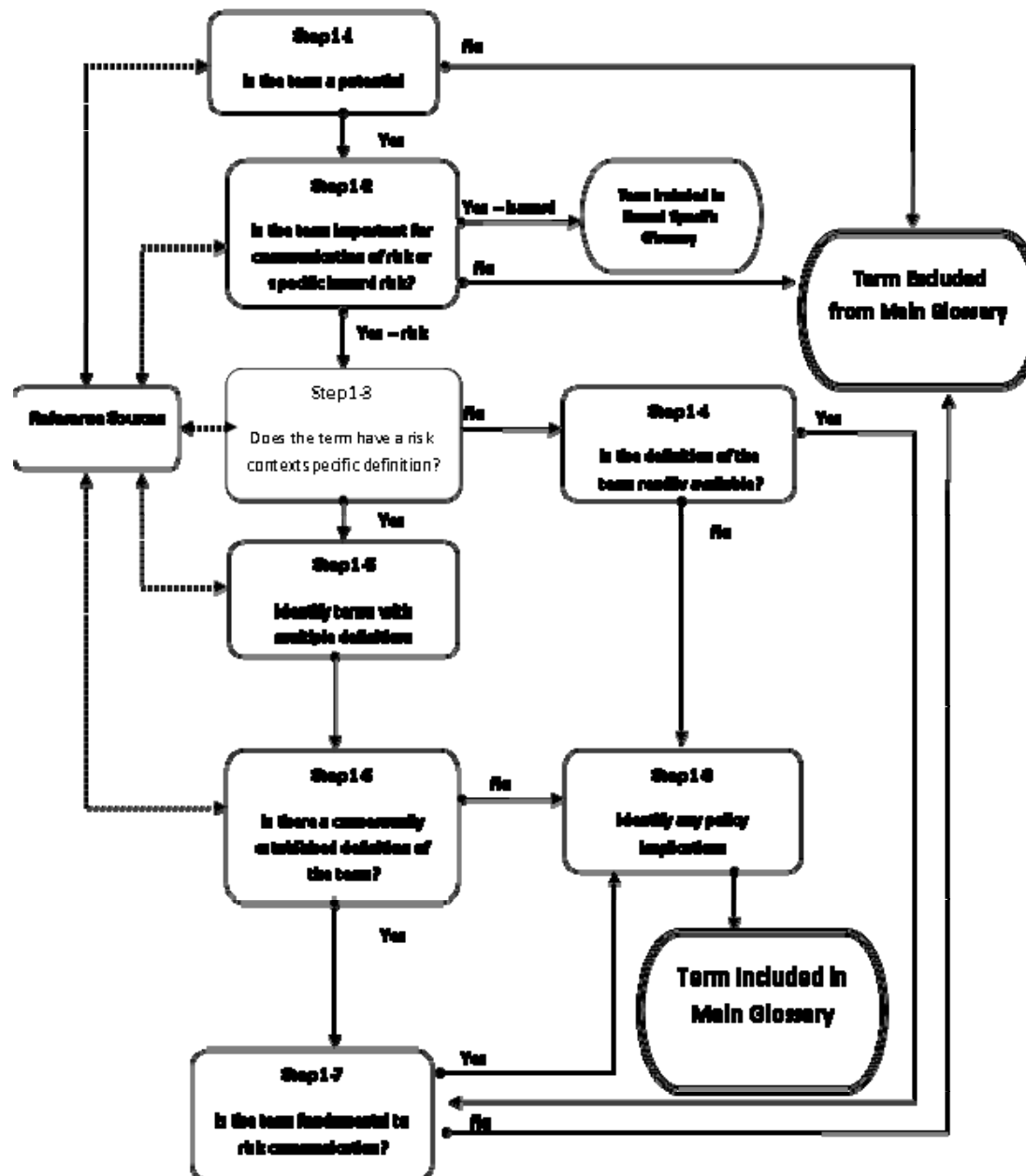
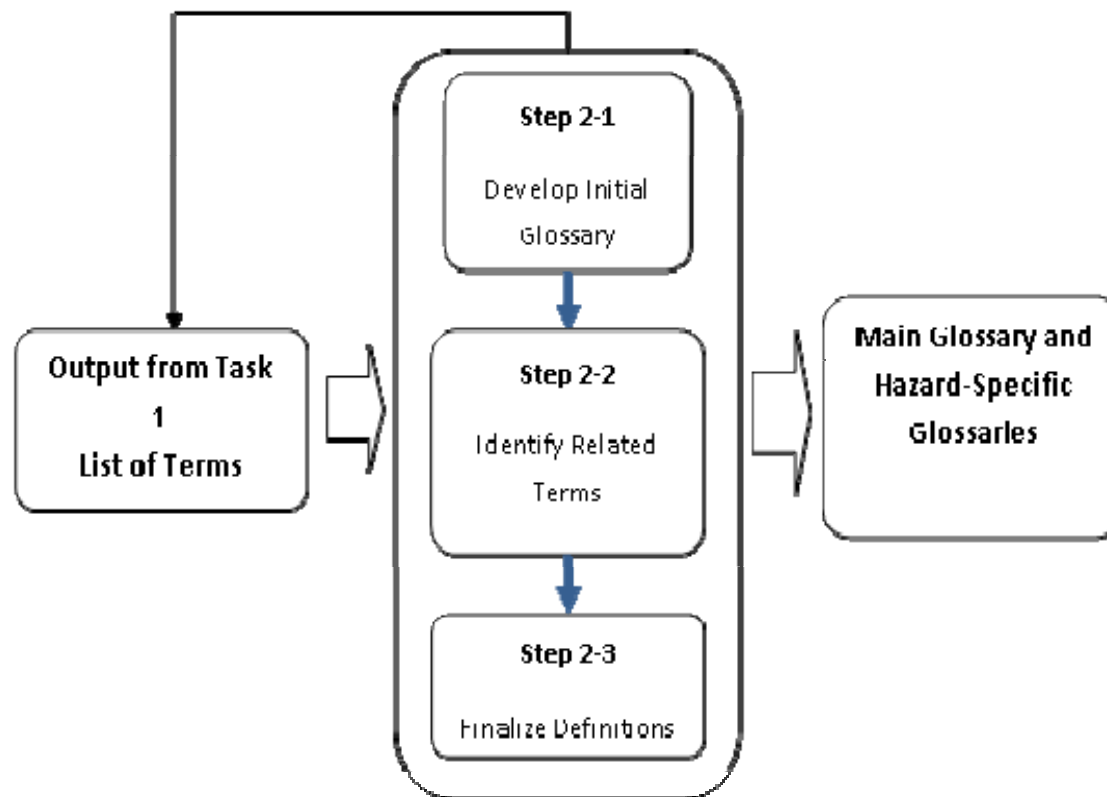


