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DOE STANDARD

DEVELOPMENT OF PROBABILISTIC RISK ASSESSMENTS FOR NUCLEAR SAFETY APPLICATIONS

U.S. Department of Energy
Washington, DC 20585

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Foreword

The Department of Energy (DOE) has taken action to improve the infrastructure and use of risk assessment methodologies for nuclear safety applications. These actions include revising DOE Policy, *Nuclear Safety Policy*, dated February 2011, establishing a Risk Assessment Technical Working Group to provide advice and expert technical guidance in matters concerning risk assessments[1], and developing this Probabilistic Risk Assessment (PRA) Standard which provides criteria and guidance for the development of PRAs at DOE nuclear facilities.

This Standard was developed by a team of DOE and industry risk assessment experts. It establishes a process for planning and performing PRAs in a controlled manner with DOE review and approval. This Standard also provides a set of references to guide project teams in performing PRAs.

DOE technical standards do not establish requirements; however, all or part of the provisions within this Standard may become requirements under the following circumstances: (1) they are explicitly stated to be requirements in an applicable DOE requirements document, or (2) the organization makes a commitment to meet a standard in a contract or in an implementation plan or program plan of a DOE requirements document. Throughout this Standard, the word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and, the word "may" is used to denote permission, but not a requirement or a recommendation. To satisfy this Standard, all applicable "shall" statements need to be met.

Comments in the form of recommendations, pertinent data, and lessons learned that may improve this document should be sent to:

James O'Brien, Director
Office of Nuclear Safety, HS-30
U.S. Department of Energy
19901 Germantown Road
Germantown, MD 20874
Phone: (301) 903-1408, Facsimile: (301) 903-6172
Email: james.o'brien@hq.doe.gov

[1] DOE has chosen to use the term "probabilistic risk assessment" in this Standard to cover quantitative risk assessments where frequency and consequence are evaluated in an integrated manner.

1. INTRODUCTION

The Department of Energy (DOE) has taken action to improve the infrastructure and use of risk assessment methodologies for nuclear safety applications. These actions include revising DOE Policy (P) 420.1, *Nuclear Safety Policy*, dated February 2011, establishing a Risk Assessment Technical Working Group to provide advice expert technical guidance in the matters concerning risk assessments,[1] and developing this Probabilistic Risk Assessment (PRA) Standard which provides criteria and guidance for the development of PRAs at DOE nuclear facilities.

The purpose of this Standard is to provide a consistent approach and process for planning, executing, and using PRAs in nuclear safety applications. This Standard is a process standard that builds upon and refers to recognized industry standards and guidance for details on how specific aspects of the process may be implemented. This Standard provides a set of references to guide project teams in the performance of probabilistic risk assessments.

The DOE Nuclear Safety Policy supports the use of quantitative and probabilistic risk assessment methodologies to supplement qualitative/deterministic methods when such use is supported by industry practices and risk data. DOE's review has indicated that such methodologies may be useful in the following areas (discussed more fully in Section 5 of this Standard):

- Aiding in the evaluation of alternatives that comply with DOE nuclear safety requirements;
- Supporting the unreviewed safety question (USQ) process;
- Supplementing traditional safety assessment methods; and,
- Evaluating changes to DOE safety requirements.

In general, application of PRA methodologies can enhance the quality of hazard and risk analyses and the decisions that are made based on these analyses.

2. APPLICABILITY AND SCOPE

This Standard was developed to support the use of PRAs in nuclear safety applications. It is based on standards, guides, and best practices used in the chemical, nuclear, and aerospace industries. PRAs developed using this Standard can be used to complement qualitative/deterministic methods for developing hazards assessments, hazard controls, and safety management programs.

This Standard also addresses the use of PRAs to assist in developing a documented safety analysis (DSA) required by 10 C.F.R. Part 830, *Nuclear Safety Management.* This Standard can also be used to support risk-informed decision making.

This Standard is not intended for analysis that simply employs a subset of PRA techniques, such as event-tree or fault-tree analysis, related to DOE nuclear facilities. The scope, complexity, and intended use of such analyses and calculations should be considered in determining the applicability of this Standard. The Standard should be applied in complex analyses where the

[1] DOE has chosen to use the term "probabilistic risk assessment" in this standard to cover quantitative risk assessments where frequency and consequences are evaluated in an integrated manner.

results are used as a significant input to decisions regarding the selection of or adequacy of safety controls, or whether to screen events and scenarios from further safety analysis.

This Standard is not intended for use in evaluating the safety of nuclear explosive operations. Ensuring safety for nuclear explosive operations conducted by DOE, the National Nuclear Security Administration (NNSA) and their respective contractors is the subject of a separate, specialized set of directives and standards (e.g., DOE Order 452.2D, *Nuclear Explosive Safety,* dated April 2009).

Throughout this Standard, the word "shall" is used to denote a requirement; the word "should" is used to denote a recommendation; and, the word "may" is used to denote permission, but not a requirement or a recommendation. To satisfy this Standard, all applicable "shall" statements need to be met.

3. OVERVIEW OF STANDARD

Section 4 of this Standard identifies general elements required for the development and use of PRAs. Section 5 identifies potential uses of PRA in DOE nuclear safety applications and describes considerations for developing a PRA. Appendix A contains a glossary of terms used in this Standard.

Appendix B is designed around three tables of industry reference standards. Table B-1 provides industry standards that are directly applicable to implementing the various sections of the standard. Table B-2 is arranged by topical area and provides industry standards that apply to specific PRA processes (e.g., peer reviews) and technical subject matter (e.g., fault tree analysis, human reliability analysis). Table B-3 provides an example of the contents of a comprehensive PRA standard using a table of contents of the PRA performance standard for advanced non-light water reactor (LWR) PRAs.[2]

4. DEVELOPMENT OF PRAs

This section provides the criteria and guidance for developing a PRA. Industry standard references supporting these key elements are discussed in Table B-1 of Appendix B.

4.1 PRA Plan

Prior to developing a new PRA for nuclear safety applications, the responsible project manager shall develop a plan for the application of PRA techniques to the needs of the project. The PRA plan shall address the following elements:

- Overview;
- PRA approach;
- Anticipated outcomes and intended use of information; and,
- PRA technical adequacy and peer review approach.

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[2] ASME/ANS-RA-S-1.4-2013, "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants," ASME, New York, NY and ANS, La Grange Park, IL, has been approved for trial use in pilot applications and is scheduled for publication in December 2013; it closely parallels the standard for LWRs (ASME/ANS RA-Sb-2013).

These main elements, along with associated subtopics, are discussed in the following sections.

4.1.1 Overview

4.1.1.1 Statement of Issue

The PRA plan provides a statement of the issue that has brought about the need to apply PRA techniques. The PRA plan shall address the topics in subsections 4.1.1.2 through 4.1.1.5 below.

4.1.1.2 Purpose, Objectives, and Scope

The purpose section describes why the PRA is being performed. The objectives section describes decisions to be supported or needs to be met by the PRA. The scope section describes the boundaries of the systems and the activities to be analyzed, including as applicable:

- Facility structures, systems, and components (SSCs) and operating states;
- Internal events and hazards;
- External events and hazards; and,
- Accident phenomena and progression.

The PRA's scope may be narrow or broad depending on the application. The PRA plan provides the rationale for the scope of the intended application.

4.1.1.3 Principal Assumptions, Limitations, and Methods and Models

The PRA plan describes the principal assumptions (explicit and implicit) upon which the PRA's methods and models are based, the rationale for the assumptions, the selection of methods and models, and any limitations of the PRA and on the use of the PRA's results.

4.1.1.4 Relationship to the Safety Basis Documents

The relationship between the anticipated PRA results and the nuclear facility safety basis and safety design basis documents (e.g., the DSA and Preliminary DSA) shall be described. The PRA plan describes the process used to identify: (a) the key PRA assumptions which require protection by appropriate mechanisms; (b) the use of PRA results to inform selection of safety controls to be included in the safety basis; and (c) applicability of the USQ process relative to maintaining the PRA (see Section 5 of this Standard).

