

ASME/ANS RA-Sa-2009

Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications

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The American Society of Mechanical Engineers
Three Park Avenue, New York, NY 10016-5990

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ASME/ANS RA-Sa-2009

Following approval by the ASME/ANS RA-S Committee and ASME, and after public review, ASME/ANS RA-Sa-2009 was approved by the American National Standards Institute on February 2, 2009.

Addenda to the 2008 edition of ASME/ANS RA-S are issued in the form of replacement pages. Revisions, additions, and deletions are incorporated directly into the affected pages. It is advisable, however, that this page, the Addenda title and copyright pages, and all replaced pages be retained for reference.

SUMMARY OF CHANGES

This is the first Addenda to be published to ASME/ANS RA-S-2008. This Standard has been revised in its entirety.

Replace or insert the pages listed. Changes given below are identified on the pages by a margin designator, **(a)**, placed next to the affected area.

<i>Page</i>	<i>Location</i>	<i>Change</i>
iii, iii.1	Contents	Updated to reflect Addenda
1	Part 1	Revised in its entirety
40	Part 2	Revised in its entirety
115	Part 3	Revised in its entirety
131	Part 4	Revised in its entirety
215	Part 5	Added
257	Part 6	Added
271	Part 7	Added
286	Part 8	Added
301	Part 9	Added
315	Part 10	Added

SPECIAL NOTE:

The Interpretations to ASME/ANS RA-S, volume 2, are included in this addenda beginning with page I-5 for the user's convenience.

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FOREWORD

The ASME Board on Nuclear Codes and Standards (BNCS) and American Nuclear Society (ANS) Standards Board mutually agreed in 2004 to form a Nuclear Risk Management Coordinating Committee (NRMCC). This committee was chartered to coordinate and harmonize Standards activities related to probabilistic risk assessment (PRA) between the two Standards development organizations (SDO). A key activity resulting from NRMCC was the development of PRA Standards structured around the Levels of PRA (i.e., Level 1, Level 2, Level 3) to be jointly issued by the two societies.

The scope of the initial issue of the ASME RA-S standard included Level 1 and Large Early Release Frequency (LERF) for internal events at power. In parallel with the development of ASME RA-S, ANS was developing companion PRA Standards covering external events, internal fire, and low power and shutdown conditions. These Standards are ANSI/ANS-58.21–2003, ANSI/ANS-58.23–2007, and ANS-58.22 (in development), respectively. ANS-58.22 will be added once it is approved as a revision or addendum. The three existing Standards are assembled together as a revision to ASME RA-S. Consequently, this revision to ASME RA-S is being issued with the revised identity of ASME/ANS RA-S–2008.

A major objective of the combined Standard is to ensure consistency in format, organization, language, and level of detail of the Standard. In assembling the component Standards the following criteria were used:

- (a) the requirements in the Standards would not be revised or modified
- (b) no new requirements would be included
- (c) the numbering scheme of the technical requirements would be preserved
- (d) the common requirements across the Standards would be consolidated into a single place
- (e) the commentary and nonmandatory requirements would be retained

Implementation of the consensus process for this Standard revealed that preserving the exact same requirements from the component Standards created certain technical issues that will need to be addressed in a revision or addendum of ASME/ANS RA-S–2008.

During the development of the ASME RA-S and the ANS companion, titled PRA Standards for Internal Fires, External Events, and Low Power and Shutdown Conditions, concerns were raised by stakeholder organizations and SDOs with respect to stability and consistency in requirements between the Standards. Thus, a key objective of this Standard is to improve consistency and foster stability by enabling future changes to be applied across the various PRA scopes that previously existed as separate Standards. It is anticipated that efficiencies and improvements will result from maintaining, interpreting, and implementing one PRA Standard as opposed to four separate Standards. Additionally, the identification of common processes in general requirements sections for such areas as PRA configuration control, peer review, maintenance versus upgrade, and use in risk-informed applications can now be provided, which further supports consistency and stability. Using a single committee responsible for this Standard provides a single point of response to inquiries and places the expertise necessary to address and coordinate activities in a single cognizant group supported by responsible technical societies. In addition, this Standard is intended to determine the technical adequacy of a PRA such that the PRA can be used in decision making.

The Committee on Nuclear Risk Management (CNRM) operates under procedures accredited by the American National Standards Institute (ANSI) as meeting the criteria of consensus procedures for American National Standards. The initial Standard was approved by the ASME Board on Nuclear Codes and Standards and subsequently approved by ANSI on April 9, 2008.

CNRM is responsible for ensuring that this Standard is maintained and revised as necessary following its original publication. This includes appropriate coordination with and linkage to other Standards under development for related risk-informed applications.

PART 1

GENERAL REQUIREMENTS FOR

A LEVEL 1 PRA, INCLUDING

LARGE EARLY RELEASE

FREQUENCY

(a)

Section 1-1 Introduction

1-1.1 OBJECTIVE

This Standard sets forth the requirements for probabilistic risk assessments (PRAs) used to support risk-informed decisions for commercial light water reactor nuclear power plants and prescribes a method for applying these requirements for specific applications.

1-1.2 SCOPE AND APPLICABILITY

This Standard establishes requirements for a Level 1 PRA of internal and external hazards for all plant operating modes (low power and shutdown modes will be included at a future date). In addition, this Standard establishes requirements for a limited Level 2 PRA sufficient to evaluate large early release frequency (LERF). The only hazards explicitly excluded from the scope are accidents resulting from purposeful human-induced security threats (e.g., sabotage). This Standard applies to PRAs used to support applications of risk-informed decision-making related to design, licensing, procurement, construction, operation, and maintenance. These requirements are written for operating power plants. They may be used for plants under design or construction, for advanced LWRs, or for other reactor designs, but revised or additional requirements may be needed.

This version of the PRA Standard provides specific requirements for the following hazard groups:

- (a) Internal Events (Part 2)
- (b) Internal Floods (Part 3)

- (c) Internal Fires (Part 4)
- (d) Seismic Events (Part 5)
- (e) High Winds (Part 7)
- (f) External Floods (Part 8)

Technical requirements for PRAs of other external hazards that may be relevant on a plant-specific basis or for specific applications are provided in Part 9. In addition to providing technical requirements for detailed PRAs of these hazards, this Standard provides requirements for screening and conservative analyses of external hazards (Part 6), and technical requirements for seismic margin analysis are provided in Part 10.

Many of the technical requirements in Part 2 are fundamental requirements for performing a PRA for any hazard group, and are therefore relevant to Parts 3 through 9 of this Standard. They are incorporated by reference in those requirements that address the development of the plant response to the damage states created by the hazard groups addressed in Parts 3 through 9. Their specific allocation to Part 2 is partially a historical artifact of the way this PRA Standard was developed, with the at-power internal events (including internal floods) requirements being developed first, and those of the remaining hazard groups being developed later. However, it is also a reflection of the fact that a fundamental understanding of the plant response to a reasonably complete set of initiating events (as defined in 1-2.2) provides the foundation for modeling the impact of various hazards on the plant. Hence, even though Part 2 is

given a title associated with the internal events hazard group it is understood that the requirements in this Part are applicable to all the hazard groups within the scope of the PRA.

1-1.3 STRUCTURE FOR PRA REQUIREMENTS

1-1.3.1 PRA Elements

The technical requirements for the PRA model are organized by their respective PRA technical elements. The PRA elements define the scope of the analysis for each Part of the Standard. This Standard specifies technical requirements for the PRA elements listed in Table 1-1.3-1.

1-1.3.2 High-Level Requirements

A set of objectives and HLRs is provided for each PRA Element in the Technical Requirements section of each respective Part of this Standard. The HLRs set forth the minimum requirements for a technically acceptable baseline PRA, independent of an application. The HLRs are defined in general terms and present the top level logic for the derivation of more detailed SRs. The HLRs reflect not only the diversity of approaches that have been used to develop the existing PRAs, but also the need to accommodate future technological innovations.

1-1.3.3 Supporting Requirements

A set of SRs is provided for each HLR (that is provided for each PRA Element) in the Technical Requirements section of each respective Part of this Standard.

This Standard is intended for a wide range of applications that require a corresponding range of PRA capabilities. Applications vary with respect to which risk metrics are employed, which decision criteria are used, the extent of reliance on the PRA results in supporting a decision, and the degree of resolution required for the factors that determine the risk significance of the subject of the decision. In developing the different portions of the PRA model, it is recognized that not every item, for example, system model, will be or need be developed to the same level of detail, same degree of plant-specificity, or the same degree of realism.

Although the range of capabilities required for each portion of the PRA to support an application falls on a continuum, three levels are defined and labeled either Capability Category I, II, or III, so that requirements can be developed and presented in a manageable way. Table 1-1.3-2 describes, for three principal attributes of PRA, the bases for defining the Capability Categories. This table was used to develop the SRs for each HLR.

