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DOE STANDARD

DEVELOPMENT AND USE OF PROBABILISTIC RISK ASSESSMENTS IN DEPARTMENT OF ENERGY NUCLEAR SAFETY APPLICATIONS

FOR INTERIM USE AND COMMENT

AREA SAFT

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Washington, DC 20585

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Foreword

There have been significant developments with regard to the risk assessment and risk informed decision making, as it applies to nuclear and other safety areas, since the Department of Energy (DOE) developed its approach to managing nuclear safety. The developments and associated technical insights may be of use to DOE in its efforts to continuously improve safety performance at its nuclear facilities.

The Department has taken several actions to provide an infrastructure for providing appropriate controls and support for use of risk assessments and risk informed decision making as it applies to nuclear safety, including establishing a Risk Assessment Technical Experts Working Group, revising its Nuclear Safety Policy to explicitly address the use and control of risk assessments, and developing this DOE Technical Standard for Control and Use of Probabilistic Risk Assessment[1].

This Standard was developed by a team of DOE and industry risk assessment experts. It is being issued for interim use and comment to allow DOE to take advantage of the insights it provides for control of risk assessments while it is being finalized and improved based upon lessons learned during pilot applications.

DOE technical standards, such as this, do not establish requirements. However, all or part of the provisions within this DOE technical standard shall be implemented under the following circumstances:

- They are explicitly stated to be requirements in a DOE requirements document.
- The organization makes a commitment to meet a standard in a contract or in an implementation plan or program of a DOE requirements document.

Throughout this Standard, the word “shall” is used to denote actions that must be performed if the objectives of this Standard are to be met. If the provisions in this Standard are made requirements through one of the two ways discussed above, then the “shall” statements become requirements. It is not appropriate to consider that “should” statements would automatically be converted to “shall” statements, as this action would violate the consensus process used to approve this Standard.

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

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[1] DOE has chosen to utilize the term probabilistic risk assessments in this standard to cover all quantitative risk assessments where frequency and consequence are evaluated in an integrated manner.

1. INTRODUCTION

The Department has taken several actions to provide an infrastructure for providing appropriate controls and support for use of risk assessments and risk informed decision making as it applies to nuclear safety including establishing a Risk Assessment Technical Experts Working Group, revising its Nuclear Safety Policy to explicitly address the use and control of risk assessments, and developing this DOE Technical Standard for Control and Use of Probabilistic Risk Assessment for interim use and comment.[1]

The purpose of this Standard is to provide guidance and criteria for a standard approach to utilization of probabilistic risk assessments (PRAs) in nuclear safety applications. This supports the Department's policy to design, construct, operate, and decommission its nuclear facilities in a manner that ensures adequate protection of workers, the public, and the environment.

To better inform decision-makers, DOE's nuclear safety decision-making processes can be supplemented and strengthened through application of quantitative and probabilistic risk assessment methodologies; such methodologies may be useful in:

- Aiding the evaluation of alternatives that comply with DOE nuclear safety requirements.
- Supporting the unreviewed safety question (USQ) process.
- Augmenting traditional safety assessment methods.
- Evaluating changes to DOE safety requirements.
- In general enhancing the quality, transparency, and credibility of analytical results and decisions that are made.

2. APPLICABILITY AND SCOPE

This Standard was developed to support use of PRAs in nuclear safety applications. It is based on standards, guides, and best practices from high-risk industry (chemical, nuclear, and aerospace) on use of risk assessments when used to support risk-informed decision-making in safety applications.

This Standard also addresses the use of risk assessments to support meeting DOE nuclear safety requirements specified in 10 Code of Federal Regulation (CFR) 830, *Nuclear Safety Management,* related to development and maintenance of documented safety analyses (DSAs) when used to support risk-informed decision-making related to the safety analysis results.

In this Standard the term "shall" is utilized when referring to an action required by 10 CFR 830. DOE technical standards, such as this, do not establish requirements. However, all or part of the provisions within this DOE technical standard shall be implemented under the following circumstances:

- They are explicitly stated to be requirements in a DOE requirements document.

[1] DOE has chosen to utilize the term probabilistic risk assessments in this standard to cover all quantitative risk assessments where frequency and consequence are evaluated in an integrated manner.

- The organization makes a commitment to meet a standard in a contract or in an implementation plan or program of a DOE requirements document.
- Throughout this Standard, the word “shall” is used to denote actions that must be performed if the objectives of this Standard are to be met. If the provisions in this Standard are made requirements through one of the two ways discussed above, then the “shall” statements become requirements. It is not appropriate to consider that “should” statements would automatically be converted to “shall” statements, as this action would violate the consensus process used to approve this Standard.

3. OVERVIEW OF STANDARD

Section 4 of this Standard identifies elements required for the development and use of PRAs in general. Section 5 then identifies elements associated with use of PRA in specific DOE nuclear safety applications. Appendix A is a glossary of risk terms. Appendix B contains a list of key references, organized by the topics in Section 4 of this Standard, as well as a topical list of useful references.

This Standard is primarily a process standard which then refers to recognized industry standards and guidance for details on how specific aspects of the process can be implemented.

4. KEY ELEMENTS IN DEVELOPMENT OF PRAs

This section provides the elements in 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 performing any PRA, the project shall develop a plan for the application of PRA techniques to the needs of the project. The PRA plan shall address the following elements:

- Statement of the Issue;
- Risk Assessment Approach;
- Results, Conclusions, and Uses; and
- Quality Assurance and Peer Review.

These main elements, along with associated subtopics, are discussed in the four following sections. All PRA activities shall be conducted according to the PRA plan.

4.1.1 Statement of the Issue

The PRA plan shall address the statement of the issue which has brought about the project’s need to apply PRA techniques. The PRA plan shall address the following topics:

4.1.1.1 Purpose, Objectives, and Scope of the PRA

The purpose shall include a discussion why the PRA is being performed. The objectives should describe underlying decisions to be supported or needs to be addressed by the PRA. The scope of the analysis including the boundaries of the systems and the activities to be analyzed shall be defined in terms of the following, as applicable:

- Plant SSCs and operating states
- Internal events and hazards
- External events and hazards
- Accident phenomena and progression
- Selected consequence metric

The scope may be very narrow or broad depending on the application. The plan shall justify the adequacy of the scope for the intended application.