4.1.1.5 Risk Metrics

The PRA plan describes the risk metrics and risk significance criteria to be used in carrying out the PRA and provides a technical rationale for these criteria. Example metrics include the frequency of exceeding specified design or radiological dose limits, probability of exceeding established safety criteria, and individual risk from radiological or chemical exposures (see definition in Appendix A of this Standard). The manner in which the selected risk metrics and the supporting quantitative risk analyses will complement insights gained through qualitative/deterministic processes[3] (to support risk-informed decisions) shall be documented in the PRA plan and in the discussion of PRA results (see also Section 4.4 of this Standard).

4.1.2 PRA Approach

Several methodologies (methods and models) for performing a PRA have been developed and are described in industry guides and standards (see Appendix B of this Standard). Each particular methodology offers specialized schemes and tools for analyzing facilities or processes. The methodology that is adopted should be systematic and consistent with PRA practice.

The PRA approach shall be described in the plan by addressing the topics below.

4.1.2.1 Detailed Assumptions

Based upon the principal assumptions (see Section 4.1.1.3 of this Standard), the plan identifies detailed assumptions that influence the strategies and the methods or models that form the basis of the PRA approach to the extent that such assumptions can be identified in the PRA planning phase.

4.1.2.2 Data Parameter Estimation and Analysis

The plan defines the process for estimation of data parameters to be used in the analysis, including the development of data parameter estimates derived from facility-specific and generic data sources, and, consistent with established industry standards, acquired through collection of service experience data, expert elicitation, and expert judgment.

Typical data parameters of interest include:

- Initiating event frequencies and uncertainties;
- Component failure rates or failure probabilities and uncertainties;
- Common cause failure parameters and uncertainties;
- Human error rates and uncertainties; and,
- Event or phenomena (e.g., gas ignition) probabilities.

The plan describes the process of collecting and analyzing information in order to estimate data parameters selected for the PRA models, including sources of information used to obtain estimates of the PRA data parameters (examples of which are listed above) of various events. The approach to characterizing and quantifying uncertainties in the estimates of data parameters is described, including the method for combining information from different sources.

The plan describes the process for data analysis, parameter estimation, and sensitivity analysis. The plan also addresses the strategy for any database development, including the service data collection process, identification of data sources, methods for combining information from different sources (e.g., Bayes' methods), and characterization of uncertainty in the parameter estimates.

[3] See DOE P 420.1 and DOE/HS-0006, *Technical Basis for U.S. Department of Energy Nuclear Safety Policy 420.1*, July 2011.

4.1.2.3 Methodology Description

The plan describes the methodology to be used for developing the PRA. The plan identifies applicable industry standards or guides for the performance of the PRA and provides the technical rationale for their use in terms of relevance, applicability, completeness, and level of detail. An example of PRA methodology elements for non-light water reactors is provided in Section 3 of Table B-3 of Appendix B in this Standard.[4]

4.1.2.4 Schedule and Resources

The plan describes the schedule and resource requirements necessary for the development, conduct, and peer review of the PRA**.**

4.1.3 Anticipated Outcomes and Intended Use of Information

The PRA plan shall address the topics in subsections 4.1.3.1, 4.1.3.2, and 4.1.3.3 below.

4.1.3.1 Outcomes

The plan describes the results to be produced and expectations for documentation of analysis and results. Further sample guidance can be found on this topic in the ASME/ANS Standards for PRA (see Appendix B).

4.1.3.2 Interpretation of Results

The plan describes how the PRA risk metric results will be evaluated and compared with established risk significance criteria in order to support risk-informed decisions. This stage of the process includes a delineation of the principal contributors to risk and an evaluation of uncertainties, including a quantification of uncertainty in the estimate of risk and a sensitivity analysis of sources of uncertainty that are not quantified.

4.1.3.3 Impact on Safety Basis

The plan describes how the results of the PRA will interface with the existing safety basis or be included in the Safety Design Strategy for new facilities (required by DOE Standard 1189-2008, *Integration of Safety into the Design Process*). The plan also describes the approach for balancing and reconciling insights derived from deterministic and probabilistic inputs to risk-informed decision making to the extent possible at the planning stage.

[4] These methodology elements are similar to those used in light water reactor (LWR) PRA but are not dependent on LWR risk metrics such as core damage frequency and as such are more relevant to DOE non-reactor facilities.

4.1.4 Technical Adequacy[5] and Peer Review Approach

The plan shall describe the approach for assuring the technical adequacy of PRA development and provide the basis for the approach. The approach to technical adequacy shall include:

- Use of qualified personnel adequately trained in applicable PRA and DOE safety subject matter,
- Completeness and transparency of documentation,
- Use of appropriate PRA methods and procedures,
- PRA model/facility configuration control and performance monitoring,
- Applicable DOE quality assurance (QA) requirements, and
- PRA technical and peer reviews.

4.1.4.1 PRA Team and Review Personnel

The plan describes the necessary disciplines and qualifications of the team to perform the PRA. The team shall include personnel experienced in DOE's nuclear safety process and requirements. The approval authority for the PRA plan shall assign qualified personnel to review the plan for technical adequacy.

4.1.4.2 Completeness and Transparency of Documentation

The PRA shall be documented in a manner that facilitates its application, upgrade, and peer review. The PRA shall provide objective evidence that the requirements of this Standard, and any supporting PRA performance standards, have been met.

4.1.4.3 Procedures

The plan describes guides and procedures employed to ensure consistency and coherence among tasks performed by different analysts.

4.1.4.4 Model/Facility Configuration Control and Performance Monitoring

The plan describes the approach and procedures for assuring adequate configuration control of the PRA model for consistency with the facility design and the actual as-built facility, both during initial performance of the PRA and its use in any ongoing risk management programs. This includes a process for monitoring system performance and verification of PRA inputs to ensure their validity during the duration of the PRA application.

[5] The term "technical adequacy" is used in the field of PRA to refer to strategies for technical quality that parallel those employed in quality assurance programs. These requirements for PRA technical adequacy are not to be confused with "quality assurance" as defined in DOE QA requirements. Any changes made to DOE facilities arising from the PRA are also required to satisfy DOE quality assurance requirements. In addition, the PRA will need to employ applicable QA procedures as part of its approach to PRA technical adequacy, based on the scope and applications of the PRA.

4.1.4.5 Applicable DOE QA Requirements

The plan identifies applicable DOE QA requirements (including, where appropriate, DOE requirements for QA records and audits, the use of verified computer programs, document logs, corrective action programs, and the use of procedures) and describes how they will be met.

4.1.4.6 Technical and Peer Reviews

The plan describes the process used to perform internal team checks and technical reviews. The plan also describes how personnel familiar with the facility will review assumptions regarding the design, operation and maintenance of the facility used in the development of PRA models and data analysis.

The plan describes the peer review process, identifies applicable standards and guides used to perform the peer review, and provides the rationale for the approach. The peer review process should be commensurate with the PRA's complexity and importance to safety. The PRA plan should identify whether peer reviews will be conducted at intermediate stages during development and conduct of the PRA. The scope of peer review may range from a single subject matter expert to a formal external review (see Section 4.5 of this Standard), depending on the scope, complexity and intended use of the PRA.

The peer review team's collective experience and expertise shall encompass the technical subject matter of the PRA and the design, maintenance, and operation of the type of facility being assessed.

4.1.5 Plan Approval

The plan shall be reviewed against this Standard and approved by DOE. The level of DOE review and approval should be commensurate with the purposes and uses of the PRA. For nuclear safety applications related to the development of DSAs, the DOE approval authority for the PRA plan should be the DOE approval authority for the DSA.

4.2 Performance

The PRA shall be performed in accordance with the PRA plan. In particular, appropriate industry standards, guides, and practices shall be implemented consistent with the PRA plan. Substantive (i.e., non-administrative) changes identified to be necessary during implementation of the PRA plan shall be documented and approved by DOE.

4.3 Completeness Criteria

PRA models shall be developed in a manner that meets the following completeness criteria:

(a) Models identify and quantify the significant contributors to risk (see definition of risk significance provided in Appendix A, *Glossary*, of this Standard) according to the risk metrics and scope selected.
(b) Models provide a comprehensive treatment of dependencies that are necessary and sufficient to provide a realistic assessment of risk levels and to identify the significant contributors to risk. These dependencies should include physical, functional and human-

caused dependencies that may increase or significantly influence the frequency and probability of multiple failures or unavailabilities.