The intent of the delineation of the Capability Categories within the SRs is generally that the degree of scope and level of detail, the degree of plant-specificity, and the degree of realism increases from Capability Category I to Capability Category III. However, the Capability

Categories are not based on the level of conservatism (i.e., tendency to overestimate risk due to simplifications in the PRA) in a particular aspect of the analysis. The level of conservatism may decrease as the Capability Category increases and more detail and more realism are introduced into the analysis. However, this is not true for all requirements and should not be assumed. Specific examples where a lower Capability Category may be less conservative are those requirements associated with the treatment of spurious operations in Fire PRA. As the Capability Category increases, the depth of the analysis required also increases. Hence, for a system train that is analyzed with less spurious operation considerations such as in Capability Category I, increasing the depth of the analysis in this case for Capability Categories II and III will identify additional spurious operations that will increase risk and thus the lower Capability Category will yield a lower (less conservative) estimated risk. Realism, however, does increase with increasing a Capability Category.

The boundaries between these Capability Categories can only be defined in a general sense. When a comparison is made between the capabilities of any given PRA and the SRs of this Standard, it is expected that the capabilities of a PRA's elements or portions of the PRA within each of the elements will not necessarily all fall within the same Capability Category, but rather will be distributed among all three Capability Categories. (There may be PRA elements, or portions of the PRA within the elements that fail to meet the SRs for any of these Capability Categories.) While all portions of the PRA need not have the same capability, the PRA model should be coherent. The SRs have been written so that, within a Capability Category, the interfaces between portions of the PRA are coherent (e.g., requirements for event trees are consistent with the definition of initiating event groups).

When a specific application is undertaken, judgment is needed to determine which Capability Category is needed for each portion of the PRA, and hence which SRs apply to the applications.

For each Capability Category, the SRs define the minimum requirements necessary to meet that Capability Category. Some SRs apply to only one Capability Category and some extend across two or three Capability Categories. When a SR spans multiple Capability Categories, it applies equally to each Capability Category. When necessary, the differentiation between Capability Categories is made in other associated SRs. The interpretation of a SR that spans multiple Capability Categories is stated in Table 1-1.3-3.

1-1.4 RISK ASSESSMENT APPLICATION PROCESS

The use of a PRA and the Capability Categories that are needed for each part of the PRA and for each of the PRA Elements will differ among applications. Section

Table 1-1.3-1 PRA Elements Addressed by Standard

Hazard Type	Hazard Group	PRA Elements
Internal Hazards	Internal Events	Initiating Events Analysis (IE) Accident Sequence Analysis (AS) Success Criteria (SC) Systems Analysis (SY) Human Reliability Analysis (HR) Data Analysis (DA) Quantification (QU) LERF Analysis (LE)
	Internal Flood	Internal Flood Plant Partitioning (IFPP) Internal Flood Source Identification and Characterization (IFSO) Internal Flood Scenarios (IFSN) Internal Flood-Induced Events (IFEV) Internal Flood Accident Sequences and Quantification (IFQU)
	Internal Fires	Plant Boundary Definition and Partitioning (PP) Fire PRA Equipment Selection (ES) Fire PRA Cable Selection (CS) Qualitative Screening (QLS) Fire PRA Plant Response Model (PRM) Fire Scenario Selection and Analysis (FSS) Fire Ignition Frequency (IGN) Quantitative Screening (QNS) Circuit Failure Analysis (CF) Postfire Human Reliability Analysis (HRA) Fire Risk Quantification (FQ) Seismic/Fire Interactions (SF) Uncertainty and Sensitivity Analyses (UNC)
External Hazards	Seismic Events	Probabilistic Seismic Hazard Analysis (SHA) Seismic Fragility Analysis (SFR) Seismic Plant Response Analysis (SPR)
	High Winds	High Wind Hazard Analysis (WHA) High Wind Fragility Analysis (WFR) High Wind Plant Response Analysis (WPR)
	External Floods	External Flood Hazard Analysis (XFHA) External Flood Fragility Analysis (XFFR) External Flood Plant Response Analysis (XFPR)
	Other External Hazards	External Hazard Analysis (XHA) External Hazard Fragility Analysis (XFR) External Hazard Plant Response Analysis (XPR)

Table 1-1.3-2 Bases for PRA Capability Categories

Attributes of PRA	I	II	III
1. Scope and Level of Detail: The degree to which the scope and level of detail of the plant design, operation, and maintenance are modeled.	Resolution and specificity sufficient to identify the relative importance of the contributors at the system or train level (and for Fire PRA, at a fire area level), including associated human actions [Notes (1) and (2)].	Resolution and specificity sufficient to identify the relative importance of the significant contributors at the component level (and for Fire PRA, at a physical analysis unit level including fire protection program and design elements) including associated human actions, as necessary [Notes (1), (2), (3), and (4)].	Resolution and specificity sufficient to identify the relative importance of the contributors at the component level (and for Fire PRA, for specific locations within fire areas or physical analysis units, including fire protection program and design elements) including associated human actions, as necessary [Notes (1), (2), (3), and (4)].
2. Plant-specificity: The degree to which plant-specific information is incorporated such that the as-built and as-operated plant is addressed.	Use of generic data/models acceptable except for the need to account for the unique design and operational features of the plant.	Use of plant-specific data/models for the significant contributors.	Use of plant-specific data/models for all contributors, where available.
3. Realism: The degree to which realism is incorporated such that the expected response of the plant is addressed.	Departures from realism will have moderate impact on the conclusions and risk insights as supported by good practices [Note (5)].	Departures from realism will have small impact on the conclusions and risk insights as supported by good practices [Note (5)].	Departures from realism will have negligible impact on the conclusions and risk insights as supported by good practices [Note (5)].

Table 1-1.3-2 Bases for PRA Capability Categories (Cont'd)**NOTES:**

- (1) The terms "fire area" and "physical analysis unit" are defined in 1-2.2. Fire areas are defined in the context of regulatory compliance documentation. Physical analysis units are subdivisions of a fire area used for the purposes of the Fire PRA.
- (2) The Fire PRA capability categories are distinguished, in part, based on the level of resolution provided in the analysis results. There is a gradation in resolution from fire areas for Capability Category I to specific locations within a fire area or physical analysis unit for Capability Category III. This distinction should not be confused with the task of plant partitioning (see 3-1.7.1). A Capability Category III Fire PRA could, for example, partition the plant at a fire area level and yet resolve fire risk contributions to the level of specific fire scenarios within each fire area. This approach would satisfy the intent of the Capability Category III basis in this regard. The term "fire protection program and design elements" as used here is intended to broadly encompass fire protection systems, features, and program provisions implemented in support of fire protection defense-in-depth. The term is intended to encompass active systems such as fire detection and suppression systems, passive features such as fire barriers, programmatic elements such as administrative controls, as well as other aspects of the fire protection program such as the manual fire brigade and post-fire safe shutdown.
- (3) The definition for Capability Categories II and III is not meant to imply that the scope and level of detail includes identification of every component and human action, but only those needed for the function of the system being modeled.
- (4) The term "fire protection program and design elements" as used here is intended to broadly encompass fire protection systems, features, and program provisions implemented in support of fire protection defense-in-depth. The term is intended to encompass active systems such as fire detection and suppression systems, passive features such as fire barriers, and programmatic elements such as administrative controls, as well as other aspects of the fire protection program such as the manual fire brigade and postfire safe shutdown.
- (5) Differentiation from moderate, to small, to negligible is determined by the extent to which the impact on the conclusions and risk insights could affect a decision under consideration. This differentiation recognizes that the PRA would generally not be the sole input to a decision. A moderate impact implies that the impact (of the departure from realism) is of sufficient size that it is likely that a decision could be affected; a small impact implies that it is unlikely that a decision could be affected, and a negligible impact implies that a decision would not be affected.

