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NUREG-2201

Probabilistic Risk Assessment and Regulatory Decisionmaking:

Some Frequently Asked Questions

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Manuscript Completed: April 2016
Date Published: September 2016

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ABSTRACT

Probabilistic risk assessment (PRA) is an important decision-support tool at the U.S. Nuclear Regulatory Commission. The availability of experiential data for accidents, including those at the Fukushima Dai-ichi nuclear power plant, raises natural questions regarding the need for and utility of PRA, which is, at heart, a systems modeling-based analytical approach. This report addresses these questions using the format of frequently asked questions (FAQs). The FAQs are organized into four topic categories: regulatory decisionmaking, PRA basics, core damage frequency state of knowledge, and large early release frequency state of knowledge. For each FAQ, the report provides both a brief answer and a supplementary discussion elaborating on that answer. The report also includes two appendices, which provide additional technical details on quantifying uncertainty and updating data based on experience.

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FOREWORD

The U.S. Nuclear Regulatory Commission (NRC) has had a long history of using probabilistic risk assessment (PRA) models and results in support of its regulatory processes and decisions. In the late 1970s, shortly after the completion of the seminal Reactor Safety Study (WASH-1400), the NRC staff was using insights from probabilistic analyses in its consideration of such diverse topics as the likelihood of loss-of-coolant accidents, the reliability of direct current power supplies, and the effectiveness of alternate containment designs. By the early 1980s, the NRC staff was considering risk arguments in support of licensee requests to extend equipment outage times. In 1985, the Commission used information from licensee-sponsored PRAs to inform its decision to allow continued operation of the Indian Point power plants. In 1995, the Commission formally adopted its PRA Policy Statement, which promoted the use of PRA technology in the NRC's activities. Consistent with this policy statement, the NRC developed and implemented numerous risk-informed programs, (i.e., programs that use risk information as one input for regulatory decisionmaking). Currently, the NRC is continuing to explore opportunities for expanding its risk-informed activities.

In 2011, following the Fukushima Dai-ichi reactor accidents, some critics raised concerns regarding the realism of the PRAs that provide the technical information needed in the NRC's risk-informed programs. Two principal complaints raised were (1) PRA-based estimates of the likelihood of major accidents were significantly smaller than simple statistical estimates based on international events (notably the accidents at Three Mile Island, Chernobyl, and Fukushima), and (2) PRAs did not predict the accident scenario experienced at Fukushima. In 2013, concerned that the first complaint was based on an oversimplified analysis and that the second reflected a lack of understanding of the nature of PRA models, NRC Commissioner George Apostolakis [1] initiated a project to address these complaints. Recognizing that direct rebuttals without background information concerning both regulatory decisionmaking and PRA would likely not be effective for a broad audience, he directed the project team [2] to develop some "frequently asked questions" (FAQs) and participated actively in the development process. In March 2014, he presented a speech at the NRC's Regulatory Information Conference (RIC) based on the project results to date.[3]

This report provides the FAQs developed by Commissioner Apostolakis and his team, some of which have been updated to reflect the authors' perspectives and the NRC staff comments, as well as a few additional FAQs. The FAQs are not necessarily the "last word" on the topics addressed. In some cases, they provide the authors' views on current topics whose resolution is still evolving. Furthermore, as with all FAQs, the FAQs in this report do not provide detailed discussions of the topics, nor does the list of FAQs completely cover all topics of potential interest. The FAQs do provide concise, broadly oriented discussions for each topic and references for further investigation.

This report should be useful to PRA analysts (experienced as well as novice), decisionmakers, and staff members curious about some of the fundamental concepts behind PRA and risk-informed decisionmaking.

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1 The Honorable George Apostolakis served as an NRC Commissioner from March 29, 2010, through June 30, 2014.

2 The project team members were Nathan Siu, Nanette Gilles, and Belkys Sosa. The project was also aided by several NRC staff, acknowledged in this report.

3 Apostolakis, G.E., "Global statistics vs. PRA results: which should we use?" Regulatory Information Conference (RIC) 2014, March 11–13, 2014.

ACKNOWLEDGMENTS

The authors gratefully acknowledge the project vision and early guidance from Commissioner G. Apostolakis, the information provided by J. Circle, D. Coe, K. Coyne, D. Dube, F. Ferrante, S. Laur, D. Marksberry, F. Schoefer, A. Szabo, M. Tobin, T. Wellock, and S.-M. Wong, and the helpful comments provided by staff reviewers.

ACRONYMS AND ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
ASP	Accident Sequence Precursor
ANS	American Nuclear Society
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
Bq	becquerel(s)
BWR	boiling water reactor
CCF	Common Cause Failure
CCDP	conditional core damage probability
CDF	core damage frequency
CFR	*Code of Federal Regulations*
Ci	curie(s)
Cs-137	cesium-137
CSNI	Committee on the Safety of Nuclear Installations
EPRI	Electric Power Research Institute
ESREL	European Safety and Reliability Conference
FAQ	frequently asked question
GAO	General Accounting Office
hr	hour
I-131	iodine-131
IAEA	International Atomic Energy Agency
ICOSSAR	International Conference on Structural Safety & Reliability
IEEE	Institute of Electrical and Electronics Engineers
INES	International Nuclear and Radiological Event Scale
IPE	individual plant examination
IPEEE	individual plant examination(s) of external events
JANTI	Japan Nuclear Technology Institute
LAR	license amendment request
LERF	large early release frequency
LP&S	low power and shutdown
MAAP	Modular Accident Analysis Program
MCi	megacurie(s)
MELCOR	Methods for Estimation of Leakages and Consequences of Releases
MSPI	Mitigating Systems Performance Index
MWe	megawatt(s) electric

NEA	Nuclear Energy Agency
NFPA	National Fire Protection Association
NOED	Notice of Enforcement Discretion
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
OECD	Organisation for Economic Co-operation and Development
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PSAM	probabilistic safety assessment and management
PTC	PTC, Inc. (formerly Parametric Technology Corporation)
PTS	pressurized thermal shock
PWR	pressurized-water reactor
QHO	quantitative health objective
R&D	research and development
RES	NRC Office of Nuclear Regulatory Research
RG	regulatory guide
RIC	Regulatory Information Conference
RISC	risk-informed safety class
ROP	Reactor Oversight Process
RY	reactor-year(s)
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
SBO	station blackout
SOARCA	State-of-the-Art Reactor Consequence Analyses
SPAR	Standardized Plant Analysis Risk
SSCs	structures, systems, and components
TBq	terabecquerel(s)
TEPCO	Tokyo Electric Power Company
TMI	Three Mile Island
TMI-2	Three Mile Island Unit 2
WGOE	Working Group on Operating Experience
yr	year

1. INTRODUCTION

A regulatory decisionmaker charged with ensuring the safe operation of U.S. nuclear power plants is often confronted with problems in which decisions have to be made based on what is currently known about the risks associated with different decision alternatives. For major decisions, the available information is usually both complex and subject to large uncertainties. These uncertainties arise, in part, because the decision problem usually requires consideration of rare events for which significant empirical data are lacking. Uncertainties also arise because of the diversity of the regulated facilities.

Probabilistic risk assessment (PRA) is an important decision-support tool at the U.S. Nuclear Regulatory Commission (NRC).[1] Following the March 11, 2011, Fukushima Dai-ichi reactor accidents, there have been a number of published estimates on the frequency of major reactor accidents based on global experience. The availability of experiential data for accidents raises natural questions regarding the need for and utility of PRA, which is, at heart, a systems modeling-based analytical approach. In this report, these questions are addressed via the format of frequently asked questions (FAQs).

The FAQs in this report are organized into four topic categories: regulatory decisionmaking, PRA basics, core damage frequency (CDF) state of knowledge, and large early release frequency (LERF) state of knowledge. For each FAQ, the report provides both a brief answer and supplementary discussion elaborating on that answer. The report also includes two appendices, which provide additional technical details on key topics.

[1] As discussed later in this report, NRC uses PRA results in a risk-informed decisionmaking framework. In particular, PRA results are not the sole basis for NRC decisionmaking.

2. REGULATORY DECISIONMAKING

At the U.S. Nuclear Regulatory Commission (NRC), probabilistic risk assessment (PRA) is used as a decision-support tool. Before delving into the details of PRA, it is important to discuss the decisionmaking context for PRA. This section provides the following frequently asked questions (FAQs) addressing the reasons PRAs are used at the NRC.

R1. What information does the NRC use in regulatory decisionmaking?
R2. What is the NRC's definition of the term "risk"?
R3. What is the relation between safety and risk?
R4. What are the NRC's safety goals?
R5. What is the relationship of surrogate risk measures to the safety goals?
R6. What is the NRC's philosophy regarding the use of risk information?
R7. What is the NRC's policy regarding the use of PRA?
R8. Where does the NRC use PRA in regulatory decisionmaking?
R9. Why does the NRC use PRA in regulatory decisionmaking?
R10. If nuclear power plant accident statistics are available, is there a need for PRA?

R1. What information does the NRC use in regulatory decisionmaking?

NRC decisionmakers need evaluations of options that typically cover a broad range of technical and nontechnical factors. The technical evaluation needs to appropriately incorporate the current state of knowledge regarding the risk associated with each decision option.

As with any decisionmaking process, nuclear regulatory decisionmaking requires the clear identification of decision options and the evaluation of these options. For major decisions, the evaluation typically involves the consideration of a broad variety of technical and nontechnical decision factors in an open, deliberative process as described in NUREG-2150 (NRC, 2012a) and reproduced in Figure 2-1.

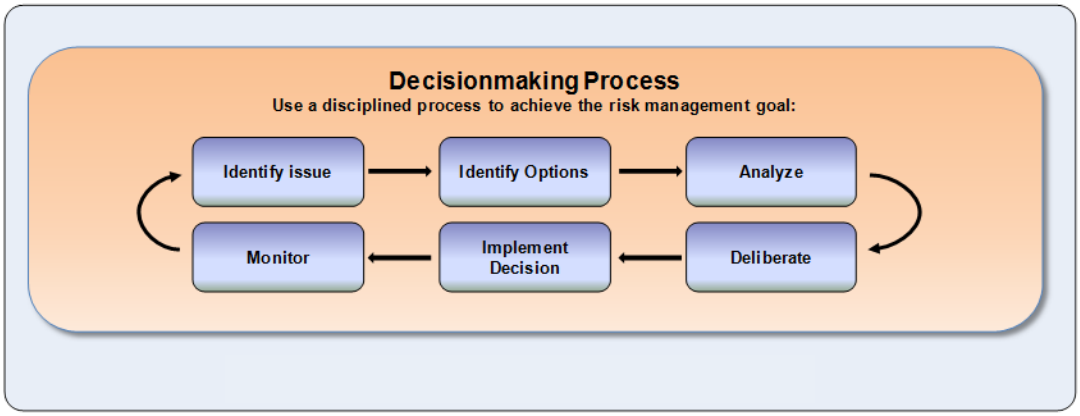

Figure 2-1 The regulatory decisionmaking process (NRC, 2012a)

Regarding technical factors, an important aspect of the NRC's decisionmaking is that it proactively deals with possible but not-yet-experienced events. The management of the risk [1] associated with such events, whether done using implicit considerations of risk (as is the case with prescriptive, conservative approaches with built-in but unquantified safety margins) or using explicit considerations of risk (as in risk-informed approaches), needs to be based on what is currently known about these events (including their likelihood, potential consequences, and causative factors). In some cases, the events of interest are rare (i.e., empirical data are sparse or even nonexistent), and the always-present need to address uncertainty, whether because of the diversity of the licensed population or to limitations in the technical community's state of knowledge, becomes even more important.

R2. What is the NRC's definition of the term "risk"?

The risk associated with a facility or operation is the combined set of answers to three questions: "What can go wrong?" "How likely is it?" and "What are the consequences?"

The dictionary includes several definitions of the term "risk." The "risk triplet" in the answer above was first articulated by Kaplan and Garrick (1981); the NRC has used it ever since (e.g., NRC, 1998a, 2013c). The risk triplet highlights the importance of qualitative outputs from a risk assessment, most importantly the descriptions of accident sequences (the answer to the question "What can go wrong?"). It also differentiates high-probability, low-consequence events

[1] See FAQ R2 for further discussion of the term "risk."

from low-probability, high-consequence events. Both of these features are often important in managing risk.

It should be recognized that, even within the PRA community (and in some NRC documents), risk is commonly characterized as the product of probability and consequence.[2] This definition, which long predates the Kaplan and Garrick triplet definition, has the virtue of enabling simple comparisons with numerical safety goals. However, it does not provide the full breadth of information coming from the triplet definition.

FAQ R6 addresses the NRC's philosophy regarding the use of risk information in regulatory decisionmaking.

R3. What is the relation between safety and risk?

In general, an activity is perceived to be safe if its perceived risks are judged to be acceptable. Quantitative risk estimates provide an important measure of nuclear power plant safety, but do not embody the full range of considerations that enter into the NRC's judgments regarding reasonable assurance of adequate protection.

The term "safety" has different connotations to different people. Then-NRC Chairman S. Jackson provides an NRC perspective on the relation between safety and risk:

> As commonly understood, safety means freedom from exposure to danger, or protection from harm. In a practical sense, an activity is deemed to be safe if the perceived risks are judged to be acceptable. The Atomic Energy Act of 1954, as amended, establishes "adequate protection" as the standard of safety on which NRC regulation is based. In the context of NRC regulation, safety means avoiding undue risk or, stated another way, providing reasonable assurance of adequate protection for the public in connection with the use of source, byproduct and special nuclear materials. (NRC, 1997a)

SECY-99-246 further expands on this relationship, emphasizing that risk is one of a number of different indicators of safety:

> Quantitative (absolute) risk estimates serve as an important measure of plant safety, but do not embody the full range of considerations that enter into the judgment regarding adequate protection. The judgment regarding adequate protection derives from a more diverse set of considerations, such as acceptable design, construction, operation, maintenance, modification, and quality assurance measures, together with compliance with NRC requirements including, license conditions, orders, and regulations. (NRC, 1999)

FAQ R4 describes the NRC's safety goals, which employ both qualitative and quantitative representations of risk.

[2] More precisely, this definition considers the sum of the probability-consequence products for all scenarios considered. This sum is the expected (i.e., average) value of the "risk" as defined by the triplet definition.

R4. What are the NRC's safety goals?

The NRC's safety goals broadly define an acceptable level of radiological risk. As a result, the safety goals partially address a fundamental regulatory concern: How safe is safe enough?

The safety goals, which are provided in the NRC's Safety Goal Policy Statement (NRC, 1986), broadly define an acceptable level of radiological risk. The NRC has defined two qualitative safety goals which are supported by two quantitative objectives. These two supporting objectives are based on the principle that nuclear risks should not be a significant addition to other societal risks. The qualitative safety goals are:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant (NPP) operation such that individuals bear no significant additional risk to life and health.

- Societal risks to life and health from NPP operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative objectives are used to determine achievement of the safety goals:

- The risk to an average individual in the vicinity of an NPP of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks to which members of the U.S. population are generally exposed.

- The risk to the population in the area near an NPP of cancer fatalities that might result from NPP operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

R5. What is the relationship of surrogate risk measures to the safety goals?

Surrogate risk measures provide an approximate method for determining when the safety goals are met. Because surrogate risk measures are easier to compute than quantitative risk estimates, they are useful when making certain types of risk-informed decisions.

The NRC uses two surrogate risk measures that are directly related to the safety goals: core damage frequency (CDF) and large early release frequency (LERF). These terms are defined in the PRA standard endorsed by the NRC (ASME and ANS, 2009) and in NUREG-2122 (NRC, 2013c):

- Core damage [3] frequency (CDF): The frequency [4] of accidents that cause uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects. CDF is the surrogate risk measure for individual latent cancer

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[3] For a definition and discussion of the term "core damage," see FAQ C1.
[4] For a definition and discussion of the term "frequency," see FAQ B3.

fatality risk (the second quantitative objective used to determine achievement of the safety goals).

- Large early release [5] frequency (LERF): The frequency of a rapid, unmitigated release of airborne fission products from the containment to the environment that occurs before effective implementation of offsite emergency response and protective actions, such that there is a potential for early health effects. LERF is the surrogate risk measure for individual prompt fatality risk (the first quantitative objective used to determine achievement of the safety goals).

R6. What is the NRC's philosophy regarding the use of risk information?

When using risk information in regulatory decisionmaking, the NRC employs a risk-informed approach. This means that risk information is not the sole basis for decisions.

SECY-98-0144 (NRC, 1998a) defines a risk-informed decision as one in which the results and findings of risk assessments are considered together with other factors (integrated decisionmaking, Figure 2-2). Regulatory Guide (RG) 1.174 (NRC, 1998b), including its latest revision (NRC, 2011a), elaborates on these other factors (compliance with current regulations, defense in depth, safety margins, and monitoring) in discussing an acceptable approach to risk-informed changes in a plant's licensing basis. Subsequent to the initial publication of RG 1.174 in 1998, its general philosophy and specific criteria have been adopted in other risk-informed regulatory applications.

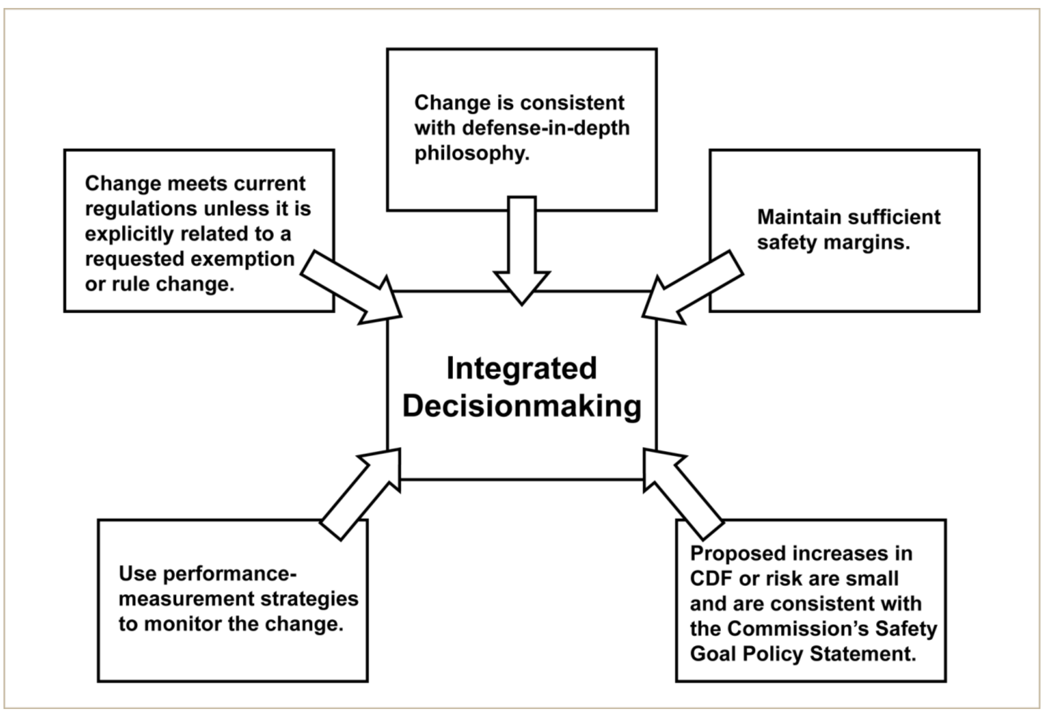

Figure 2-2 Risk-informed, integrated decisionmaking (NRC, 1998b)

[5] For a definition and discussion of the term "large early release," see FAQ L1.

R7. What is the NRC's policy regarding the use of PRA?

The NRC encourages the use of PRA in all nuclear regulatory matters to the extent supported by the state of the art in terms of methods and data.

The NRC's 1995 PRA Policy Statement (NRC, 1995) was issued to encourage (but not to prescribe or require) increased staff and industry use of PRA methods, and to help ensure that potential applications of PRA could be implemented in a consistent and predictable manner that promoted regulatory stability and efficiency. The provisions of the policy statement are as follows:

1. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

2. PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state of the art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support a proposal for additional regulatory requirements in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 50.109, "Backfitting." Appropriate procedures for including PRA in the process for changing regulatory requirements should be developed and followed. It is, of course, understood that the intent of this policy is that existing rules and regulations shall be complied with unless these rules and regulations are revised.

3. PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

4. The Commission's safety goals for NPPs and subsidiary numerical objectives [6] are to be used with appropriate consideration of uncertainties in making regulatory judgments on the need for proposing and backfitting new generic requirements on NPPs.

The *Federal Register* notice publishing the policy statement also includes considerable discussion on the history, benefits, and current uses of PRA; and the rationale and process for developing the policy statement. It also summarizes public comments received during the development of the policy statement and presents staff responses.

FAQ R8 discusses the implementation of this Policy Statement.

R8. Where does the NRC use PRA in regulatory decisionmaking?

The NRC currently uses PRA models and results in its nuclear reactor regulatory activities concerning (1) the development of regulations and guidance, (2) licensing decisions and certification of reactor designs, (3) oversight of licensee operations and facilities, and (4) the evaluation of operational experience.