4.1.1.2 Principal Assumptions and Limitations

The PRA shall include a description of the principal assumptions upon which the PRA methods and models are based, and any limitations on the use of the PRA's results.

4.1.1.3 Relationship to the Safety Basis

To address the relationship between the PRA results and the safety basis (e.g., the Final DSA or Preliminary DSA), the PRA Plan shall describe the process used to identify the key PRA assumptions which require protection by the TSR; safety controls to be included in the TSR, based on PRA results, for safety class or safety significant structures, systems, and components (SSCs); and commitments to maintain the PRA for use in supporting the unreviewed safety question process, as applicable (see Section 5).

4.1.1.4 Applicable Approvals

The plan should be reviewed by the appropriate management for which it is being developed. For example, if the PRA is being used to support the development of the Preliminary DSA as part of a new nuclear facility project, then the approval authority for the project should review and approve the plan; alternatively, changes to an existing DSA would require review by the approval authority.

4.1.1.5 Risk Metrics

The user shall select and provide the rationale for risk metrics used. Example metrics may include the frequency of exceeding specified design limits, probability of exceeding established safety criteria, and individual risk from radiological or chemical exposures (see definition in Appendix A).

4.1.2 PRA Approach

Several methodologies (methods and models) for performing a PRA have been developed and are described in industry standards (see Appendix B). Each particular methodology offers specialized schemes and tools for analyzing the subject facilities or processes. However, when properly applied, all methodologies are systematic and provide a disciplined approach to the evaluation of safety or risk.

The PRA approach should be described in the plan by addressing the following topics:

4.1.2.1 Forming the Team

The plan shall define the disciplines and qualifications of the team necessary to perform the PRA, and it shall include personnel experienced in DOE's nuclear safety process and requirements.

4.1.2.2 Detailed Assumptions

Based upon the principle assumptions, the plan shall identify detailed assumptions that influence the strategies and the method or model that form the basis of the approach.

4.1.2.3 Data Quality Objectives

The plan shall define the quality objectives for the data to be used in the analysis, including data derived through expert elicitation and engineering judgment. The process of collecting and analyzing information in order to estimate various parameters for the PRA models shall be described, including sources used to obtain the probabilities of various events (such as component failure rates, initiator frequencies, human failure probabilities, and characterization of physical phenomena).

Typical quantities of interest are:

- Initiating Events (IEs) Frequencies
- Component Failure Rates or Failure Probabilities
- Human Error Rates
- Event or Phenomena (e.g., gas ignition) Probabilities

Data quality objectives shall also be identified for parameter estimation techniques and for the results of sensitivity analysis. The plan shall also address the strategy for any data base development for collecting and making available input data (operational data) and its sources.

4.1.2.4 PRA Methodology Development

The PRA methodology to be developed shall be described, including identification of the applicable industry standards or guides that are being applied. An example of PRA methodology elements for commercial light water reactors are listed in Table 1-1.3-1 from the ASME/ANS PRA Standard 2009 Addendum A.

4.1.2.5 Schedule and Resources

The plan shall define the schedule and resource requirements necessary for the development, conduct, and peer review (see below) of the PRA**.**

4.1.2.6 Peer Review Approach

The plan shall define the peer review process and the applicable standards and guides used to perform the review. The peer review process should be commensurate with the PRA's complexity and importance to safety, and the process 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).

4.1.3 Results, Conclusions, and Uses

The PRA plan shall address the following topics:

4.1.3.1 Outcomes

The plan shall indicate what results are to be produced. Further sample guidance based on light water reactors can be found on this topic in the ASME/ANS PRA Standard for PRA (see Appendix B).

4.1.3.2 Interpretation of Results

The plan shall indicate how the PRA results will be compared with established metrics, interpreted, and used to support the decisions.

4.1.3.3 Impact on Safety Basis

The plan shall clearly identify how the results of the PRA will interface with the existing safety basis (per DOE Standard 3009 for existing Category 1, 2 or 3 Nuclear Facilities) or be included in the Safety Design Strategy for new facilities (required by DOE Standard 1189, *Integration of Safety into the Design Process*).

4.1.4 Quality Assurance and Peer Review Plans

The PRA plan shall identify the applicable DOE QA requirements and describe how they will be met including DOE requirements for QA records and audits, the use of verified computer programs, document logs, a corrective action program, and the use of procedures, in addition to the following topics:

4.1.4.1 Documentation

The PRA shall be documented in a manner that facilitates the PRA application, upgrades, and peer review. The plan shall also identify the quality assurance requirements and associated procedures for documentation of the methodology (methods and models), its use, and results.

4.1.4.2 Configuration Controls

The plan shall identify the quality assurance requirements and associated procedures for configuration control, both during initial use of the PRA and its maintenance to support future use, as applicable.

4.1.4.3 Technical Adequacy

The plan shall identify the quality assurance requirements and associated procedures for assuring the technical adequacy of the PRA.

4.1.4.4 Peer Review

The plan shall describe quality assurance requirements for the peer review process (as further described in Section 4.4).

4.1.4.5 Performance Monitoring

The plan should describe the processes to ensure appropriate corrective actions are taken and that assumptions are maintained and remain valid.

4.2 PRA Performance

The PRA shall be performed in accordance with the PRA plan. Changes identified as necessary during implementation of the Plan shall be documented in accordance with established procedures. In particular, appropriate industry standards, guides, and practices shall be implemented as described in the PRA plan.

4.3 PRA Documentation

The first several elements are the same as those documented in the PRA plan, but will need to be updated to reflect any changes made during the PRA performance. The following key elements of the PRA shall be documented:

- The project's purpose and objective.
- The appropriateness of the results in meeting the PRA's objective.
- A clear and concise tabulation of all known limitations and constraints associated with the analysis.