Table 1-1.3-3 Interpretation of Supporting Requirements

SR Spans	Peer Review Finding	Interpretation of the Supporting Requirement
All Three Capability Categories (I/II/III)	Meets SR	Capable of supporting applications in all Capability Categories
	Does not meet SR	Does not meet minimum standard
Single Capability Category (I, II, or III)	Meets individual SR	Capable of supporting applications requiring that Capability Category or lower
	Does not meet any SR	Does not meet minimum standard
Lower Two Capability Categories (I/II)	Meets SR for CC I/II	Capable of supporting applications requiring Capability Category I or II
	Meets SR for CC III	Capable of supporting applications in all Capability Categories
	Does not meet SR	Does not meet minimum standard
Upper Two Capability Categories (II/III)	Meets SR for CC II/III	Capable of supporting applications in all Capability Categories
	Meets SR for CC I	Capable of supporting applications requiring Capability Category I
	Does not meet SR	Does not meet minimum standard

1-3 describes the activities to determine whether a PRA has the capability to support a specific application of risk-informed decision making. Three different PRA Capability Categories were described in 1-1.3. PRA capabilities are evaluated for applicable parts of a PRA and each associated SR, rather than by specifying a Capability Category for the whole PRA. Therefore, only those parts of the PRA required to support the application in question need the Capability Category appropriate for that application. For a given application, supplementary analyses may be used in place of, or to augment, those aspects of a PRA that do not fully meet the requirements in the Technical Requirements section of each respective Part of this Standard. Requirements for supplementary analysis are outside the scope of this Standard.

1-1.5 PRA CONFIGURATION CONTROL

Section 1-5 provides requirements for configuration control of a PRA (i.e., maintaining and upgrading a plant-specific PRA) such that the PRA reflects the as-built, as-operated facility to a degree sufficient to support the application for which it is used.

1-1.6 PEER REVIEW REQUIREMENTS

Section 1-6 provides the general requirements for a peer review to determine if the PRA methodology and its implementation meet the requirements of the Technical Requirements section of each respective Part of this Standard. Scope-specific requirements are contained in the Peer Review Section of the respective Parts of this Standard.

1-1.7 ADDRESSING MULTIPLE HAZARD GROUPS

The technical requirements to determine the technical adequacy of a PRA for different hazard groups to support applications are presented in Parts 2 through 10. The approaches to modeling the plant damage resulting from different hazard groups vary in terms of the degree of realism and the level of detail achievable by the state of the art. For example, there are uncertainties that are unique to the modeling of the different hazards and their effect on the plant, and the assumptions made in dealing with these uncertainties can lead to varying degrees of conservatism in the estimates of risk. Furthermore, because the analyses can be resource intensive, it is normal to use screening approaches to limit the number of detailed scenarios to be evaluated and the number of mitigating systems credited while still achieving an acceptable evaluation of risk. These screening approaches are unique to each hazard group.

For many applications, it is necessary to consider the combined impact on risk from those hazard groups for which it cannot be demonstrated that the impact on the decision being made is insignificant. This can be done by using a single model that combines the PRA models for the different hazard groups, or by combining the results from separate models. In either case, when combining the results from the different hazard groups, it is essential to account for the differences in levels of conservatism and levels of detail so that the conclusions drawn from the results are not overly biased or distorted. To support this objective, the Standard is structured so that requirements for the analysis of the PRA results, including identification of significant contributors, identification and characterization of sources of uncertainty,

and identification of assumptions are included in each Part separately.

In some cases, the requirements for developing a PRA model in Parts 3 through 10 refer back to the requirements of Part 2. The requirements of Part 2 should be applied to the extent needed given the context of the modeling of each hazard group. In each Part, many of the requirements that differentiate between Capability Categories, either directly, or by incorporating the requirements of Part 2, do so on the basis of the treatment of significant contributors and significant accident sequences/cutsets for the hazard group being addressed. Because, as discussed above, there are differences in the way the PRA models for each specific hazard group are developed, the requirements are best treated as being self-contained for each hazard group separately when determining significant contributors and significant accident sequences/cutsets. In other words, these are identified with respect to the CDF and LERF for each hazard group separately. While there is a need in some applications to assess the significance with respect to the total CDF or LERF, this assessment has to be done with a full understanding of the differences in conservatism and level of detail introduced by the modeling approaches for the different hazard groups, as well as within each hazard group.

To determine the Capability Category at which the SRs have been met, it is necessary to have a definition of the term “significant.” Consequently, the term “significant” is used in various definitions in this Standard and is thereby explicitly incorporated into specific supporting requirements (SRs). Generally, the philosophy used in Capability Category II ensures a higher level of realism for significant contributors. This manifests itself in SRs related to the scope of plant-specific data, detailed

HRA (versus screening values), CCF treatment, documentation, and others.

The only consequence of not meeting the Standard definition of significant for a specific SR is that the PRA would not meet Capability Category II for that SR. Thus, in the context of an application, if a hazard group is a small contributor, it should be acceptable to meet Capability Category I by using screening HEPs, not using plant-specific data for equipment reliability, etc. The applicable portion of the PRA will simply be considered as meeting Capability Category I for that specific SR for that hazard group.

Additionally, from a practical standpoint, PRA models are generally developed on a hazard group basis (i.e., a fire PRA, a seismic PRA, a high wind PRA, etc.). While they may be integrated into a single model with multiple hazards, the development is done on a hazard group basis. In Capability Category II, this Standard strives to ensure that the more significant contributors to each hazard group are understood and treated with an equivalent level of resolution, plant specificity, and realism, so as to not skew the results for that hazard group. The definitions also acknowledge that there may be cases where the proposed quantitative definition is inappropriate (e.g., the hazard group risk is very low or bounding methods are used).

To summarize, the definitions that use the term “significant” simply help to define how much realism is necessary to meet Capability Category II of some SRs. They are NOT intended to be definitions of what is significant in a particular application. Indeed, in the context of a specific application, they may be either too loose or too restrictive, depending on what is being evaluated. In the context of this Standard, the decisions on applying these definitions and/or defining what is significant to a decision would be addressed in the Risk Assessment Application Process (see Section 1-3).

Section 1-2

Acronyms and Definitions

The following definitions are provided to ensure a uniform understanding of acronyms and terms as they are specifically used in this Standard.

1-2.1 ACRONYMS

AC: alternating current	GIP: generic implementation procedure
ACRS: Advisory Committee for Reactor Safeguards	HCLPF: high confidence of low probability of failure
ADS: automatic depressurization system	HEP: human error probability
ANS: American Nuclear Society	HFE: human failure event
AOPs: abnormal operating procedures	HLR: High Level Requirement
AOT: allowed outage time	HPCI: high pressure coolant injection
ARI: alternate rod insertion	HRA: human reliability analysis
ASEP: accident sequence evaluation program	HVAC: heating, ventilation, and air conditioning
ATWS: anticipated transient without scram	I&C: instrumentation and control
BWR: boiling water reactor	IE: initiating event
CCDP: conditional core damage probability	IPE: individual plant examination
CCF: common cause failure(s)	IPEEE: individual plant examination of external events
CCW: component cooling water	ISLOCA: interfacing systems loss of coolant accident
CDF: core damage frequency	LCO: limiting condition of operation
CDFM: conservative deterministic failure margin	LERF: large early release frequency
CEUS: central and eastern U.S.	LLNL: Lawrence Livermore National Laboratory
CLERP: conditional large early release probability	LOCA: loss of coolant accident
DBE: design-basis earthquake	LOOP: loss of offsite power (also referred to as "LOSP")
DC: direct current	LWR: light water reactor
DID: defense-in-depth	MCR: main control room
DOE: U.S. Department of Energy	MMI: modified Mercalli intensity
DW: drywell	MOV: motor operated valve
ECCS: emergency core cooling system	NEI: Nuclear Energy Institute
EDG: emergency diesel generator	NFPA: National Fire Protection Association
EOPs: emergency operating procedures	NPP: nuclear power plant
EPRI: Electric Power Research Institute	NPSH: net positive suction head
ES: equipment selection (fire PRA technical element)	NRC: Nuclear Regulatory Commission
FHA: fire hazards analysis (or assessment)	NSSS: nuclear steam supply system
FIVE: fire-induced vulnerability evaluation	OBE: operating-basis earthquake
FPRA: fire probabilistic risk assessment	P&IDs: piping and instrumentation drawings (or diagrams)
FRSS: fire risk scoping study (NUREG/CR-5088 [1-1])	PCS: power conversion system
FSAR: Final Safety Analysis Report	PDS: plant damage state
	PGA: peak ground acceleration
	PMF: probable maximum flood
	PORV: power (or pilot) operated relief valve
	PRA: probabilistic risk assessment

PSHA: probabilistic seismic hazard analysis
PWR: pressurized water reactor
QA: quality assurance
RAI: request for additional information
RCIC: reactor core isolation cooling
RCP: reactor coolant pump
RCS: reactor coolant system
RES: Office of Nuclear Regulatory Research (of the NRC)
RG: regulatory guide (an NRC issued communication)
RLE: review level earthquake
RPT: reactor pump trip
RPV: reactor pressure vessel
RRS: required response spectrum
RWST: refueling water storage tank
SAR: safety analysis report
SBO: station blackout
SDP: significance determination process
SEL: seismic equipment list
SFPE: Society of Fire Protection Engineers
SGTR: steam generator tube rupture
SLCS: standby liquid control system
SM: safety margin
SMA: seismic margin assessment
SME: seismic margin earthquake
SORV: stuck open relief valve
SQRT: seismic qualification review team
SR: Supporting Requirements
SRA: senior reactor analyst
SRP: standard review plan
SRT: seismic review team
SSA: safe shutdown analysis
SSCs: structures systems and components
SSE: safe shutdown earthquake
SSSEL: safe shutdown equipment list
SSHAC: Senior Seismic Hazard Analysis Committee
SSI: soil-structure interaction
SW: service water
THERP: Technique for Human Error Rate Prediction (see NUREG/CR-1278 [1-2])
TS: Technical Specifications
UHS: uniform hazard response spectrum