(c) Models use industry-accepted methods for identifying and quantifying the risk of common cause failures that may adversely impact the reliability of redundant safety systems.

(d) Models shall not screen out events based solely on assessments of low frequency or low probability without consideration of the consequences of the event (to establish the event’s overall risk significance). The PRA may screen out events only when it can be shown that inclusion of the event would not impact the characterization of risk-significant accident sequences or basic events. The rationale supporting any screening shall be documented.

4.4 Results and Documentation

The first several elements listed below are the same as those documented in the PRA plan, but will need to be updated to reflect experience gained during performance of the PRA. The following key elements of the PRA shall be documented in a final report submitted to DOE:

(a) The project’s purpose and objective.
(b) Evidence that the requirements in this Standard and its supporting standards have been met; justification for any requirements not met.
(c) A description of whether and how the PRA results meet the PRA objectives.
(d) A clear and concise tabulation of known limitations and constraints associated with the analysis.
(e) A clear and concise tabulation of the assumptions used, especially with respect to mission success criteria.
(f) Delineation of, and justification for, risk metrics and significance criteria used to interpret the results.
(g) Justification of completeness in identifying and screening out of risk contributors (i.e., justification that screened out facility operating states, hazards groups, initiating events, accident sequences, human actions, and failure modes, etc.) do not individually or collectively make a significant contribution to risk.
(h) Identification of data sources and data analysis methods, including the approach for the integration of estimates from different sources and for characterization and quantification of uncertainties.
(i) Identification of significant contributors to analyzed risk metrics, including facility operating states, hazards, initiating events, accident sequences, end states, basic events, and human actions.
(j) Characterization and quantification of uncertainties (including parameter uncertainties, uncertainties in phenomena and human performance, and modeling uncertainties) in the evaluation of analyzed risk metrics, and the results of sensitivity studies to evaluate sources of uncertainty that are not quantified.
(k) Risk insights relative to deterministic analyses contained in the relevant DSA.
(l) Results of actions taken to ensure technical adequacy, and steps necessary to protect assumptions used in the PRA.
(m) Disposition and resolution of peer review comments.
(n) Results and conclusions.

4.5 Assuring Technical Adequacy

The approach to assuring technical adequacy of the PRA is documented as part of the PRA plan (see Section 4.1.4 of this Standard). If the approach to technical adequacy changes during the course of performing the PRA, those changes shall be documented, justified, and approved (same approval authority as plan approval per Section 4.1.5 of this Standard) prior to implementation.

4.5.1 Implementing Technical Adequacy Requirements

Implementing technical adequacy requirements for the PRA shall address the approach defined by the PRA plan, including:

(a) Use of qualified personnel adequately trained in the subject matter of the PRA and applicable DOE requirements.
(b) Completeness and transparency of documentation.
(c) Use of appropriate methods and procedures.
(d) Model/facility configuration control and performance monitoring.
(e) Applicable DOE QA requirements.
(f) Technical and peer reviews.

4.5.2 Performing a Peer Review

The peer review process should be commensurate with complexity of the analysis and importance to safety of the expected PRA results. In the simplest case, peer review may entail an independent review by a single qualified subject matter expert. More complex facilities or analyses, especially those dealing with large potential radiological exposures or releases, warrant more comprehensive peer reviews.

4.5.2.1 Peer Review Process

Where the detailed peer review process is warranted and employed, the peer review shall:

(a) Use a documented peer review process.
(b) List the review topics to ensure completeness, consistency, and uniformity.
(c) Review the appropriateness of the PRA model.
(d) Review assumptions and assess validity and appropriateness.
(e) Ensure that data sources are documented and appropriately used.
(f) Review treatment and propagation of uncertainties.
(g) Review whether the PRA appropriately represents facility design and operations.
(h) Review the use of industry standards.
(i) Evaluate the manner in which the insights gained through the PRA are integrated with and/or complement the results of DSA deterministic analyses.
(j) Review the process used to ensure that technical adequacy requirements were met.
(k) Review results of each PRA technical element for reasonableness.
(l) Review PRA maintenance, update and configuration control processes.

4.5.2.2 Peer Review Team

The peer review team shall be independent of the user organization (i.e., with no conflicts of interest that could affect objectivity).

The review team shall include personnel who:

(a) Have collective expertise and experience that encompasses the technical subject matter of the PRA,
(b) Are knowledgeable of the peer review process,
(c) Are knowledgeable of the DOE's nuclear safety processes and requirements, and
(d) Are knowledgeable of the design, operation and maintenance of the facility being analyzed.

4.5.2.3 Peer Review Results

Results of the peer review shall be documented in a report that addresses the following elements:

(a) The peer review process.
(b) The scope of the peer review (i.e., what was reviewed by the peer review team).
(c) The findings on technical adequacy of the PRA.
(d) A discussion of where the PRA does not meet desired characteristics and attributes.
(e) The assessment of the significance of any vulnerabilities and deficiencies identified.
(f) The qualifications of individual team members and discussion of why the team's collective expertise was adequate for the review.
(g) A discussion of the independence of the peer review team and freedom of individual members from conflicts of interest.

5. USES OF PRAs IN DOE NUCLEAR SAFETY APPLICATIONS

DOE P 420.1 allows the use of PRA when employed to supplement DOE's qualitative/ deterministic processes and supported by industry practices and availability of risk data.

In determining whether a facility-specific PRA should be pursued, the following are relevant considerations:

- Purpose for the PRA (risk-informed decisions, complex accident phenomena and progression, surveillance and maintenance of safety SSCs, equipment performance and reliability, etc.), scope of the PRA (full facility vs. partial, accident frequencies and/or risk consequence analysis), and availability of data to support the PRA;
- Complexity of facility processes (number and complexity of SSCs, scope of facility functions and operations), and relevant phenomena (such as fires);
- Magnitude of unmitigated dose consequences; and,
- Programmatic importance of the facility (mission critical), and facility design life-cycle stage (new, existing, major modifications).

DOE has determined that PRA insights may be used to supplement traditional analytic approaches; examples are provided below.

5.1 Evaluating Alternative Compliance Approaches

Insights resulting from PRAs may assist decision makers in evaluating alternative courses of action, each of which comply with DOE's nuclear safety requirements for design, operations, and decommissioning. For example, PRAs may be used to inform the selection of safety SSCs and specific administrative controls.

5.2 Supporting the USQ Process

The USQ process requires evaluation of proposed facility changes and the significance of new safety information. Results from a PRA may support USQ determination, assessment of a potential inadequacy of the safety analysis, preparation of a justification for continued operation, evaluation of safety margins, and selection of compensatory measures.

5.3 Supplementing Traditional Safety Methods

A PRA methodology can augment existing DOE safety assessment methods by (a) providing input to the definition and selection of design basis events; (b) prioritizing efforts for addressing safety issues, based on risk information;(c) assessing uncertainties in semi-quantitative analyses; (d) evaluating lessons learned and operating experience, exposing and evaluating sources of uncertainty in safety analyses; and, (e) testing the sensitivity of analytical results against key assumptions. PRA results can enhance DOE decisions on defense-in-depth by providing information on the importance of each control making up the defense-in-depth strategy and by providing risk insights needed to balance prevention and mitigation strategies. In addition, PRAs can inform the design process, especially for complex, high-hazard facilities.

5.4 DOE Evaluation of Changes to Safety Requirements

PRA results can be used to support risk-informed decision making for DOE safety requirements in the following areas:

- Proposed rulemaking to impose new requirements directed at improving safety;
- Proposed orders or rules directed at increasing effectiveness, or furthering the Department's strategic goals other than safety, but which raise safety questions; and,
- Proposed exemptions or changes to existing orders or rules that might cause increases in risk.

Appendix A
Glossary

The following is a glossary of risk assessment terms used at DOE and in general industry. The source of the definition is noted. Different definitions can be found from different sources; however, these definitions are the most appropriate to DOE nuclear safety applications.