[6] See the Commission's Safety Goal Policy Statement (USNRC, 1986). See also FAQs R4 and R5.

The NRC conducts a wide variety of regulatory activities in performing its mission (see Figure 2-3). Per its 1995 PRA Policy Statement (NRC, 1995)—see FAQ R7—the agency has been increasing its use of PRA technology in regulatory activities, especially in activities involving commercial power reactors. Some notable examples of reactor-related activities supported by PRA models and results are as follows.[7]

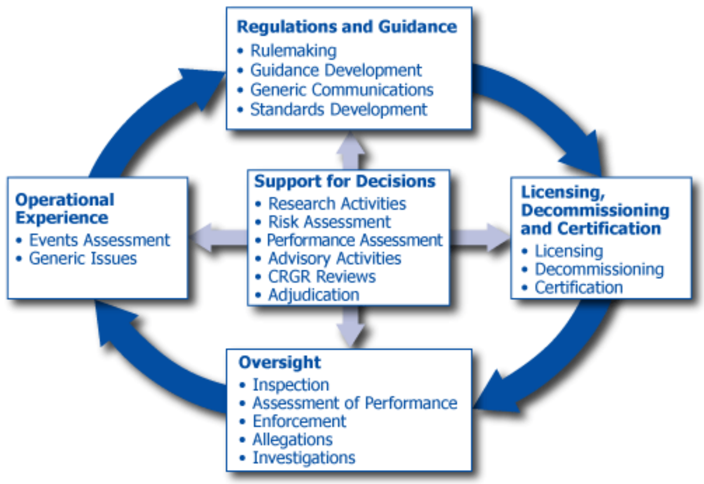

Figure 2-3 NRC's regulatory activities
(http://www.nrc.gov/about-nrc/regulatory.html)

Regulations and Guidance

- Potential new or amended regulations: As part of its process for considering whether to introduce additional regulatory requirements on licensees under the backfit provisions of 10 CFR 50.109, the "Backfit Rule" (CFR, 2007a), PRA results are used in a regulatory analysis to determine whether the new requirements could lead to a substantial safety improvement, and to assess the potential benefits of different regulatory options (e.g., reduced expected public dose from accidents) which are then monetized and compared against the cost of these options. Guidelines for performing regulatory analysis are provided in NUREG/BR-0058 (NRC, 2004) and NUREG/BR-0184 (NRC, 1997b).

- Station blackout protection: 10 CFR 50.63, the "Station Blackout (SBO) Rule" (CFR, 2007b), requires that NPPs be capable of withstanding an SBO (complete loss of alternating current power to the essential and nonessential electric switchgear buses) for a specified, plant-specific duration and of maintaining core cooling during that period. This rule, which, when promulgated in 1988, provided new requirements for licensees, was prompted by PRA findings regarding the risk significance of SBO scenarios.

- Maintenance management: 10 CFR 50.65, the "Maintenance Rule" (CFR, 2007c), requires that licensees assess and manage the risk of maintenance activities (including, but not limited to, surveillance, post-maintenance testing, and corrective and preventive maintenance). Licensees typically use their PRAs in their evaluations to comply with this

[7] Additional details can be found on the NRC's Web site (see, for example, http://www.nrc.gov/about-nrc/regulatory/risk-informed/rpp.html) and in a recent international survey of PRA activities (OECD, 2012).

requirement. This requirement was introduced following the results of the NRC inspections which found a lack of consideration of plant risk in prioritizing, planning, and scheduling maintenance activities.

- Combustible gas control: 10 CFR 50.44 (CFR, 2003), which provides requirements for post-accident combustible gas control, was amended to relax or remove the requirements for certain reactor containment types. This change was prompted by the results of PRAs which showed that the then-current requirements would not be effective for risk-significant scenarios.

- Fire protection: 10 CFR 50.48(c) (CFR, 2007d), which is commonly referred to by a National Fire Protection Association (NFPA) standard endorsed by the rule NFPA 805 (NFPA, 2001), provides a voluntary option by which a licensee can use fire PRA to support the development of a risk-informed, performance-based fire protection program. This option, which was developed recognizing the value of PRA in focusing attention on risk-significant issues, provides licensees with an alternative, nondeterministic approach for achieving compliance with the NRC's fire protection requirements.

- Reactor pressure vessel protection: 10 CFR 50.61a, the "Pressurized Thermal Shock (PTS) Rule" (CFR, 2010), provides licensees with an optional method to assess the limiting level of radiation-induced embrittlement of the reactor pressure vessel beyond which they cannot continue operation without further plant-specific evaluation. This method was developed with the support of PRAs indicating that the risk of vessel cracking because of embrittlement was much lower than previously estimated (NRC, 2007a) and that some degree of relaxation in associated requirements could reduce regulatory burden without jeopardizing safety.

- Special treatment of structures, systems, and components (SSCs): 10 CFR 50.69 (CFR, 2004) allows a licensee to use a plant-specific PRA to support the grouping of SSCs into four classes used to determine the required degree of special treatment (e.g., beyond-commercial grade quality assurance) for the SSCs. These classes are Risk-Informed Safety Class (RISC)-1 (safety-related SSCs that perform safety-significant functions), RISC-2 (nonsafety-related SSCs that perform safety-significant functions), RISC-3 (safety-related SSCs that perform low safety-significant functions), and RISC-4 (nonsafety-related SSCs that perform low safety-significant functions). The process is risk-informed; the PRA results are not used as the sole basis for categorization.

- New reactor certification and licensing: 10 CFR 52.47 (CFR, 2009) requires that an application for standard design certification contain, among other things, a description of the plant-specific PRA and its results. A similar requirement is provided for combined license applicants in 10 CFR 52.79 (CFR, 2013). Additional requirements related to the maintenance and upgrading of the PRA, which must be met by the time of initial fuel load and throughout subsequent plant operation, are contained in 10 CFR 50.71 (CFR, 2007e).

- PRA technical adequacy: Regulatory Guide (RG) 1.200 (NRC, 2009a) provides an approach for determining the acceptability of a PRA when used in a regulatory application. It establishes the attributes and characteristics of a technically acceptable PRA and endorses consensus PRA standards and the industry peer review process.

Licensing, Decommissioning, and Certification

- Changes in plant licensing basis: RG 1.174 (NRC, 2011a) provides an acceptable general approach by which a licensee can use risk information to support a voluntary change from a plant's current (and accepted) licensing basis to a new licensing basis. The RG is written quite broadly, and the general principles discussed in the RG have been adopted in many NRC risk-informed application-specific regulatory guides. For example, RG 1.177 (NRC, 2011b) supports risk-informed changes to plant Technical Specifications regarding "completion times" (i.e., the time by which service work on an SSC needs to be completed) and "surveillance frequencies" (which prescribe how often SSCs should be tested). The RG identifies both deterministic and risk considerations (see FAQ R6). Other NRC risk-informed regulatory guides that build off of the guidance in RG 1.174 include RG 1.178 (NRC, 2003) for inservice inspections, RG 1.201 (NRC, 2006a) for categorization of SSCs under 10 CFR 50.69, and RG 1.205 (NRC, 2009b) for fire protection. In situations in which licensees apply for non-risk-informed license amendments (i.e., amendment applications not supported with risk information), the NRC staff still may perform a risk evaluation to determine whether the proposed amendment has the potential to significantly affect risk, but only if special circumstances are identified by the staff (NRC, 2007b).

- Environmental reviews: When submitting applications for plant license renewals and combined licenses, or certified designs, applicants are required, as part of their environmental assessments, to identify and evaluate potentially cost-beneficial severe accident mitigation alternatives (SAMAs) or severe accident mitigation design alternatives (SAMDAs), respectively. Although not required, the applicants typically use plant-specific PRAs to support these evaluations. The NRC staff's reviews of SAMA analyses are documented in published supplements to NUREG-1437 (NRC, 2013b), and SAMDA reviews can be found in the Environmental Assessment developed as part of the design-specific 10 CFR 52 appendix.

Oversight

- Reactor Oversight Process: In the NRC's Reactor Oversight Process (ROP) (NRC, 2006b), PRA results are used to support inspection planning, and, as discussed in the NRC's Inspection Manual Chapter 0609 (NRC, 2011c), PRA tools (including the licensee's PRA and the NRC's Standardized Plant Analysis Risk (SPAR) models (NRC, 2014a)) are used to determine the risk significance of inspection findings. The Mitigating Systems Performance Index, one of the performance indicators used in the ROP, is a risk-informed performance indicator that uses licensee PRA results in its computation (NRC, 2005).

- Incident investigation: Following the NRC's Management Directive 8.3 (NRC, 2001), the NRC staff uses PRA models to support decisions regarding the appropriate response to a reported incident. Conditional core damage probability (CCDP) is calculated and is considered along with other factors (including uncertainty of the results) when determining the type of inspection team (an incident investigation team, an augmented inspection team, or a special inspection team) to send with a higher CCDP generally leading to a larger, more thorough inspection.

- Notice of Enforcement Discretion: As discussed in the NRC's Inspection Manual Chapter 0410 (NRC, 2012b), the NRC staff uses risk models to support decisions regarding whether enforcement discretion is warranted for unanticipated temporary non-compliances with various license conditions.

Operational Experience

- Accident precursors: The NRC's Accident Sequence Precursor Program (NRC, 2014a) uses SPAR models and other sources of information to evaluate operational events (both actual events and degraded conditions) at U.S. NPPs to identify, document, and rank these events. This program was mandated by Congress, and the NRC provides an annual report.

- Emergent issues: Following the General Accounting Office's (GAO's) audit of the NRC's actions (including its use of risk information) in response to reactor pressure vessel head degradation at the Davis-Besse Nuclear Power Station plant (GAO, 2004), the NRC developed and implemented a process to document risk-informed decisions (including decisions to shut down a plant) dealing with emergent issues. The guidance, which is provided in the Office of Nuclear Reactor Regulation's Office Instruction LIC-504 (NRC, 2010a), follows the general principles of risk-informed decisionmaking described in RG 1.174 and places a strong emphasis on documenting the decision so that factors driving the decision are identified and suitably qualified to address uncertainties.

- Generic issues: The NRC uses PRA results to support the assessment and disposition of potential safety issues that can affect more than one plant. The Generic Issues Program is described in NUREG-0933 (NRC, 2011d). The approach used to determine whether further regulatory assessment and action should be recommended is described in the NRC's Management Directive 6.4 (NRC, 2009c) and the NRC Office of Nuclear Regulatory Research's Office Instruction TEC-002 (NRC, 2010b).

Decision Support

- SPAR models. The NRC maintains a fleet of Level 1, at-power, internal events PRA models covering all U.S. commercial operating reactors. There are also a limited number of models addressing fires and external hazards, low power and shutdown operations, and some new reactors. These models are used in a variety of regulatory applications, as discussed above. A discussion of the models is provided by Appignani, Sherry, and Buell (2008). SECY-14-0107 (NRC, 2014a) provides a recent status report.

- Consequence analyses. The NRC uses PRA results to identify accident scenarios potentially worthy of detailed examination in accident consequence analyses, such as the recent State-of-the-Art Reactor Consequence Analyses documented in NUREG-1935 (NRC, 2012d). Such analyses also provide tools and results that can be useful for subsequent PRAs.

- Risk management methods. Recently, the NRC has published NUREG-2150 (NRC, 2012a), which proposes a risk management regulatory framework that could be used to improve consistency among the NRC's reactors, materials, waste, fuel cycle, and transportation programs and discusses implementing such a framework for specific program areas.

R9. Why does the NRC use PRA in regulatory decisionmaking?

Past PRAs have provided valuable safety perspectives and supported effective improvements. Using a top-down approach that starts with the definition of a decision problem, derives the quantitative measures of risk important to that problem, and develops plant-specific models to assess those measures, PRA enables an open, integrated treatment of diverse safety issues and the evaluation of the impact of potential changes consistent with the current state of knowledge.

Starting with WASH-1400 (NRC, 1975), NPP PRAs have provided important, actionable safety insights and lessons. WASH-1400 pointed out, contrary to the prevailing view of the time, that some accidents (e.g., a loss-of-coolant accident resulting from a small pipe break) less severe than a plant's design-basis accident (a loss-of-coolant accident resulting from a large pipe break) could be more important to risk (Beckjord, Cunningham, and Murphy, 1993). Thus, it was important to explicitly identify, analyze, and ensure appropriate defenses against such scenarios. WASH-1400 also showed the risk significance of so-called "non-safety" systems (e.g., a plant's auxiliary feedwater system), human error, and the failures of multiple, redundant components because of a single cause (common-cause failures). Thus, these systems and failures deserved a high level of attention.

Subsequent PRAs performed by the nuclear industry (e.g., see (Garrick, 1989) and (Gaertner, True, and Wall, 2003)) and by the NRC (e.g., NUREG-1150 (NRC, 1990)) demonstrated the risk significance of scenarios involving widespread loss of electrical power (the so-called "station blackout" scenarios), scenarios involving the loss of the plant's ultimate heat sink, and scenarios triggered by hazards originating within the plant (notably fires and floods) and those outside (notably earthquakes, high winds, and floods). Importantly, these studies showed that, although broad, fleet-wide statements could be made about potentially risk-significant scenarios, the important scenarios for a particular plant were highly plant-specific. Technical details in the plant design and operation were (and continue to be) important factors in determining the dominant contributors to risk.

Especially in the early years before the widespread use of PRA, the lessons derived from PRA were not always consistent with then-current wisdom. Moreover, it was recognized that the PRA models, as with any engineering analysis models, were (and remain) imperfect representations of reality. Nevertheless, industry and regulatory decisionmakers took major actions suggested and supported by PRAs.

These early PRAs were generally performed for anticipatory reasons, and not in reaction to actual accidents, incidents, or inspection findings. Nevertheless, their results and insights prompted numerous actions. Arguably, this can be attributed to some key features of a PRA model:

- Top-down. A PRA modeling effort generally starts with the specification of a decision problem (or class of problems) to be solved. Once the problem is specified, the factors affecting the decision (notably, the measures of risk—"risk metrics"—in the case of risk-informed decisions) can be identified and the models needed to assess these factors developed. Thus, the PRA model is focused on issues relevant to the safety decision.

- Engineering-oriented. Nuclear power plant PRA models are structured to directly represent plant SSCs and plant responses to abnormal conditions (including operator actions). Thus, the scenarios generated by a PRA (each of which represents a "story") can be understood by people directly responsible for plant safety who are not necessarily specialists in PRA modeling.

- Integrated. Severe accident scenarios typically involve a wide variety of events and phenomena. Inputs are often needed from a variety of diverse engineering and scientific disciplines, including plant operations (e.g., to identify expected and potential hardware and crew responses to abnormal events), human factors (e.g., to identify the potential effects of different plant conditions), thermal hydraulics (e.g., to assess plant transient behavior and determine the equipment performance required to avoid fuel damage), civil engineering (e.g., to assess the performance of plant structures under seismic or post-core damage conditions), the geosciences (e.g., to assess the likelihood and magnitude of external hazards such as earthquakes or floods), and specialized disciplines (e.g., to model the effects of fire and explosions). A PRA model provides a single, consistent framework needed to weigh and combine the inputs from these different disciplines. Moreover, it enables decisionmaking across very different scenarios (e.g., a scenario initiated by a reactor coolant system pipe break versus one initiated by a turbine-generator oil fire versus another one initiated by a hurricane storm surge). This decisionmaking typically involves PRA-suggested solutions that are effective across a wide range of scenarios, as well as tradeoffs between scenario-specific solutions.

- Systematic. Although other less formal processes (e.g., brainstorming) can be used to identify and characterize potentially important accident scenarios, PRA tools and techniques provide the support needed for a systematic assessment of such complex facilities as NPPs. No practical engineering analysis approach can guarantee that all potentially important scenarios are identified—all tools rely on our current state of knowledge—but the PRA modeling process, which asks the fundamental question "what can go wrong?", at least prompts the analyst to look for possibilities that might otherwise be overlooked. PRA models also, by their nature, include combinations of identified possibilities that could easily be missed by less systematic approaches.

- Sufficiently realistic. In general, PRA models are aimed at assessing accidents that are sufficiently rare that they cannot be validated against accident statistics. As discussed in FAQ R7, the NRC's 1995 PRA Policy Statement requires that PRA evaluations used in support of regulatory decisions should be as realistic as practicable. In this context, the term "realistic" means that conservatisms are not deliberately introduced into the PRA model to compensate for uncertainties. The NRC's confidence that PRA models can be sufficiently realistic for decisionmaking purposes stems from (1) the models' use of available, relevant information, including empirical data and research results (e.g., regarding the causes and likelihood of human error), in identifying and quantifying model sub-elements (e.g., the failure rate of a particular component), (2) the formal and informal PRA model reviews performed prior to and during model application, and (3) the existence of various activities and programs (e.g., inspections, operational experience reviews, accident precursor studies) that have identified issues in PRA modeling approaches as well as with specific models.

- Supportive of "what-if" analyses. As with other systems analysis methods, PRA provides a mechanism for proactively assessing events that have not yet occurred and for plant changes that have not yet been made, and for exploring the impact of different modeling conditions and assumptions on the results of such assessments. Thus, a PRA model provides the means to assess the risk impact of different decision options.

- Openness. Again, as with other systems analysis methods, a well-documented PRA model enables a reviewer to trace how model inputs are transformed into outputs, to identify important modeling features and assumptions, to see whether and how specific issues are modeled, and to identify "what-if" analyses to explore the potential implications of alternate modeling assumptions. The understanding resulting from such explorations (and from answers to questions stemming from exploration) can improve confidence in the model and its results.

In general, as stated in its PRA Policy Statement (NRC, 1995), the NRC believes that the use of PRA leads to an improved regulatory process by enhancing safety decisionmaking, making more efficient use of agency resources, and reducing unnecessary burden on licensees. Comparing a probabilistic approach with a deterministic approach, the Policy Statement provides the following:

> The NRC has generally regulated the use of nuclear material based on deterministic approaches. Deterministic approaches to regulation consider a set of challenges to safety and determine how those challenges should be mitigated. A probabilistic approach to regulation enhances and extends this traditional, deterministic approach, by: (1) Allowing consideration of a broader set of potential challenges to safety, (2) providing a logical means for prioritizing these challenges based on risk significance, and (3) allowing consideration of a broader set of resources to defend against these challenges.

It should be emphasized again that, in the NRC's risk-informed decisionmaking environment (see FAQ R6), PRA results do not constitute the sole basis for decisionmaking. Information from a variety of sources (including traditional engineering analyses and expert panels) is also used as appropriate to the decision at hand. Further, as discussed in RG 1.174 (NRC, 2011a), monitoring and inspection programs are used to ensure that PRA model assumptions remain appropriate.

R10. If nuclear power plant accident statistics are available, is there a need for PRA?

Statistical estimates for nuclear power plant accident rates are based on modeling assumptions that are usually inappropriate for regulatory decisionmaking. PRA remains necessary to bring in important design-, site-, and plant-specific information not addressed by such estimates and to evaluate the impact of potential changes.

Nuclear power plant accidents are rare events, and any lessons following an accident should be seriously considered when taking actions aimed at preventing future such occurrences. In recent years, there have been attempts to derive statistical estimates of NPP accident rates based solely on accident data. Examples are studies published by Cochran (2011, 2012), Lelieveld, Kunkel, and Lawrence (2012), Kaiser (2012), and Gallucci (2012) following the March 2011 Fukushima Dai-ichi reactor accidents.[8]

[8] These statistically derived estimates of CDF are further discussed in FAQ C2. FAQ C3 discusses PRA-derived CDFs. Similarly, FAQ L2 discusses statistically derived estimates of LERF, and FAQ L3 discusses PRA-derived estimates.

However, it is important to recognize that these estimates are based on a strong modeling assumption: all plants in the analyzed group are of identical design and are operating under the same conditions. In mathematical terms, all of the plants in the analyzed group over the analyzed time period are assumed to be “exchangeable.” Only when this assumption of exchangeability is valid can past accidents be used to say something about the future behavior of plants in that group.

The assumption of exchangeability of plants is not valid because of two major reasons: plant risk is heavily dependent on plant-specific details (e.g., design and operations), and major safety improvements have been made over time in reaction to analyses of hypothetical accidents (such as in PRAs) or actual accidents.

Broad discussions of plant changes spurred by PRAs can be found in a number of sources, (e.g., (Garrick, 1989) and (Gaertner, True, and Wall, 2003)). Numerous plant-specific discussions of PRA-prompted changes can be found in licensee-submitted risk-informed license amendment requests (LARs), which are publicly available through the NRC’s Agencywide Documents Access and Management System (ADAMS),[9] and in the NRC staff’s reviews of licensee analyses of potentially cost-effective SAMAs performed to support plant license renewal applications. These latter reviews are published in plant-specific supplements to NUREG-1437 (NRC, 2013b).

Of course, PRAs employ modeling assumptions as well, and it is important to validate these assumptions with empirical information when such information becomes available. However, as discussed in FAQ R10, PRA is valuable for decision support because it brings in a broader range of information (including non-statistical, phenomenological information regarding accident causation and progression), identifies and analyzes possibilities not yet seen in actual incidents or accidents, and enables the identification and analysis of potential improvements.