Figure 2-1 Process to Identify terms for glossary

Development of Definitions



Development of Definitions

- Step 1: Develop initial definition
- Step 2: Identify related terms
 - Cross reference of terms
 - Grouping of terms
- Step 3: Finalize definition
 - Modify definition if needed
 - Develop commentary



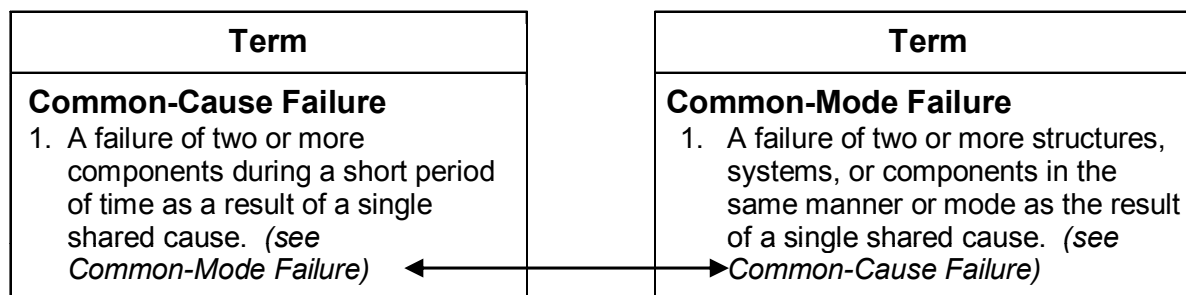
Format and Structure

- Definition provided in plain English
- Comments provided to expand discussion on the term.
- Commentary includes
 - Definition in a risk-context
 - Different definitions of the terms
 - How the term has been and should be used
 - How terms relate to other terms