Term	Definition and Source
Deterministic Analysis	An analysis that specifies and applies a set of conservative deterministic rules and requirements for the design and operation of facilities or for the planning and conduct of activities. This conservative approach provides a way of compensating for uncertainties in the performance of equipment and the performance of personnel, by providing a large safety margin. [adapted from the IAEA Safety Standard GSR Part4, *Safety Assessment for Facilities and Activities*, Section 4.54]
Probabilistic Risk Assessment (PRA)	A qualitative and quantitative assessment of the risk associated with plant or facility operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as *release category frequency* and its effects on the health of the public [also referred to as a probabilistic safety assessment (PSA) or quantitative risk assessment (QRA)]. [adapted from ANS/ASME JCNRM RA-S-1.4, *Advanced Non LWR PRA Standard*, 2013 (hereafter ASME/ANS RA-S-1.4-2013)]
PRA Application	A documented analysis based in part or whole on a plant or facility-specific PRA that is used to assist in decision making with regard to the design, licensing, procurement, construction, operation, or maintenance of a nuclear facility. [adapted from ASME/ANS RA-S-1.4-2013]
PRA Maintenance	The update of the PRA models to reflect plant or facility changes such as modifications, procedure changes, or facility performance (data). [adapted from ASME/ANS-RA-Sb-2013, Addenda to ASME/ANS RA-S–2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications]
PRA Upgrade	The incorporation into a PRA model of a new methodology or significant 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. [adapted from ASME/ANS RA-S-1.4-2013]
Probabilistic Method	A technique which uses distributions of parameters (including uncertainty and randomness) to perform an analysis. Results are expressed in terms of probabilistic distributions, which quantify uncertainty. [from DOE-HBK-1188, *Glossary of Environment, Safety and Health Terms*]
Quantitative Risk Assessment (QRA)	See PRA
Risk	Frequency and consequences of an event, as expressed by the "risk triplet" that answers the following three questions: (1) What can go wrong? (2) How likely is it? and (3) What are the consequences if it occurs? [adapted from ASME/ANS RA-S-1.4-2013]

Term	Definition and Source
Risk Assessment Tools/Techniques	Analytical methodologies, approaches, representations, and criteria, including computer-based techniques that may be used in a risk assessment activity. Examples include failure modes and effects analyses, fault trees, event trees, risk bins, mathematical models for consequence estimation, complementary cumulative distribution functions, and risk curves. [from DOE Information Notice, June 2010, *Risk Assessment in Support of DOE Nuclear Safety*]
Risk Evaluation	A process of comparing the results of risk analysis with risk criteria to determine whether the risk and/or its magnitude are acceptable or tolerable. [from ISO 31000, *Risk Management - Principles and Guidelines*: 2009]
Risk Identification	A process of finding, recognizing and describing risks. Risk identification involves the identification of risk sources, events, their causes and their potential consequences. Risk identification can involve historical data, theoretical analysis, informed and expert opinions, and stakeholder's needs. [from ISO 31000]
Risk-informed Decision Making	An approach to decision making that makes use of risk insights while maintaining deterministic and probabilistic principles. These principles include the following: 1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption 2. The proposed change is consistent with a defense-in-depth philosophy. 3. The proposed change maintains sufficient safety margins. 4. When proposed changes result in an increase in risk, the increases should be small and consistent with the intent of the DOE Nuclear Safety Policy. 5. The impact of the proposed change should be monitored using performance measurement strategies. [adapted from NRC Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 1, 2002]
Risk Metric/Risk Significance Criteria	Terms of reference against which the significance of risk is evaluated. Risk metrics and the risk significance criteria to which they are compared are based on organizational objectives and external and internal context (environment in which the organization seeks to achieve its objectives). [from ISO 31000] Risk metrics are the quantities (conditions or states of concern, e.g. large release frequency) measured in a PRA, while the risk significance criteria are thresholds of concern against which the levels of risk are measured. Example risk metrics include frequency of: public or worker dose limits or constraints, meeting cost to benefit goals, system failure modes, radiological health effects, core damage or other facility damage, a radioactive material release level or specific health or safety detriment. Risk significance criteria, in addition to risk metric thresholds, can include criteria on contributions to risk such as significant basic events and significant event sequences.

Term	Definition and Source
Significant Basic Event	A basic event that contributes significantly to the computed risks for the total risk for risk for a specific hazard group. For internal events, this includes any basic event that has a Fussell-Vesely importance greater than 0.005 or a risk achievement worth (RAW) importance greater than 2. For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate, and, if used, should be justified. [adapted from ASME/ANS RA-S-1.4-2013]
Significant Risk Contributor	Implies a significant contributor to a given risk metric which may be expressed as the total integrated risk, or risk associated with a specified part of the risk model, as defined by: the source(s) of radioactive material; plant or facility operating state(s); hazard group(s); event sequence(s); event sequence family(ies); or release category(ies). Contributors may be defined in terms of initiating events, initiating event families, event sequences, event sequence families, plant or facility damage states, release categories, basic events, or other defined elements of the PRA model. [adapted from ASME/ANS RA-S-1.4-2013]
Significant Event Sequence	One of the set of event sequences, defined at the functional or systematic level, that, when rank-ordered by decreasing frequency, aggregate to a specified percentage of the release category frequency, or that individually contribute more than a specified percentage of the release category frequency or other risk metric calculated in the PRA. Depending on the context, significance may be measured in terms of the total integrated risk, or for the risk associated with a specific source of radioactive material, plant or facility operating state and hazard group. For this version of the Standard, the aggregate percentage is 95% and the individual percentage is 1%. Event sequence significance can be measured relative to each separate release category frequency. For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate, and, if used, should be justified. [adapted from ASME/ANS RA-S-1.4-2013]
Screening	A process that eliminates items from further consideration based on their insignificant contribution to the frequency or probability of an accident sequence or its consequences. [adapted from ASME/ANS RA-S-1.4-2013]
Screening Criteria	The values and conditions used to determine whether an item is an insignificant contributor to the frequency or probability of an accident sequence or its consequences. [adapted from ASME/ANS RA-S-1.4-2013]
Technical Adequacy	Term used in PRA for the set of activities to achieve quality as described in this Standard.

Appendix B
Key References Listed by Section 4 Topical Areas

The purpose of this Appendix is to provide references that offer guidance on how to plan, perform, and apply PRAs to risk-informed decision making in a manner that meets the requirements of this Standard. The references provided are drawn from PRA applications at DOE facilities, chemical and process industries, the aerospace industry, and the commercial nuclear power industry. The references include example standards used in developing and applying PRAs, procedure guides that may be used to guide PRA development; and guides and standards for applying PRAs in risk-informed decision-making. The list is not exhaustive; it is a representative set that may be useful in applying this Standard. The user of this Standard is responsible for providing the rationale for the applicability of any referenced guides and standards, as set forth in the requirements of this Standard. The references are up to date at the point this Standard was issued; however, the user of this Standard should check whether more recent versions are available and may use the more recent versions.

In Table B-1, the application of several key references to each of the topical areas in Section 4 of this Standard is described. These key references including nuclear safety policies and quantitative safety goals employed at DOE-owned and NRC-licensed facilities, guides and standards used at NASA and NRC-licensed facilities for risk-informed decision making, and the ASME/ANS PRA standard developed for commercial light water and non-light water reactor nuclear power plants.

A more extensive list of references is provided in Table B-2, organized into the following topical areas:

- Standards for PRA and Risk-Informed Decision Making
- Guidance for Risk-Informed Decision Making
- Non-Reactor PRA Applications
- Guidance for PRA Peer Reviews
- Guidance for PRA Methodology
- PRA Methods for Special Topics
    - Fault Tree Analysis
    - Database Development and Analysis
    - Common Cause Failure Analysis
    - Human Reliability Analysis
    - Internal Flooding PRA
    - Internal Fire PRA
    - External Event Screening
    - Aircraft Crash Analysis
    - Seismic PRA
    - External Flooding PRA
    - High Winds PRA
    - Expert Elicitation

- Probabilistic Treatment of Phenomena
- Quantification and Treatment of Uncertainties

One of the more comprehensive references is ASME/ANS RA-S-1.4-2013, *Advanced Non LWR PRA Standard.* This document's Table of Contents, along with pertinent subsections, is provided in Table B-3. However, the specific information provided in this Standard regarding subject matter for which specific DOE guidance is available (e.g., external events discussed in DOE Standard 3014-2006, *Accident Analysis for Aircraft Crash into Hazardous Facilities*) needs to be interpreted in light of that authoritative guidance.