[9] Currently, risk-informed LARs can be found using the advanced ADAMS search capability that can look for specific terms in the document content (e.g., “CDF”) as well as key words in the document title (e.g., “license amendment request”).

3. PRA BASICS

In discussions of accident statistics and probabilistic risk assessments (PRAs), one sometimes hears statements directly or implicitly indicating misconceptions regarding PRA. This section provides the following frequently asked questions (FAQs) addressing the philosophy and practice of PRA.

B1. **What is PRA?**
B2. **Do PRAs predict the future?**
B3. **If a PRA quantifies likelihoods in terms of probabilities, what are frequencies?**
B4. **What is the meaning of the percentiles for CDF reported by some PRAs?**
B5. **In situations in which the CDF is highly uncertain, is the mean CDF a meaningful quantity for regulatory decisionmaking?**
B6. **What are aleatory and epistemic uncertainties and why do we distinguish between the two?**
B7. **What is the most valuable output of a PRA?**
B8. **Are PRAs supposed to be conservative?**
B9. **Is PRA a mature analysis technology?**
B10. **How are PRAs updated to reflect operational experience and other new information?**
B11. **What is the current international status of nuclear power plant PRA and risk-informed decisionmaking?**
B12. **Is PRA limited to event tree/fault tree analysis?**

B1. What is PRA?

"PRA" is an acronym standing for "Probabilistic Risk Assessment." It is also called "Probabilistic Safety Assessment" (PSA). PRA answers three fundamental questions: What can go wrong? How likely is it? And, what are the consequences?

PRA [1] is an engineering analysis process that models a nuclear power plant (NPP) as a system. By answering the three fundamental risk questions (see FAQ R2), it provides both qualitative and quantitative information. It builds on and represents the current state of knowledge regarding (1) what design, operation, and environmental (physical and regulatory) aspects of the plant are important to risk, what elements are not important, and why, and (2) what is the magnitude of the risk associated with the overall plant.

An NPP PRA model will, for some plant operational mode (e.g., at-power operation, low power conditions, shutdown conditions), typically delineate and analyze a large number of accident scenarios, starting with an "initiating event," (i.e., a disturbance to plant normal operations such as the loss of coolant), an internal hazard (e.g., a fire or flood starting within the plant boundaries), or an external hazard (e.g., an earthquake or external flood), and ending with some level of consequence.

Regarding consequences, the definition of PRA is sufficiently broad to encompass a wide variety of analysis endpoints. Thus, depending on the needs of the decision problem at hand, a PRA can assess consequences in terms of core condition (this is the endpoint of a Level 1 PRA), radioactive material release from the plant (a Level 2 PRA), or offsite effects (a Level 3 PRA).

The general framework for performing an NPP PRA was first proposed by Farmer in 1967 (Farmer, 1967). This framework emphasized a probabilistic approach involving the treatment of a wide spectrum of accident scenarios (thereby avoiding the need to distinguish between "credible" and "incredible" accidents), the use of logic models to identify these scenarios, the use of reliability engineering methods (including statistical analysis for component performance) to quantify scenario likelihoods, the use of expert judgment to assess the performance of structures, and the calculation of public health consequences from potential releases. This framework was, with some improvements, implemented in the U.S. Nuclear Regulatory Commission's (NRC's) Reactor Safety Study (WASH-1400), the first comprehensive NPP PRA (NRC, 1975). Other notable past U.S. PRA studies [2] include the first industry-sponsored Indian Point and Zion PRAs (notable from a technical perspective for their systematic treatment of uncertainties, internal hazards, and external events; the Indian Point study is also notable from a regulatory perspective for its use in an adjudicatory process),[3] the industry's individual plant examinations (IPEs) and individual plant examinations of external events (IPEEE) performed in response to the NRC's Generic Letter 88-20 (NRC, 1988) and supplements (notably NRC, 1991), the NRC's risk study of five plants documented in NUREG-1150 (NRC, 1990), and the NRC's follow-on study of low-power and shutdown (LP&S) risk at two plants (Chu, et al., 1993), (Whitehead, et al., 1994).

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[1] The international community uses the term "probabilistic safety assessment" (PSA) instead of PRA (IAEA, 2010). For all intents and purposes, these terms are equivalent.
[2] See FAQ B11 for a discussion of international PSAs.
[3] Remarks on these proprietary studies are provided by Garrick (1984, 1989). The NRC-sponsored reviews of these studies are documented in NUREG/CR-2934 (Kolb, et al., 1982) and NUREG/CR-3300 (Berry, et al., 1984). The findings and recommendations of the Atomic Safety Licensing Board and the Commission's ultimate decisions can be found in the NRC's public records (ASLB, 1983), (NRC, 1985).

Since the performance of the initial IPE and IPEEE studies, most U.S. plants have updated their PRA studies in support of various risk-informed applications (primarily involving license amendment requests) and to support the identification and analysis of potentially cost-effective severe accident mitigation alternatives in support of plant license renewal applications. The results of these studies are discussed in FAQs C3 and L3.

B2. Do PRAs predict the future?

No, PRAs do not make predictions. PRAs identify potential accident sequences and produce statements regarding their likelihood and consequences, thereby informing our current state of knowledge on which decisions are based.

As a simple analogy, consider a single roll of two dice (which is known to be fair), and assume the sum of the two dice face values will be analyzed. A probabilistic model will conclude that the most likely outcome is a “7” (with probability (6/36) = 0.17), and the least likely outcomes are “2” and “12” (each with probability (1/36) = 0.028). The probabilistic model does not suggest that the next roll will be “7.” It only says that this value is the most likely to occur (based on current knowledge). In addition, even though a “7” is the most likely outcome, its probability is fairly low. All this information would be useful to a gambler who might be willing to bet on the outcome of a roll. The intent is not to predict the outcome of that roll but to inform the gambler’s state of knowledge regarding the likelihood of the potential outcomes.

Similarly, an NPP PRA enumerates the numerous ways an accident can occur and indicates which scenarios are more likely than others (based on current information). If the next accident involves the most likely scenario, this does not mean the PRA was “right.” Similarly, if the next accident involves a much more unlikely scenario, this does not necessarily mean the PRA was “wrong.” In both cases, a detailed comparison of the causal mechanisms built into the PRA with those underlying the actual accident will shed more light on the PRA’s representation of reality, but it must be recognized that any scenario included in the PRA is believed to be possible.

The converse is not necessarily true (i.e., that scenarios not included in the PRA are believed to be impossible). Scenarios can be excluded in situations in which they are believed to be unimportant relative to the decision problem at hand. Also, of course, scenarios can be excluded because of limitations in the state of knowledge when the PRA was developed (assuming that the PRA had been properly reviewed). Incompleteness of our state of knowledge is a fact that is readily acknowledged by decisionmakers. The thousands of accident sequences that PRAs identify help to reduce this incompleteness, but do not eliminate it.

B3. If a PRA quantifies likelihoods in terms of probabilities, what are frequencies?

The frequency of an event is the average number of occurrences of that event or accident condition per unit of time (usually 1 year in PRAs).

The frequency of an event does not imply that the event will occur at regular intervals. Thus, for example, if the loss of offsite power frequency at a plant is 5×10^{-2}/yr, this does not mean that the plant will suffer a loss of offsite power once every 20 years. It means that the statistical average of the time between such events (where this time interval is a random variable) is 20 years.[4]

Note that for very small event frequencies (e.g., 10^{-4}/yr), the associated average time between events (10,000 years in this case) is a theoretical concept. The practical value of the event frequency is twofold: (1) it can be used to compute the probability of occurrence of an event in a given time interval, and (2) it can be used to compute the frequency (and probability) of multiple, jointly occurring events (such as those in an accident scenario).

Appendix A provides a detailed discussion on the concept of frequency as it is used in PRAs and on the mathematical connection between event frequencies and probabilities.

B4. What is the meaning of the percentiles for CDF reported by some PRAs?

The percentiles for CDF indicate the assessor's degree of uncertainty in the value of CDF.

The Pth percentile for core damage frequency (CDF), CDF_P, is 100 times the probability that the true value of CDF is less than CDF_P. Thus, for example, the 95th percentile for CDF, CDF_{95}, is that value for which there is a 0.95 probability that the value of CDF is less than CDF_{95}. PRAs that report CDF percentiles often report the 5th and 95th percentiles as indicators of the potential range of CDF.

Notes—

- The CDF percentiles are indicative of the probability distribution that quantifies the assessor's state of knowledge regarding CDF (i.e., the assessor's epistemic uncertainties).[5] Closer percentile values indicate a smaller degree of uncertainty in the range of possible values for CDF.

- The uncertainties in CDF because of specific contributors or scenarios can be very large (perhaps with 5th-to-95th percentile ranges spanning multiple orders of magnitude). However, the total plant CDF, which is the sum of the CDFs from all contributors, will have a significantly smaller spread, as conceptually illustrated in Figure 3-1. FAQ C7 provides more discussion on the magnitude of uncertainties in total CDF.

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[4] Note that the definition of frequency provided in NUREG-2122 (USNRC, 2013c) refers to the "expected" number of occurrences. Recognizing that the terms "expected value," "average," and "mean" are synonymous in the field of probability and statistics, NUREG-2122 is mathematically correct. This report uses the term "average" to avoid any implications that a particular value is actually expected in a broad sense.

[5] See FAQ B6 for a discussion of epistemic (and also aleatory) uncertainties.

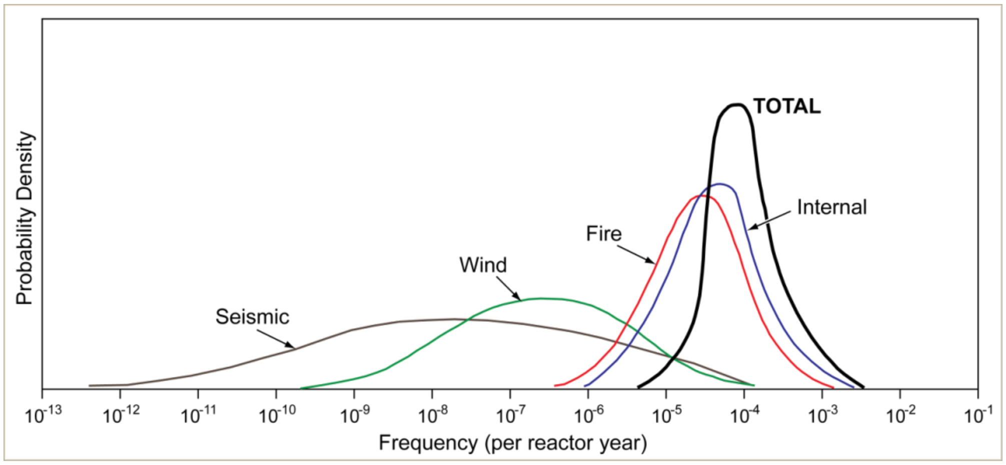

Figure 3-1 Illustrative probability density functions for CDF (adapted from Garrick, 2014)[6]

B5. In situations in which the CDF is highly uncertain, is the mean CDF a meaningful quantity for regulatory decisionmaking?

The mean CDF is a useful metric for risk-informed decisionmaking because it is relatively simple to compute and use, and it provides some accounting for uncertainties. This usefulness does not rely on the intuitive meaningfulness of the metric.

The mean value is a mathematical quantity that, although precisely defined,[7] has no everyday intuitive meaning.

Consider a situation in which we are estimating the mean CDF for a plant whose CDF is described by a lognormal distribution with 5th and 95th percentiles of 10^{-6}/reactor-year (RY) and 10^{-4}/RY, respectively. In this situation, as shown in Figure 3-2, the CDF mode equals 1.4×10^{-6}/RY, the CDF median equals 1.0×10^{-5}/RY, and the CDF mean equals 2.7×10^{-5}/RY. In this case, the mean value corresponds to the 76th percentile of the CDF distribution. It is neither the most likely value (i.e., the mode), nor an indication of the "middle" of the distribution (i.e., the median [8]). For situations involving larger degrees of uncertainty, the mean value will correspond to even higher percentiles of the distribution.

Although the mean value does not have an intuitive meaning, it is still useful in decisionmaking. First, and not to be discounted, it is clearly defined, is easily computable, and is a scalar (i.e., a

[6] The probability density function for an uncertain variable *X* can be viewed as representing the probability that *X* "equals" any selected value *x*. More formally, the function is proportional to the probability that *X* takes on a value in the interval (*x*,*x*+*dx*), where dx is an infinitesimally small value.

[7] For a positively valued random variable *X*, the mean value (also called the average or expected value) is defined by:

$$\text{mean value of } X \equiv \int_0^\infty x \cdot f_X(x)dx$$

where $f_X(x)$ is the probability density function for *X*.

[8] More precisely, the median is the value for which there is equal likelihood that the uncertain variable—CDF in our case—is greater or lesser. In other words, as shown in Figure 3-2, the area under the probability density function curve to the left of the median equals 0.50, as does the area under the curve to the right of the median.

single value). The last characteristic means that it can be easily compared with simple criteria, which allows decisionmakers to make informed decisions without needing to consider the full results of a detailed uncertainty analysis.[9] Also, because it is computed using the entire probability distribution, the mean value provides some reflection of the uncertainty quantified by that distribution. (Note that the median is not affected by the length of distribution tails (i.e., the possibility of large deviations from the distribution center), and the mode, in most practical situations, represents only one point on the distribution.) Finally, in formal theories of decisionmaking, it can be shown that the mean value is the theoretically appropriate metric to use under certain conditions.

Regulatory Guide (RG) 1.174 (NRC, 2011a) emphasizes the use of the mean value, stating that "Because of the way the acceptance guidelines … were developed, the appropriate numerical measures to use in the initial comparison of the PRA results to the acceptance guidelines are mean values."

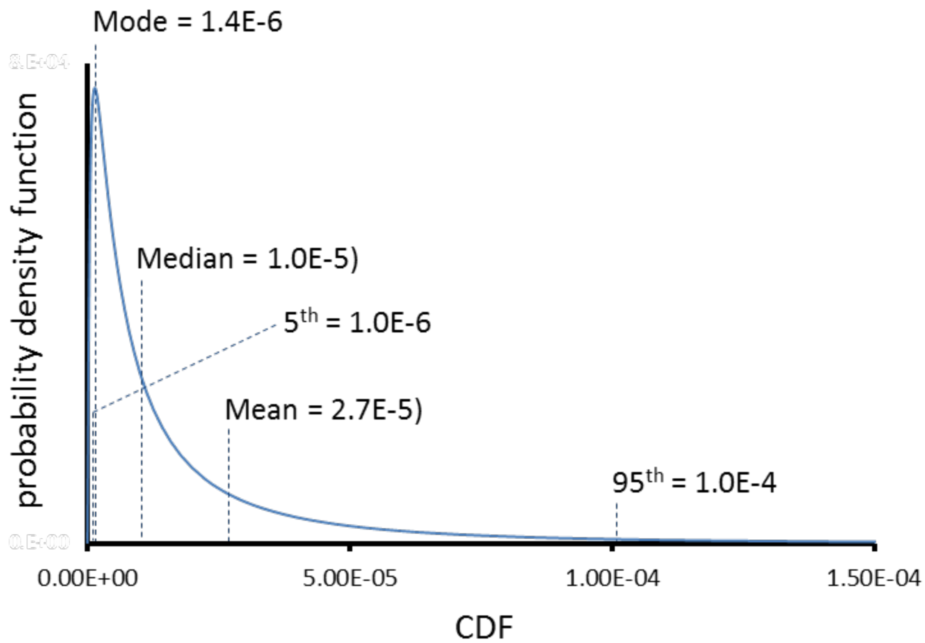

Figure 3-2 Example CDF distribution with key characteristics

B6. What are aleatory and epistemic uncertainties and why do we distinguish between the two?

Aleatory uncertainty, also referred to as "stochastic," "statistical," or "random" uncertainty, is uncertainty that arises because of inherent variability in a modeled process. Epistemic uncertainty, also referred to as "state-of-knowledge" uncertainty, is uncertainty that arises because of limitations in the state of knowledge of the assessor. The distinction between aleatory and epistemic uncertainties is a modeling decision and not absolute. However, it is useful for ensuring the clarity of PRA models and therefore in interpreting and communicating the results of these models.

[9] During the development of RG 1.174, different treatments of uncertainty were discussed. Some involved the use of two criteria (e.g., the mean value and the 90th percentile). The staff eventually decided to use a single criterion based on the mean value.

The conceptual distinction between aleatory and epistemic uncertainties, discussed in a PRA context by Apostolakis (1978, 1990, 1994),[10] is well-established by longstanding modeling conventions in many parts of a PRA model. Thus, for example, the number of times a pump will fail on demand in a series of repeatable trials, which is unknown prior to executing the trials, is treated as an aleatory variable. On the other hand, the underlying failure rate of the pump, whose precise value is unknown even after the trials are executed (because a different set of trials could lead to a different set of successes and failures), is treated as an epistemic parameter. Similarly, the number of core damage accidents over a set period of time is treated as an aleatory variable, whereas the underlying CDF is an epistemic parameter estimated using a PRA model.

There are situations in which the distinction is less clear cut (Siu, 1999), (Siu et al., 2000). As pointed out by Apostolakis (1990 and 1994), the assessor must take care to clearly articulate the "model of the world," the model which, among other things, defines what the assessor chooses to treat as "random" and unknowable versus what is deterministic and knowable. Through this articulation, the assessor can precisely state what the PRA's statements on uncertainty are referring to. Rather than generic references to "the uncertainties in the analysis," the assessor can point to uncertainties in specific quantities, thereby aiding the communication of results to decisionmakers.

Appendix A provides additional discussion on aleatory and epistemic uncertainties.

B7. What is the most valuable output of a PRA?

The relative value of the qualitative (accident sequences) and quantitative (frequencies) outputs of a PRA depends on the information needs of the decision problem at hand; neither one nor the other can be stated to be more important independently of the decision context.

In discussions of the strengths and weaknesses of PRA, one sometimes hears, "The quantitative results are meaningless—only the insights matter." Conversely, discussions of PRA results sometimes focus too readily on the numerical results (e.g., the mean CDF) and ignore the valuable qualitative information conveyed by accident scenarios.

Clearly, both of these points of view are limited. Quantitative results play an explicit role in those risk-informed regulatory applications that employ numerical guidelines, particularly for CDF and large early release frequency, following the lead established by RG 1.174 (NRC, 2011a). Numerical results also enable comparisons of risk contributions across disparate scenarios (e.g., scenarios initiated by plant hardware failures versus those initiated by external natural phenomena). On the other hand, neglect of qualitative information and insights ignores PRA's value in conveying an understanding of sources of risk, underlying reasons, and possible sources of improvement. PRA results are valuable when they can support effective decisionmaking.

[10] The terminology "aleatory" and "epistemic" uncertainties was introduced to the nuclear PRA community by Apostolakis (1994) and Budnitz et al. (1997), and later echoed in Regulatory Guide 1.174 (USNRC, 1998b). Earlier discussions in the PRA literature variously refer to "statistical" and "frequency" distributions in representing the former and "state-of-knowledge" distributions in representing the latter.

B8. Are PRAs supposed to be conservative?

PRAs are expected to be realistic. Although some conservative assumptions are inevitable, analysts should strive to avoid bias and to characterize uncertainty. This helps ensure that decisionmakers are presented with a reasonable representation of the current state of knowledge regarding risk. The decisionmaker may decide to make conservative decisions to account for uncertainties.

NRC's principles of good regulation (see NRC, 2014b) include the following statement: "Final decisions must be based on objective, unbiased assessments of all information…." With respect to PRA, the Advisory Committee on Reactor Safeguards states, "A PRA should be done realistically. The proper time to add an appropriate measure of conservatism is when its results are used in the regulatory process. If the PRA itself is done with conservative assumptions … and is then used in a conservative regulatory decisionmaking process, self-deception can result, or resources can be squandered." (ACRS, 1991). The need for realism is re-emphasized in the NRC's 1995 PRA Policy Statement: "PRA evaluations in support of regulatory decisions should be as realistic as practicable…" (NRC, 1995). Thus, the role of the analyst is to provide the decisionmaker with the best possible information. It is important to recognize that realism includes explicit consideration of uncertainties, given the possibly large uncertainties in any assessment of rare events. Characterizations of risk that ignore these uncertainties are likely to be misleading.

In practice, PRAs are often performed in an iterative fashion to focus analytical resources on the most important scenarios. In this iterative approach, scenarios are initially analyzed using conservative modeling assumptions, and scenarios that appear to be important are further analyzed using progressively more realistic assumptions. This approach increases computational efficiency, but care is needed when using the model and results in situations for which the analysis was not originally performed (e.g., when performing conditional analyses involving the assumed failure of particular components).

B9. Is PRA a mature analysis technology?

PRA methods, models, tools, and data are sufficiently mature to support risk-informed decisionmaking at the NRC.