1-2.2 DEFINITIONS

accident class: a grouping of severe accidents with similar characteristics (such as accidents initiated by a transient with a loss of decay heat removal, loss of coolant accidents, station blackout accidents, and containment bypass accidents).

accident sequence: a representation in terms of an initiating event followed by a sequence of failures or successes of events (such as system, function, or operator performance) that can lead to undesired consequences, with a specified end state (e.g., core damage or large early release).

accident sequence analysis: the process to determine the combinations of initiating events, safety functions, and system failures and successes that may lead to core damage or large early release.

accident sequence, significant: see *significant accident sequence*.

adversely affect: in the context of fire PRA, to impact, via fire, plant equipment items and cables leading to equipment or circuit failure (including spurious operation of devices).

aleatory uncertainty: the uncertainty inherent in a non-deterministic (stochastic, random) phenomenon. Aleatory uncertainty is reflected by modeling the phenomenon in terms of a probabilistic model. In principle, aleatory uncertainty cannot be reduced by the accumulation of more data or additional information. (Aleatory uncertainty is sometimes called “randomness.”)

as-built, as-operated: a conceptual term that reflects the degree to which the PRA matches the current plant design, plant procedures, and plant performance data, relative to a specific point in time.

NOTE: At the design certification stage, the plant is neither built nor operated. For these situations, the intent of the PRA model is to reflect the “as-designed, as-to-be-built, and as-to-be-operated” plant.

assumption: a decision or judgment that is made in the development of the PRA model. An assumption is either related to a source of model uncertainty or is related to scope or level of detail. An assumption related to a model uncertainty is made with the knowledge that a different reasonable alternative assumption exists. A reasonable alternative assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being made. An assumption related to scope or level of detail is one that is made for modeling convenience. An assumption is labeled “key” when it may influence (i.e., have the potential to change) the decision being made. Therefore, a key assumption is identified in the context of an application.

at power: those plant operating states characterized by the reactor being critical and producing power, with

automatic actuation of critical safety systems not blocked and with essential support systems aligned in their normal power operation configuration.

availability: the complement of unavailability.

basic event: an event in a fault tree model that requires no further development, because the appropriate limit of resolution has been reached.

bounding analysis: analysis that uses assumptions such that the assessed outcome will meet or exceed the maximum severity of all credible outcomes.

cable: referring solely to “electric cables,” a construction comprising one or more insulated electrical conductors (generally copper or aluminum). A cable may or may not have other physical features such as an outer protective jacket, a protective armor (e.g., spiral wound or braided), shield wraps, and/or an uninsulated ground conductor or drain wire. Cables are used to connect points in a common electrical circuit and may be used to transmit power, control signals, indications, or instrument signals.

cable failure mode: the behavior of an electrical cable upon fire-induced failure that may include intracable shorting, intercable shorting, and/or shorts between a conductor and an external ground. (See also *hot short*.)

circuit failure mode: the manner in which a conductor fault is manifested in the circuit. Circuit failure modes include loss of motive power, loss of control, loss of or false indication, open circuit conditions (e.g., a blown fuse or open circuit protective device), and spurious operation.

common cause failure (CCF): a failure of two or more components during a short period of time as a result of a single shared cause.

community distribution: for any specific expert judgment, the distribution of expert judgments of the entire relevant (informed) technical community of experts knowledgeable about the given issue.

component: an item in a nuclear power plant, such as a vessel, pump, valve, or circuit breaker.

composite variability: the composite variability includes the aleatory (randomness) uncertainty (β_R) and the epistemic (modeling and data) uncertainty (β_U). The logarithmic standard deviation of composite variability, β_C , is expressed as $(\beta_R^2 + \beta_U^2)^{1/2}$.

concurrent hot short: the occurrence of two or more hot shorts such that the shorts overlap in time (e.g., a second hot short occurs before a prior hot short has self-mitigated or has been mitigated by an operator action).

containment bypass: a direct or indirect flow path that may allow the release of radioactive material directly to the environment bypassing the containment.

containment challenge: severe accident conditions (e.g., plant thermal hydraulic conditions or phenomena) that

may result in compromising containment integrity. These conditions or phenomena can be compared with containment capability to determine whether a containment failure mode results.

containment failure: loss of integrity of the containment pressure boundary from a core damage accident that results in unacceptable leakage of radio nuclides to the environment.

containment failure mode: the manner in which a containment radionuclide release pathway is created. It encompasses both those structural failures of containment induced by containment challenges when they exceed containment capability and the failure modes of containment induced by human failure events, isolation failures, or bypass events such as ISLOCA.

containment performance: a measure of the response of a nuclear plant containment to severe accident conditions.

core damage: uncover 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.

core damage frequency (CDF): expected number of core damage events per unit of time.

damage criteria: those characteristics of the fire-induced environmental effects that will be taken as indicative of the fire-induced failure of a damage target or set of damage targets.

damage target: see *target*.

damage threshold: the values corresponding to the damage criteria that will be taken as indicative of the onset of fire-induced failure of a damage target or set of damage targets.

deaggregation: determination of the functional contribution of each magnitude-distance pair to the total seismic hazard. To accomplish this, a set of magnitude and distance bins are selected, and the annual frequency of exceeding selected ground motion parameters from each magnitude-distance pair is computed and divided by the total probability.

demonstrably conservative analysis: analysis that uses assumptions such that the assessed outcome will be conservative relative to the expected outcome.

dependency: requirement external to an item and upon which its function depends and is associated with dependent events that are determined by, influenced by, or correlated to other events or occurrences.

design-basis earthquake (DBE): a commonly employed term for the safe shutdown earthquake (SSE), defined separately below.

diagnosis: examination and evaluation of data to determine either the condition of an SSC or the cause of the condition.

distribution system: piping, raceway, duct, or tubing that carries or conducts fluids, electricity, or signals from one point to another.

electrical overcurrent protective device: an active or passive device designed to prevent current flow from exceeding a predetermined level by breaking the circuit when the predetermined level is exceeded (e.g., fuse or circuit breaker).

end state: the set of conditions at the end of an accident sequence that characterizes the impact of the sequence on the plant or the environment. In most PRAs, end states typically include success states (i.e., those states with negligible impact), plant damage states for Level 1 sequences, and release categories for LERF sequences.

epistemic uncertainty: the uncertainty attributable to incomplete knowledge about a phenomenon that affects our ability to model it. Epistemic uncertainty is reflected in ranges of values for parameters, a range of viable models, the level of model detail, multiple expert interpretations, and statistical confidence. In principle, epistemic uncertainty can be reduced by the accumulation of additional information. (Epistemic uncertainty is sometimes also called “modeling uncertainty.”)

equipment: a term used to broadly cover the various components in a nuclear power plant. Equipment includes electrical and mechanical components (e.g., pumps, control and power switches, integrated circuit components, valves, motors, fans, etc.), and instrumentation and indication components (e.g., status indicator lights, meters, strip chart recorders, sensors, etc.). Equipment, as used in this Standard, *excludes* electrical cables.

equipment qualification: the generation and maintenance of data and documentation to demonstrate that equipment is capable of operating under the conditions of a qualification test, or test and analysis.

evaluator expert: an expert who is capable of evaluating the relative credibility of multiple alternative hypotheses, and who is expected to evaluate all potential hypotheses and bases of inputs from proponents and resource experts, to provide both evaluator input and other experts’ representation of the community distribution.

event tree: a logic diagram that begins with an initiating event or condition and progresses through a series of branches that represent expected system or operator performance that either succeeds or fails and arrives at either a successful or failed end state.

event tree top event: the conditions (i.e., system behavior or operability, human actions, or phenomenological events) that are considered at each branch point in an event tree.

expert elicitation: a formal, highly structured, and documented process whereby expert judgments, usually of multiple experts, are obtained.

expert judgment: information provided by a technical expert, in the expert’s area of expertise, based on opinion, or on an interpretation based on reasoning that includes evaluations of theories, models, or experiments.

exposed structural steel: structural steel elements that are not protected by a passive fire barrier feature (e.g., fire-retardant coating) with a minimum fire-resistance rating of 1 hr.

external event: an event originating outside a nuclear power plant that directly or indirectly causes an initiating event and may cause safety system failures or operator errors that may lead to core damage or large early release. Events such as earthquakes, tornadoes, and floods from sources outside the plant and fires from sources inside or outside the plant are considered external events. (See also *internal event*.) By historical convention, LOOP not caused by another external event is considered to be an internal event.

extremely rare event: one that would not be expected to occur even once throughout the world nuclear industry over many years (e.g., $<1\text{E-6/r-yr}$).

facilitator/integrator: a single entity (individual, team, company, etc.) who is responsible for aggregating the judgments and community distributions of a panel of experts to develop the composite distribution of the informed technical community (herein called “the community distribution”).

failure mechanism: any of the processes that results in failure modes, including chemical, electrical, mechanical, physical, thermal, and human error.

failure mode: a specific functional manifestation of a failure (i.e., the means by which an observer can determine that a failure has occurred) by precluding the successful operation of a piece of equipment, a component, or a system (e.g., fails to start, fails to run, leaks).