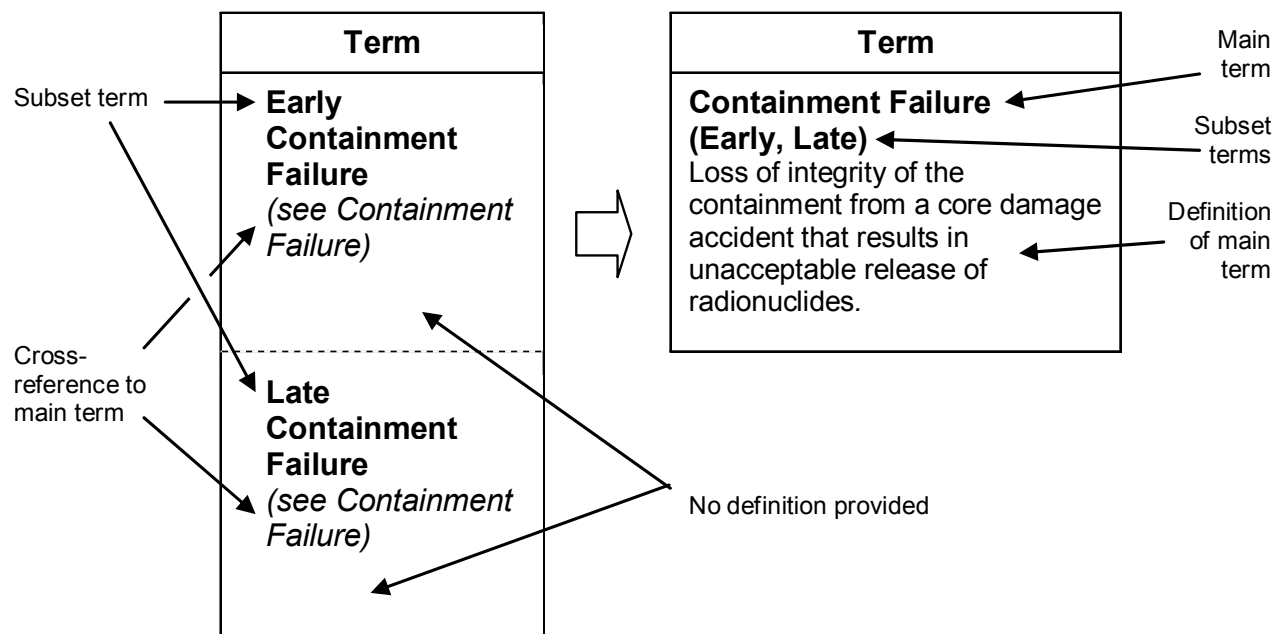
Format and Structure

	Term	Comments
Main term	Common-Cause Failure	
Main term definition	1. A failure of two or more components during a short period of time as a result of a single shared cause. (see <i>Common-Mode Failure</i>)	In a PRA, common-cause failure (CCF) is a special form of dependent failure that reflects such things as a common manufacturer, environment, or maintenance. A CCF is considered if it is within the time period that a system or component is required to operate to successfully perform its function (i.e., mission time). In a PRA, the mission time is typically modeled as 24 hours.
Related term		
Commentary on relationship for the two terms		This term is sometimes used interchangeably (and incorrectly) with common-mode failure (CMF). CCF only accounts for the structures, systems, or components that fail because of the same, single cause and not if they ultimately fail in the same manner, which is CMF.