Although judging the maturity of a technical field is a subjective matter, some rigor can be introduced by a systematic consideration of key factors. Different authors have identified some characteristics they consider to be indicators of maturity. Stetkar, Shack, and Nourbakhsh (2011), in a discussion of the current status of efforts to risk-inform plant fire protection programs, distinguish between the maturity of an analysis technology (which dictates what level of analysis is possible) and the maturity of the application of that technology (which indicates what is happening in the field). They also tie the notion of maturity to the number of experienced analysts performing fire PRAs. Budnitz (1998) provides similar indicators in a discussion of the state of seismic PRA, referring to the number of practitioners (or groups of practitioners), the degree of practice, and the state of technical development of the field (including the availability of detailed guidance for new practitioners). Budnitz emphasizes the use of the technology in support of practical decisionmaking as an important indicator of maturity. Cornell (1981), in a discussion on the state of structural safety engineering, describes characteristic situations associated with the different stages of development of a technical field based on his observations from a number of fields (including geotechnical engineering, structural dynamics, and finite element analysis). Table 3-1 provides a summary of Cornell's discussion, grouping his situations into one of three categories of indicators involving the field's practitioners, research agenda, and applications.

The above discussion indicates that NPP PRA is, on the whole, in an intermediate-to-late stage of maturity.[11] PRA results have been used to support major decisions, starting with the Commission's 1985 decision to allow continued operation of the Indian Point Plants (NRC, 1985), continuing with plant changes identified in the IPEEE program (NRC, 2002), and more recently with staff approvals of licensee-requested fire protection program transitions (Hamzehee, 2014). These show that the technology is being used in practical applications. Further, it appears that the field has many of the characteristics shown in the third column of Table 3-1 and a few of those in the second column.

It is important to recognize that the existence of reducible uncertainties (with an associated need for research and development (R&D)) does not necessarily imply that the discipline is immature. As indicated in Table 3-1, even mature technologies have needs for R&D.

It is also important to recognize that the issues of PRA maturity and realism, although related, are actually separate. The concept of maturity addresses the relative state of development of a technical discipline. On the other hand, in a PRA context, the concept of realism addresses the degree to which an analysis represents the technical and organizational system relevant to the decision problem. The analytical technology (i.e., methods, models, tools, and data) of a less mature discipline could, but need not, produce unrealistic analysis results. Conversely, a more mature discipline could, for practical reasons, employ technology with known weaknesses, only requiring that the weaknesses be understood and appropriately addressed in the decisionmaking process.

[11] There are variations within the field. For example, the treatment of internal events scenarios is more mature than the treatment of some external hazards (e.g., tsunamis).

Table 3-1 Indicators of Stages of Technical Maturity (adapted from Cornell, 1981)

	Developmental Stage		
	Early (Infancy, Emerging)	**Intermediate** (Adolescent, Developing)	**Late** (Mature, Stable)
Practitioners	• Small research community • Small number of practitioners • Strong personality influences, competing schools of thought	• Larger number of practitioners • Larger number of experienced researchers	• Many well-trained and experienced practitioners • Recognize limits of applicability of methods • Can adapt methods to new situations • Can work with researchers to identify important issues
Research Agenda	• Driven by perceived needs • Problem selection affected by personal choice (e.g., because of ease of formulation or solution)	• New practice-driven research problems • Some consensus positions for some broadly defined problem areas • Some unproductive research lines abandoned • Incomplete coverage of topics	• Most research driven by needs of practice • More abstract research addresses needs clearly identifiable by all concerned
Applications	• Local applications (addressing small parts of larger problems) • No broader framework	• Fast growth • Developing vocabulary • Optimistic views on new methods; limitations not well understood	• Vocabulary has evolved • General framework exists • Little “selling” of area

B10. How are PRAs updated to reflect operational experience and other new information?

Over the years, with the accumulation of operational experience from actual events and lessons from analyses, PRA models have been expanded to include new accident scenarios, and the values of PRA model parameters have been re-estimated. PRA standards and guidance help ensure that PRAs used to support decisionmaking reflect current knowledge.

PRA models can be characterized by (1) their structure (often represented using event trees and fault trees) which identifies possible accident scenarios and (2) the numerical values of their parameters (e.g., component failure rates). An important example of PRA model structure changes spurred by actual events is provided by the introduction of detailed models for fire scenarios following the 1975 Browns Ferry fire (Scott, 1976). WASH-1400 (NRC, 1975), which was published as a draft report in 1974 and as a final report in 1975, provided only a cursory analysis of the risk implications of that fire. As lessons from the fire became available, detailed fire PRA methods and models were developed and used in the industry-sponsored Zion and Indian Point studies (Apostolakis, Kazarians, and Bley, 1982).

An example of model structure changes spurred by analyses is provided by the introduction of analyses for accident scenarios occurring during LP&S operation. Consideration of these scenarios was prompted by a French study (Lanore, 1990), (Brisbois, et al., 1991), which showed that the CDF during LP&S operations could, under some conditions, be comparable to the CDF associated with full-power operations. In reaction to this finding, the NRC sponsored LP&S analyses of two plants (Chu, et al., 1993), (Whitehead, et al., 1994) as a follow-up to the well-known NUREG-1150 (at-power) study (NRC, 1990). Nowadays, the need to consider LP&S risk is internationally recognized.

Regarding the numerical values of PRA model parameters, the NRC has performed a number of studies assessing the statistical performance of selected nuclear components based on operational experience and comparing the results of these assessments against values used in the NUREG-1150 and IPE studies. These component studies, which are documented in the NUREG-1715 series of reports, (e.g., NRC, 2000) indicate general consistency between the statistical estimates and the parameter values used in the PRAs (recognizing that there are uncertainties in both).[12] This is not surprising because PRA component failure rates are derived using operating experience, as explained below. The assumption of exchangeability is approximately valid at the component level.

To provide up-to-date parameter estimates consistent with the latest experience, Bayesian updating statistical methods, as discussed in current PRA standards (ASME and ANS, 2009) and described in guidance documents, (e.g., NUREG/CR-6823 (Atwood, et al., 2003)), are routinely used both in plant-specific PRAs and in generic studies performed to develop industrywide estimates for such parameters as initiating event frequencies. An example of the latter is provided in NUREG/CR-6928 (Eide, et al., 2007). An interesting observation is that initiating event frequencies used in early PRAs, and which were derived from expert judgment, turned out to be higher than estimates produced by statistical analysis.

B11. What is the current international status of nuclear power plant PRAs and risk-informed decisionmaking?

PRAs, often referred to as PSAs internationally, are widely used to support safety-related decisionmaking. Increasingly, these PRAs are Level 2 studies that cover internal events, internal hazards (notably fires and floods), and external hazards.

A recent survey of probabilistic safety assessment (PSA) uses is provided by a Nuclear Energy Agency (NEA) survey of NEA members and participating nonmembers (OECD, 2012).

As described in the NEA report and summarized by Lanore, Siu, and Amri (2012), the performance of PSA is widely encouraged in the surveyed countries. In many countries, PSAs are required for operating plants, typically as part of the documentation required for periodic safety reviews. Moreover, most of the surveyed countries require PSAs for new plants. Several of the countries have defined and use reference values for risk and associated metrics (e.g., CDF), and several countries have developed PSA standards and guidance documents.

[12] The NRC has also sponsored a series of studies investigating the statistical performance of selected systems typically modeled in PRAs. These system studies are documented in the NUREG/CR-5500 series of reports (e.g., Poloski, et al., 1998). As discussed by Rasmuson, et al. (1998), the statistical- and PRA-based estimates for system reliability were reasonably consistent for four of the five system types compared; the comparison for the fifth system was hindered by limited operational data.

The report observes that all operating plants in the surveyed countries have a Level 1 internal events PSA and that many have Level 2 (or at least a "Level 1+"[13]) PSA. In several cases, the PSAs address events during LP&S operation as well as at-power operation, and many address internal hazards (fires and floods) and external hazards (e.g., earthquakes, high winds, floods) as appropriate to their sites. In general, many of the surveyed countries are moving to develop Level 2 PSAs that (1) address all modes of operation (at-power, low-power, and shutdown conditions) and all initiating events (including those caused by internal and external hazards) and (2) are regularly updated. The report also observes a wide range of PSA applications, including design evaluations, operational decisions (e.g., regarding the timing of refueling outages and the performance of maintenance), and the analysis of operational events.

B12. Is PRA limited to event tree/fault tree analysis?

The definition of PRA is sufficiently broad to accommodate any systematic engineering analysis that answers the three fundamental questions: What can go wrong? How likely is it? What are the consequences?

Most current U.S. PRA efforts are focused on Level 1 PRA. These analyses typically use logic-based models (event trees and fault trees) to define accident scenarios and to compute the probabilities of these scenarios. However, PRAs can employ alternate modeling approaches as long as these are appropriate to the needs of the decision problem. Thus, for example, Level 2 PRAs make heavy use of phenomenological models when developing event trees that delineate possible scenarios following core damage, and Level 3 PRAs typically use Monte Carlo simulation approaches to assess the potential offsite consequences of radiological releases. NUREG/CR-2300 (ANS and IEEE, 1983) provides an early, but nevertheless still quite useful, compilation of modeling approaches.

Within the PRA research community, some alternate approaches have been suggested to improve the fidelity of current PRAs in analyzing complex accident scenarios. Many of these alternate approaches, referred to as "dynamic PRA" (or "dynamic PSA" in the international community), are focused on more detailed modeling of the behavior of a plant and its operators over time (in a probabilistic context). The potential capabilities of these approaches are discussed by Siu in a survey article (Siu, 1994), and the subject continues to be a matter of active interest in international PRA conferences and workshops (e.g., Smidts and Aldemir, 2007). To date, partly because of modeling and computational complexities, there have been no dynamic PRA applications affecting regulatory decisions. With continuing increases in available computational power and with progress in the treatment of key technical issues (e.g., regarding the modeling of operating crew behavior), ongoing research efforts may result in practical tools aimed at current regulatory needs (e.g., Coyne, et al., 2012).

[13] A Level 1+ PSA is a PSA that groups core melt sequences into different categories based on containment behavior but does not include actual calculations of the amount, composition, and timing of radioactive material release.

4. CDF—CURRENT STATE OF KNOWLEDGE

CDF, the frequency of core damage, is a metric used widely in risk-informed decisionmaking. The following frequently asked questions (FAQs) address the definition of core damage and our current state of knowledge regarding the numerical value of CDF for U.S. plants.

C1. What is "core damage"?
C2. What do global accident statistics tell us about core damage frequency (CDF)?
C3. What are U.S. PRA studies telling us about CDF for operating plants?
C4. Are PRA estimates for CDF "better" than global statistical estimates?
C5. What is ΔCDF? How is it used in regulatory decisionmaking?
C6. What does our current state of knowledge regarding CDF tell us about the probability of a future accident anywhere in the United States?
C7. How large are the uncertainties in CDF?

C1. What is core damage?

In a risk-informed regulatory context, "core damage" is a state in which the nuclear fuel in the core is damaged to an extent that radioactive material released from the fuel, should it escape to the environment, could significantly affect public health and safety.

The preceding definition is from NUREG-2122 (NRC, 2013c).[1]

Past probabilistic risk assessments (PRAs) have considered a variety of conditions to represent the end point of a Level 1 analysis (see Table 4-1 for some examples), ranging from the onset of damage through core melt.

Table 4-1 Example Definitions of Core Damage and Core Melt

PRA	Terms	Notes
NUREG-1150 draft (NRC, 1987a)	Severe core damage, core melt	• A severe core damage accident occurs when reactor conditions have degraded sufficiently to threaten loss of core cooling. • Core melt (large portions of the fuel becoming molten and penetrating the reactor pressure vessel) would occur if the accident is not terminated.
NUREG-1150 final (NRC, 1990)	Core damage	Core damage accidents involve core uncovery with reflooding not imminently expected. Operationally, • Pressurized-Water Reactors (PWRs): reactor water level drops to a point at the top of the active fuel. • Boiling Water Reactors (BWRs): reactor water level drops to a point 2 feet (0.6 meters) above the bottom of the active fuel
Individual Plant Examinations, as summarized in NUREG-1560 (NRC, 1997c)	Core damage, core melt	• Core damage: "uncovery and heatup of the reactor core because of a loss of core cooling to the point where prolonged clad oxidation and fuel damage is anticipated." • Core melt: "severe damage to the reactor fuel and core internal structures following the onset of core damage, including the melting and relocation of core materials." • Notes that submittals have used several definitions of core damage (e.g., involving peak cladding temperature, oxidation levels, or water level in the vessel), but all would release a substantial amount of gap activity (equivalent or greater than the design basis).

As pointed out by NUREG-2122 (NRC, 2013c), the terms "core damage" and "core melt" are sometimes incorrectly considered as being synonymous; the former refers to a less severe state than the latter.[2] However, some past PRAs have argued that, for practical purposes, this difference does not affect the results for risk-significant scenarios because of the relatively short time gap between the two states. In other words, the PRA model lacks sufficient detail to make a significant distinction, effectively assuming that scenarios severe enough to cause core damage are likely to cause core melt before any realistic chance of terminating the accident.

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[1] This definition is adapted from that provided in the ASME/ANS PRA Standard: "uncovery and heatup of the reactor core to the point at which prolonged clad oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects." (ASME and ANS, 2009)

[2] Core melt occurs after the onset of core damage. Core damage does not necessarily indicate that the reactor fuel has melted, only that radioactive material could be released from the core into the reactor vessel.

NUREG-1560 (NRC, 1997c) generally concurs, but notes that the differences can sometimes matter, depending on whether and how the performance of various systems is credited following core uncovery.

NUREG/CR-6042 reports that during the Three Mile Island (TMI) accident, there was a gap of about 40 minutes between the time of initial fuel uncovery (and initial fuel damage) and initial fuel melting. There was an additional delay of some 80 minutes before large-scale fuel movement (Haskin, et al., 2002). In the case of the Fukushima Dai-ichi 1, 2, and 3 accidents, MELCOR analyses sponsored by the U.S. Department of Energy (Gauntt, et al., 2012) indicate that there may have been a 1–2-hour gap between initial fuel damage and initial fuel movement, and an additional 2–3-hour delay until major fuel movement for Units 1 and 3. (The core debris in Unit 2 does not appear to have moved to the lower portion of the reactor pressure vessel.) The results of these analyses appear to be generally consistent with data and analysis results from the Tokyo Electric Power Co. (TEPCO, 2012a) and more recent analysis results sponsored by the Electric Power Research Institute (EPRI) (Luxat and Gabor, 2013). The treatment of the recovery activities following the onset of core damage will lead to more realistic assessments of risk.

Regardless of the precise formulation, and even discounting the potentially important difference between core damage and core melt,[3] all of these definitions indicate a state of widespread fuel damage (potential or actual), which is an important precursor to potentially significant radiological releases and subsequent public health and safety impacts.

C2. What do global statistics tell us about core damage frequency (CDF)?

Recent estimates for CDF, based on worldwide statistics for reactor accidents, range from about 1×10^{-4} to 2×10^{-3} per reactor-year. These point estimates do not reflect the diversity of designs and operational practices in the worldwide fleet.

After the reactor accidents at Fukushima Dai-ichi plant, statistical estimates for CDF have been developed by a variety of authors, including Cochran (2011, 2012), Gallucci (2012), Kaiser (2012), and Lelieveld, Kunkel, and Lawrence (2012). These estimates, shown in Table 4-2, are derived by (1) defining the population of reactors of interest (e.g., all reactors, all power reactors, all boiling water reactors (BWRs)) and (2) taking the ratio of the number of reactor accidents experienced by that population and the number of years of operating experience for the population:

$$CDF= \frac{N}{T} \qquad \text{(C2-1)}$$

where

N = number of reactor accidents

[3] NUREG/CR-7177 (Krall, et al., 2014) documents a recent investigation of the potential effect of different accident sequence end state definitions.

T = total number of operating reactor-years[4]

Underlying this simple formula is the key assumption that reactor accidents are exchangeable events (i.e., that they are independent events generated from a population of nominally identical reactors). As discussed below and in FAQ R10, satisfying this assumption for nuclear power plants is problematic.

Consider, for example, the upper end CDF estimate of 2×10^{-3} per reactor-year (Cochran, 2011). This estimate is derived by focusing on BWRs with Mark I or Mark II containments and by treating the Fukushima Dai-ichi accidents as three independent events. The estimate treats all U.S. and international BWRs with Mark I or Mark II containments as being the same. Clearly, this is a strong simplification, as it does not account for important differences across BWR design types (which affect, for example, postaccident cooling requirements, and specify the cooling systems and operational procedures provided to meet these requirements) and across sites (including such design features as the spatial arrangement of equipment and the number of emergency diesel generators provided, and environmental features such as the site-specific external hazards). Within the United States alone, current PRAs indicate that these differences can lead to over an order-of-magnitude difference in CDF estimates.[5] The wide variety of physical and operational (including regulatory) environments for plants worldwide can lead to even larger variations.

Table 4-2 Some Recent Statistical Point Estimates of CDF

Source	CDF Estimate(s)	Notes*
Cochran, 2011	8×10^{-4}/RY (World power reactors) 2×10^{-3}/RY (World BWRs with Mark I or II containments)	1, 2
Cochran, 2012	4×10^{-4}/RY (World power reactors)	3
Gallucci, 2012	1×10^{-4}/RY (U.S. power reactors, inland sites) 3×10^{-4}/RY (World power reactors, coastal sites)	4
Kaiser, 2012	2×10^{-4}/RY (World power reactors)	5
Lelieveld, Kunkel, and Lawrence, 2012	2×10^{-4}/RY (World power reactors)	6

*Notes—

1. Power reactor estimate based on 11 events involving reactors connected to the grid at some point (Sodium Reactor Experiment, Fermi 1, Chapelcross 2, St. Laurent A-1 and A-2, TMI, Chernobyl, Greifswald 5, Fukushima Dai-ichi 1–3) and 14,400 reactor-years of operation.
2. BWR Mark I and II estimate based on three events (Fukushima Dai-ichi 1–3) and 1,900 reactor-years of operation.
3. Estimate inferred from data presented in (Cochran, 2012): six events (after eliminating events indicated as being questionable): Fermi 1, TMI, Chernobyl, and Fukushima Dai-ichi 1–3. Operational experience is 15,400 reactor-years.
4. Considers estimated conditional core damage probabilities developed by the U.S. Accident Sequence Precursor Program (NRC, 2014a) in addition to six events (TMI, St. Laurent A-2, Chernobyl, and Fukushima Dai-ichi 1–3) with an International Atomic Energy Agency (IAEA) International Nuclear and Radiological Event Scale (INES) rating of Level 4 or higher. Uses

[4] Because reactors cannot operate at full power all the time (shutdowns, both scheduled and unscheduled, are needed for maintenance and refueling), the number of operating years for a reactor is shorter than the number of calendar years the reactor has been available for operation. For the rough estimation purposes of this report (see Appendix B for details), the numerical difference between operating years and calendar years is relatively small and is ignored.

[5] For the studies discussed in FAQ C3, the smallest reported BWR CDF is about 4×10^{-6}/RY; the largest is about 8×10^{-5}/RY. (Both of these estimates include contributions from external, as well as internal, hazards.)

operational experience of 14,000 reactor-years and different approaches to select a point estimate from estimated ranges.

5. Considers three events (TMI, Chernobyl, and Fukushima Dai-ichi) with an INES rating of Level 5 or higher. Treats Fukushima as one event. Develops separate estimates for 1979, 1986, and 2011, but the separate estimates are quite similar. Reports 95% confidence interval of [4.4×10^{-5}/RY to 6.2×10^{-4}/RY] for 2011.
6. Considers four events (Chernobyl, Fukushima Dai-ichi 1–3) with an INES rating of Level 7 and 14,500 reactor-years of operation. Paper makes a minor adjustment to the direct statistical estimate (3×10^{-4}/RY), judging that this latter estimate is "high-biased."

A second simplification underlying Equation (C2-1) is its assumption that the risk-significant features of the plants do not change over time. This assumption neglects numerous plant changes made over the years, some of which were prompted by events (e.g., TMI), and others which were prompted by analyses (including PRAs). FAQ C3 indicates that there has been a general decrease in CDF from the time of the individual plant examination/individual plant examination of external events (IPE/IPEEE) analyses to the present.

In summary, statistical estimates for CDF accident frequency estimates based on global accident statistics require the strong and arguable modeling assumption of exchangeability across plants and over time and are too coarse for formal regulatory decisionmaking purposes. Post-Fukushima global estimates for CDF have not played a significant role in the U.S. Nuclear Regulatory Commission's (NRC's) post-Fukushima response or in other recent regulatory activities. For example, in a response to a public comment received during the Davis-Besse Nuclear Power Station license renewal process, the staff states—

> Basing CDF on global statistical estimates ignores the variations in plant design, variations in operating procedures and variations in regulatory requirements. In addition, basing CDF on global statistical estimates ignores the lessons learned from past accidents including both design and procedure changes. The NRC staff disagrees that the applicant's SAMA analysis is inadequate because the CDF is not estimated generically from direct experience. The SAMA analysis for license renewal … should be evaluated on a site-specific basis. (NRC, 2015)

It should be noted that, because global statistical estimates provide an independent perspective on PRA-based estimates, they are not entirely devoid of value. In particular, large differences between global estimates and PRA-based estimates, in combination with knowledge of the causal factors and chains of events involved in actual accidents, can provide a useful spur to the PRA community to examine its standard modeling practices in light of these differences.