- A clear and concise tabulation of all the assumptions used in the PRA, especially with respect to mission success criteria.
- Justification of the omission of any failure modes.
- Identification of data sources.
- Identification of key parameters that greatly influence the numerical results of the PRA.
- Results of activities undertaken (e.g., sensitivity studies) to ensure that the results of the PRA would not be negated if an alternative parameter value or modeling assumption is employed.
- Relationship to DSA deterministic analyses and the supplemental insights that were obtained.
- Results of activities undertaken to ensure technical quality.
- Results and conclusions.

4.4 Quality Assurance and Peer Review

The QA requirements in 10 CFR 830, Part A, apply to PRA used to inform nuclear safety decisions, including requirements for:

> Criterion 2—Management/Personnel Training and Qualification: (1) Train and qualify personnel to be capable of performing their assigned work.
>
> Criterion 4—Management/Documents and Records. Perform work consistent with technical standards, administrative controls, and other hazard controls adopted to meet regulatory or contract requirements, using approved instructions, procedures, or other appropriate means.
>
> Criterion 6—Performance/Design. (1) Design items and processes using sound engineering/scientific principles and appropriate standards.
>
> Criterion 9—Assessment/Management Assessment. Ensure managers assess their management processes and identify and correct problems that hinder the organization from achieving its objectives.
>
> Criterion 10—Assessment/Independent Assessment

4.4.1 Implementing the Quality Assurance Requirements

Implementing the quality assurance requirements for the PRA shall address:

- Qualification of personnel performing the analysis and peer review.
- Procedures for control of documentation, including revisions.
- Provisions for independent review, verification, or checking of calculations (peer review) and information used in the analyses.
- Methods for documentation and maintenance of records.

4.4.2 The Peer Review

This peer review process needs to be commensurate with complexity and importance to safety of the PRA results. In the simplest case, it may entail an independent review by a qualified SME.

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

- Use a documented process.
- List the review topics to ensure completeness, consistency, and uniformity.
- Review the appropriateness of the PRA model.
- Review assumptions and data inputs and assesses their validity and appropriateness.
- Review the treatment and propagation of uncertainties.
- Review whether the PRA appropriately represents plant design and operations.
- Review of the utilization of industry standards.
- Evaluate the manner in which the insights gained through the PRA are integrated with and/or complement the results of DSA deterministic analyses.
- Review the process utilized to ensure quality assurance requirements were implemented.
- Review results of each PRA technical element for reasonableness.
- Review PRA maintenance and update processes.

4.4.2.1 Peer Review Team

The Peer Review Team shall be:

- Independent with no conflicts of interest that can affect the team's objectivity.
- Experts in all the technical elements of a PRA including integration with PDSA/DSA.
- Experts in the technical element assigned to review.
- Knowledgeable of the plant design, operation and maintenance.
- Knowledgeable of the DOE nuclear safety process and requirements.
- Knowledgeable of the peer review process.

4.5 Peer Review Results

The peer review results shall be documented. The following shall be described in the report:

- The peer review process.
- The scope of the peer review performed (i.e., what was reviewed by the peer review team).
- Where PRA does not meet desired characteristics and attributes.
- An assessment of the significance of vulnerabilities and deficiencies.
- The qualifications of peer review team.

5. USES OF PRAs IN DOE NUCLEAR SAFETY APPLICATIONS

Section 4 (above) provides general criteria and guidance for effective management of PRAs in nuclear safety applications. This section will provide background on DOE's processes for hazard and accident analysis at nuclear facilities, as it relates to the calculation of accident likelihoods and use of PRA. Section 5 then describes some potential ways that PRAs can be used to supplement DOE's semi-quantitative hazard and accident analysis process to support nuclear safety decisions.

5.1 Background

DOE's requirements for the analysis of facility hazards and development of safety controls are contained in 10 CFR 830 Subpart B, Appendix A, *General Statement of Safety Basis Policy*. Pertinent to the subject of this Standard, Appendix A states, in part: "…the documented safety analysis for a complex, high-hazard facility may be quite elaborate and more quantitative." This quote alludes to the semi-quantitative nature of the analysis process described in DOE Standard 3009 and incorporated into 10CFR830 Part B as a safe harbor.

Related to performance of analyses to determine the likelihood and consequences of potential accidents, DOE Standard 3009 is somewhat equivocal; for example it states that "…the graded approach ranges from a hazard analysis to a detailed quantitative analysis where formally qualified event trees, and/or fault trees, form the bases for physical phenomena modeling and engineering analysis..." However, when it moves on to discuss the calculation of estimates of the likelihood of accidents, DOE Standard 3009 states that "hazard analysis…moves beyond basic hazard identification to…estimation of likelihood of accidents…," this "in no way connotes the level of effort of a probabilistic or quantitative risk assessment."

Further, in discussing binning frequency of occurrence, DOE Standard 3009 is more definitive, stating that "detailed probability calculations are not required." It concludes with the observation that the: "principal purpose of the accident analysis is to identify any safety-class SSCs, SACs and TSRs needed for protection of the public"; safety-related uses of PRA should be similarly focused. The Department has determined that there are a number of areas where PRA insights can supplement its traditional approaches; examples are discussed below.

5.2 Evaluating Alternative Compliance Approaches

Insights resulting from PRAs can assist decision makers in evaluating alternative courses of action, each of which comply with DOE nuclear safety requirements, during design, operations, and decommissioning.

5.3 Supporting the USQ Process (PISA Process)

The USQ process requires frequent evaluations of safety adequacy; these include USQ determination, PISA evaluations, and determination of the need and acceptability of continued operations (e.g., JCOs). Results from PRA can provide additional insight and perspective to the assessment of the adequacy of safety margins.