NOTE: In the context of Fire PRA, *spurious operation* is also considered a failure mode above and beyond failures that preclude successful operation.

failure modes and effects analysis (FMEA): a process for identifying failure modes of specific components and evaluating their effects on other components, subsystems, and systems.

failure probability: the likelihood that an SSC will fail to operate upon demand or fail to operate for a specific mission time.

failure rate: expected number of failures per unit time, evaluated, for example, by the ratio of the number of failures in a population of components to the total time observed for that population.

fault tree: a deductive logic diagram that depicts how a particular undesired event can occur as a logical combination of other undesired events.

figure of merit: the quantitative value, obtained from a PRA analysis, used to evaluate the results of an application (e.g., CDF or LERF).

fire analysis tool: as used in this Standard, “fire analysis tool” is broadly defined as any method used to estimate or calculate one or more physical fire effects (e.g., temperature, heat flux, time to failure of a damage target, rate of flame spread over a fuel package, heat release rate for a burning material, smoke density, etc.) based on a predefined set of input parameter values as defined by the fire scenario being analyzed. Fire analysis tools include, but are not limited to, computerized compartment fire models, closed-form analytical formulations, empirical correlations such as those provided in a handbook, and lookup tables that relate input parameters to a predicted output.

fire area: a portion of a building or plant that is separated from other areas by rated fire barriers adequate for the fire hazard (RG 1.189 [1-3]). (Note that a rated fire barrier is a fire-barrier with a fire-resistance rating.)

fire barrier: a continuous vertical or horizontal construction assembly designed and constructed to limit the spread of heat and fire and to restrict the movement of smoke (NFPA 805 [1-4]).

fire compartment:¹ a subdivision of a building or plant that is a well-defined enclosed room, not necessarily bounded by rated fire barriers. A fire compartment generally falls within a fire area and is bounded by noncombustible barriers where heat and products of combustion from a fire within the enclosure will be substantially confined. Boundaries of a fire compartment may have open equipment hatches, stairways, doorways, or unsealed penetrations. This is a term defined specifically for fire risk analysis and maps plant fire areas and/or zones, defined by the plant and based on fire protection systems design and/or operations considerations, into compartments defined by fire damage potential. For example, the control room or certain areas within the turbine building may be defined as a fire compartment (This definition is derived from NUREG/CR-6850-EPRI TR-1011989 [1-5]). In this Standard, “physical analysis unit” is used to represent all subdivisions of a plant for Fire PRA. Physical analysis units include fire compartments.

fire-induced initiating event: that initiating event assigned to occur in the FPRA plant response model for a given fire scenario (adapted from NUREG/CR-6850-EPRI TR-1011989 [1-5]).

¹ It is noted that the term “fire compartment” is used in other contexts, such as general fire protection engineering, and that the term’s meaning as used here may differ from that implied in an alternate context. However, the term also has a long history of use in Fire PRA and is used in this Standard based on that history of common Fire PRA practice.

fire-induced initiating event: that initiating event assigned to occur in the FPRA plant response model for a given fire scenario (adapted from NUREG/CR-6850-EPRI TR-1011989 [1-5]).

fire modeling: as used in this Standard, “fire modeling” refers to the process of exercising a fire analysis tool including the specification and verification of input parameter values, performance of any required supporting calculations, actual application of the fire analysis tool itself, and the interpretation of the fire analysis tool outputs and results.

fire PRA plant response model: a representation of a combination of equipment, cable, circuit, system, function, and operator failures or successes, of an accident that when combined with a fire-induced initiating event can lead to undesired consequences, with a specified end state (e.g., core damage or large early release).²

fire protection defense-in-depth: the principle of providing multiple and diverse fire protection systems and features that will, collectively, prevent fires from starting; detect rapidly, control, and extinguish promptly those fires that do occur; and provide protection for SSCs important to safety so that a fire that is not promptly extinguished by the fire suppression activities will not prevent the safe shutdown of the plant (derived from 10CFR50, Appendix R, Sec. II.A [1-6]).

fire protection design elements: any aspect of the fire protection program that is supported by specific design requirements and/or analyses.

fire protection feature: administrative controls, fire barriers, means of egress, industrial fire brigade personnel, and other features provided for fire protection purposes (NFPA 805 [1-4]).

fire protection program: the integrated effort involving equipment, procedures, and personnel used in carrying out all activities of fire protection. It includes system and facility design, fire prevention, fire detection, annunciation, confinement, suppression, administrative controls, fire brigade organization, inspection and maintenance, training, quality assurance, and testing (RG 1.189 [1-3]).

fire protection program element: any specific aspect or provision included as a part of the fire protection program.

fire protection system: fire detection, notification, and fire suppression systems designed, installed, and maintained in accordance with the applicable NFPA codes and Standards (NFPA 805 [1-4]).

fire-resistance rating: the time, in minutes or hours, that materials or assemblies have withstood a fire exposure

² This definition has been adapted to suit fire PRA needs from the definition of “accident sequence.” A variety of equivalent terms has been used in other fire PRA related documents including, but not limited to, postfire safe shutdown model, fire PRA model, and postfire plant response model.

as established in accordance with an approved test procedure appropriate for the structure, building material, or component under consideration (NFPA 805 [1-4]).

fire scenario: a set of elements that describes a fire event. The elements usually include a physical analysis unit, a source fire location and characteristics, detection and suppression features to be considered, damage targets, and intervening combustibles.

fire scenario selection: the process of defining a fire scenario to be analyzed in the Fire PRA that will represent the behavior and consequences of fires involving one or more fire ignition sources. Fire scenario selection includes the identification of a fire ignition source (or set of fire ignition sources); secondary combustibles and fire spread paths; fire damage targets, detection and suppression systems and features to be credited; and other factors that will influence the extent and timing of fire damage.

fire suppression system: generally refers to permanently installed fire protection systems provided for the express purpose of suppressing fires. Fire suppression systems may be either automatically or manually actuated. However, once activated, the system should perform its design function with little or no manual intervention.

fire wrap: a localized protective covering designed to protect cables, cable raceways, or other equipment from fire-induced damage. Fire wraps generally provide protection against thermal damage.

fractile hazard curves: a set of hazard curves used to reflect the uncertainties associated with estimated hazard. A common family of hazard curves used in describing the results of a probabilistic seismic hazard analysis (PSHA) consists of curves of fractiles of the probability distributions of estimated seismic hazard as a function of the level of ground motion parameter.

fragility: fragility of an SSC is the conditional probability of its failure at a given hazard input level. The input could be earthquake motion, wind speed, or flood level. The fragility model used in seismic PRA is known as a double lognormal model with three parameters, which are the median acceleration capacity, the logarithmic standard deviation of the aleatory (randomness) uncertainty in capacity, and the logarithmic standard deviation of the epistemic (modeling and data) uncertainty in the median capacity.

front-line system: a system (safety or nonsafety) that is capable of directly performing one of the accident mitigating functions (e.g., core or containment cooling, coolant makeup, reactivity control, or reactor vessel pressure control) modeled in the PRA.