Table B-1 Identification of Industry Guides and Standards for Implementing DOE PRA Standard

Section of Standard	Topics	Applicable Industry Guides and Standards	Discussion
4.1 PRA Plan			
4.1.1 Statement of Issue	How to frame PRA application in context of a risk informed decision making process.	• Section 2 RG 1.174 • Section 1 NASA-2010b	• This process is designed to preserve deterministic principles and ensure changes in risk are small • This process is designed for NASA space missions and includes criteria for when PRA is applied
	How to frame statement of the problem as a risk-informed decision.	• Section 2.1 RG 1.174 • Sections 3.1 of NASA-2010b	• Problem framed in terms of specific changes to licensing basis of reactors; risk metrics in this application are changes in Core Damage Frequency (CDF) and Large Early Release Frequency (LERF) • Problem framed in terms of risk of space missions; includes selection of risk metrics (performance measures)
4.1.2 PRA Assessment Approach	How to structure the PRA for the application and facility life cycle.	• Section 2.2 RG 1.174 • Section 1.2 – 1.2.2 of RG 1.200 • Section 3 of ASME/ANS RA-S-1.4-2013 • Section 3.2 NASA-2010b • IAEA-2002	• Includes evaluation of deterministic criteria and using PRA to evaluate changes in selected risk metrics • Discusses technical characteristics and attributes of internal event PRAs • Includes flow chart for deciding which parts of the PRA model are important to decision, what PRA capabilities are required, and what requirements in standard are needed • Includes structuring alternatives, using graded approach to PRA with alternative risk metrics selected for the decision • Addresses PRA for non-reactor facilities
4.1.3 Anticipated Results and Uses of Information	How to select risk metrics and establish risk significance criteria.	• DOE-1991 and 2010 • NRC-1986a • Section 2.2.4 of RG 1.174 • Section 3.3.1 of NASA-2010b • NUREG-1860 • NUREG-2150	• DOE nuclear safety policy with quantitative safety goals • NRC equivalent of DOE-1991 providing safety goals for commercial nuclear power plants • Risk significance criteria for changes in CDF and LERF are presented based on baseline CDF and LERF values; these are risk significance criteria rather than risk acceptance criteria • Criteria are expressed as risk tolerance levels which are not fixed but tailored to the application • Risk significance criteria based on risk metrics of frequency and doses resulting from accidents for use with deterministic safety criteria based on preservation of barriers to release • Guidance on the use of PRA results and incorporation of risk insights into regulatory decision making for reactor and non-reactor facilities
	How to evaluate results based on risk acceptance and deterministic criteria.	• Section 2.2.6 of RG 1.174 • Section 3.3.2 of NASA-2010b	• Framed as “integrated decision making” and includes both probabilistic and deterministic elements. • Uses a deliberative process to make the decision, and document the results and the rationale for the decision

Table B-1 Identification of Industry Guides and Standards for Implementing DOE PRA Standard

Section of Standard	Topics	Applicable Industry Guides and Standards	Discussion
	What is done after the risk-informed decision is initially made?	• Section 2.3 of RG 1.174 • Section 4 of NASA-2010b	• Includes an implementation part that defines how decision is implemented and a monitoring program to ensure there are no unexpected downsides to the change • Framed in terms of a continuous risk management program that monitors and adjusts decisions to manage risk levels
4.1.4 PRA Technical Adequacy and Peer Review Approach	What is the scope of the PRA Technical Adequacy Plan?	• Section 2.5 of RG 1.174 • Section 3 of RG 1.200 • Sections 1.6 and 5 of ASME/ANS RA-S-1.4-2013 • Sections 1.7 and 6 of ASME/ANS RA-S-1.4-2013 • Section 4 of ASME/ANS RA-S-1.4-2013	• Includes use of qualified personnel, procedures to guide the work, independent and peer reviews, and PRA configuration and control • Describes how technical adequacy of a PRA is assured; major topics include: risk contributors, modeling, assumptions/ approximations • Addresses technical requirements for risk assessment, including development process and expert judgment
4.2 PRA Performance	What are the available guides and standards for performing a PRA?	• AICE-2000 and Chem-2005 • ASME/ANS RA-S-1.4-2013 and NRC-2009 • NRC-1983a • NASA-2002b and NASA-2010a • IAEA-2002	• PRA methodology for chemical and process industries • Requirements for PRAs for risk-informed applications; tailored to operating non-LWR plants and based on generic risk metrics for release of radioactive material; most requirements applicable to non-reactor PRAs; PRA scope covered under continuous expansion • General methodology for PRAs on nuclear power plants • PRA methodology for space applications • PRA for non-reactor facilities
	What are the specific guides and standards for treating special topics in PRA?	• See references in Table B-2	• PRA guides and standard for special topics such as fault tree analysis, database development, external events, expert elicitation, and many other special topics
4.3 PRA Completeness Criteria	What is the necessary and sufficient level of completeness to achieve industry best practice for technical adequacy	• Section 4 of ASME/ANS RA-S-1.4-2013	• These criteria are reflected in the definitions of risk significant accident sequences, basic events, and contributors provided in the glossary. Section 4 of ASME/ANS RA-S-1.4-2013 includes criteria for screening and quantification of initiating events, plant or facility operating states, accident sequences, basic events, and human actions.
4.4 PRA Results and Documentation	What are the available guides and standards to prepare the documentation for the PRA and its application(s)?	• Section 3 of RG 1.174 • Section 3.2.2 and 3.3.2 of NASA-2010b • ASME/ANS RA-S-1.4-2013 and NRC-2009 • IAEA-2002	• Focus is on documentation of the risk-informed evaluation of a proposed decision for regulatory approval • Section 3.2.2 covers documentation of the evaluation and 3.3.2 covers documenting the decision following deliberation • Documentation requirements are developed specifically for each element of the PRA scope and are intended to be sufficient to support PRA applications and peer review • Documentation for non-reactor facility PRA

Table B-1 Identification of Industry Guides and Standards for Implementing DOE PRA Standard			
Section of Standard	**Topics**	**Applicable Industry Guides and Standards**	**Discussion**
4.5 PRA Technical Adequacy and Peer Review			
4.5.1 Implementing the Technical Adequacy Requirements	What are the available guidance for use of qualified personnel, documentation, procedures, PRA model configuration control, PRA maintenance, updates, and upgrades?	• ASME/ANS RA-S-1.4-2013 • ASME/ANS-RA-Sb-2013	• Section 1.4 provides guidance for use of expert judgment, when required, to augment other data sources. • Sections 1.6 and 5 provide general requirements for configuration control; • Appendix 1-A provides guidance for PRA maintenance, PRA upgrades, and associated peer reviews
4.5.2 The Peer Review	What are the available guides and standards to plan and conduct and document the independent reviews and peer review results?	• Section 2.2 of RG 1.174 • ASME/ANS RA-S-1.4-2013 • NEI-00-02, NEI-05-04, NEI-07-12	• Describes expectations for the peer review process, personnel qualifications, and documentation of results. • Sections 1.7 and 6 provide general requirements for peer review, the each PRA technical area • Nuclear industry guides for performing PRA peer reviews
4.5.3 Peer Review Results		• ASME/ANS RA-S-1.4-2013	• Section 6 describes expectations for peer review results