5.4 Supplementing the Traditional Safety Methods

The PRA methodology can augment existing DOE safety assessment methods by: prioritizing safety challenges based on risk, assessing uncertainties in semi-quantitative analyses, evaluating lessons learned and operating experience, or testing the sensitivity of analytical results to key assumptions. PRA results can enhance DOE decisions on Defense in Depth (DID) by providing data and information on the importance of each control making up the DID strategy. In addition, they can also inform the design process, especially for complex, high-hazard facilities.

5.5 Evaluating Changes to DOE Safety Requirements

Risk-informed decision making for DOE safety requirements can be enhanced in the following areas:

- Proposed rulemaking to impose new requirements directed at improving safety.
- Proposed safety basis conditions of approval.
- Proposed orders or rules directed at increasing effectiveness, or furthering the Department's strategic goals other than safety, but which raise safety questions.
- Proposed exemptions or changes to existing orders or rules that might cause increases in risk.
- Applying risk information to safety basis decisions where existing guidance is silent.

Appendix A
Glossary

The following is a glossary of risk assessment terms utilized 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 related to DOE nuclear safety applications. The terms are organized from the more general to the more specific.

Term	Definition and Source
Deterministic Analysis	Analyses that use deterministic methods exercising mathematical models in which a single set of assumptions (i.e., scenario, model, and model input parameters) is used to calculate a single value of model output. [adapted from definition of deterministic methods,in NCRP 152, *Performance Assessment of Near-surface Facilities for Disposal of Low-Level Radioacitve Waste,* National Council for Radiation Protection and Measurement report 152, 2005]
PRA Application	*PRA application:* a documented analysis based in part or whole on a plant-specific PRA that is used to assist in decision making with regard to the design, licensing, procurement, construction, operation, or maintenance of a nuclear power plant. ASME/ANS-2009
PRA maintenance	*PRA maintenance:* the update of the PRA models to reflect plant changes such as modifications, procedure changes, or plant performance (data). ASME/ANS-2009
PRA Upgrade	*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. ASME/ANS-2009
PRA, Probabilistic Risk Assessment	*probabilistic risk assessment (PRA):* a qualitative and quantitative assessment of the risk associated with activities, operation and maintenance that is measured against risk metrics, such as, public or worker dose, cost to benefit goals, system reliability, cancer risk, core damage or other facility damage or a radioactive material release or specific health or safety detriments, [also referred to as a probabilistic safety assessment (PSA)]. [adapted from ASME/ANS-2009]
Probabilistic method	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	A numerical assessment of the probability and impact of the identified risks. For this standard PRA is inclusive of QRA.

Term	Definition and Source
Quality Assurance	QUALITY ASSURANCE. All those actions that provide confidence that quality is achieved. See 10 CFR Part 830 and DOE O 414,1C, (DOE-HBK-1188) The ASME/ANS-2009 risk assessment standard uses use "technical adequacy" to address quality assurance.
Risk	**Risk** – The potential to cause harm, expressed as a measure that combines the probability of an event and the consequences should an event occur (e.g., expected frequency of an undesired event in events per unit time). DOE-IN-2010
Risk Analysis	**Risk Analysis:** A process to comprehend the nature of risk and to determine the level of risk. ISO 13000
Risk Assessment	**risk assessment:** overall process of risk identification, risk analysis and risk evaluation: ISO 31000
Risk assessment tools/techniques	**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. DOE-IN-2010
Risk Evaluation	**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. ISO 13000
Risk Identification	**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. ISO 13000
Risk Metric	**T**erms of reference against which the significance of a **risk** is evaluated. Risk metrics or criteria are based on organizational objectives and **external** and **internal context** **(**environment in which the organization seeks to achieve its objectives). ISO 31000 Example metrics include public or worker dose limits or constraints, cost to benefit goals, system reliability, cancer risk levels, core damage or other facility damage, a radioactive material release levels or specific health or safety detriments. The metric may simply be a comparison of alternative actions or controls to select the most cost beneficial that can meet all health and safety requirements or is most optimal.
Risk Assessment Criteria	**See Risk Metric**
Risk Criteria	**See Risk Metric**
Screening	*Screening:* a process that eliminates items from further consideration based on their negligible contribution to the probability of an accident or its consequences. ASME/ANS-2009
Screening Criteria	*Screening Criteria:* the values and conditions used to determine whether an item is a negligible contributor to the probability of an accident sequence or its consequences. ASME/ANS-2009
Traditional Safety Assessment Methods	DOE's approach to safety as prescribed in 10 CFR Part 830, DOE directives and Standards such as DOE-STD-3009 which does not require the use of probabilistic risk assessment, although its use to supplement nuclear safety is not prohibited. (based on discussion in DOE-IN-2010)

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, aero-space industry, and the commercial nuclear power industry. These include example standards used in developing and applying PRAs, PRA procedure guides that may be used to guide the PRA development, as well as guides and standards for applying PRAs in risk-informed decision making. This is not an exhaustive list of references, but rather a representative set that may be useful in applying this standard. The user of this standard is responsible to provide the rational for the applicability of any referenced guides and standards, as set forth in the requirements of this standard.

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 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 existing commercial LWR nuclear power plants.
A more extensive list of references is provided in Table B-2 that has been 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 the ASME/ANS PRA standard whose Table of Contents is 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 in DOE-STD-3014) 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 designed to preserve deterministic principles and ensure changes in risk are small • This processed design 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 CDF and LERF • Problem framed in terms of risk of space missions; includes selection of risk metrics (performance measures
4.1.2 Risk Assessment Approach	How to structure the PRA for the application.	• Section 2.2 RG 1.174 • Section 1.2 – 1.2.2 of RG 1.200 • Section 1-1.3 of ASME/ANS-2009 • Section 3.2 NASA-2010b	• Includes evaluation of deterministic criteria and using PRA to evaluate changes in CDF and LERF • Discusses technical characteristics and attributes of internal event PRA's • 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
4.1.3 Results, Conclusion and Uses	How to establish risk acceptance criteria.	• DOE-1991 and 2010 • NRC-1986a • Section 2.2.4 of RG 1.174 • Section 3.3.1 of NASA-2010b	• DOE nuclear safety policy with quantitative safety goals • NRC equivalent of DOE-1991 for commercial nuclear power plants • 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
	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
	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