Fussell-Vesely (FV) importance measure: for a specified basic event, Fussell-Vesely importance is the fractional contribution to the total of a selected figure of merit for all accident sequences containing that basic event. For

PRA quantification methods that include nonminimal cutsets and success probabilities, the Fussell-Vesely importance measure is calculated by determining the fractional reduction in the total figure of merit brought about by setting the probability of the basic event to zero.

ground acceleration: acceleration at the ground surface produced by seismic waves, typically expressed in units of g , the acceleration of gravity at the Earth's surface.

harsh environment: an abnormal environment (e.g., high or low temperature, humidity, corrosive conditions) expected as a result of postulated accident conditions appropriate for the design basis or beyond design basis accidents.

hazard: an event or a natural phenomenon that poses some risk to a facility. Internal hazards include events such as equipment failures, human failures, and flooding and fires internal to the plant. External hazards include events such as flooding and fires external to the plant, tornadoes, earthquakes, and aircraft crashes.

hazard group: a group of similar hazards that are assessed in a PRA using a common approach, methods, and likelihood data for characterizing the effect on the plant. Typical hazard groups considered in a nuclear power plant PRA include internal events, internal floods, seismic events, internal fires, high winds, external flooding, etc.

HCLPF capacity: refers to the High Confidence of Low Probability of Failure capacity, which is a measure of seismic margin. In seismic PRA, this is defined as the earthquake motion level at which there is a high (95%) confidence of a low (at most 5%) probability of failure. Using the lognormal fragility model, the HCLPF capacity is expressed as $A_m \exp[-1.65\beta_R + \beta_{UI}]$. When the logarithmic standard deviation of composite variability β_c is used, the HCLPF capacity could be approximated as the ground motion level at which the composite probability of failure is at most 1%. In this case, HCLPF capacity is expressed as $A_m \exp[-2.33\beta_c]$. In deterministic SMAs, the HCLPF capacity is calculated using the CDFM method.

high energy arcing fault: electrical arc that leads to a rapid release of electrical energy in the form of heat, vaporized copper, and mechanical force.

high-hazard fire source: a fire source that can lead to fires of a particularly severe and challenging nature. High-hazard fire sources would include, but are not limited to, the following: catastrophic failure of an oil-filled transformer, an unconfined release of flammable or combustible liquid, leaks from a pressurized system containing flammable or combustible liquids, and significant releases or leakage of hydrogen or other flammable gases.

high winds: tornadoes, hurricanes (or cyclones or typhoons as they are known outside the U.S.), extratropical (thunderstorm) winds, and other wind phenomena depending on the site location.

hot short: individual conductors of the same or different cables coming in contact with each other where at least one of the conductors involved in the shorting is energized resulting in an impressed voltage or current on the circuit being analyzed.

human error (HE): any human action that exceeds some limit of acceptability, including inaction where required, excluding malevolent behavior.

human error probability (HEP): a measure of the likelihood that plant personnel will fail to initiate the correct, required, or specified action or response in a given situation, or by commission performs the wrong action. The HEP is the probability of the human failure event.

human failure event (HFE): a basic event that represents a failure or unavailability of a component, system, or function that is caused by human inaction, or an inappropriate action.

human reliability analysis (HRA): a structured approach used to identify potential human failure events and to systematically estimate the probability of those events using data, models, or expert judgment.

ignition frequency: frequency of fire occurrence generally expressed as fire ignitions per reactor-year.

ignition source: piece of equipment or activity that causes fire (RG 1.189 [1-3]).

initiating event: an event either internal or external to that which perturbs the steady state operation of the plant by challenge plant control and safety systems whose failure could potentially lead to core damage or release of airborne fission products. These events include human-caused perturbations and failure of equipment from either internal plant causes (such as hardware faults, floods, or fires) or external plant causes (such as earthquakes or high winds).

initiator: see *initiating event*.

integrator: a single entity (individual, team, company, etc.) who is ultimately responsible for developing the composite representation of the informed technical community (herein called “the community distribution”). This sometimes involves informal methods such as deriving information relevant to an issue from the open literature or through informal discussions with experts, and sometimes involves more formal methods.

intensity: a measure of the impact of a hazard.

intercable (as in “intercable conductor-to-conductor short circuit”): electrical interactions (shorting) between the conductors of two (or more) separate electrical cables. (See also *intracable*.)

interfacing systems LOCA (ISLOCA): a LOCA when a breach occurs in a system that interfaces with the RCS, where isolation between the breached system and the RCS fails. An ISLOCA is usually characterized by the overpressurization of a low-pressure system when subjected to RCS pressure and can result in containment bypass.

internal event: an event originating within a nuclear power plant that directly or indirectly causes an initiating event and may cause safety system failures or operator errors that may lead to core damage or large early release. By historical convention, loss of offsite power is considered to be an internal event, and internal fire is considered to be an external event except when the loss is caused by an external hazard that is treated separately (e.g., seismic-induced LOOP). Internal floods have sometimes been included with internal events and sometimes considered as external events. For this Standard, internal floods are considered to be separate from internal events. (See also *external event*.)

intracable (as in “intracable conductor-to-conductor short circuit”): electrical interactions (shorting) between the conductors of one multiconductor electrical cable. (See also *intercable*.)

key safety functions: the minimum set of safety functions that must be maintained to prevent core damage and large early release. These include reactivity control, reactor pressure control, reactor coolant inventory control, decay heat removal, and containment integrity in appropriate combinations to prevent core damage and large early release.

large early release: the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of off-site emergency response and protective actions such that there is a potential for early health effects.

large early release frequency (LERF): expected number of large early releases per unit of time.

LERF analysis: evaluation of containment response to severe accident challenges and quantification of the mechanisms, amounts, and probabilities of subsequent radioactive material releases from the containment.

level 1 analysis: identification and quantification of the sequences of events leading to the onset of core damage.

licensee-controlled area: areas of the plant site that are directly controlled by the nuclear power plant licensee.

master logic diagram: summary fault tree constructed to guide the identification and grouping of initiating events and their associated sequences to ensure completeness.

may: used to state an option to be implemented at the user’s discretion.

mission time: the time period that a system or component is required to operate in order to successfully perform its function.

multicompartment fire scenario: a fire scenario involving targets in a room or fire compartment other than, or in addition to, the one where the fire was originated.

multiple spurious operations: concurrent spurious operations of two or more equipment items.

mutually exclusive events: a set of events where the occurrence of any one precludes the simultaneous occurrence of any remaining events in the set.

operating-basis earthquake (OBE): that earthquake for which those features of the nuclear power plant necessary for continued operation without undue risk to health and safety are designed to remain functional. In the past, the OBE was commonly chosen to be one-half of the safe shutdown earthquake (SSE).

operating time: total time during which components or systems are performing their designed function.

peak ground acceleration (PGA): maximum value of acceleration displayed on an accelerogram; the largest ground acceleration produced by an earthquake at a site.

performance shaping factor (PSF): a factor that influences human error probabilities as considered in a PRA's human reliability analysis and includes such items as level of training, quality/availability of procedural guidance, time available to perform an action, etc.

physical analysis units: the spatial subdivisions of the plant upon which the Fire PRA is based. The physical analysis units are generally defined in terms of fire areas and/or fire compartments under the plant partitioning technical element.

plant: a general term used to refer to a nuclear power facility (for example, "plant" could be used to refer to a single unit or multiunit site).

plant damage state (PDS): group of accident sequence end states that have similar characteristics with respect to accident progression, and containment or engineered safety feature operability.

plant-specific data: data consisting of observed sample data from the plant being analyzed.

point estimate: estimate of a parameter in the form of a single number.

post-initiator human failure events: human failure events that represent the impact of human errors committed during response to abnormal plant conditions.

PRA application: a documented analysis based in part or whole on a plant-specific PRA that is used to assist in decision making with regard to the design, licensing, procurement, construction, operation, or maintenance of a nuclear power plant.

PRA maintenance: the update of the PRA models to reflect plant changes such as modifications, procedure changes, or plant performance (data).