As a technical side note, even if it could be assumed that plants are actually exchangeable, it is important to recognize that Equation (C2-1) only provides a point-estimate, and that there is a significant degree of uncertainty in the true value of CDF because the empirical data are sparse. For example, following the standard methods described in NUREG/CR-6823 (Atwood et al., 2003) and using the event data provided in Cochran (2012), it can be shown that the 5th and 95th percentiles for the worldwide CDF are about 2×10^{-4}/reactor-year (RY) and 7×10^{-4}/RY, respectively, as shown in Figure 4-1.[6] (Per Table 4-2, the point estimate is 4×10^{-4}/RY.) Using the same approach with just U.S. light water reactor data (one core damage accident—TMI—in about 3,840 RY), the estimate for the CDF of currently operating U.S. plants is about

[6] Details on the calculations underlying Figures 4-1 and 4-2 are provided in Appendix B.

3×10^{-4}/RY; the 5th and 95th percentiles are about 1×10^{-5}/RY and 8×10^{-4}/RY, respectively, as shown in Figure 4-2.[7]

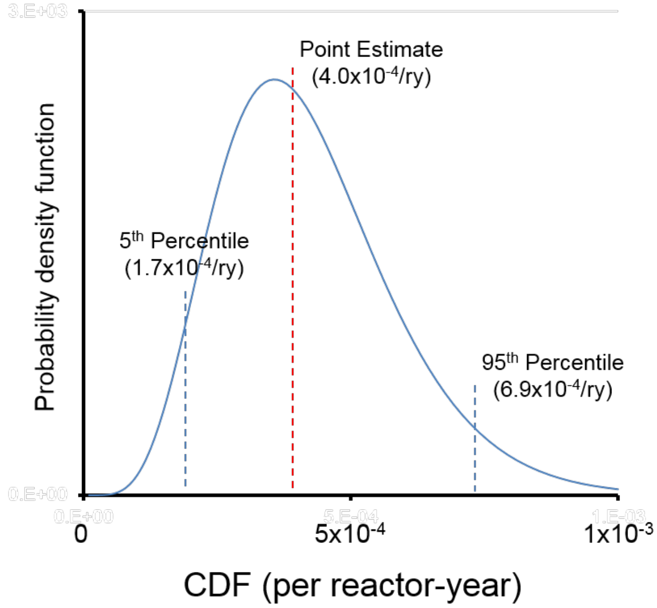

Figure 4-1 Uncertainty band about Cochran (2012) estimate

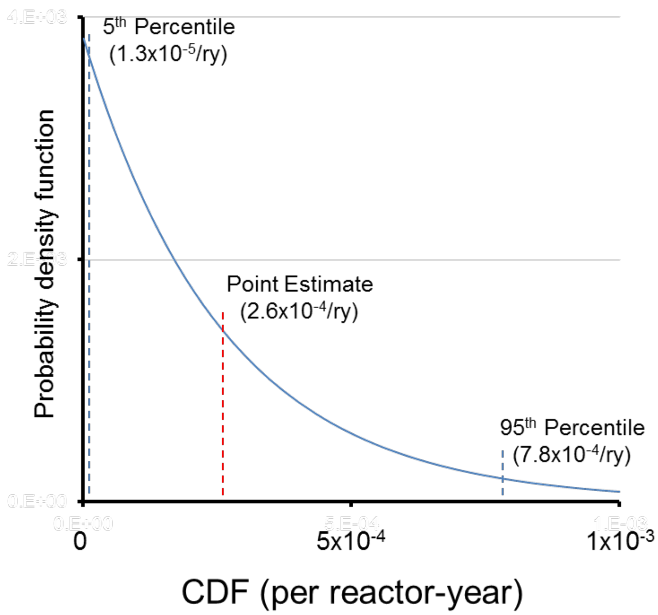

Figure 4-2 Statistical estimate for CDF (U.S. plants only)

[7] The "reverse J-shape" in this figure shows that when very small numbers of events have been observed, extremely small values of CDF cannot be ruled out. In fact, smaller values are more likely than larger ones.

C3. What are U.S. PRA studies telling us about mean CDF for operating plants?

Current point estimates for CDF (for potential accidents occurring during power operation including external as well as internal initiators) range from about 4×10^{-6} to 1×10^{-4} per reactor-year. The CDF estimates and the dominant contributors to CDF are plant specific. The range in quantitative and qualitative results reflects the diversity of plant configurations (design and operations), the particular plant configuration at the time the PRA was performed, and the modeling approach used in the PRA.

Figure 4-3 illustrates the results of a review of recent licensee point estimates of CDF.[8] This review identified estimates for 61 units that included CDF contributions from internal-event, internal-hazard, and external-event scenarios. Most of these estimates were developed to support various risk-informed license amendment requests (LARs) to change requirements governing such things as allowed equipment outage times, containment leak rate testing, or plant fire protection. These LARs can be found using the NRC's Agencywide Documents Access and Management System.

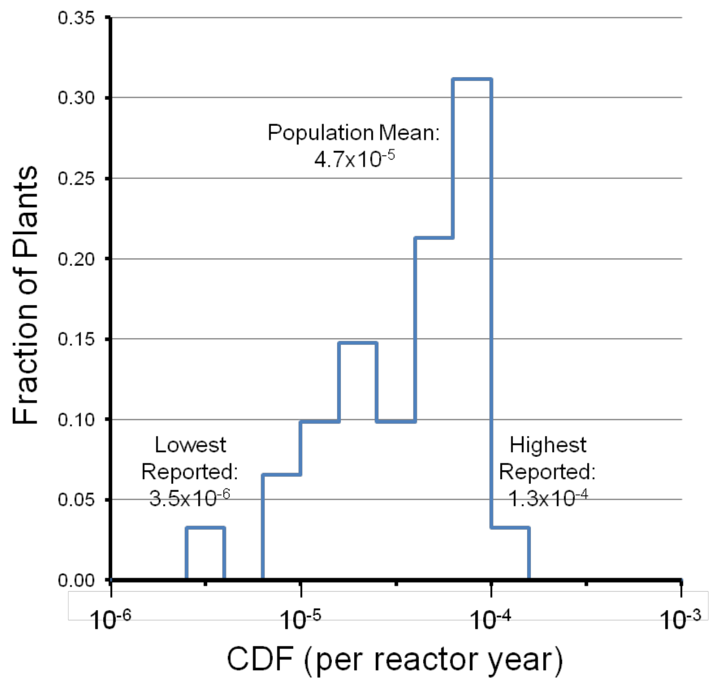

Figure 4-3 Distribution of recent point estimates for total CDF, U.S. plants (Sources: LAR submittals and SAMA analyses)

A few of the CDF estimates identified in the review were developed to support analyses of potentially cost-beneficial severe accident mitigation alternatives (SAMAs). These analyses are performed as part of the Environmental Impact Analyses conducted by licensees applying for operating license extensions. The NRC staff's reviews of these SAMA analyses, documented in plant-specific supplements to NUREG-1437 (NRC, 2013b), provide both the total reported CDF and the key contributors to that total. Importantly, the reviews also typically provide a discussion of the sources of difference between the current CDF estimate and the estimate provided by the

[8] See FAQ C7 for a discussion of uncertainties.

licensee's IPE and IPEEE submittals. These differences, some of which involve increases in CDF, arise because of both plant changes and PRA modeling improvements.

The earliest result included in Figure 4-3 is from a 2002 analysis. However, most of the results are more recent; over 80 percent of the results are from 2008 or later. The figure shows that the estimates range from around 4×10^{-6}/RY to 1×10^{-4}/RY. The median is about 5×10^{-5}/RY, and the population mean is also about 5×10^{-5}/RY. It is important to recognize that these statistics are characteristics of the population variability distribution for U.S. plant mean CDFs; they are not the same as the percentiles for CDF discussed in FAQs B4 and C2. The latter represent epistemic (i.e., state-of-knowledge) uncertainties in the CDF for a particular plant.[9]

Similar to the IPE and IPEEE analyses, the LAR and SAMA analyses typically do not provide quantitative estimates of the uncertainty associated with their CDF estimates.

Overall, it is important to recognize that—

- As discussed in FAQ R10, past PRAs have consistently shown that potential vulnerabilities (and therefore plant risk) are highly plant specific.

- Design and operational changes addressing lessons identified by PRAs can lead to significant changes in CDF. This is illustrated by (1) a general decrease in CDFs from the IPE and IPEEE analyses to the present (see Figure 4-4), some of which can be attributed to plant improvements as discussed earlier, and (2) the small CDF values reported for new reactor designs (which have benefitted from the lessons of past PRAs). Current new reactor estimates range from around 4×10^{-8} to 2×10^{-6} per RY.

- The above estimates for total CDF are developed by adding the CDFs estimated for different accident scenarios.

- The CDF contributions from accidents caused by internal hazards (e.g., floods, fires) and external events (e.g., earthquakes, high winds, and external floods) can be significant. When reviewing CDF estimates, the reader should ensure that the estimates come from PRAs with the same scope.

[9] In the case of FAQ C2, the "particular plant" is any plant, since the assumption of exchangeability treats all plants as being alike.

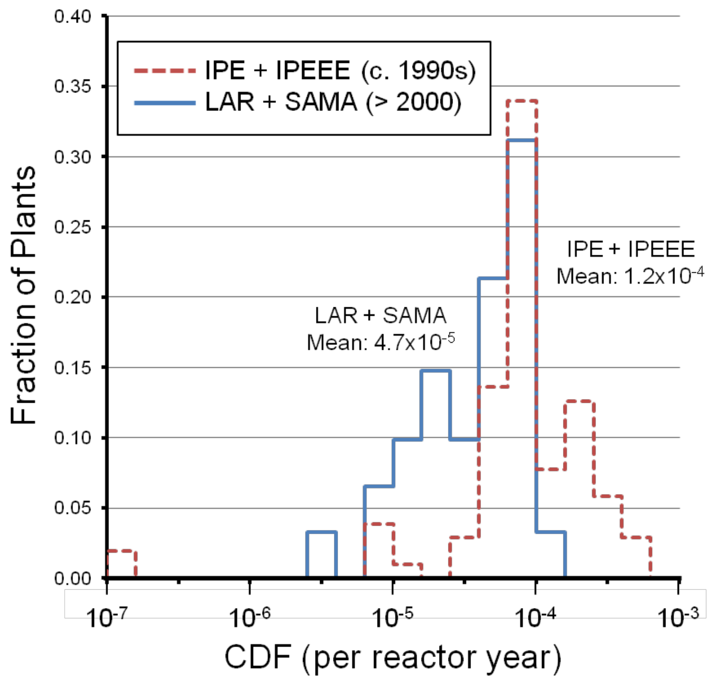

Figure 4-4 Comparison of recent and past estimates for total CDF, U.S. plants (Sources: IPE, IPEEE, LAR, and SAMA analyses)

C4. Are PRA estimates for CDF "better" than global statistical estimates?

PRA estimates are well-suited for regulatory decisionmaking support because they reflect the current state of knowledge regarding plant design, operation, and regulation.

Regulatory decisionmaking needs to be based on information that accurately reflects the current situation. Global statistical estimates, with their fundamental (and flawed) assumption of exchangeability, do not account for differences between plants and changes over time (including, for example, the numerous changes that have occurred following the Three Mile Island Unit 2 accident). Thus, these numerical estimates do not provide an adequate basis for regulatory decisionmaking.

FAQ R10 provides additional discussion on the benefits of PRA as a decision support tool.

C5. What is ΔCDF? How is it used in regulatory decisionmaking?

ΔCDF, read "delta CDF," is the result obtained by subtracting one CDF estimate (typically the CDF for a plant before a potential change) from another CDF estimate (typically the CDF after the potential change).

ΔCDF provides an important measure of the difference between two decision options. Moreover, because the "before" and "after" estimates are based on the same PRA model (save for those portions of the model that have been modified to reflect the "after" conditions), ΔCDF is less sensitive to the effect of many PRA uncertainties than the before/after estimates of CDF themselves.

ΔCDF plays an explicit role in several regulatory applications. Example applications include the evaluation of licensee-proposed risk-informed changes to a plant's licensing basis, as discussed in Regulatory Guide (RG) 1.174 (NRC, 2011a); the NRC's Significance Determination Process (NRC, 2011c); the NRC's evaluation of potential generic issues discussed in NUREG-0933 (NRC, 2011d); and the NRC's evaluation of proposed regulatory backfits per the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.109, "Backfitting" (CFR, 2007a).

Figure 4-5 shows how both ΔCDF and CDF are used in RG 1.174 to determine regions of acceptability and of regulatory attention. Particularly noteworthy is the assignment of a particular value of ΔCDF (1×10^{-6}/RY) to indicate when a change in CDF is considered to be "small."

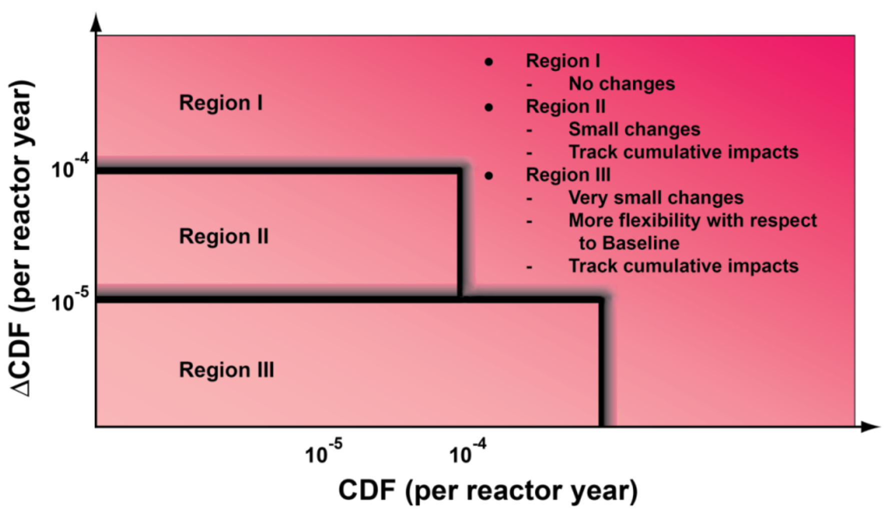

Figure 4-5 Acceptance guidelines for CDF from RG 1.174 (NRC, 2011a)

Figure 4-5 also shows that the boundaries between the different regions of acceptability and attention are not crisp. This reflects a fundamental precept of risk-informed decisionmaking: the numerical results of a PRA, although important, should not provide the sole basis for a decision.

C6. What does our current state of knowledge regarding CDF tell us about the probability of a future core damage accident anywhere in the United States?

Given current conditions, the probability of a core damage accident anywhere in the United States is about 5×10^{-3} per year.

The frequency of a core damage accident anywhere in the United States is the sum of the CDFs for all U.S. plants. For the currently operating fleet, and assuming an average CDF derived from LARs and SAMA analyses that addressed internal events, internal hazards, and external events, this frequency is roughly 5×10^{-3}/yr. The probability of such an accident in the next T years is, for most conditions of interest [10], the product of this frequency times T (assuming that the situation remains reasonably constant (i.e., that there are no significant changes in the plants, in the external environment, or in the PRA community's state of knowledge)). Thus, the probability of an accident in the next year is 0.005 [11], and the probability of an accident in the next 10 years is 0.05. These results are, of course, based on the key assumption that, in the future, the fleet will be the same as it is today, without improvements because of operating experience, new technologies, regulatory changes, or substantive changes in the number of plants. Given the significant changes in regulatory requirements following the Fukushima accident, this assumption is clearly not true.

To provide a perspective on the importance of this assumption, consider the results of the IPE and IPEEE studies, which were completed in the middle 1990s. Using the results of studies that provided CDF estimates for internal hazards and external events as well as internal events,[12] the total accident rate for a 100-reactor fleet was roughly 1×10^{-2}/yr. This translates to a 10-year nationwide probability of 0.1. The reduction in probability is because of both actual plant changes and to PRA modeling improvements since the completion of the IPE and IPEEE studies (see FAQ C3).

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[10] See Appendix A for additional discussion.

[11] As a historical note, WASH-1400 (USNRC, 1975) estimated the total accident rate for a 100-reactor fleet to be around 5×10^{-3}/yr. The similarity with the estimate provided above is largely coincidental, as WASH-1400 did not provide quantitative estimates for the CDF contributions due to external events (notably seismic events) or internal hazards (notably internal fires and floods).

[12] A number of IPEEE studies used "margins" approaches that did not require the estimation of seismic or fire CDF.

C7. How large are the uncertainties in CDF?

The reported ranges for CDF from past full-scope PRAs are typically an order of magnitude or greater. These ranges typically represent the uncertainties associated with the uncertainties in PRA model parameters; they typically do not include contributions from uncertainties in the PRA models themselves, and do not include contributions from scenarios outside the scope of the analyses (e.g., security-related scenarios or scenarios involving aging).

Even considering the experience from the Fukushima Dai-ichi reactor accidents in 2011, core damage events remain unlikely. With relatively little empirical evidence for key PRA model inputs, the uncertainties in PRA results are considerably larger than those associated with traditional engineering analyses.

Limited information is available to provide quantitative estimates of the uncertainty in the results of current PRAs. The current PRA ASME/ANS standard requires the characterization (but not quantification) of uncertainties. Although this helps ensure a qualitative understanding of the sources and effect of parameter and model uncertainties,[13] it does not provide a direct answer to the question.

Quantitative information regarding uncertainties in past PRA estimates is available from some past full-scope studies.[14] Figure 4-6 shows the results for the NRC-sponsored NUREG-1150 (NRC, 1990) study,[15] and for some industry-sponsored studies as reported by Garrick (1989). All of the latter are for pressurized-water reactors (PWRs). It can be seen that the ratio of the 95th to 5th percentile is nearly a factor of 10 or greater for most of the studies.[16]

It should be noted that all of the studies whose results are shown in Figure 4-6 were performed in the 1980s. Since that time, the plants have made numerous changes,[17] and new information from experiments (e.g., regarding the likelihood of fire-induced spurious operations), analyses (e.g., on the seismic hazards faced by plants in the central and Eastern United States) and events (e.g., the Fukushima Dai-ichi reactor accidents) has come to light. The effect of these changes (some of which will tend to reduce the plant CDF and others which will tend to increase it) on the uncertainty in full-scope PRA estimates is not clear.

[13] As discussed in NUREG-1855 (NRC, 2013a), PRA uncertainties are typically classified as parameter, model, or completeness uncertainties. The effect of parameter uncertainties on the overall CDF estimate can be estimated in a straightforward manner using standard PRA tools and techniques. Model uncertainties, although quantifiable in principle, are typically treated using sensitivity studies consistent with current guidance. Completeness uncertainties, some of which arise from scope limitations and others of which are from “unknown unknowns,” are typically not quantified.

[14] A full-scope PRA addresses scenarios triggered by internal hazards (e.g., internal fires, internal floods) and external hazards (e.g., earthquakes, external floods, high winds) as well internal events.

[15] The NUREG-1150 studies did not report the distributions for the sum of the internal events, fire, and seismic CDFs. The approximate totals shown in Figure 4-6 are based on lognormal fits to the separate hazard distributions. (The actual distributions are not lognormal, but the approximation is believed to be good enough for the purpose of this FAQ.)

[16] The reasons for the apparently systematic difference between the NRC- and industry-sponsored studies would require indepth review of the individual studies and is beyond the scope of this report.

[17] Many of the plant-specific supplements to NUREG-1437 (NRC, 2013b) provide detailed information regarding changes to the plant or to the plant’s PRA, and the effect of these changes on the estimated CDF (principally, the internal events CDF).

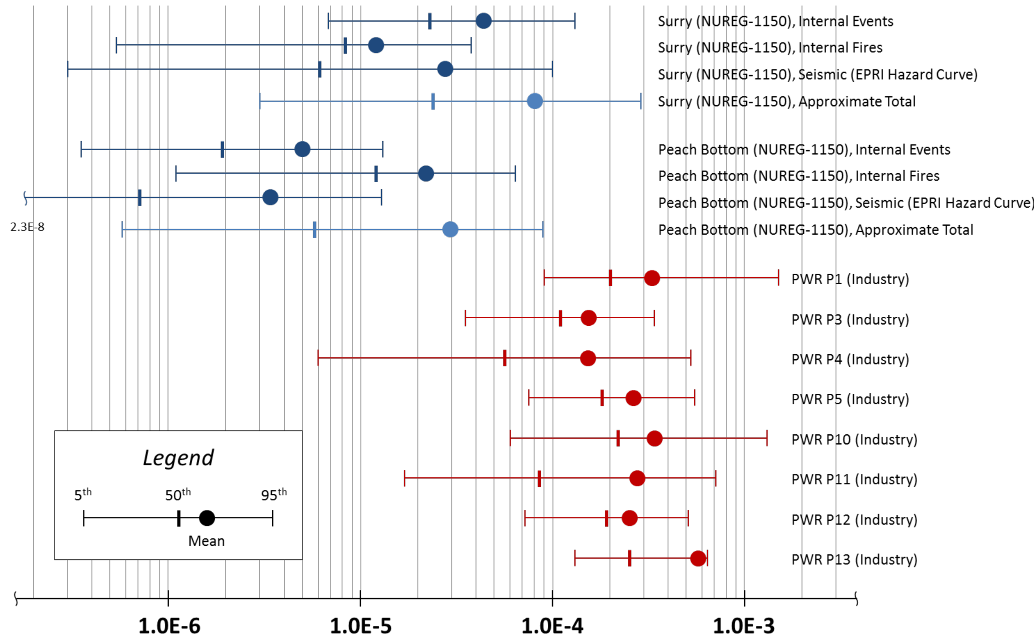

Figure 4-6 Uncertainties in CDF from some past full-scope PRAs

It should also be noted that published estimates of uncertainty typically represent the effect of parameter uncertainties. Model uncertainties are typically treated through sensitivity analyses (e.g., see NUREG-1150's separation of the seismic risk estimates produced using hazard curves developed by Lawrence Livermore National Laboratory and those produced using hazard curves developed by EPRI) or qualitative discussions.[18]

Overall, the uncertainties in PRA results involve orders of magnitude. The reader should keep such uncertainties in mind when reviewing published results that show multiple significant digits.