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries		
Reference ID	**Reference**	**Topic**
Guides and Standards for PRA and Risk Informed Decision Making		
DOE-1991	U.S. Department of Energy, SEN-35-91, "Nuclear Safety Policy," September 9, 1991.	Includes quantitative safety goals similar to those in NRC-1986a for DOE facilities, as well as criteria for management, technical competence, oversight, and safety culture;
DOE- 2011a	U.S. Department of Energy, "Department of Energy Nuclear Safety Policy," DOE P 420.1, 2011.	Draft revision to DOE-1991, includes same safety goals
DOE-IN-2010	U.S. Department of Energy, Risk Assessment in Support of Nuclear Safety, DOE Information Notice, June 2010.	Describes DOE expectations with regard to DOE's use of risk assessment use to better inform Nuclear Safety decisions.
NRC-1986a	U.S. NRC, "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," Federal Register, Vol. 51, p. 30028 (51 FR 30028), August 4, 1986.	Risk metrics and risk significance criteria (safety goals and Quantitative Health Objectives) for NPP accidents
ASME/ANS-RA-Sb-2013	ASME/ANS RA-Sb-2013, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum B to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, September 2013.	PRA Standards for LWRs
ASME/ANS RA-S-1.4-2013	ASME/ANS RA-S-1.4-2013, "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants", ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, December 2013.	PRA Standard for non-LWRs using generalized risk metrics not specific to any reactor type
ISO-31000	International Organization for Standardization, ISO 31000:2009(E), Risk management – Principles and guidelines, first edition 2009.	Describes process and terms for integrating risk management into decision making through an organizations overall operations and activities.
ISO-31010	International Organization for Standardization, IEC/ISO 31010, Risk Management – Risk assessment techniques, Edition 1, November 2009.	Describes the general risk assessment process and specific risk assessment techniques and tools that can be used to support risk management and inform decisions.
NASA-2008a	NPR 8000.4A, Agency Risk Management Procedural Requirements, December 2008) and NPR 7120.5, (NASA Space Flight Program and Project Management Handbook, 2010.	NASA Risk Management Requirements
NASA-2008b	NASA, NASA General Safety Program Requirements, NASA General Safety Program Requirements (w/Change 9 dated 2/08/13), NPR 8715.3C, 2008.	NASA Safety Requirements
NUREG-2150	U.S. NRC, A Proposed Risk Management Regulatory Framework, NUREG-2150, 2012.	A framework for future revisions to NRC risk-informed decision making in both reactor and non-reactor facilities regulated by the NRC
NRC-2009	U.S. NRC, Regulatory Guide 1.200, Revision 1, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, March 2009.	NRC Guide on Industry Standards
NUREG-1860	U.S. NRC, Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, NUREG-1860, 2007.	Guidance for performing PRA on advanced non-LWRs and proposed risk significance criteria based on frequency and dose for individual event sequences
NRC-0800	U.S. NRC, Standard Review Plan for the Review of the Safety Analysis Reports for Nuclear Power Plants, NUREG-0800, Section 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decision Making: General Guidance."	Review guidance

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries		
Reference ID	**Reference**	**Topic**
ANSI/ANS-2.27	ANSI/ANS-2.27-2008: " Criteria for Investigations of Nuclear Facility Sites for Seismic Hazard Assessments"	Seismic Hazard Analysis
ANSI/ANS-2.29	ANSI/ANS-2.29-2008, "Probabilistic Seismic Hazard Analysis."	Seismic Hazard Analysis
Non-Reactor PRA Applications		
AICE-2000	American Institute of Chemical Engineers, Guidelines for Chemical Process Quantitative Risk Analysis, Second Edition, 2000.	Chemical Industry PRA Procedures
DOD-1997	"Assess the Safety of Planned Demilitarization Operations for Chemical Weapons at Tooele, Anniston, and Others," Anniston Chemical Agent Disposal Facility, Phase 1 Quantitative Risk Assessment, (May 1997).	Chemical Weapon PRA
IAEA-2002	International Atomic Energy Agency, Procedures For Conducting Probabilistic Safety Assessment For Non-Reactor Nuclear Facilities, IAEA TECDOC-1267, January 2002.	Non-reactor facilities with radiological hazards
WTP-2009a	WTP 2007 Operations Risk Assessment Report, B-ORA07, Rev 1, SARACon, Inc., May 28, 2009.	PRA of WTP
Guidance for Risk-Informed Decision Making		
NRC-2008b	U.S. NRC, Risk-Informed Decisionmaking for Nuclear Materials and Waste Applications, Revision 1, 2008.	
EPRI-1995a	Electric Power Research Institute, PSA Applications Guide, EPRI TR-105396, 1995.	Risk-informed Decision Process
NASA-2010b	NASA, NASA Risk-Informed Decision Making Handbook, NASA/SP-2010-576, 2010.	Risk Informed Decision Process
RG-1.174	U.S. NRC, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Regulatory Guide 1.174, Nov 2002.	Risk-informed Decision Process
RG-1.175	U.S. NRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," Regulatory Guide 1.175, August1998.	Risk-informed IST
RG-1.177	U.S. NRC, An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications, Regulatory Guide 1.177, August 1998.	Risk-informed TS
RG-1.178	U.S. NRC, An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Inspection of Piping, Regulatory Guide 1.178, September 2003.	Risk-informed ISI
Guidance for PRA Technical Adequacy and Peer Reviews		
DOE-2011	U.S. Department of Energy, Quality Assurance Program Guide, DOE G 414.1-2B, 2011.	Guidance for DOE QA Programs
NEI-00-02	Nuclear Energy Institute, "Probabilistic Risk Assessment Peer Review Process Guidance," NEI 00-02, Revision 1, May 2006.	Peer review procedures
NEI-05-04	Nuclear Energy Institute, "Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard," NEI 05-04, Revision 2, November 2008.	Peer review procedures
NEI-07-12	Nuclear Energy Institute, Washington, DC, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," NEI 07-12, Draft Version H, Revision 0, November 2008.	Peer review procedures
NRC-1993	U.S. NRC, Software Quality Assurance Program and Guidelines, (NUREG/BR-0167).	Guidance for software quality assurance programs

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries		
Reference ID	**Reference**	**Topic**
Guidance for PRA Technical Adequacy and Peer Reviews		
DOE-2011b	U.S. Department of Energy, Quality Assurance Program Guide, DOE G 414.1-2B, Admin. Chg. 2, 2011.	Guidance for DOE QA Programs
Guidance for PRA Methodology		
Chem-2005	"Collar Hazards with a Bow-Tie," J. Philley, Chemical Processing Journal, Jan. 23, 2006 (http://www.chemicalprocessing.com/articles/2005/612.html)	Hazards analysis method for chemical industry
NASA- 2011	NASA, Probabilistic Risk Assessment Procedures Guide for NASA Managers and Practitioners, second edition, NASA/SP-2011-3421, 2011.	NASA PRA Procedures
NASA-2010a	NASA, Technical Probabilistic Risk Assessment (PRA) Procedures for Safety and Mission Success for NASA Programs and Projects, NPR 8705.5A, 2010.	NASA PRA Procedures
NRC-1975	U.S. NRC, WASH-1400, The Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, NUREG-75/014, 1975.	LWR PRA Case Study; first PRA on LWR power plants
NRC-1983a	U.S. NRC, PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, NUREG/CR-2300, 1983	PRA Procedures Guide extensively used in commercial nuclear plants
NRC-1983b	U.S. NRC, Interim Reliability Evaluation Program Procedures Guide, NUREG/CR-2728, 1983.	PRA Procedures
NRC-1985b	U.S. NRC, Probabilistic Safety Analysis Procedures Guide, NUREG/CR-2815, 1984.	PRA Procedures
NRC-1990a	U.S. NRC, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, NUREG 1150, 1990.	PRA Procedures
NRC-2003a	U.S. NRC, Issues and Recommendations for Advancement of PRA Technology for Risk-Informed Decision Making, NUREG/CR-6813, 2003	Technical Issues in PRA for commercial nuclear plants for US. NRC ACRS
PRA Methods for Special Topics – Fault Tree Analysis		
NASA-2002a	NASA, Fault Tree Handbook with Aerospace Applications, Version 1.1, 2002	Fault tree Procedures for Aerospace
NRC-1981b	U.S. NRC, Fault Tree Handbook, NUREG-0492, 1981.	Fault tree Procedures
NRC-1998f	U.S. NRC, Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment, NUREG/CR-5485, 1998.	CCF Modeling in Fault trees
PRA Method for Special Topics – Database Development and Analysis		
WTP-2009b	WTP RAMI Database, 24590-WTP-DBRA-IT-05-0007, and 24590-WTP-DBMP-IT-07-0015.	PRA Data for WTP PRA
WSRC-1998	Westinghouse Savannah River Company, "Savannah River Site Generic Database Development", WSRC-TR-93-262, Rev. 1, 1998.	Failure rate estimates for Savannah River
NASA-2009	NASA, Bayesian Inference for NASA Probabilistic Risk and Reliability Analysis, NASA/SP-2009-569, 2009.	Bayes methods for analyzing data and treatment of uncertainties in NASA PRAs
EGG-1990	Idaho National Engineering Laboratory, "Generic Component Failure Databases for Light Water and Liquid Sodium Reactor PRAs," EGG-SSRE-8875, 1990.	Generic failure rate data for LWRs and liquid metal reactors
NRC-1994	U.S. NRC, Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR), Vols. 1–5, NUREG/CR-4639, Revision 4, 1994.	Generic data for PRA
NRC-1997b	U.S. NRC, Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1986, NUREG/CR-5496, 1997.	Loss of offsite power data for LWRs
NRC-1998b	U.S. NRC, Modeling Time to Recover and Initiate Even Frequency for Loss-of-Offsite	Power recovery data for LWRs