PRA upgrade: the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. This could include items such as new human error analysis methodology, new data update methods, new approaches to quantification or truncation, or new treatment of common cause failure.

pre-initiator human failure events: human failure events that represent the impact of human errors committed during actions performed prior to the initiation of an accident (e.g., during maintenance or the use of calibration procedures).

prior distribution (priors): in Bayesian analysis, the expression of an analyst's prior belief about the value of a parameter prior to obtaining sample data.

probabilistic risk assessment (PRA): a qualitative and quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public [also referred to as a probabilistic safety assessment (PSA)].

probability of exceedance (as used in seismic hazard analysis): the probability that a specified level of ground motion for at least one earthquake will be exceeded at a site or in a region during a specified exposure time.

probability of nonsuppression: probability of failing to suppress a fire before target damage occurs.

proponent expert: an expert who advocates a particular hypothesis or technical position.

raceway: an enclosed channel of metal or nonmetallic materials designed expressly for holding wires, cables, or bus bars, with additional functions as permitted by code. Raceways include, but are not limited to, rigid metal conduit, rigid nonmetallic conduit, intermediate metal conduit, liquid-tight flexible conduit, flexible metallic tubing, flexible metal conduit, electrical nonmetallic tubing, electrical metallic tubing, underfloor raceways, cellular concrete floor raceways, cellular metal floor raceways, surface raceways, wireways, and busways (RG 1.189 [1-3]).

randomness (as used in seismic-fragility analysis): the variability in seismic capacity arising from the randomness of the earthquake characteristics for the same acceleration and to the structural response parameters that relate to these characteristics.

rare event: one that might be expected to occur only a few times throughout the world nuclear industry over many years (e.g., $<1E-4/r\text{-yr}$).

reactor-operating-state-year: an equivalent calendar year of operation in a particular plant operating state. See Note (3) in Table 2-2.1-2(c).

reactor-year: a calendar year in the operating life of one reactor, regardless of power level. See Note (3) in Table 2-2.1-2(c).

recovery: restoration of a function lost as a result of a failed SSC by overcoming or compensating for its failure. Generally modeled by using HRA techniques.

reliability: the complement of unreliability.

repair: restoration of a failed SSC by correcting the cause of failure and returning the failed SSC to its modeled functionality. Generally modeled by using actuarial data.

repair time: the period from identification of a component failure until it is returned to service.

required time: the time needed by operators to successfully perform and complete a human action.

resource expert: a technical expert with knowledge of a particular technical area of a PRA.

response: a reaction to a cue for action in initiating or recovering a desired function.

response models: represent post-initiator control-room operator actions, following a cue or symptom of an event, to satisfy the procedural requirements for control of a function or system.

response spectrum: a curve calculated from an earthquake accelerogram that gives the value of peak response in terms of acceleration, velocity, or displacement of a damped linear oscillator (with a given damping ratio) as a function of its period (or frequency).

review level earthquake (RLE): an earthquake larger than the plant SSE and is chosen in seismic margin assessment (SMA) for initial screening purposes. Typically, the RLE is defined in terms of a ground motion spectrum.

NOTE: A majority of plants in the eastern and midwestern U.S. have conducted SMA reviews for an RLE of 0.3g PGA anchored to a median NUREG/CR-0098 spectrum [1-7].

risk: probability and consequences of an event, as expressed by the “risk triplet” that is the answer to the following three questions:

- (a) What can go wrong?
- (b) How likely is it?
- (c) What are the consequences if it occurs?

risk achievement worth (RAW) importance measure: for a specified basic event, risk achievement worth importance reflects the increase in a selected figure of merit when an SSC is assumed to be unable to perform its function due to testing, maintenance, or failure. It is the ratio or interval of the figure of merit, evaluated with the SSC’s basic event probability set to one, to the base case figure of merit.

risk-relevant consequences: the fire-induced failure of any risk-relevant target, or the fire-induced creation of environmental conditions that may complicate or preclude credited postfire operator actions.

risk-relevant damage targets: any equipment item or cable whose operation is credited in the Fire PRA plant response model or whose operation may be required to support a credited postfire operator action.

risk-relevant ignition source: any ignition source considered in the Fire PRA fire scenario definitions that could cause a fire that might induce a plant initiating event or adversely affect one or more damage targets.

risk significant equipment: equipment associated with a significant basic event. (See also *significant basic event*.)

safe shutdown earthquake (SSE): that earthquake for which certain SSCs are designed to remain functional. In the past, the SSE has been commonly characterized by a standardized spectral shape anchored to a PGA value.

safe shutdown equipment list (SSEL): the list of all SSCs that require evaluation in the seismic-margins-calculation task of an SMA. Note that this list can be different from the seismic equipment list (SEL) used in a seismic PRA.

safe stable state: a plant condition, following an initiating event, in which RCS conditions are controllable at or near desired values.

safety function: function that must be performed to control the sources of energy in the plant and radiation hazards.

safety systems: those systems that are designed to prevent or mitigate a design-basis accident.

screening: a process that eliminates items from further consideration based on their negligible contribution to the probability of an accident or its consequences.

screening criteria: the values and conditions used to determine whether an item is a negligible contributor to the probability of an accident sequence or its consequences.

secondary combustible: combustible or flammable materials that are not a part of the fire ignition source that may be ignited if there is fire spread beyond the fire ignition source.

seismic equipment list (SEL): the list of all SSCs that require evaluation in the seismic-fragilities task of a seismic PRA. Note that this list can be different from the SSEL used in an SMA.

seismic margin: seismic margin is expressed in terms of the earthquake motion level that compromises plant safety, specifically leading to severe core damage. The margin concept can also be extended to any particular structure, function, system, equipment item, or component for which “compromising safety” means sufficient loss of safety function to contribute to core damage

either independently or in combination with other failures.

seismic margin assessment (SMA): the process or activity to estimate the seismic margin of the plant and to identify any seismic vulnerabilities in the plant. This is described further in Part 10 and Nonmandatory Appendix 10-A.

seismic source: a general term referring to both seismogenic sources and capable tectonic sources. A seismogenic source is a portion of the Earth assumed to have a uniform earthquake potential (same expected maximum earthquake and recurrence frequency), distinct from the seismicity of the surrounding regions. A capable tectonic source is a tectonic structure that can generate both vibratory ground motion and tectonic surface deformation such as faulting or folding at or near the Earth's surface. In a probabilistic seismic hazard analysis (PSHA), all seismic sources in the site region with a potential to contribute to the frequency of ground motions (i.e., the hazard) are considered.

seismic spatial interaction: an interaction that could cause an equipment item to fail to perform its intended safety function. It is the physical interaction of a structure, pipe, distribution system, or other equipment item with a nearby item of safety equipment caused by relative motions from an earthquake. The interactions of concern are

- (a) proximity effects
- (b) structural failure and falling
- (c) flexibility of attached lines and cables

severe accident: an accident that involves extensive core damage and fission product release into the reactor vessel and containment, with potential release to the environment.

severity factor: severity factor is the probability that fire ignition would include certain specific conditions that influence its rate of growth, level of energy emanated, and duration (time to self-extinguishment) to levels at which target damage is generated.

shall: used to state a mandatory requirement.

should: used to state a recommendation.

significant accident progression sequence: one of the set of accident sequences contributing to large early release frequency resulting from the analysis of a specific hazard group that, when rank-ordered by decreasing frequency, sum to a specified percentage of the large early release frequency, or that individually contribute more than a specified percentage of large early release frequency for that hazard group. For this version of the Standard,³ the

summed percentage is 95% and the individual percentage is 1% of the applicable hazard group. (See Part 2 Requirements LE-C3, LE-C4, LE-E5, LE-C10, LE-C12, LE-D1, LE-D4, LE-D5, LE-D7, and LE-E2.) For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate, and, if used, should be justified.

significant accident sequence: one of the set of accident sequences resulting from the analysis of a specific hazard group, defined at the functional or systematic level, that, when rank-ordered by decreasing frequency, sum to a specified percentage of the core damage frequency for that hazard group, or that individually contribute more than a specified percentage of core damage frequency. For this version of the Standard,³ the summed percentage is 95% and the individual percentage is 1% of the applicable hazard group. (See Part 2 Requirements IE-B3, HR-H1, QU-B2, QU-C1, QU-D1, QU-D5, and QU-F2.) For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate, and, if used, should be justified.

significant basic event: a basic event that contributes significantly to the computed risks for a specific hazard group. For internal events,⁴ this includes any basic event that has an FV importance greater than 0.005 or a RAW importance greater than 2. (See Part 2 Requirements DA-C13, DA-D1, DA-D3, DA-D5, DA-D8, HR-D2, and HR-G1.) For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate, and, if used, should be justified.

significant containment challenge: a containment challenge that results in a containment failure mode that is represented in a significant accident progression sequence.

significant contributor: in the context of

(a) an internal events accident sequence/cutset, a significant basic event or an initiating event that contributes to a significant sequence

(b) accident sequences/cutsets for hazard groups other than internal events, the following are also included: the hazard source, hazard intensity, and hazard damage scenario; for example, for Fire PRA, fire ignition source, physical analysis unit, or fire scenario that contributes to a significant accident sequence would also be included

(c) an accident progression sequence, a contributor that is an essential characteristic (e.g., containment failure mode, physical phenomena) of a significant accident progression sequence, and if not modeled would lead to the omission of the sequence

³ Alternative criteria may be appropriate for specific applications. In particular, an alternative definition of "significant" may be appropriate for a given application where the results from PRA models for different hazard groups need to be combined.