[18] A number of early industry-sponsored fire PRAs addressed fire model uncertainties through an error factor calibrated using limited experimental data (Siu and Apostolakis, 1982). This approach has not been used in more recent PRAs.

5. LERF—CURRENT STATE OF KNOWLEDGE

LERF, the frequency of a large early release, is another metric used in risk-informed decisionmaking. The following frequently asked questions (FAQs) address the definition of "large early release" and our current state of knowledge regarding the numerical value of LERF for U.S. plants.

L1. What is a "large early release"?
L2. What do accident statistics tell us about large early release frequency (LERF)?
L3. What have past PRA studies told us about LERF for operating plants?

L1. What is a large early release?

In a risk-informed regulatory context, a "large early release" is an event involving a rapid, unmitigated release of airborne fission products from the containment to the environment that occurs before effective implementation of offsite emergency response, and protective actions, such that there is a potential for early health effects.

The preceding definition is from NUREG-2122 (USNRC, 2013c), which mirrors the definition provided in the ASME/ANS Probabilistic Risk Assessment (PRA) Standard (ASME and ANS, 2009).

The concept that timing, as well as the magnitude of an accidental radiological release, is a significant factor in determining public health risk (particularly, the likelihood of prompt fatalities [1]) was recognized in early PRAs, starting from WASH-1400 (USNRC, 1975). However, these PRAs tended to tie the notion of "early" with in-plant accident processes. Thus, for example, NUREG-1150 defined "early containment failure" as a failure "occurring before or within a few minutes of reactor vessel breach" (for pressurized-water reactors) and a failure "occurring before or within 2 hours of vessel breach" (for boiling water reactors). The introduction of offsite actions into the term "large early release" appears to have been made in Draft Regulatory Guide (RG) DG-1061 (USNRC, 1997d), the predecessor to RG 1.174 (USNRC, 1998b).[2]

At present, as discussed in an international survey report (OECD, 2009), there are no quantitative criteria attached to the term "large early release." As discussed in that report, some countries have defined "large release" in terms of specific amounts of radionuclides. For example, Finland and Canada define a large release as involving 100 terabecquerels (TBq [3]) of cesium-137 (Cs-137). The United Kingdom defines a large release as involving 10^4 TBq of iodine-131 (I-131) or 200 TBq of cesium-137 or 200 TBq of other isotopes. To provide some perspective, Table 5-1 presents estimated release timing and amounts for the Three Mile Island, Chernobyl, and Fukushima Dai-ichi reactor accidents.

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1 Prompt fatalities (also referred to as early fatalities) are "deaths from the acute effects of radiation that may occur within a few months of the exposure" (NRC, 2013c).

2 DG-1061 (NRC, 1997d) states that "LERF [Large Early Release Frequency] is being used as a surrogate for the early fatality QHO [the Quantitative Health Objective of the NRC's Safety Goal Policy Statement (NRC, 1986)]. It is defined as the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation."

3 A TBq is a unit of radioactivity. 1 TBq = 10^{12} becquerels (Bq) = 27 curies (Ci) = 2.7×10^{-5} megacuries (MCi).

Table 5-1 Estimated Release Amounts and Timing for Major Nuclear Power Plant Accidents

	Release Magnitude (TBq/MCi)	Time to Core Damage (hr)	Time to Release (hr)	Time to Start Evacuation (hr)
TMI-2	I-131: $0.55/1.5\times10^{-5}$	2	3	54
Chernobyl-4	I-131: $1.2\times10^{6}/32$ Cs-137: $8.5\times10^{4}/2.3$	~0	~0	10.5
Fukushima Dai-ichi	I-131: $1.3\times10^{5}/3.5$ Cs-137: $1.1\times10^{4}/0.3$	Unit 1: 4 Unit 2: 77 Unit 3: 42	Unit 1: 15 Unit 2: 109 Unit 3: 66	6

Sources—
1. TMI-2: (Rogovin, 1980)
2. Chernobyl-4: (OECD, 1996), (USNRC, 1987b)
3. Fukushima Dai-ichi: (GoJ, 2011), (Gauntt, et al., 2012), (JANTI, 2012), (TEPCO, 2012b)

L2. What do accident statistics tell us about large early release frequency (LERF)?

Classifying Chernobyl as the world's single instance of a large, early release, a global statistical point estimate for LERF for the world's reactors is about 7×10^{-5} per reactor-year. This estimate assumes that all plants are exchangeable. Meaningful statistical estimates of LERF for U.S. plants cannot be made independently of U.S. PRA estimates.

To date, the authors have been unable to identify any open literature estimates of LERF based on global statistics.[4] However, such estimates can be made using the same modeling assumptions and Bayesian updating approach used to estimate core damage frequency (CDF) (see FAQ C2 and Appendix B).

At the world level, the statistical evidence is one large, early release [5] in about 15,200 reactor-years (RY) of operation. The resulting mean estimate is about 7×10^{-5}/RY with a 5th percentile of 3×10^{-6}/RY and a 95th percentile of 2×10^{-4}/RY (see Figure 5-1). Just as with previous global estimates of CDF, this estimate assumes that Chernobyl is exchangeable with U.S. reactors, an assumption that is clearly not valid.

[4] Lelieveld, Kunkel, and Lawrence (2012) do estimate the frequency of accidents with INES rating of Level 7, which can be equated to the frequency of large release (but not large, early release). Smythe (2011) presents an analysis of the frequency of different magnitude incidents and accidents, considering facilities (both military and civilian) for nuclear fuel processing and reprocessing, as well as nuclear power plants.

[5] It appears that, by all practical measures, the Chernobyl release was both large and early. However, it should also be noted that classifying Chernobyl as a large, early release per the definition of large, early release in FAQ L1 requires the judgment that there was a potential for early (i.e., acute) offsite fatalities, since there were no such effects from the actual event despite a 10-hour delay until evacuation (OECD, 2002).

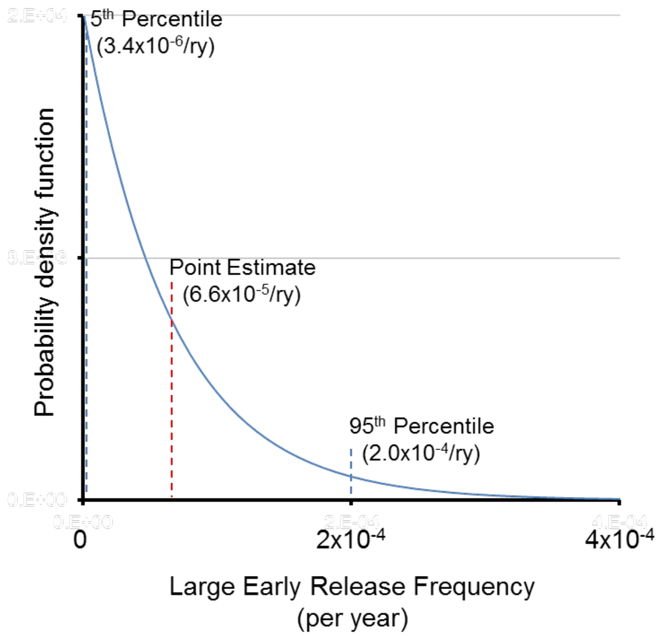

Figure 5-1 Uncertainty band about point estimate for LERF (world reactors)

At the U.S. level, the statistical evidence is zero large, early releases in about 3,840 RY of operation. The resulting mean estimate is about 2×10^{-5}/RY with a 5th percentile of 2×10^{-9}/RY and a 95th percentile of 1×10^{-4}/RY.[6] It should be noted that although any Bayesian estimates of accident frequency based on a small number of observed events are sensitive to the assessor's mathematical representation of knowledge prior to the treatment of observed events (i.e., to the "prior distribution"), the estimates are especially uncertain when there have been no observed events.

It should be strongly emphasized that the numerical results provided in this FAQ are provided only for perspective. In addition to the cautions regarding the key assumption of exchangeability underlying any global statistical estimate, additional caution is warranted because the results are heavily influenced by the assessor's prior distribution. Should a decisionmaking need arise when the experiential evidence is weak, the assessor needs to develop (e.g., through formal expert elicitation) a prior distribution that reasonably represents the actual state of knowledge regarding the unknown quantity.

L3. What have past U.S. PRA studies told us about LERF for operating plants?

Current point estimates for LERF (for potential accidents occurring during power operation, and considering external as well as internal initiators) range from about 8×10^{-8}/RY to 3×10^{-5}/RY. The LERF estimates and the dominant contributors to LERF are plant specific. The range in quantitative and qualitative results reflects the diversity of plant configurations (design and operations), the particular plant configuration at the time the PRA was performed, and the modeling approach used in the PRA.

[6] Appendix B provides details on the calculations behind these results. It should be cautioned that, in this case, the 5th percentile is especially sensitive to mathematical modeling assumptions.

Figure 5-2 illustrates the spectrum of recent licensee estimates of LERF. This figure is based on estimates for 55 units that included LERF contributions from internal-event, internal-hazard, and external-event scenarios. Similar to the CDF estimates discussed in FAQ C3, these plant-specific estimates were developed to support various risk-informed license amendment requests (LARs).[7]

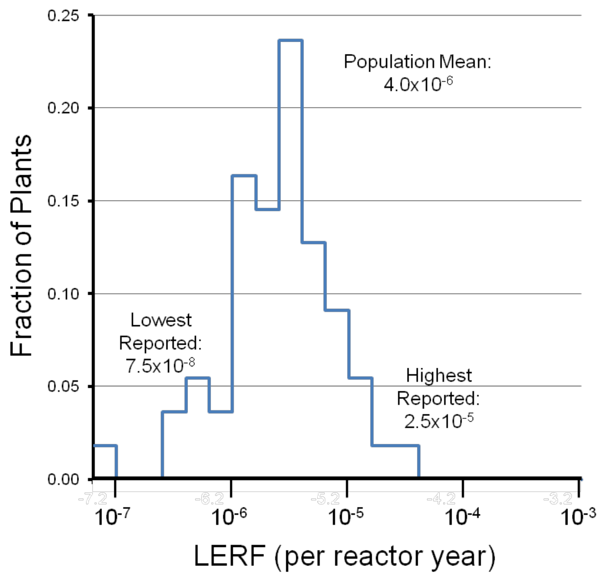

Figure 5-2 Distribution of recent point estimates for total LERF, U.S. plants (from LAR submittals)

The earliest result included in Figure 5-2 is from a 2002 analysis. However, most of the results are more recent; about 85 percent of the results are from 2008 or later. The figure shows that the estimates range from around 8×10^{-8}/RY to 3×10^{-5}/RY. The median is about 3×10^{-6}/RY and the population mean is about 4×10^{-6}/RY.[8] In general, the LAR analyses do not provide quantitative estimates of the uncertainty associated with their LERF estimates.

Many of the observations and cautions provided in FAQ C3 regarding the interpretation of CDF estimates apply to LERF as well:

- As discussed in FAQ R10, past PRAs have consistently shown that potential vulnerabilities (and therefore plant risk) are highly plant specific.

- Design and operational changes addressing lessons identified by PRAs can lead to significant changes in CDF. This is illustrated by a general downward shift in the population distribution for LERF, at least for internal events scenarios, as shown in (Figure 5-3).

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[7] In general, the SAMA analyses do not report estimates for LERF, as this is not a metric used in the evaluation of the different SAMAs.

[8] Similar to the discussion in FAQ C3, these statistics are characteristics of the population variability distribution for U.S. plant mean LERFs; they are not the same as the percentiles for risk metric for a particular plant.

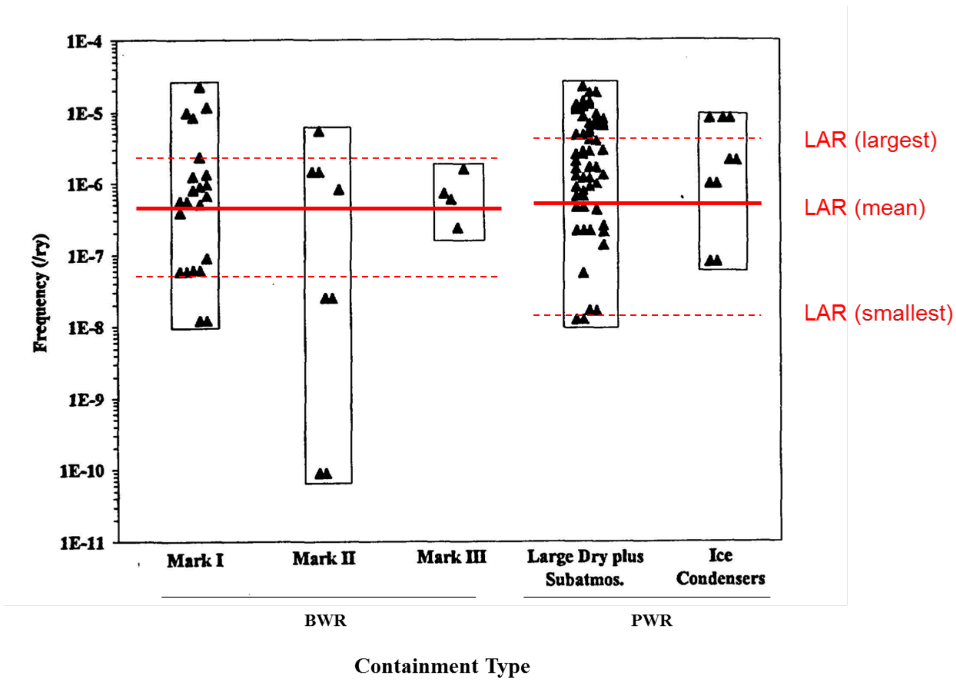

Figure 5-3 Comparison of LAR and IPE estimates for total LERF, U.S. plants (internal events only) (IPE results from NUREG-1560)

- The above estimates for total LERF are developed by adding the LERFs estimated for different accident scenarios.

- The LERF contributions from accidents caused by internal hazards (e.g., floods, fires) and external events (e.g., earthquakes, high winds, and external floods) can be significant. When reviewing LERF estimates, the reader should ensure that the estimates come from PRAs with the same scope.

6. CONCLUSION

In keeping with its 1995 Probabilistic Risk Assessment (PRA) Policy Statement (USNRC, 1995), and as discussed in NUREG-2150 (USNRC, 2012a), the U.S. Nuclear Regulatory Commission (NRC) is increasing its use of the results and insights from PRA models in its regulatory activities. Regarding activities involving nuclear power plants, as discussed in this report's frequently asked questions, the PRA models are well-suited for regulatory decision support because—

- Regulatory decisionmaking concerning safety must be based on the current state of knowledge regarding plant risk.

- Plant risk is affected by such plant-specific factors as design, operation, and environment (physical and regulatory).

- PRA models reflect the current state of knowledge regarding these factors and can be used to assess the impact of potential changes (including changes prompted by regulatory action).

However, PRA models, as with any models, are imperfect representations of reality.[1] Furthermore, unlike many models, PRA models also have to address the possibility of unlikely situations for which empirical data are (fortunately) sparse or even nonexistent. It is, therefore, important from a model validity viewpoint, as well as a broader safety viewpoint, to look for lessons from accidents and other significant operational events if and when such events occur.

Just as the definition of risk has qualitative aspects and quantitative aspects, the lessons learned from operational events can be both qualitative and quantitative. Both can be instructive. From a qualitative perspective, PRA models do not predict the future; instead, they evaluate and assess potential accident scenarios to inform the decisionmakers' current state of knowledge. It is still useful to determine if the observed chain of events during an accident—

- is typically addressed as part of the current PRA state-of-practice, or, if not,
- can be addressed with current PRA methods, models, and tools

(Note that there should be no expectation that a PRA will reproduce the exact chain of events observed. PRA models treat accident scenarios at a discrete, macro level—each scenario represents a bundle of possibilities involving variations in the precise timing and manner of component behaviors, with the latter usually being idealized as being either "successes" or "failures.")

From a quantitative perspective, as discussed in this report, accident frequency estimates based on global accident statistics require the strong and arguable modeling assumption of exchangeability across plants and over time and are too coarse for formal regulatory decisionmaking purposes. (Note that global statistics have not played a significant role in the NRC's post-Fukushima actions.) However, because such estimates provide an independent perspective on PRA-based estimates, they are not entirely devoid of value. In particular, large

[1] British statistician G.E.P. Box is credited with the well-known aphorism "All models are wrong, but some are useful" (Box and Draper, 1987). While not providing a charter for sloppy work, this statement provides an important reminder for analysts and decisionmakers.

differences between global estimates and PRA-based estimates, in combination with knowledge of the causal factors and chains of events involved in actual accidents, can provide a useful spur to the PRA community to examine its standard modeling practices in light of these differences.

Of course, care must be exercised when drawing lessons from major events. Such events are rare occurrences and represent a very small sample of accident possibilities. Statistically based estimates are subject to large uncertainties, and qualitative lessons can be strongly affected by the particulars of a single event. (Consider the lessons that might have been drawn had the March 2011 Tōhoku earthquake led to the same tsunami height at Fukushima Dai-ni but a much lesser tsunami height at Fukushima Dai-ichi.) PRA models are currently being used by the NRC's Accident Sequence Precursor Program to identify noteworthy operational events (e.g., see USNRC, 2014a). As demonstrated by NUREG/CR-6738 (Nowlen, Kazarians, and Wyant, 2001) in its review of major fire events, PRA's structured approach to representing accident scenarios can be used to draw broad qualitative lessons across events. It appears that further exploitation of PRA models to identify, organize, and prioritize lessons during operational experience reviews, such as those conducted following the Fukushima Dai-ichi reactor accidents (e.g., see OECD, 2014) could be fruitful.

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APPENDIX A

THEORETICAL FRAMEWORK FOR PRA TREATMENT OF UNCERTAINTIES

A.1 Introduction

Current nuclear power plant (NPP) probabilistic risk assessments (PRAs) address both (1) "aleatory uncertainties" (i.e., uncertainties arising because of randomness in the occurrence of key events (e.g., initiating events that trigger accidents, equipment failures, core damage accidents)) and (2) "epistemic uncertainties" regarding that randomness, (i.e., uncertainties in the state of knowledge regarding characterizations of randomness (Apostolakis, 2004), (NRC, 2009d)). Following the general approach used by WASH-1400 (NRC, 1975) and formalized by Kaplan and Garrick (1981), event frequencies (per unit time or per demand) are used to characterize randomness. When uncertainties in these frequencies are quantified, this is typically done using probability distributions.

This appendix elaborates on this two-level treatment. In particular, it provides a tutorial discussion of the mathematical framework underlying the notion of frequency (as used in PRA models [1]) and of the concepts of aleatory and epistemic uncertainties in the context of this framework. More detailed discussions can be found in Apostolakis (1978, 1990, 1994), Siu and Kelly (1998), Atwood, et al. (2003), and NUREG-1855 (NRC, 2013a).

A.2 Mathematical Definition of Event Frequency

Of the several dictionary definitions of the term "frequency," one quite close to the concept as addressed by PRAs is "the number of times any action or occurrence is repeated in a given period" (Webster, 1977). This definition is both clear and intuitive. However, it can be read as implying that (1) the notion of frequency is meaningful only when events actually occur (and are therefore countable), and (2) events occur with some regularity (as they would in the case of periodic, physical processes).

In many risk-informed decisionmaking applications, the likelihood of occurrence of events of interest may be such that actual occurrences, although possible and potentially worthy of protective measures, may not be expected over the period of time relevant to a particular decision. In addition, even in a thought experiment involving multiple trials over an extremely long observation period, because of random variations in the factors affecting event occurrence, one would not expect to see the same number of events in each trial.

To address these issues, PRAs use the concept of frequency not as a direct measure of the number of events in a given class, but as a measure of the rate of occurrence of a class of random (but not necessarily likely) events.