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries		
Reference ID	**Reference**	**Topic**
	Power Incidents at Nuclear Power Plants, NUREG/CR-5032, 1988.	
NRC-1999a	U.S. NRC, Rates of Initiating Events at U.S. Nuclear Power Plants, NUREG/CR-5750, 1999.	Initiating event data for LWRs
NRC-2003b	U.S. NRC, Handbook of Parameter Estimation for Probabilistic Risk Assessment, NUREG/CR-6823, 2003.	Data analysis methodology
NRC-2007	U.S. NRC, Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, NUREG/CR-6928, 2007.	Generic failure rate data for LWRs
NRC-2008	U.S. NRC, Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process, NUREG-1829, 2008.	Initiating event data for LWR LOCAs
Fleming-2004b	Fleming, K. N., "Markov Models for Evaluating Risk Informed In-Service Inspection Strategies for Nuclear Power Plant Piping Systems", comma outside quotes Reliability Engineering and System Safety, Vol. 83, No. 1 pp.:27-45, 2004.	Passive Component Reliability
PRA Methods for Special Topics – Common Cause Failure Analysis		
NRC-1987b	U.S. NRC, "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," NUREG/CR-4780, 1987.	PRA methods for CCF
NRC-1998d	U.S. NRC, Common-Cause Failure Parameter Estimations, NUREG/CR-5497, 1998.	CCF Parameter Estimates
NRC-1998e	U.S. NRC, Common Cause Failure Database and Analysis System, Vols. 1–4, NUREG/CR-6268, 1998.	CCF Data Analysis Method
DOE-1996b	U.S. Department of Energy, Office of Field Management and Office of Project and Fixed Asset Management, "Project Reviews." Good Practice Guide, GPG-FM-015, 1996.	Project Reviews
PRA Methods for Special Topics – Human Reliability Analysis		
WSRC-1994	Westinghouse Savannah River Company, Savannah River Site Human Error Data Base Development for Nonreactor Nuclear Facilities, WSRC-TR-93-581, 1994.	Human error rates from Savannah River Service Data
EPRI-2009	Electric Power Research Institute, HRA Calculator 4.1.1, Human Reliability Analysis, EPRI Product # 1020436, 2009.	HRA PRA Methodology
NRC-1983c	U.S. NRC, Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, 1983.	HRA PRA Methodology
NRC-1987c	U.S. NRC, Accident Sequence Evaluation Program Human Reliability Analysis Procedure, NUREG/CR-4772, 1987.	HRA PRA Methodology
NRC-2000	U.S. NRC, Technical basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA), NUREG-1624, Revision 1, 2000.	HRA PRA Methodology
NRC-2005a	U.S. NRC, Good Practices for Implementing Human Reliability Analysis (HRA), NUREG-1792, 2005.	HRA PRA Methodology
NRC-2006	U.S. NRC, "Evaluation of Human Reliability Analysis Methods Against Good Practices," NUREG-1842, 2006.	HRA PRA Methodology
WSMS-2009	Westinghouse Safety Management Solutions, Human Reliability Analysis, WSMS-SAE-M-09-0014, 2009.	HRA PRA Methodology
PRA Methods for Special Topics – Internal Flooding PRA		
EGG-1991	Idaho National Engineering Laboratory, Component External Leakage and Rupture Frequency Estimates, EGG-SSRE--9639, 1991.	Pipe Failure Data for flood PRA

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries		
Reference ID	**Reference**	**Topic**
EPRI-2009	Electric Power Research Institute, "Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment, EPRI Product # 1019194, 2009.	PRA Procedures for internal flood PRA
EPRI-2010	Electric Power Research Institute, Fleming, K. N. and B. O. Y. Lydell, "Pipe Rupture Frequencies for Internal Flooding PRAs", Revision 2, EPRI Product # 1021086, 2010.	Pipe Failure Data for flood PRA
Fleming-2004a	Fleming, K. N. and B. O. Y. Lydell, "Database Development and Uncertainty Treatment for Estimating Pipe Failure Rates and Rupture Frequencies," Reliability Engineering and System Safety, 86: 227–246, 2004.	Pipe Failure Data for flood PRA
PRA Methods for Special Topics – Internal Fire PRA		
EPRI-1992	Electric Power Research Institute, Fire-Induced Vulnerability Evaluation (FIVE), EPRI TR-100370, 1992.	Fire PRA for LWRs
EPRI-1995b	Electric Power Research Institute, Fire PRA Implementation Guide, EPRI TR-105928, 1995.	Fire PRA for LWRs
ERI-1997	Energy Research, Inc., "Review of the EPRI Fire PRA Implementation Guide," ERI/NRC 97-501, 1997.	Fire PRA for LWRs
EPRI-2005	Electric Power Research Institute and U.S. NRC, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, EPRI TR-1011989 and NUREG/CR-6850, 2005.	Fire PRA for LWRs
NFPA-805	National Fire Protection Association, NFPA 805: "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants".	
NRC-2004a	U.S. NRC, Fire Dynamics Tools (FDTs): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program, NUREG-1805, 2004.	Fire PRA for LWRs
PRA Methods for Special Topics – External Events Screening		
NRC-1989b	U.S. NRC, Recommended Procedures for the Simplified External Event Risk Analyses for NUREG-1150, NUREG/CR-4840, 1989.	PRA Procedures For Screening of external events
NRC-1992	U.S. NRC, Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development," NUREG/CR-4839, 1992.	PRA Procedures For Screening of external events
NRC-1998a	U.S. NRC, Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150, NUREG/CR-4840, 1988.	PRA Procedures For Screening of external events
PRA Methods for Special Topics – Aircraft Crash		
DOE-1996b	U.S. Department of Energy, Accident Analysis for Aircraft Crash Into Hazardous Facilities, DOE-STD-3014-96, 1996.	DOE Standard for aircraft crashes
PRA Methods for Special Topics – Seismic PRA		
USGS-2009	U.S. Geological Survey, Implementation of the SSHAC Guidelines for Level 3 and 4 PSHAs--experience gained from actual applications, File Report 2009-1093. [http://pubs.usgs.gov/of/2009/1093/]	
NRC-1985a	U.S. NRC, Simplified Seismic Probabilistic Risk Assessment: Procedures and Limitations, NUREG/CR-43311985.	Simplified Seismic PRA method
Budnitz-1998	R. J. Budnitz, "Current Status of Methodologies for Seismic Probabilistic Safety Analysis," Reliability Engineering and Systems Safety, Vol. 62, pp. 71–88 (1998).	Seismic PRA method
EPRI-1991	Electric Power Research Institute, A Methodology for Assessment of Nuclear Power Plant Seismic Margin, EPRI NP-6041-SL, Rev. 1, 1991.	Seismic PRA method