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

4.1.4 Quality Assurance	What is the scope of the Quality Assurance Program?	• Section 2.5 of RG 1.174 • Section 3 of RG 1.200 • Section 1.4 of ASME/ANS-2009	• QA includes use of qualified personnel, procedures to guide the work, independent and peer reviews, documentation of the PRA application, QA records, and corrective action program • 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-2009 and NRC-2009 • NRC-1983a • NASA-2002b and NASA-2010a	• PRA methodology for chemical and process industries • Requirements for PRAs for risk-informed applications; tailored to operating LWR plants and focused on risk metrics of CDF and LERF for baseline PRAs; most requirements applicable to non-LWR and non-reactor PRAs; PRA scope covered under continuous expansion • General Methodology for PRAs on nuclear power plants • PRA methodology for space applications
	What are the specific guides and standards for treating special topics in PRA?	• See references in Table 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 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-2009 and NRC-2009	• 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
4.4 QA and Peer Review			
4.4.1 Quality Assurance	What are the available guidance for PRA model configuration control, PRA maintenance, updates, and upgrades?	• ASME/ANS-2009	• Section 1-1.5 provides general requirements for configuration control; Appendix 1-A provides guidance for PRA maintenance, PRA upgrades, and associated peer reviews
4.4.2 The Peer Review 4.4.3 Peer Review Results	What are the available guides and standards to plan and conduct and document the independent reviews?	• Section 2.2 of RG 1.174 • ASME/ANS-2009 • NEI-00-02, NEI-05-04, NEI-07-12	• Describes expectations for the peer review process, personnel qualifications, and documentation of results. • Section 1-1.6 provides general requirements for peer review, the each part of the Standard, Parts 2-10 has specific peer review requirements for each hazard group within the PRA scope, e.g. internal events (Part 2), Internal floods (Part 3), … • Nuclear industry guides for performing PRA peer reviews

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries

Reference ID	Reference	Topic
Standards for PRA and Risk Informed Decision Making		
DOE-1991	SEN-35-91, U.S. Department of Energy, 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-2010	DOE-P420.X (Draft), Department of Energy Nuclear Safety Policy	Draft revision to DOE-1991, includes same safety goals
DOE-IN-2010	DOE Information Notice, *Risk Assessment in Support of Nuclear Safety,* DOE Office of Nuclear Safety Policy and Assistance, June 2009.	Describes DOE expectations with regard to DOE's use of risk assessment use to better inform Nuclear Safety decisions.
NRC-1986a	USNRC, "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," Federal Register, Vol. 51, p. 30028 (51 FR 30028), August 4, 1986.	Risk acceptance criteria (safety goals and Quantitative Health Objectives) for NPP accidents
ASME/ANS-2009	ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RA-S-2008, ASME, New York, NY, American Nuclear Society, La Grange Park, Illinois, February 2009.	PRA Standards for LWRs
ISO-13000	International Standard, ISO 13000:2009(E), *Risk management – Principles and guidelines,* first edition 11/15/2009*,*	Describes process and terms for integrating risk management into decision making through an organizations overall operations and activities.
ISO-13010	International Standard, IEC/ISO 13010, *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 Risk Management Procedural Requirements, December 2008) and NPR 7120.5D (NASA Space Flight Program and Project Management Requirements)	NASA Risk Management Requirements
NASA-2008b	NPR 8715.3C, NASA General Safety Program Requirements, March 2008	NASA Safety Requirements
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
NRC-0800	U.S. NRC, NUREG-0800, "Standard Review Plan for the Review of the Safety Analysis Reports for Nuclear Power Plants," Section 19, "Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance," U.S. Nuclear Regulatory Commission, Washington, DC.	Review guidance

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries

BSR/ANS-227	BSR/ANS-2.27: American Nuclear Society, “Guidelines for Investigations of Nuclear Facility Sites for Seismic Hazard Analysis” (Draft Standard)	Seismic Hazard Analysis
BSR/ANS-229	BSR/ANS-2.29: American Nuclear Society, “Probabilistic Seismic Hazards Analysis” (Draft Standard)	Seismic Hazard Analysis
Non-Reactor PRA Applications		
AICE-2000	Guidelines for Chemical Process Quantitative Risk Analysis, Second Edition, American Institute of Chemical Engineers, New York, NY. 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
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-2005b	Risk-Informed Decision-Making for Nuclear Materials and Waste Applications. Draft for Trial Use. MAY 11, 2005	
EPRI-1995a	EPRI TR-105396, PSA Applications Guide; D.True, et al.; August 1995; Publisher: The Electric Power Research Institute (EPRI), 3412 Hillview Avenue, Palo Alto, CA 94304	Risk-informed Decision Process
NASA-2010b	NASA/SP-2010-576, NASA Risk-Informed Decision Making Handbook, April 2010	Risk Informed Decision Process
RG-1.174	U.S. NRC, Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” U.S. Nuclear Regulatory Commission, Washington, DC. , Nov 2002	Risk-informed Decision Process
RG-1.175	U.S. NRC, Regulatory Guide 1.175, “An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing,” U.S. Nuclear Regulatory Commission, Washington, DC., Aug-1998	Risk-informed IST
RG-1.177	U.S. NRC, Regulatory Guide 1.177, “An Approach for Plant-Specific, Risk-Informed Decision making: Technical Specifications,” U.S. Nuclear Regulatory Commission, Washington, DC. , Aug 1998	Risk-informed TS
RG-1.178	U.S. NRC, Regulatory Guide 1.178, “An Approach for Plant-Specific, Risk-Informed Decision making: In service Inspection of Piping,” U.S. Nuclear Regulatory Commission, Washington, DC. , September 2003	Risk-informed ISI
Guidance for PRA Peer Reviews		
NEI-00-02	NEI 00-02, “Probabilistic Risk Assessment Peer Review Process Guidance,” Revision A3, Nuclear Energy Institute, Washington, DC, March 20, 2000.	Peer review procedures
NEI-05-04	NEI 05-04, “Process for Performing Follow-On PRA Peer Reviews Using the ASME PRA Standard,” Revision 2, Nuclear Energy Institute, Washington, DC, November 2008.	Peer review procedures
NEI-07-12	NEI 07-12, “Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines,” Draft Version H, Revision 0, Nuclear Energy Institute, Washington, DC, November 2008.	Peer review procedures
Guidance for PRA Methodology		
Chem-2005	*Process Hazard Analysis, Bow-Tie Methodology*, J. Philley, (2005); (http://www.chemicalprocessing.com/articles/2005/612.html)	Hazards analysis method for chemical industry
NASA-2002b	NASA, Probabilistic Risk Assessment Procedures Guide for NASA Managers and Practioners, Version 1.1, August 2002	NASA PRA Procedures