[1] FAQ B3, following NUREG-2122 (NRC, 2013c), defines frequency as the expected number of occurrences of an event expressed per unit time. Although less formal, this definition is consistent with the mathematical framework, as discussed later in this appendix. A benefit of the mathematical framework is that it provides a solid, theoretical basis for the simpler definition.

To explain this approach, it is useful to start with the following three observations:

1. At a detailed modeling level, PRAs deal with some physical variables that are, in principle, observable. Example variables include the time to failure of a pump, the number of times a relief valve operates prior to failure, and the time at which an operator takes a particular action at a given point in an accident sequence.

2. Because of limitations in resources, lack of knowledge, or both, PRAs treat some of these variables as being the results of random processes. For such variables, if one employs a thought experiment involving a number of repeatable trials, one can envision observing a distribution of values (e.g., an empirical histogram).

3. PRAs treat the remaining variables as being deterministic (if perhaps imprecisely known). For such variables, if one employs a thought experiment involving a number of repeatable trials, one can envision observing a single value (or, at least, a range of values sufficiently small that representing the range as a point value is adequate for practical applications).

It is important to recognize that there is no fundamental principle dictating whether a variable must be modeled as being random or deterministic. The analyst needs to decide if the notion of repeatable trials makes sense for the problem being addressed. For many situations, PRA modeling conventions developed over years of practical use are well established.[2] In other more novel applications, considerable effort may be needed to develop a satisfactory approach. (See, for example, Siu, 1999 and Siu, et al., 2000 for discussions of the treatment of uncertainties in the consideration of reactor pressure vessel failure caused by pressurized thermal shock.) In general, the analyst needs to clearly define the PRA "model of the world" (i.e., the model of the physical situation at hand) and to treat uncertainties in a manner consistent with that model (Apostolakis, 1990, 1994).

Focusing on the first two observations (the third will be discussed in the following section), the standard PRA model of the world for random events occurring over time is a simple one.[3] In this model, it is assumed that—

(i) The physical process leading to event occurrences is not changing over time.

(ii) The likelihood of an event occurring in a given time period does not affect the likelihood of a separate event occurring in a separate and distinct time period.

(iii) The likelihood of separate events occurring at exactly the same time is vanishingly small.

It should be emphasized that these are modeling assumptions. For example, Assumption (i) does not hold over long (multiyear) time scales for NPPs because of changes (e.g., plant changes prompted by PRA results) made over time. The modeling assumption is that, at the particular point in time of the analysis, the underlying physical processes are reasonably

[2] For example, hardware component failures are typically modeled using simple one-parameter probabilistic models of the type discussed in this appendix. (One can envision much more complicated models involving the mechanistic simulation of component performance under a variety of conditions.)

[3] The discussion in this section focuses on events occurring over time. The treatment of events occurring on a demand basis is similar, as briefly discussed later in this appendix. Additional details are provided in Siu and Kelly (1998) and Atwood et al. (2003).

constant. Regarding Assumption (ii), dependencies between events over time are treated using different models. The modeling assumption is that the PRA model captures all of the important dependencies. A similar point holds regarding Assumption (iii).[4]

Using the three modeling assumptions above, the mathematical laws of probability can be used to show that the probability of observing *R* events in a time period *T* is governed by the Poisson distribution:

$$P(R|\lambda,T)=\frac{(\lambda T)^{R}}{R!}e^{-\lambda T} \tag{A.1}$$

Where P{R|λ,T}, read as the “probability of R given λ and T,” is the probability of observing exactly *R* events in a time interval of length *T*, assuming λ (which has units of inverse time) has a known value. It should be emphasized that, according to this probability model, there is not one “true value” of *R*; different values of *R* (i.e., different numbers of observed events) can arise with different probabilities. The single parameter determining the likelihood of *R* is λ. This parameter is called the “event frequency.” Matching intuition, as either λ or *T* (or both) increase, the likelihood of events also increases (see Figure A-1).

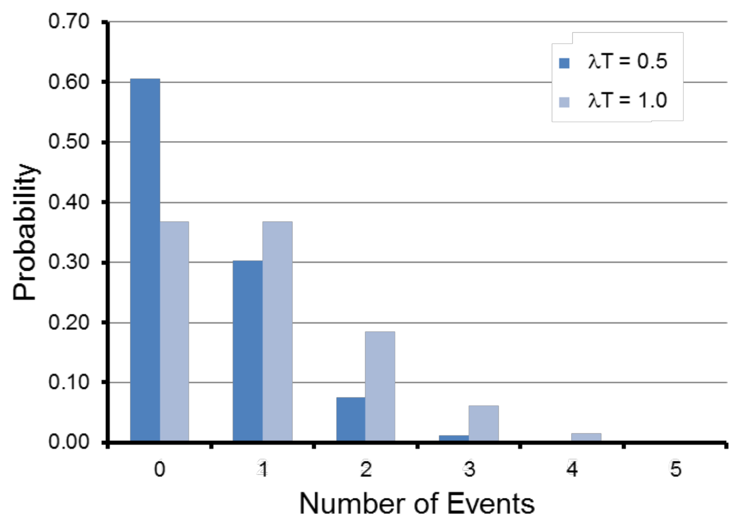

Figure A-1 Effect of increasing λT on event occurrences

Because portraying the full distribution of *R* as in Figure A-1 can be cumbersome, simple metrics are typically used to represent key aspects of the distribution. One such metric is the

[4] Note that parametric common-cause failure models are typically used in current PRAs to treat “simultaneous” component failure events. Arguably, a similar approach would be appropriate for treating “simultaneous” macro-scale events such as the multiple core damage events at Fukushima Dai-ichi. Section A.3 provides additional discussion on such an approach.

average value of R, also called the "mean" and "expected" value of R. By definition, this average value is a weighted sum over all possible values of R:[5]

$$\bar{R} \equiv \sum_{R=0}^{\infty} R \cdot P(R|\lambda,T) \tag{A.2}$$

Working through the math, it can be shown that Equation A.2 reduces to a very simple result:

$$\bar{R} = \lambda T \tag{A.3}$$

or

$$\lambda = \frac{\bar{R}}{T} \tag{A.4}$$

Equation A.4 provides the relationship between the formal notion of frequency discussed in this appendix and the definition given in NUREG-2122.

> To recap—
>
> - PRAs use the Poisson distribution to model the occurrence of events occurring randomly over time.
> - The event frequency is the single parameter of the Poisson distribution.
> - The event frequency is numerically equal to the ratio of the average number of events over a time period T divided by T.

A.3 Additional Topics

A.3.1 Implications of Poisson Model for Event Timing

The preceding discussion has centered on modeling the number of events (R). In PRAs, this is useful for treating events in the detailed PRA models (e.g., initiating events, equipment failures). However, for risk management purposes, the time to the first occurrence of a major event (e.g., a core damage accident) may be of greater importance.[6]

[5] It is important to recognize that the term "expected value," as used in PRA (and in statistics), refers to a mathematically defined quantity. In some cases (e.g., when the random variable follows a Gaussian distribution, where the average value is also the most likely value), this mathematical quantity coincides with everyday notions of expectation. In many other cases of interest to PRA, the average value does not have an intuitive meaning, and should only be viewed as one metric characterizing the random variable's distribution.

[6] Of course, should such a major event occur, the Poisson model would have to be reevaluated, or at least reapplied to new circumstances, since such an event could precipitate changes that would invalidate modeling assumption (i).

It can be shown that for a Poisson process, if T_1 is the time to the first event, then the distribution of T_1 is exponentially distributed.[7] Thus, the cumulative distribution function for T_1 (i.e., the probability that T_1 is less than or equal to some fixed value, denoted by "t") is given by:

$$F_{T_1}(t) \equiv P\{T_1 \leq t\} = 1 - e^{-\lambda t} \tag{A.5}$$

If the product λt is small (say, less than 0.1), then the cumulative distribution for *T* can be approximated very simply:

$$F_{T_1}(t) \approx \lambda t \tag{A.6}$$

For example, if λ equals 10^{-2} per year, the probability that T_1 is less than 1 year (which is equivalent to saying that there will be at least one event in 1 year) is approximately 0.01.

Using Equation A.5, it can be seen that as λ increases, the probability of observing the first event by a specified time also increases (albeit at a decreasing rate). This is illustrated in Figure A-2.

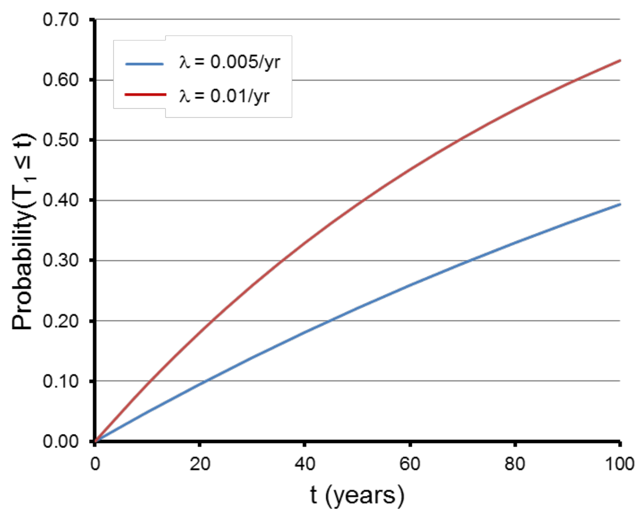

Figure A-2 Effect of increasing λ on cumulative distribution for first occurrence time

As with the random variable *R*, simple metrics are useful for characterizing the full distribution of T_1. In this case, the average (or "mean" or "expected") value of T_1, often called the "return period," is given by:

$$\overline{T_1} \equiv \int_0^{\infty} t \cdot f_{T_1}(t)dt = \frac{1}{\lambda} \tag{A.7}$$

It is important to recognize that this simple result does not mean that, in an experiment involving repeated trials, events will occur at regular intervals of 1/λ. In fact, for a Poisson process,

[7] The converse is also true. If the time to first occurrence is exponentially distributed, then the number of events is Poisson-distributed.

smaller values of T_1 are more likely to be observed than larger values. To show this, consider the probability density function for T_1 (which is proportional to the probability that T_1 "equals" t [8]). This density function is the derivative of the cumulative distribution function given in Equation A.5:

$$f_{T_1}(t) = \frac{d}{dt} F_{T_1}(t) = \lambda e^{-\lambda t} \quad (A.8)$$

The "reverse J-shape" of this function, which increases with smaller values of *t*, is illustrated in Figure A-3.

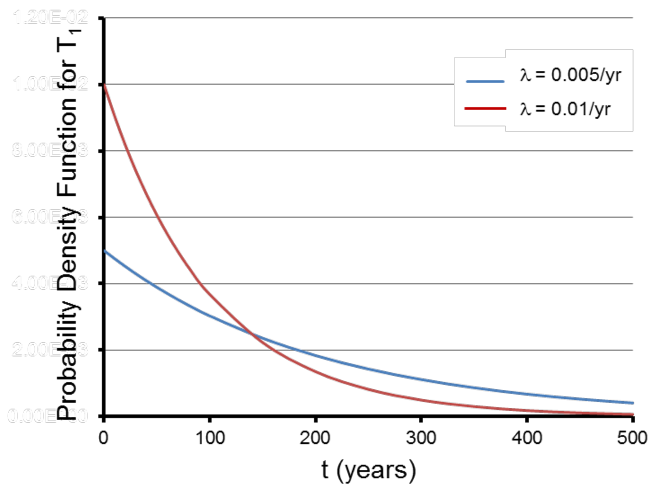

Figure A-3 Probability density functions for T_1

An illustrative time trace for a Poisson process is shown in Figure A-4. In this figure, each diamond represents an event occurrence, and the distance between two diamonds is the time interval between the occurrences. In Figure A-4, the time intervals were obtained by Monte Carlo sampling off of an exponential distribution. It can be seen that, even though the events are independent, the time trace is characterized by clusters of events, with occasional longer gaps.

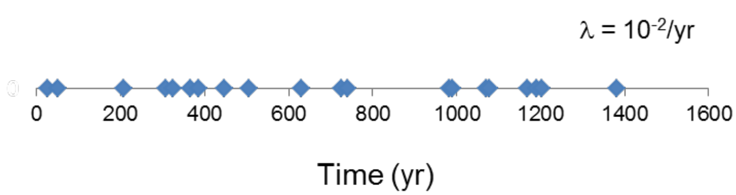

Figure A-4 Example time trace for a Poisson process

Figure A-4 shows that a Poisson model does not imply regularity. If an event has not occurred over a period of time, this does not mean that an event is "due." The chance that an event will

8 More precisely, the probability density function is proportional to the probability that T_1 lies in the interval {t, t+dt}, where dt is an infinitesimally small increment.

occur in the next year is the same chance that would have been computed at the beginning of the event-free period, and the same chance that would be computed years down the road.

The plot also shows that an observation of uneven spacing of intervals between events (whether increasing or decreasing) does not a priori invalidate the Poisson model.

Finally, it should be observed that most of the time intervals shown in Figure A-4 are less than the mean value of T_1 (100 years in this example). For an exponential distribution, the mean value corresponds to the 63rd percentile of T_1. Unlike the mean value for a normal distribution, the mean value for an exponential distribution has no intuitive meaning. It is only a mathematical index that, through an integration that includes the probability distribution function for the random variable (see Equation A.7), provides some treatment of the uncertainty in the random variable.

A.3.2 Treatment of Single Accident Sequences

PRA accident sequences typically comprise an initiating event and a subsequent set of events representing the success or failure of plant systems, structures, and components in accomplishing a defined mission. The initiating events and some of the subsequent events can occur randomly in time and are modeled using the Poisson distribution as discussed in the preceding section. Other subsequent events are modeled as events that occur "on demand," (i.e., in response to some trigger (e.g., a command to operate)). As discussed below, under standard modeling assumptions, the preceding discussion of frequency and event occurrences applies to accident sequences as well as basic events.[9]

Consider a basic event representing the failure of a component (e.g., an emergency diesel generator) to start on demand following an initiating event (e.g., a loss of offsite power). Denote the probability of the event occurrence (i.e., the failure) by *p*. Assuming that—

(i) There are only two possible outcomes (success or failure) given a demand.

(ii) The physical process leading to event occurrences is not changing over time, so the failure probability (*p*) and the success probability (1−*p*) are constant from demand to demand.

(iii) The likelihood of an event occurring on a given demand does not affect the likelihood of a separate event occurring on a different demand (i.e., event occurrences across separate demands are statistically independent).

the process of generating events is called a "Bernoulli process"[10] and the probability of observing *R* events in *N* demands is given by the binomial distribution:

$$P(R|p,N)=\frac{N!}{R!(N-R)!}p^R(1-p)^{N-R} \tag{A.9}$$

[9] As defined in NUREG-2122 (NRC, 2013c), a "basic event" is "[a]n element of the PRA model for which no further decomposition is performed because it is at the limit of resolution consistent with available data." Initiating events and component failures are examples of events that can be treated as basic events.

[10] Also known as the "coin flip process," where *p* is equal to 0.5 if the coin is fair

Using the definition of average value:

$$\bar{R} \equiv \sum_{R=0}^{\infty} R \cdot P(R|p,N) \tag{A.10}$$

it can be shown that the average number of events is given by:

$$\bar{R} = Np \tag{A.11}$$

Consider now an accident sequence (a compound event) comprising an initiating event (generated by a Poisson process with frequency λ_I) and a demand failure event (generated by a Bernoulli process with probability p_B).[11] Using the mathematical laws governing probabilities, it can be shown that accident sequence occurrences follow a Poisson distribution with overall frequency λ_I p_B (Ross, 2014):

$$P(R|\lambda_I,p_B,T)=\frac{(\lambda_I p_B T)^R}{R!}e^{-\lambda_I p_B T} \tag{A.12}$$

A similar result can be developed for accident sequences composed of an initiating event and an arbitrary number of successes and failures following that initiating event.

Thus, the simple relationship for frequencies of events given in Equation A.4 and the exponential distribution for the timing of events given in Equation A.5 hold for accident sequences as well as basic events as long as Assumptions (i) through (vi) are valid. Conversely, if the assumptions are not valid, these equations no longer apply.

A.3.3 Treatment of Multiple Accident Sequences

In a typical NPP PRA model, core damage can result from thousands or even millions of accident sequences. As discussed below, as long as the accident sequences can be modeled using the Poisson distribution, the occurrence of core damage can be modeled using the Poisson distribution with a frequency (i.e., core damage frequency (CDF)) equal to the sum of the frequencies of the contributing accident frequencies.

Consider the simplified example event tree shown in Figure A-5, in which initiating event I can, depending on the particular sequence, be followed by subsequent events A through C. Core damage can result from accident sequences 2, 4, and 5. The frequencies of these sequences are as shown in Figure A-5. (Note that the event probabilities are, in general, dependent on preceding events.)[12]

[11] This generating process is sometimes called a “filtered Poisson” process.

[12] This is not a contradiction of Assumption (vi). That assumption applies to each particular event singly (e.g., event A) and not to combinations of different events (e.g., A and B).

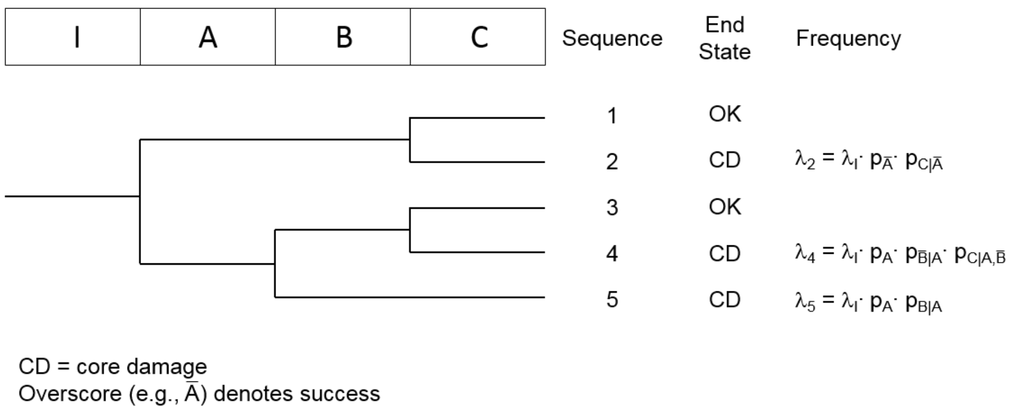

Figure A-5 Example event tree

Figure A-6 shows an equivalent "state-transition diagram" representation of the event tree. This representation emphasizes that the transition from its initial "OK" state to core damage (CD) can occur via one of three paths, and that the (random) times to core damage along each path, T_{CD2}, T_{CD4}, and T_{CD5}, respectively, are governed by the frequencies of the paths (λ_2, λ_4, and λ_5). Core damage will occur by the path which happens to be the quickest.[13] Thus, the overall time to core damage, T_{CD}, is the minimum of the three possible core damage times:

$$T_{CD} = \min(T_{CD2}, T_{CD4}, T_{CD5}) \qquad (A.13)$$

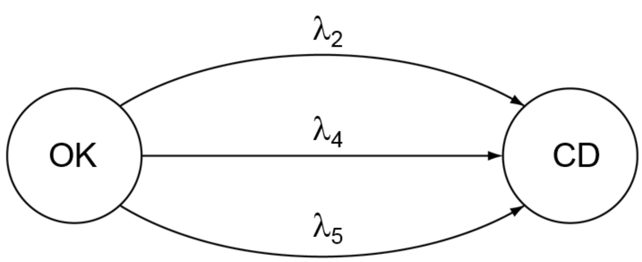

Figure A-6 Equivalent state-transition diagram for example

Using (1) the fact that the path-dependent core damage times are exponentially distributed (see Equation A.5) and (2) Equation A.13, it can be shown that T_{CD} is also exponentially distributed with frequency parameter CDF = $\lambda_2 + \lambda_4 + \lambda_5$:

$$F_{T_{CD}}(t) \equiv P\{T_{CD} \leq t\} = 1 - e^{-(\lambda_2+\lambda_4+\lambda_5)t} \qquad (A.14)$$

It can be seen that this result can be generalized to larger event trees and to full PRA models with multiple initiating events.

[13] In the statistical literature for health care applications, models such as those shown in Figure A-6 are often referred to as "competing risks" models.

To recap—

- The Poisson model for the occurrence of random events over time implies an exponential distribution for the occurrence times of these events.

- The average (or "mean" or "expected") event occurrence time, also called the "return period," is equal to the inverse of the event frequency (i.e., to 1/λ).

- A characteristic time trace for Poisson-distributed events is irregular. Smaller time intervals occur more often than larger time intervals (leading to apparent clustering), and most of the time intervals are less than the average event occurrence time.

- The occurrence of an accident sequence can be modeled using the Poisson distribution as long as initiating event occurrences can be treated using the Poisson distribution and subsequent successes and failures can be treated using the binomial distribution.

- The occurrence of core damage can be modeled using the Poisson distribution with an overall CDF equal to the sum of the frequencies of the contributing accident sequences as long as the occurrence of accident sequences can be modeled using the Poisson distribution.