Crossed referenced terms

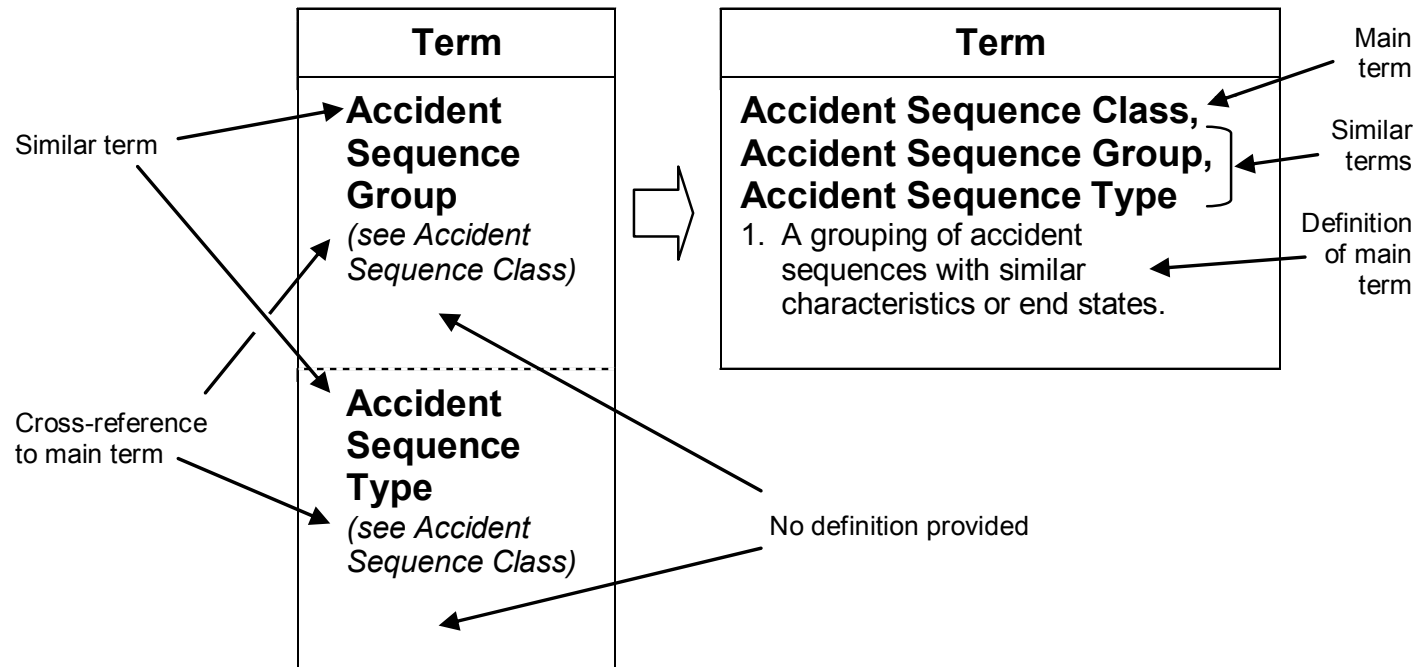


Term subset of another term



Format and Structure

Grouped terms



Appendices

- Hazard Specific Glossaries
 - Internal Fire Glossary
- PRA Technical Elements
 - Level 1 PRA: Internal events, external flood, internal fire, seismic
 - Level 2: All hazards
 - Level 3: All hazards



Examples of Terms from the Glossary

Accident Sequence, Accident Event Sequence, Accident Scenario, Event Scenario, Event Sequence, Event Tree Sequence

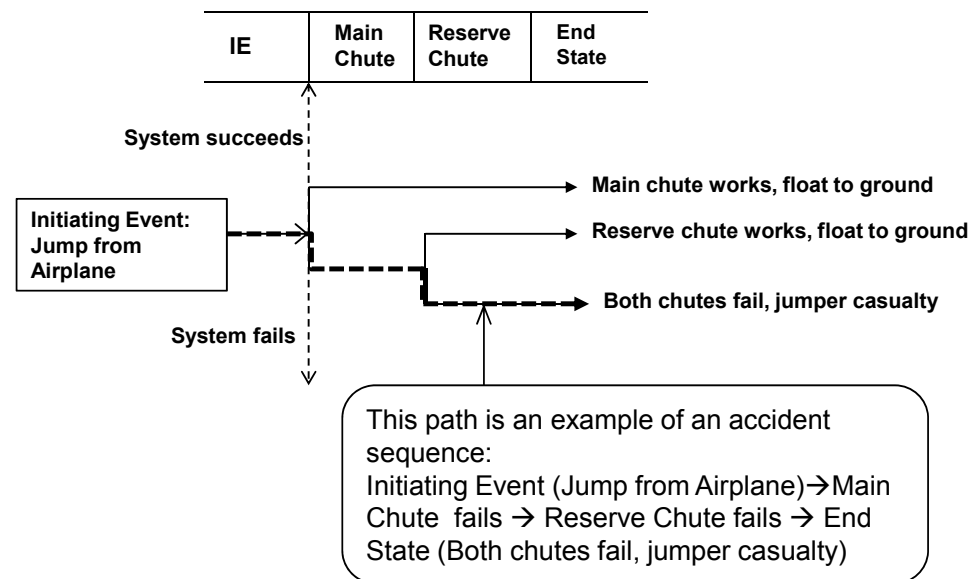
1. A series of events that can lead to undesired consequences. (see *Accident Sequence Analysis, Severe Accident, End State, Event Tree*)

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:



Assumption (Key)

1. 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. 29) 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.”

Key Assumption

(see *Assumption*)

The term key assumption is defined under “Assumption.”

Defense in Depth

1. 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. 34) 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."

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."

Deterministic (Analysis, Approach, Regulation)

1. 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. An 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. 91), 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.