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries

NASA-2010a	NPR 8707.5A, Technical Probabilistic Risk Assessment (PRA) Procedures for Safety and Mission Success for NASA Programs and Projects, June 2010	NASA PRA Procedures
NRC-1975	Wash-1400, The Reactor Safety Study, 1975 (also known as NUREG-75/014); Publisher: U.S. Nuclear Regulatory Commission (NRC), 11555 Rockville Pike, Rockville, MD 20852	LWR PRA Case Study; first PRA on LWR power plants
NRC-1983a	NUREG/CR-2300, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, 1983; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	PRA Procedures Guide extensively used in commercial nuclear plants
NRC-1983b	NUREG/CR-2728, Interim Reliability Evaluation Program Procedures Guide, March 3, 1983; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	PRA Procedures
NRC-1985b	M. McCann, J. Reed, C. Ruger, K. Shiu, T. Teichmann,A. Unione, and R. Youngblood, “Probabilistic Safety Analysis Procedures Guide,” Report NUREG/CR-2815, Vol. 2, Brookhaven National Laboratory and U.S. Nuclear Regulatory Commission (1985)	PRA Procedures
NRC-1990a	NUREG 1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, December 1990; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	PRA Procedures
NRC-2003a	Fleming, Karl N., “Issues and Recommendations for Advancement of PRA Technology for Risk Informed Decision Making”, prepared by Technology Insights for U.S. NRC Advisory Committee on Reactor Safeguards, NUREG/CR-6813, January 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, August 2002	Fault tree Procedures for Aerospace
NRC-1981b	U.S. Nuclear Regulatory Commission (USNRC), “Fault Tree Handbook,” NUREG-0492, Washington, D.C., 1981.	Fault tree Procedures
NRC-1998f	NUREG/CR-5485, Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment, November 20, 1998; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	CCF Modeling in Fault trees
PRA Method for Special Topics – Data Base 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	Roy, B. N., “Savannah River Site Generic Database Development”, WSRC-TR-93-262, Rev. 1, Westinghouse Safety Management Solutions, Aiken, SC, May 1998.	Failure rate estimates for Savanah River
NASA-2009	NASA/SP-2009-569, Bayesian Inference for NASA Probabilistic Risk and Reliability Analysis", June 2009	Bayes methods for analyzing data and treatment of uncertainties in NASA PRAs
EGG-1990	S. A. Eide et al, “Generic Component Failure Databases for Light Water and Liquid Sodium Reactor PRAs”, EGG-SSRE-8875, Idaho National Engineering Laboratory, Idaho Falls, ID, February 1990.	Generic failure rate data for LWRs and liquid metal reactors
NRC-1994	NUREG/CR-4639, Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR), Vols. 1–5, 1994; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	Generic data for PRA

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries

NRC-1997b	NUREG/CR-5496, Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1986; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	Loss of offsite power data for LWRs
NRC-1998b	NUREG/CR-5032, Modeling Time to Recover and Initiate Even Frequency for Loss-of-Offsite Power Incidents at Nuclear Power Plants, March 1988; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	Power recovery data for LWRs
NRC-1999a	NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants, Idaho National Engineering and Environmental Laboratory, Idaho Falls, February 1999; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	Initiating event data for LWRs
NRC-2003b	NUREG/CR-6823, Handbook of Parameter Estimation for Probabilistic Risk Assessment, Sandia National Laboratories, et al., September 2003; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	Data analysis methodology
NRC-2007	NUREG/CR-6928, Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants, Idaho National Laboratory, Idaho Falls, ID, February 2007; Publisher: U.S. Nuclear RegulatoryCommission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	Generic failure rate data for LWRs
NRC-2008	Tregoning, R., L. Abramson, and P. Scott, Estimating Loss-of-Coolant Accident (LOCA) Frequencies Through the Elicitation Process, NUREG-1829, U.S. Nuclear Regulatory Commission, Washington, D.C., April 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", 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	Mosleh, A., K. N. Fleming, et al., "Procedures for Treating Common Cause Failures in Safety and Reliability Studies," Pickard, Lowe and Garrick, Inc., prepared for U.S. Nuclear Regulatory Commission and Electric Power Research Institute, NUREG/CR-4780, April, 1987.	PRA methods for CCF
NRC-1998d	NUREG/CR-5497, Common-Cause Failure Parameter Estimations, 1998; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	CCF Parameter Estimates
NRC-1998e	NUREG/CR-6268, Common Cause Failure Database and Analysis System, Vols. 1–4, 1998; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852 113	CCF Data Analysis Method
DOE-1996b	DOE. 1996b. Project Reviews. Good Practice Guide, GPG-FM-015. Office of Field Management. Washington, D.C.: U.S. Department of Energy.	Project Reviews
PRA Methods for Special Topics – Human Reliability Analysis		
WSRC-1994	Savannah River Site Human Error Data Base Development for Nonreactor Nuclear Facilities (U), WSRC-TR-93-581, February 1994.	Human error rates from Savanah River Service Data
EPRI-2008	Julius, J., J. Grobbelaar, D. Spiegel and F. Rahn. HRA Calculator 4.0 – Human Reliability Analysis Calculator User's Manual, 1015358, Electric Power Research Institute, Palo Alto (CA), February 2008.	HRA PRA Methodology
NRC-1983c	NUREG/CR-1278 Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications; A. D. Swain and H. E. Guttmann; August 1983 (THERP); Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	HRA PRA Methodology