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries		
Reference ID	**Reference**	**Topic**
NRC-2012	U.S. NRC, Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies. NUREG-2117, Revision 1, 2012.	
NRC-1985c	U.S. NRC, An Approach to the Quantification of Seismic Margins in Nuclear Power Plants, NUREG/CR-4334, 1985.	Seismic PRA method
NRC-1997d	U.S. NRC, Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts, NUREG/CR-6372, 1997.	Seismic Hazard Analysis, Expert Elicitation
DOE-1998	U.S. Department of Energy, Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada, Contract DE-AC04-94AL85000, in three volumes, prepared for the U.S. Geological Survey, 1998.	Seismic hazard Analysis for Yucca Mountain
EPRI-2004	Electric Power Research Institute, CEUS Ground Motion Project Final Report, TR-100984, 2004.	Seismic Hazard Analysis
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Young-2003	R. Youngs et al., “A Methodology for Probabilistic Fault Displacement Hazard Analysis (PFDHA),” Earthquake Spectra, Volume 19, No. 1, pages 191-219, (2003).	Seismic Hazard Analysis
EPRI-1994a	Electric Power Research Institute, Methodology for Developing Seismic Fragilities, TR-103959, 1994.	Seismic Fragility method
EPRI-1994b	Electric Power Research Institute, Methodology for Developing Seismic Fragilities, TR-103959, 1994.	Seismic Fragility method
Kennedy-1984a	Kennedy, R. P. and M. K. Ravindra, “Seismic Fragilities for Nuclear Power Plant Risk Studies,” 31 Nuclear Engineering and Design, Vol. 79, No. 1, pp. 47-68, 1984.	Seismic Fragility method
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LLNL-1998	Lawrence Livermore National Laboratory, Probabilistic Flood Hazard Assessment for the N Reactor, Hanford, Washington, Report UCRL-2106, 1988.	External flood PRA
MIT-1982	E. H. Vanmarke and H. Bohnenblust, “Risk-Based Decision Analysis for Dam Safety,” Research Report R82-11, Massachusetts Institute of Technology, Department of Civil Engineering (1982).	External flood PRA
NAS-1998	National Academy of Sciences, “Estimating Probabilities of Extreme Floods, Methods and Recommended Research,” Committee on Techniques for Estimating Probabilities of Extreme Floods, Water Science and Technology Board, National Research Council, 1988.	External flood PRA
STA-1985	Stanford University Department of Civil Engineering, M.W. McCann, Jr., and G. A. Hatem, “Progress on the Development of a Library and Data Base on Dam Incidents in the U.S.,” Progress Report No. 2 to Federal Emergency Management Agency; available in an alternative form as G. A. Hatem, “Development of a Database on Dam Failures in the United States: Preliminary Results,” Engineering Thesis, 1985.	External flood PRA
WDC-1986	U.S. Geological Survey, Water Resources Division, “Feasibility of Assigning a Probability to the Probable Maximum Flood,” Work Group on Probable Maximum Flood Risk Assessment, Under the Direction of the Hydrology Subcommittee of the Interagency Advisory Committee on Water Data, U.S. Office of Water Data Coordination, 1986.	External flood PRA

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries		
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DOE-1985	L. A. Twisdale and M. B. Hardy, "Tornado Windspeed Frequency Analysis of the Savannah River Plant," Savannah River Plant Report, prepared for E. I. DuPont de Nemours and Company, Aiken, South Carolina (1985).	PRA of High Winds
NRC-1981a	L. A. Twisdale,W. L. Dunn, and B. V. Alexander, "Extreme Wind Risk Analysis of the Indian Point Nuclear Generating Station," Report No. 44T-2171, Prepared for Pickard, Lowe and Garrick, Inc., available from the U.S. Nuclear Regulatory Commission, Docket Nos. 50-247 and 50-286 (1981).	PRA of High Winds
NRC-1982 same as Reinhold-1982 below	U.S. NRC, Tornado Damage Risk Assessment, NUREG/CR-2944, 1982.	PRA of High Winds
NRC-1986b	U.S. NRC, Tornado Climatology of the Contiguous United States, NUREG/CR-44611986.	PRA of High Winds
NRC-1987a	U.S. NRC, Shutdown Decay Heat Removal Analysis of a Westinghouse 2-loop Pressurized Water Reactor, Appendix G, "Extreme Wind Analysis for the Point Beach Nuclear Power Plant," NUREG/CR-4458, 1987.	PRA of High Winds
NRC-1990b	U.S. NRC, State-of-the-Art and Current Research Activities in Extreme Winds Relating to Design and Evaluation of Nuclear Power Plants, NUREG/CR-5497, 1990.	PRA of High Winds
Ravindra-1997	M. K. Ravindra, Z. M. Li, P. Guymer, D. Gaynor, and A. DiUglio, "High Wind IPEEE of Indian Point Unit 2," Transactions of 14th International Structural Mechanics in Reactor Technology (SMiRT) Conference, August 1997, Lyon, France.	PRA of High Winds
Reinhold-1982 see NRC-1982 above	Reinhold, T.A. and Ellingwood, B., "Tornado Damage Risk Assessment," Brookhaven National Laboratory, NUREG/CR-2944, 1982.	PRA of High Winds
Twinsdale-1995	L. A. Twisdale and P. J. Vickery, "Extreme Wind Risk Assessment," Probabilistic Structural Mechanics Handbook — Theory and Industrial Applications, Chapter 20, C. Sundararajan, Editor, Chapman and Hall, New York (1995)	PRA of High Winds
PRA Methods for Special Topics – Expert Elicitation		
DOE-1996a	Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada, BA000-1717-2200-00082, U.S. Department of Energy Yucca Mountain Site Characterization Project, 1996.	PRA for Volcano Hazard at Yucca mountain
NRC-1996	U.S. NRC, Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program, NUREG/CR-1563, 1996.	PRA for High level waste application
NRC-1997d	U.S. NRC, Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts, NUREG/CR-6372, 1997.	Seismic Hazard Analysis, Expert Elicitation
see above		Case Study on Use of Expert Elicitation in Seismic Hazard Analysis
PRA Methods for Special Topics – Probabilistic Treatment of Phenomena		
BNL-2006	Brookhaven National Laboratory , "Experience Using Phenomena Identification and Ranking Technique (PIRT) for Nuclear Analysis," by David J. Diamond, BNL-76750-2006-CP, , PHYSOR-2006 Topical Meeting, Vancouver, British Columbia, Canada, September 10-14, 2006.	Method for evaluating phenomena
NRC-1998c	U.S. NRC, Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture, NUREG-1570, 1998.	Phenomenological probabilities for LWR Severe Accidents

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries		
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NRC-2004b	U.S. NRC, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events, NUREG/CR-6595, Revision 1, 2004.	Phenomenological probabilities for LWR Severe Accidents
WTP-2009c	Quantitative Risk Analysis of Hydrogen Events at WTP: Development of Event Frequency-Severity Analysis Model Document number: 24590-WTP-RPT-ENG-10-008, Rev 1	WTP PRA of Hydrogen Events
PRA Methods for Special Topics – Quantification and Treatment of Uncertainties		
Apostolakis-1981	G. Apostolakis and S. Kaplan, "Pitfalls in Risk Calculations," Reliability Engineering, Vol. 2, pp. 135–145, 1981.	Methods for treatment of uncertainty
Morgan-1990	Morgan, M. G. and M. Henrion, Uncertainty; A Guide to Dealing with Uncertainty in Quantitative Risk And Policy Analysis, Cambridge University Press, 1990.	Methods for treatment of uncertainty
NRC-1989a	U.S. NRC, "Quantifying Reactor Safety Margins: Application of Code Scaling, Applicability, and Uncertainty Evaluation Methodology to a Large-Break Loss-of-Coolant Accident," NUREG/CR-5249, Revision 4, 1989.	Phenomenological probabilities for LWR Severe Accidents
Rao-2007	Rao, K.D., Kushwaha, H.S., Verma, A. K., and Srividya, A., "Quantification of epistemic and aleatory uncertainties in level-1 probabilistic safety assessment studies," Reliability Engineering & System Safety (2007) 92, 947-956.	Methods for treatment of uncertainty
WSRC-2002	Westinghouse Savannah River Company, "Development of Probabilistic Uncertainty Analysis for SRS Performance Assessments Maintenance Plan Activities," WSRC-TR-2002-00121, 2002.	Uncertainty analysis at Savannah River Plant
Wu-2004	Wu, F-C and Tsang, Y-P, "Second-order Monte Carlo uncertainty/variability analysis using correlated model parameters; application to salmonid embryo survival risk assessment," Ecological Modeling (2004) 177, 393-414.	Methods for treatment of uncertainty

Table B-3 ASME/ANS RA S-1.4-2013 PRA Standard Table of Contents

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