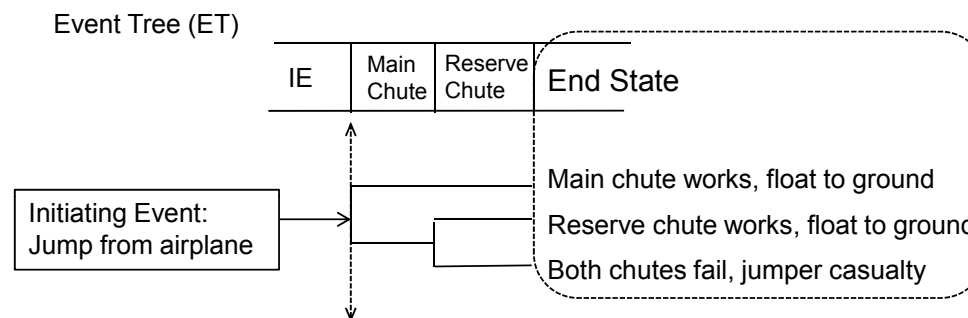
The NRC Web site Glossary (Ref. 34) defines the term deterministic as “consistent with the principles of ‘determinism,’ which hold that specific causes completely and certainly determine effects of all sorts. As applied in nuclear technology, it generally deals with evaluating the safety of a nuclear 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-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.

End State

1. 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).

Frequency (Accident Sequence, Core Damage, Initiating Event, Large Early Release, Large Release, Radioactive Material Release)

1. 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:

- Accident Sequence Frequency: 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*)
- Core Damage Frequency: The sum of the accident sequence frequencies of those accident sequences whose end state is core damage.
- Initiating Event Frequency: The frequency of an event originating from an internal or external hazard that both challenges normal plant operation and requires successful mitigation.
- Large Early Release Frequency: 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.
- Large Release Frequency: 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 health effects, regardless of its timing. (*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.

Groundshine

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|---|--|
| <p>1. Exposure from radioactive material deposited on the ground. (see <i>Exposure, Cloudshine, Inhalation, Ingestion, Skin Deposition</i>)</p> | <p>In a Level 3 PRA, for the consequence calculation, groundshine is one of the five assumed pathways by which an individual can receive doses. The pathways of exposure include: (1) direct external exposure to radioactive material in the plume (principally due to gamma radiation (cloudshine)), (2) exposure from inhalation of radioactive materials in the cloud and resuspended material deposited on the ground, (3) exposure to radioactive material deposited on the ground (groundshine), (4) radioactive material deposited onto the body surfaces (skin deposition), and (5) ingestion from deposited radioactive materials that make their way into the food and water pathway.</p> |
|---|--|

Instantaneous Conditional Probability (Core Damage, Large Early Release)

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|--|--|
| <p>1. Event probability at the specific time the plant is analyzed, given that a prior event has occurred. (see <i>Conditional Probability, Incremental Conditional Probability</i>)</p> | <p>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.</p> <p>The following are other definitions that could describe instantaneous conditional probability:</p> <ul style="list-style-type: none">• 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. <p>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.</p> |
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Level of Detail

1. 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 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.

Parameter

1. 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.

Radioactive Material

- | | |
|--|---|
| <p>1. The substance from the reactor that emits radiation. (See <i>Radionuclide</i>, <i>Fission Product</i>)</p> | <p>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 that 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 is used.</p> |
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State of Knowledge Correlation

- | | |
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| <p>1. A type of dependency that arises when the same data is used to quantify the individual probabilities of two or more basic events. (see <i>Uncertainty</i>)</p> | <p>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.</p> <p>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.</p> <p>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.”</p> |
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Next Steps

- This draft NUREG is pre-decisional and has not received internal staff review
- Draft NUREG-2122 has been issued, in parallel, for internal staff review and comment and public review and comment
 - Because of the nature of the NUREG, and since it does not contain policy information (nor does it contain any analytical or regulatory conclusions) it was decided it would be more efficient to address both staff and public comments together
- A 4 month review and comment period was specified
 - Both internal and public comments are due by September 28, 2012
 - Send all comments to Sandra Lai, RES/DRA/PRB, 301-251-7607, Sandra.Lai@nrc.gov

Next Steps (cont'd)

- In reviewing the glossary, for example:
 - Are the definitions and commentary clear and understandable?
 - Are there terms missing?
 - Is the commentary informative and helpful?
- NUREG-2122 is scheduled to be published by the end of this calendar year
- Should the staff consider an expansion to the glossary and what would be the nature of the expansion?