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries

NRC-1987c	NUREG/CR-4772, Accident Sequence Evaluation Program Human Reliability Analysis Procedure; A.D. Swain; February 1987 (ASEP); Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	HRA PRA Methodology
NRC-2000	Barriere, M. et al. Technical basis and Implementation Guidelines for A Technique for Human Event Analysis (ATHEANA), NUREG-1624, Revision 1, U.S. Nuclear Regulatory Commission, Washington (DC), May 2000.	HRA PRA Methodology
NRC-2005a	NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA),” U.S. Nuclear Regulatory Commission, Washington, DC, April 2005.	HRA PRA Methodology
NRC-2006	NUREG-1842, “Evaluation of Human Reliability Analysis Methods Against Good Practices,” U.S. Nuclear Regulatory Commission, Washington, DC, September 2006.	HRA PRA Methodology
WSMS-2009	Benhardt, H.C., Human Reliability Analysis, WSMS-SAE-M-09-0014, December 3, 2009.	HRA PRA Methodology
PRA Methods for Special Topics – Internal Flooding PRA		
EGG-1991	Eide, S.A., S.T. Khericha, M.B. Calley, D.A. Johnson and M.L. Marteeny. Component External Leakage and Rupture Frequency Estimates, EGG-SSRE--9639, Idaho National Engineering Laboratory, November 1991.	Pipe Failure Data for flood PRA
EPRI-2009	Fleming, K. N. and B. O. Y. Lydell, "Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment". EPRI, Palo Alto, CA: 2009. 1019194.	PRA Procedures for internal flood PRA
EPRI-2010	Fleming, K. N. and B. O. Y. Lydell, "Pipe Rupture Frequencies for Internal Flooding PRAs", Revision 2. EPRI, Palo Alto, CA: 2010. 1021086.	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	EPRI TR-100370, Fire-Induced Vulnerability Evaluation (FIVE), May 1992; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304	Fire PRA for LWRs
EPRI-1995b	EPRI TR-105928, Fire PRA Implementation Guide, December 1995; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304	Fire PRA for LWRs
EPRI-1997	EPRI/NRC 97-501, “Review of the EPRI Fire PRA Implementation Guide,” Letter Report to the U.S. Nuclear Regulatory Commission, August 1997; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304	Fire PRA for LWRs
EPRI-2005	EPRI TR-1011989 and NUREG/CR-6850: EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities. Electric Power Research Institute (EPRI), Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD: 2005	Fire PRA for LWRs

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries

NRC-2004	NUREG-1805, Fire Dynamics Tools (FDTs): Quantitative Fire Hazard Analysis Methods for the U.S.Nuclear Regulatory Commission Fire Protection Inspection Program, December 2004; Publisher:U.S. Nuclear Regulatory Commission (NRC), 11555 Rockville Pike, Rockville, MD 20852	Fire PRA for LWRs
PRA Methods for Special Topics – External Events Screening		
NRC-1989b	NUREG/CR-4840, “Recommended Procedures for the Simplified External Event Risk Analyses for NUREG-1150,” September 1989; Publisher: U.S. Nuclear Regulatory Commission (NRC), 11555 Rockville Pike, Rockville, MD 20852	PRA Procedures For Screening of external events
NRC-1992	M. K. Ravindra and H. Bannon, “Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development,” Report NUREG/CR-4839, Sandia National Laboratories and U.S. Nuclear Regulatory Commission (1992)	PRA Procedures For Screening of external events
NRC-1998a	M. P. Bohn and J. A. Lambright, “Procedures for the External Event Core Damage Frequency Analyses for NUREG-1150,” Report NUREG/CR-4840, SAND88-3102, Sandia National Laboratories and U.S. Nuclear Regulatory Commission (1988)	PRA Procedures For Screening of external events
PRA Methods for Special Topics – Aircraft Crash		
DOE-1996b	DOE-STD-3014-96:U.S. Department of Energy, “Accident Analysis for Aircraft Crash Into Hazardous Facilities” (1996)	DOE Standard for aircraft crashes
PRA Methods for Special Topics – Seismic PRA		
NRC-1985a	L. C. Shieh, J. J. Johnson, J. E. Wells, J. C. Chen, and P. D. Smith, “Simplified Seismic Probabilistic RiskAssessment: Procedures and Limitations,” Report NUREG/CR-4331, Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission (1985)	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	NTS Engineering, RPK Structural Mechanics Consulting, Pickard Lowe & Garrick, Woodward Clyde Consultants, and Duke Power Company, “A Methodology for Assessment of Nuclear Power Plant Seismic Margin,” Report EPRI NP-6041-SL, Rev. 1, Electric Power Research Institute (1991).	Seismic PRA method
NRC-1985c	R. J. Budnitz, P. J. Amico, C. A. Cornell, W. J. Hall, R. P. Kennedy, J. W. Reed, and M. Shinozuka, “An Approach to the Quantification of Seismic Margins in Nuclear Power Plants,” Report NUREG/CR-4334, Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission (1985).	Seismic PRA method
NRC-1997d	R. J. Budnitz, D. M. Boore, G. Apostolakis, L. S. Cluff, K. J. Coppersmith, C. A. Cornell, and P. A. Morris, “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts,” Report NUREG/CR-6372, U.S. Nuclear Regulatory Commission (1997).	Seismic Hazard Analysis, Expert Elicitation
DOE-1998	Civilian Radioactive Waste Management System Management and Operating Contractor, “Probabilistic Seismic Hazard Analyses for Fault Displacement and Vibratory Ground Motion at Yucca Mountain, Nevada,” U.S. Department of Energy, DE-AC04-94AL85000, in three volumes, prepared for the U.S. Geological Survey (1998)	Seismic hazard Analysis for Yucca Mountain
EPRI-2004	EPRI, “CEUS Ground Motion Project Final Report,” Technical Report 100984, Electric Power Research Institute (2004)	Seismic Hazard Analysis