A.4 Aleatory and Epistemic Uncertainties

The preceding discussion addresses uncertainties because of "inherent randomness." In earlier literature, they are often called "random uncertainties" or "stochastic uncertainties." Using the terminology introduced to the nuclear PRA community by Apostolakis (1994) and Budnitz et al. (1997) and adopted in many regulatory documents, (e.g., Regulatory Guide 1.174 (USNRC, 1998b, 2011a)), they are called "aleatory uncertainties." Their principal characteristic is that they are (or are modeled as being) irreducible; they are defined by the form of the probability distribution (e.g., the Poisson distribution) and the value of the distribution parameters (e.g., λ).

Note that in the examples given earlier, the variability in the uncertain variable (e.g., *R* or *T*) is observable, at least in principle. In other words, repeated observations of the variable will result in an empirical distribution of values. This provides a way to think about aleatory uncertainties—if repeated trials of an idealized thought experiment (in which the conditions are kept constant from trial to trial) will lead to a distribution of outcomes for the variable. This distribution is a measure of the aleatory uncertainties in the variable.

Another type of uncertainty addressed in PRAs is "epistemic uncertainty," which has been called "state-of-knowledge uncertainty" in earlier papers because it is because of weaknesses in the current state of knowledge of the assessor. Uncertainties in a deterministic variable whose true value is unknown are epistemic. Repeated trials of a thought experiment involving the variable will, in principle, result in a single outcome, the true value of the variable.

Unlike aleatory uncertainty, epistemic uncertainty is reducible with the collection of additional information. In PRAs, for example, it is typically assumed that the Poisson model is a good representation for the number of failures of equipment while running, (i.e., it is a good model for the aleatory uncertainties). It is assumed that there is a particular failure rate for each component. Initially, we may not have much failure data for a component, and our epistemic uncertainties in the value of the failure rate will be large. After we collect a large enough sample of failure data, we can get a very good estimate of the failure rate, (i.e., the epistemic uncertainties in the value of the failure rate will be small).

The epistemic uncertainties are quantified using probability distributions. Figure A-7 provides an example showing how the distributions are narrowed (i.e., the uncertainties are reduced) when additional information is collected.[14] In this figure, *R* represents the number of observed failures and *T* represents the total number of component-years (i.e., the cumulative years of experience for all of the components in the population). In both cases, the ratio of *R* to *T* is the same. However, because one case has 10 times the amount of data as the other, the uncertainties in the underlying value of λ are significantly less.

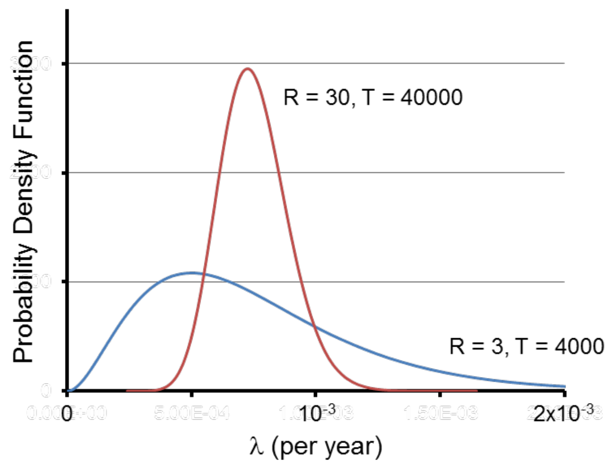

Figure A-7 Reduction in epistemic uncertainty with increased data

Figure A-8 shows how aleatory and epistemic uncertainties can be represented when dealing with event occurrence times as modeled in PRAs. (An analogous representation can be developed for variables representing the number of events in a given time period.) The heavy curves (solid and dashed) are the cumulative probability distributions quantifying the aleatory uncertainties in the event occurrence time. The light curve crossing these heavy curves is the probability density function quantifying the epistemic uncertainties in λ. As discussed in Section A.2, the aleatory distributions are conditioned on the value of λ; the three curves shown correspond to the 5th percentile (λ_{05}), mean ($\bar{\lambda}$), and 95th percentile (λ_{95}) values of λ. Note that PRA results typically focus on characterizing the epistemic uncertainty in λ; the aleatory uncertainties in the observable variable are assumed to be understood.

[14] The method for generating these distributions, given data, is discussed in Appendix B.

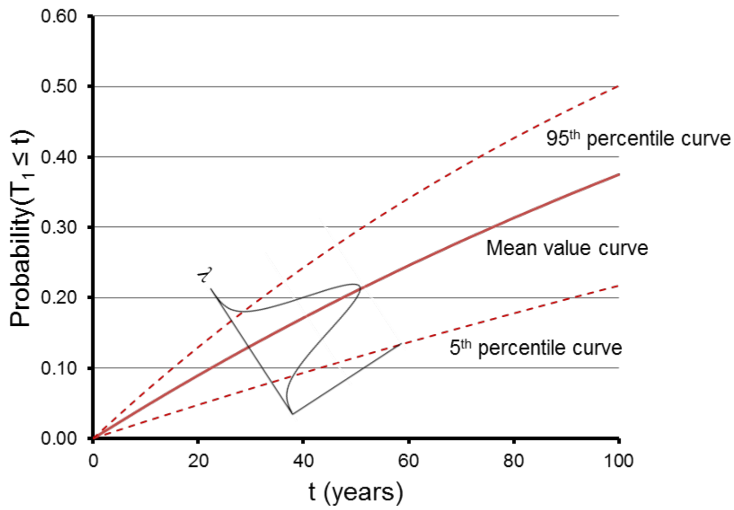

Figure A-8 Representation of aleatory and epistemic uncertainties in event occurrence time

To recap—

- Uncertainties in a variable are treated in PRAs as being aleatory when the variable is assumed to be the result of a random process (i.e., repeated trials of a thought experiment will lead to a distribution of values for the events). Typical aleatory variables are the failure times of components (while running) and the number of times a component fails on demand.

- Uncertainties in a variable are treated in PRAs as being epistemic when the variable is assumed to be deterministic (i.e., repeated trials of a thought experiment will lead to a single value for the variable). Typical epistemic variables are the rate of occurrence of initiating events and the rate of failure of components (per demand or while running).

APPENDIX B

BAYESIAN ESTIMATION OF CDF AND LERF

B.1 Introduction

Figures 4-1, 4-2, and 5-1 in the main body of this report show distributions for plant core damage frequency (CDF) and large early release frequency (LERF) derived from global statistics under a variety of assumptions. Such distributions, which are not discussed in most published estimates, quantify the epistemic uncertainty in the CDF and LERF estimates. These distributions were derived using Bayes' Theorem, as described in this appendix. Additional information on the philosophy and application of Bayesian estimation (in the context of probabilistic risk assessment (PRA)) can be found in many places, including Apostolakis (1978, 1990), Siu and Kelly (1998), and Atwood, et al. (2003).

B.2 Bayes' Theorem

The quantification of the uncertainties in many quantities of interest in PRAs involves the collection and interpretation of a variety of forms of evidence (e.g., model predictions, expert opinion, empirical data) and the application of an appropriate estimation procedure that uses this evidence. Formally, the estimation procedure involves the application of Bayes' Theorem. The general form of this theorem is:

$$\pi_1(\underline{\theta}|E) = \frac{L(E|\underline{\theta})\pi_0(\underline{\theta})}{\int_{\underline{\Theta}}^{\infty} L(E|\underline{\theta})\pi_0(\underline{\theta})d\underline{\theta}} \tag{B.1}$$

Where $\underline{\theta}$ is the vector of epistemic parameters to be estimated; E is the evidence; $\pi_1(\underline{\theta}|E)$ is the posterior epistemic distribution (i.e., the probability distribution for $\underline{\theta}$ *after* observing the evidence); $L(E|\underline{\theta})$ is the likelihood function (i.e., the aleatory probability of observing the evidence if it is known); $\pi_0(\underline{\theta})$ is the prior epistemic distribution for $\underline{\theta}$ (i.e., the probability distribution for $\underline{\theta}$ *prior to* observing the evidence); and the integral on the right-hand side of the equation is performed over all possible values of $\underline{\theta}$ (so the denominator is just a normalization constant).

While Equation B.1 may appear to be complicated, its application is straightforward in many practical cases. In this report, for example, we are interested in estimating λ, the frequency of an event, and the evidence consists of an observation of *R* events in a specified time interval *T*. As discussed in Appendix A, event occurrences are typically modeled in PRAs using the Poisson distribution (Equation A.1, reproduced as Equation B.2 below for convenience).

$$P(R|\lambda,T)=\frac{(\lambda T)^R}{R!}e^{-\lambda T} \tag{B.2}$$

Thus, after some simplification by removing constants that are common to the numerator and denominator, Equation B.1 becomes:

$$\pi_1(\lambda|R,T) = \frac{\lambda^R e^{-\lambda T}\pi_0(\lambda)}{\int_0^\infty \lambda^R e^{-\lambda T}\pi_0(\lambda)d\lambda} \tag{B.3}$$

This equation has analytical solutions for some forms of the prior distribution. For other forms, it can be solved numerically using simple tools (e.g., spreadsheets or equation solving software).

When the evidence is not in the form of *R* events in time *T*, the use of Bayes' Theorem may not be as straightforward. For example, the evidence may consist of expert estimates for λ, perhaps including uncertainties. Although Bayesian procedures have been developed to address such cases, current PRAs often employ alternative procedures (e.g., elicitation processes such as those described in NUREG/CR-6372 (Budnitz et al., 1997) to develop the probability distribution).

B.3 Non-Informative Prior Distributions for λ

In principle, the prior distribution, $\pi_0(\lambda)$ in our case, quantifies the analyst's belief regarding the possibility that the uncertain variable (λ) takes on values in different ranges. In practice, in situations for which large amounts of data are available, the precise shape of the prior distribution is unimportant; the data "overwhelms" prior beliefs in shaping the posterior distribution. In order to avoid the effort needed to develop (and potentially defend) a prior distribution in such situations, practical studies often make use of so-called "non-informative" prior distributions.

Non-informative prior distributions are mathematically defined functions constructed to represent a relative state of ignorance regarding the uncertain variable. Because there is no one unique way to define this state of ignorance, there is no one unique way to define a non-informative prior distribution. Three distributional forms found in PRA analyses and used in this report are (1) the non-informative distribution as defined by Winkler and Hays (1975), (2) the Jeffreys' Rule prior (Jeffreys, 1961), and (3) the constrained non-informative prior distribution (Atwood, 1996; Atwood et al., 2003). For our problem of estimating λ, these take the functional form:

(1) $$\pi_0(\lambda) \propto \frac{1}{\lambda} \tag{B.4}$$

(2) $$\pi_0(\lambda) \propto \frac{1}{\sqrt{\lambda}} \tag{B.5}$$

(3) $$\pi_0(\lambda) \propto \frac{1}{\sqrt{\lambda}} e^{-\frac{\lambda}{2E_0[\lambda]}} \tag{B.6}$$

where $E_0[\lambda]$ is the mean value of λ prior to observing any new data. (The constants of proportionality are unimportant for all three forms because, for each case, the constant appears in the numerator and denominator of Equation B.3.)

It is extremely important to recognize that these different prior distributions are intended for use when there are sufficient amounts of data to make the precise shape of the prior unimportant.[1] In more general cases, useful statistical estimates for an unknown quantity should be based on a prior distribution that represents the state of knowledge regarding the unknown quantity prior to observing the evidence (Apostolakis, 1978, 1990).

B.4 Posterior Distribution for λ

Using the likelihood function given by Equation B.2 and any of the prior distributions given by Equations B.4 through B.6, it can be shown that Equation B.3 has the following analytical solution:

$$\pi_1(\lambda|R,T) = \frac{\Gamma(\alpha'+\beta')}{\Gamma(\alpha')\Gamma(\beta')}\lambda^{\alpha'-1}e^{-\beta'\lambda} \tag{B.7}$$

where

$\Gamma()$ = gamma function [2]
$\alpha' = R + \alpha$
$\beta' = T + \beta$

and the parameters α and β are parameters characterizing the prior distribution, as shown in Table B-1.

Table B-1 Non-Informative Prior Distribution Parameter Values

Distribution Form	α	β
(1) Non-informative per Winkler and Hays (1975)	0	0
(2) Jeffreys prior	0.5	0
(3) Constrained non-informative	0.5	$0.5/E_0[\lambda]$

The mean value and variance of the posterior distribution are given by:

$$E_1[\lambda|R,T] = \frac{\alpha'}{\beta'} \tag{B.8}$$

$$Var_1[\lambda|R,T] = \frac{\alpha'}{\beta'^2} \tag{B.9}$$

The percentiles of the posterior distribution have to be determined by numerical integration. For example, λ_{95}, the 95th percentile, is determined by solving the implicit equation:

[1] Note that in such cases, the singular behavior of $\pi_0(\lambda)$ as λ approaches zero is not a practical concern.
[2] The gamma function is a complicated integral but can be solved using built-in functions provided in common spreadsheet and equation solving software packages.

$$0.95 = \int_0^{\lambda_{95}} \frac{\Gamma(\alpha'+\beta')}{\Gamma(\alpha')\Gamma(\beta')} \lambda^{\alpha'-1} e^{-\beta'\lambda}\, d\lambda \qquad \text{(B.10)}$$

B.5 Posterior Distributions for CDF and LERF

Using the results of Section B.4 above, posterior distributions for CDF and LERF can be determined in a generally straightforward manner.[3] The parameter values used in the calculations (which are based upon reactor-years through 2013) are shown in Table B-2; the results are shown in Figures B-1 through B-4. We used the equation solving package MathCAD 15.0 by Parametric Technology Corporation (PTC).

Table B-2 Parameter Values Used in Bayesian Estimates of CDF and LERF

		Data		Prior Distribution Parameters	
Figure	**Parameter**	**R**	**T***	**α**	**β**
B-1	CDF (Worldwide)	6**	15182	0	0
B-2	CDF (U.S.)	1	3839	0	0
B-3	LERF (Worldwide)	1	15182	0	0
B-4	LERF (U.S.)	0	3839	0 (Blue) 0.5 (Red) 0.5 (Green)	0 (Blue) 0 (Red) 5×10^4 yr (Green)

*Based on worldwide reactor-years through 2013; does not account for outages
**From Cochran (2012)

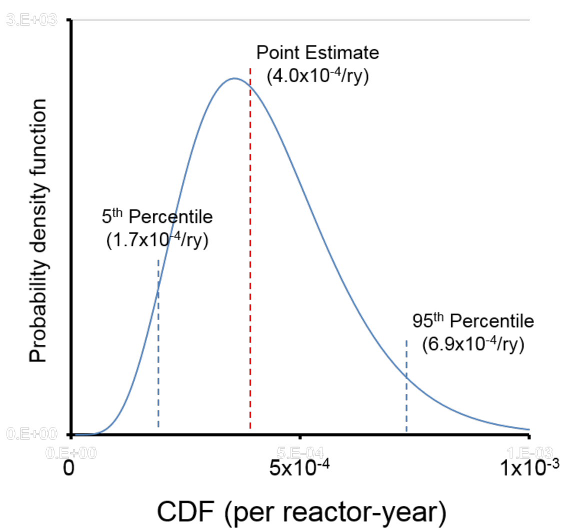

Figure B-1 Uncertainty band about Cochran (2012) estimate

Note that the prior distribution parameters underlying the blue, red, and green curves shown in Figure B-4 characterize the three different "non-informative" prior distributions discussed in Section B.3. Setting α and β to zero corresponds to Equation B.4, setting α to 0.5 and β to zero

[3] To obtain the result shown in Figure B-4, since there have been no large early releases in the United States, numerical problems arise when solving Equation B.7. In this case, we truncated the prior distribution for LERF, setting a lower value of 10^{-9}/RY and an upper value of 10^{-2}/RY.

corresponds to Equation B.5, and setting α to 0.5 and β to 5×10^4 corresponds to Equation B.6 in which the mean prior value is set to a value of 1×10^{-5}/reactor-year (RY) (a round number suggested by past PRAs).

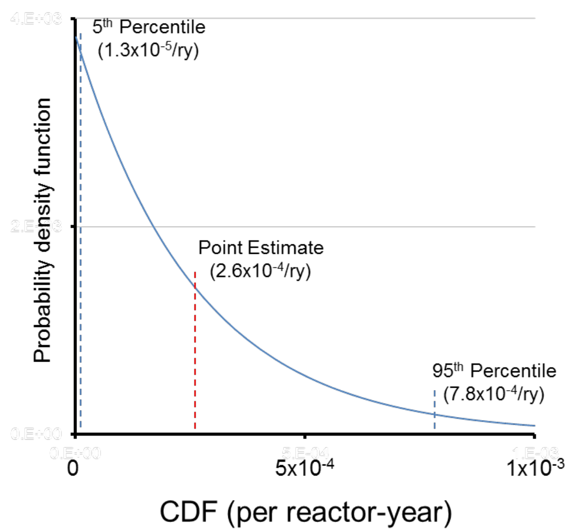

Figure B-2 Statistical estimate for CDF (U.S. plants only)

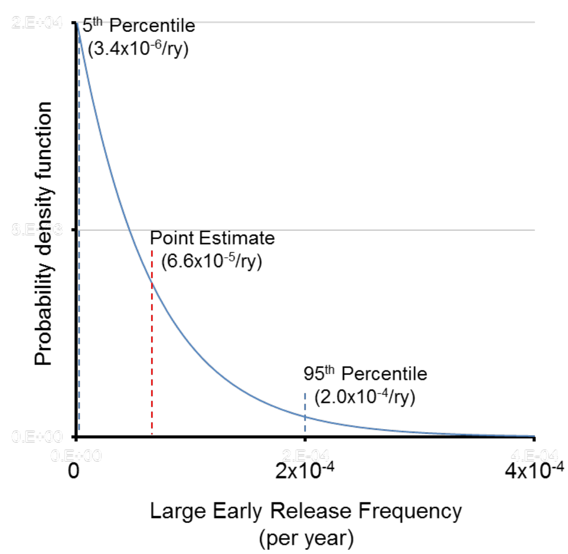

Figure B-3 Uncertainty band about point estimate for LERF (world reactors)

We caution that the results shown in Figures B-1 through B-4 are for illustration purposes only. Clearly, as demonstrated in Figure B-4, the available statistical evidence is weak and the assumption that the precise shape of the prior distribution is unimportant is incorrect for most of the data sets analyzed. (The results shown in Figure B-1 are fairly robust.) Should there be a need to develop a Bayesian estimate for CDF or LERF to support decisionmaking, this should involve the development of an informative prior (i.e., a prior distribution that reflects what is currently known about the parameter of interest).

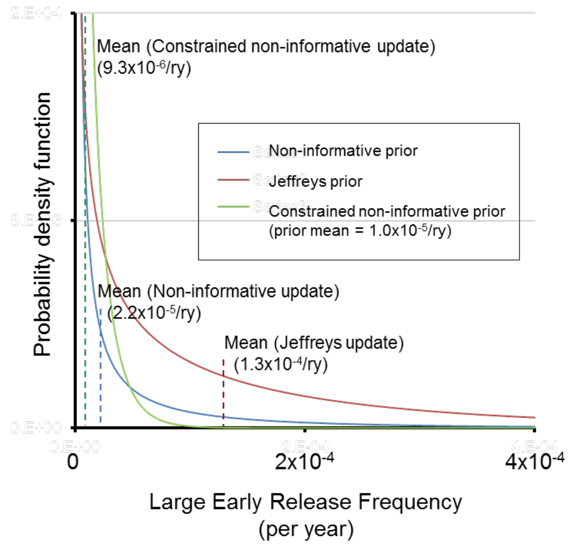

Figure B-4 Statistical estimate for LERF (U.S. plants only)

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2. TITLE AND SUBTITLE Probabilistic Risk Assessment and Regulatory Decisionmaking: Some Frequently Asked Questions	3. DATE REPORT PUBLISHED MONTH: September	YEAR: 2016
	4. FIN OR GRANT NUMBER	
5. AUTHOR(S) Nathan Siu Martin Stutzke Suzanne Dennis Donald Harrison	6. TYPE OF REPORT Technical	
	7. PERIOD COVERED (Inclusive Dates)	

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Division of Risk Assessment
Office of Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above" if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

Probabilistic risk assessment (PRA) is an important decision-support tool at the U.S. Nuclear Regulatory Commission. The availability of experiential data for accidents, including those at the Fukushima Dai ichi nuclear power plant, raises natural questions regarding the need for and utility of PRA, which is, at heart, a systems modeling-based analytical approach. This report addresses these questions using the format of frequently asked questions (FAQs). The FAQs are organized into four topic categories: regulatory decisionmaking, PRA basics, core damage frequency state of knowledge, and large early release frequency state of knowledge. For each FAQ, the report provides both a brief answer and a supplementary discussion elaborating on that answer. The report also includes two appendices, which provide additional technical details on quantifying uncertainty and updating data based on experience.

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	13. AVAILABILITY STATEMENT
PRA risk core damage regulatory decisionmaking uncertainty LERF CDF	unlimited
	14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified
	15. NUMBER OF PAGES
	16. PRICE

NUREG-2201 **Probabilistic Risk Assessment and Regulatory Decisionmaking: Some Frequently Asked Questions** **September 2016**