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries

NRC-1997c	NUREG/CR-6372, Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on the Use of Experts; R. J. Budnitz, G. Apostolakis, D. M. Boore, L. S. Cluff, K. J. Coppersmith, C. A. Cornell, and P. A. Morris; U.S. NRC and Lawrence Livermore National Laboratory, 1997; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	Seismic Hazard Analysis Expert Elicitation
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	J. W. Reed and R. P. Kennedy, “Methodology for Developing Seismic Fragilities,” Report TR-103959, Electric Power Research Institute (1994)	Seismic Fragility method
EPRI-1994b	J. W. Reed and R. P. Kennedy, “Methodology for Developing Seismic Fragilities,” Report TR-103959, Electric Power Research Institute (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; ublisher: Elsevier Science, P. O. Box 945, New York, NY 10159	Seismic Fragility method
Kennedy-1984b	R. P. Kennedy and M. K. Ravindra, “Seismic Fragilities for Nuclear Power Plant Risk Studies,” Nuclear Engineering and Design, Vol. 79, No. 1, pp. 47–68 (May 1984)	Seismic Fragility method
	PRA Methods for Special Topics – External Flooding PRA	
LLNL-1998	M. W. McCann, Jr., and A. C. Boissonnade, “Probabilistic Flood Hazard Assessment for the N Reactor, Hanford, Washington,” Report UCRL-2106, Lawrence Livermore National Laboratory (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	“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, National Academy of Sciences (1988)	External flood PRA
STA-1985	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.,” Stanford University Department of Civil Engineering, 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, Stanford University Department of Civil Engineering (1985)	External flood PRA
WDC-1986	“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 onWater Data,U.S. Office ofWater Data Coordination (1986)	External flood PRA
	PRA Methods for Special Topics – High Winds PRA	
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

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries

NRC-1982	T. A. Reinhold and B. Ellingwood, “Tornado Damage Risk Assessment,” Report NUREG/CR-2944, The Johns Hopkins University, Baltimore, Maryland (1982)	PRA of High Winds
NRC-1986b	J. V. Ramsdell and G. L. Andrews, “Tornado Climatology of the Contiguous United States,” Report NUREG/CR-4461, Battelle Pacific Northwest Laboratories and U.S. Nuclear Regulatory Commission (1986)	PRA of High Winds
NRC-1987a	J. W. Reed and W. L. Ferrell, “Extreme Wind Analysis for the Point Beach Nuclear Power Plant,”Appendix G in Shutdown Decay Heat Removal Analysis of a Westinghouse 2-Loop PWR,” Report NUREG/CR-4458, Sandia National Laboratories andU.S. Nuclear Regulatory Commission (1987)	PRA of High Winds
NRC-1990b	M. K. Ravindra and A. M. Nafday, “State-of-the-Art and Current Research Activities in Extreme Winds Relating to Design and Evaluation of Nuclear Power Plants,” Report NUREG/CR-5497, UCID-21933-Rev-1,Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission (1990)	PRA of High Winds
Ravindra-1997	M. K. Ravindra, Z. M. Li, P. Guymer, D. Gaynor, and A. DiUglio, “HighWind 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	Reinhold, T.A. and B. Ellingwood, “Tornado Damage Risk Assessment,” 1982	PRA of High Winds
Twinsdale-1995	L. A. Twisdale and P. J. Vickery, “ExtremeWind 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	BA000-1717-2200-00082, Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada, U.S. Department of Energy Yucca Mountain Project, Geomatrix Consultants, Inc., 1996; Publisher: U.S. Department of Energy Yucca Mountain Project, P.O. Box 364629, North Las Vegas, NV 89036	PRA for Volcano Hazard at Yucca mountain
NRC-1996	NUREG/CR-1563, Branch Technical Position on the Use of Expert Elicitation in the High-Level RadioactiveWaste Program; J. P. Kotra, M.P. Lee, N.A. Eisenberg, and A. R. DeWispelare; U.S. NRC Office of Nuclear Materials Safety and Safeguards, 1996; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	PRA for High level waste application
NRC-1997d	R. J. Budnitz, D. M. Boore, G. Apostolakis, L. S. Cluff, K. J. Coppersmith, C. A. Cornell, and P. A. Morris, “Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts,” Report NUREG/CR-6372, U.S. Nuclear Regulatory Commission (1997).	Seismic Hazard Analysis, Expert Elicitation
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PRA Methods for Special Topics – Probabilistic Treatment of Phenomena		
BNL-2006	Diamond, D.J., “Experience Using Phenomena Identification and Ranking Technique (PIRT) for Nuclear Analysis,” BNL-76750-2006-CP, Brookhaven National Laboratory, PHYSOR-2006 Topical Meeting, Vancouver, British Columbia, Canada, September 10-14, 2006.	Method for evaluating phenomena
NRC-1998c	NUREG-1570, Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture, March 1998; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	Phenomenological probabilities for LWR Severe Accidents

Table B-2 Reference Documents for PRA Guides and Standards in Different Industries

NRC-1999a	NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events, January 1999; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852	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; Publisher: Elsevier Applied Science, Essex, England.	Methods for treatment of uncertainty
Morgan-1990	Morgan, M. G. and M. Henrion, 1990, Uncertainty; A guide to dealing with uncertainty in quantitative risk and policy analysis, Cambridge University Press.	Methods for treatment of uncertainty
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WSRC-2002	Cook, J.R., et al., "Development of Probabilistic Uncertainty Analysis for SRS Performance Assessments Maintenance Plan Activities," WSRC-TR-2002-00121, March 2002.	Uncertainty analysis at Savanah River Plant
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