⁴ The examples in this version of the Combined Standard are focused primarily on internal events. Additional examples will be added in a future revision.

significant cutset: one of the set of cutsets resulting from the analysis of a specific hazard group that, when rank-ordered by decreasing frequency, sum to a specified percentage of the core damage frequency (or large early release frequency) for that hazard group, or that individually contribute more than a specified percentage of core damage frequency (or large early release frequency). For this version of the Standard,³ the summed percentage is 95% and the individual percentage is 1% of the applicable hazard group. Cutset significance may be measured relative to overall CDF (or LERF) or relative to an individual accident sequence CDF (or LERF) of the applicable hazard group. (See Part 2 Requirements QU-A2, QU-B2.) For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate, and, if used, should be justified.

skill of the craft: that level of skill expected of the personnel performing the associated function.

source of model uncertainty: a source is related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event). A source of model uncertainty is labeled “key” when it could impact the PRA results that are being used in a decision, and consequently, may influence the decision being made. Therefore, a key source of model uncertainty is identified in the context of an application. This impact would need to be significant enough that it changes the degree to which the risk acceptance criteria are met, and therefore, could potentially influence the decision. For example, for an application for a licensing base change using the acceptance criteria in RG 1.1.74, a source of model uncertainty or related assumption could be considered “key” if it results in uncertainty regarding whether the result lies in Region II or Region I, or if it results in uncertainty regarding whether the result becomes close to the region boundary or not.

spectral acceleration: spectral acceleration, in general, given as a function of period or frequency and damping ratio (typically 5%), is equal to the peak relative displacement of a linear oscillator of frequency, f , attached to the ground, times the quantity $(2\pi f)^2$. It is expressed in gravitational acceleration (g) or centimeters per second squared (cm/s^2).

split fraction: a unitless quantity that represents the conditional (on preceding events) probability of choosing one direction rather than the other through a branch point of an event tree.

spurious operation: the undesired operation of equipment resulting from a fire that could affect the capability to

achieve and maintain safe shutdown (RG 1.189 [1-3]).

state-of-knowledge correlation: the correlation that arises between sample values when performing uncertainty analysis for cutsets consisting of basic events using a sampling approach (such as the Monte Carlo method); when taken into account, this results, for each sample, in the same value being used for all basic event probabilities to which the same data applies.

station blackout: complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant.

statistical model: a model in which a modeling parameter or behavior is treated as a random variable with specified statistical characteristics.

success criteria: criteria for establishing the minimum number or combinations of systems or components required to operate, or minimum levels of performance per component during a specific period of time, to ensure that the safety functions are satisfied.

success path: a set of systems and associated components that can be used to bring the plant to a stable hot or cold condition and maintain this condition for at least 72 hr.

support system: a system that provides a support function (e.g., electric power, control power, or cooling) for one or more other systems.

system failure: loss of the ability of a system to perform a modeled function.

target: may refer to a fire damage target and/or to an ignition target. A fire damage target is any item whose function can be adversely affected by the modeled fire. Typically, a fire damage target is a cable or equipment item that belongs to the Fire PRA cable or equipment list and that is included in event trees and fault trees for fire risk estimation. An ignition target would be any flammable or combustible material to which fire might spread (NUREG/CR-6850-EPRI TR-1011989 [1-5]).

target set: a group of damage targets that will be assumed to suffer fire-induced damage based on the same damage criteria and damage threshold in any given fire scenario.

NOTE: The collection of target sets associated with a fire scenario often represents a subset of the damage targets present in the fire compartment but may also encompass all risk-relevant damage targets in a single physical analysis unit or a collection of damage targets in multiple physical analysis units. This definition implies that all members of any single target set will be assumed to fail when the first member of the target set fails (i.e., “. . . damage based on the same damage criteria and damage threshold”). Progressive or time-dependent states of fire damage may be represented through the definition of multiple target sets for a single fire scenario (e.g., cables in raceways directly above a fire source versus cables in raceways remote from the fire source). The level of detail associated with target set definition will generally parallel the level of detail employed in fire scenario selection and analysis (e.g., screening level analysis versus detailed analysis).

time available: the time period from the presentation of a cue for human action or equipment response to the time of adverse consequences if no action is taken.

top event: undesired state of a system in the fault tree model (e.g., the failure of the system to accomplish its function) that is the starting point (at the top) of the fault tree.

transient combustible: combustible materials that are not fixed in place or an integral part of an operating system or component (RG 1.189 [1-3]). (Note that the term “component” as used in this definition is considered interchangeable with the terms “equipment” or “piece of equipment” as those terms are used in this Standard.)

truncation limit: the numerical cutoff value of probability or frequency below whose results are not retained in the quantitative PRA model or used in subsequent calculations (such limits can apply to accident sequences/cutsets, system level cutsets, and sequence/cutset database retention).

unavailability: the probability that a system or component is not capable of supporting its function including, but not limited to, the time it is disabled for test or maintenance.

uncertainty: a representation of the confidence in the state of knowledge about the parameter values and models used in constructing the PRA.

uncertainty analysis: the process of identifying and characterizing the sources of uncertainty in the analysis, and evaluating their impact on the PRA results and developing a quantitative measure to the extent practical.

uncertainty (as used in seismic-fragility analysis): the variability in the median seismic capacity arising from imperfect knowledge about the models and model parameters used to calculate the median capacity.

uniform hazard response spectrum (UHS): a plot of a ground response parameter (for example, spectral acceleration or spectral velocity) that has an equal likelihood of exceedance at different frequencies.

unreliability: the probability that a system or component will not perform its specified function under given conditions upon demand or for a prescribed time.

verify: to determine that a particular action has been performed in accordance with the requirements of this Standard, either by witnessing the action or by reviewing records.

walkdown: inspection of local areas in a nuclear power plant where systems and components are physically located to ensure accuracy of procedures and drawings, equipment location, operating status, and environmental effects or system interaction effects on the equipment, which could occur during accident conditions.

Section 1-3

Risk Assessment Application Process

1-3.1 PURPOSE

This Section describes required activities to establish the capability of a PRA to support a particular risk-informed application. For this Section, the term “PRA” (or “PRA model”) can refer to either an integrated model that includes all relevant hazard groups or multiple PRA models that address one or more hazard groups. For a specific application, PRA capabilities are evaluated in terms of Capability Categories for individual Supporting Requirements (SRs) rather than by specifying a single Capability Category for the whole PRA. Depending on the application, the required PRA capabilities may vary over and within different Parts of this Standard. The process is intended to be used with PRAs that have had a peer review that meets the requirements of the Peer Review Section of each respective Part of this Standard.

Figure 1-3-1 shows one logical ordering for the process. However, although the specified activities are required, their order of execution may vary. As shown in the dashed-line boxes, there are five stages to the process:

(a) *Stage A.* An application is defined in terms of the structures, systems, and components (SSCs) and activities affected by the proposed change. For the particular application, the portions of a PRA affected by the plant change are determined (i.e., the relevant portions), and the hazard group(s) needed to be addressed in the application, the scope within the PRA related to the application, and risk metrics needed to support the application are identified. By using an understanding of the cause-and-effect relationship between the application and the portions of a PRA model that are particularly sensitive to the proposed change, the relative importance of each portion of the PRA necessary to support the application are determined. The SRs relevant to the different portions of a PRA within the scope, across the elements, and possibly within each element, may be required to have different Capability Categories to support the application, and some portions of a PRA may be irrelevant to the application.

(b) *Stage B.* The relevant portions of the PRA are examined to determine whether the scope and level of detail are sufficient for the application. If the relevant portions of the PRA are found lacking in one or more areas, they may be upgraded or supplemented by other analyses (Stage E).

(c) *Stage C.* An evaluation is performed to determine whether the capability requirements for the SRs from

the Standard for each relevant portion of the PRA are sufficient to support the application. If not, the SRs may be augmented with supplementary requirements as described in Stage E.

(d) *Stage D.* Each relevant portion of the PRA is compared to the appropriate SRs in the Standard for the Capability Category needed to support the application as determined in Stage A. It is determined whether the relevant portions of the PRA have adequate capability, need upgrading to meet the appropriate set of SRs, or need supplementary analyses as described in Stage E.

(e) *Stage E.* The relevant portions of the PRA, supplemented by additional analyses if necessary, are used to support the application. This activity is outside the scope of this Standard.

The scope of the activities in Fig. 1-3-1 determines how to evaluate the role of the PRA in the application and how to determine which Capability Categories are needed for each portion of the PRA to support an application. The criteria for judging the quality of any supplementary analyses that are performed in lieu of upgrading the PRA to meet a desired Capability Category are outside the scope of this Standard. Accordingly, to “meet this Standard” means that the portions of the PRA used in the application meet the High Level Requirements and SRs for a specified set of Capability Categories. The determination of how the PRA is used in the application and which Capability Categories are appropriate for each application must be made on a case-by-case basis.

1-3.2 IDENTIFICATION OF APPLICATION AND DETERMINATION OF CAPABILITY CATEGORIES (STAGE A)

1-3.2.1 Identification of Application

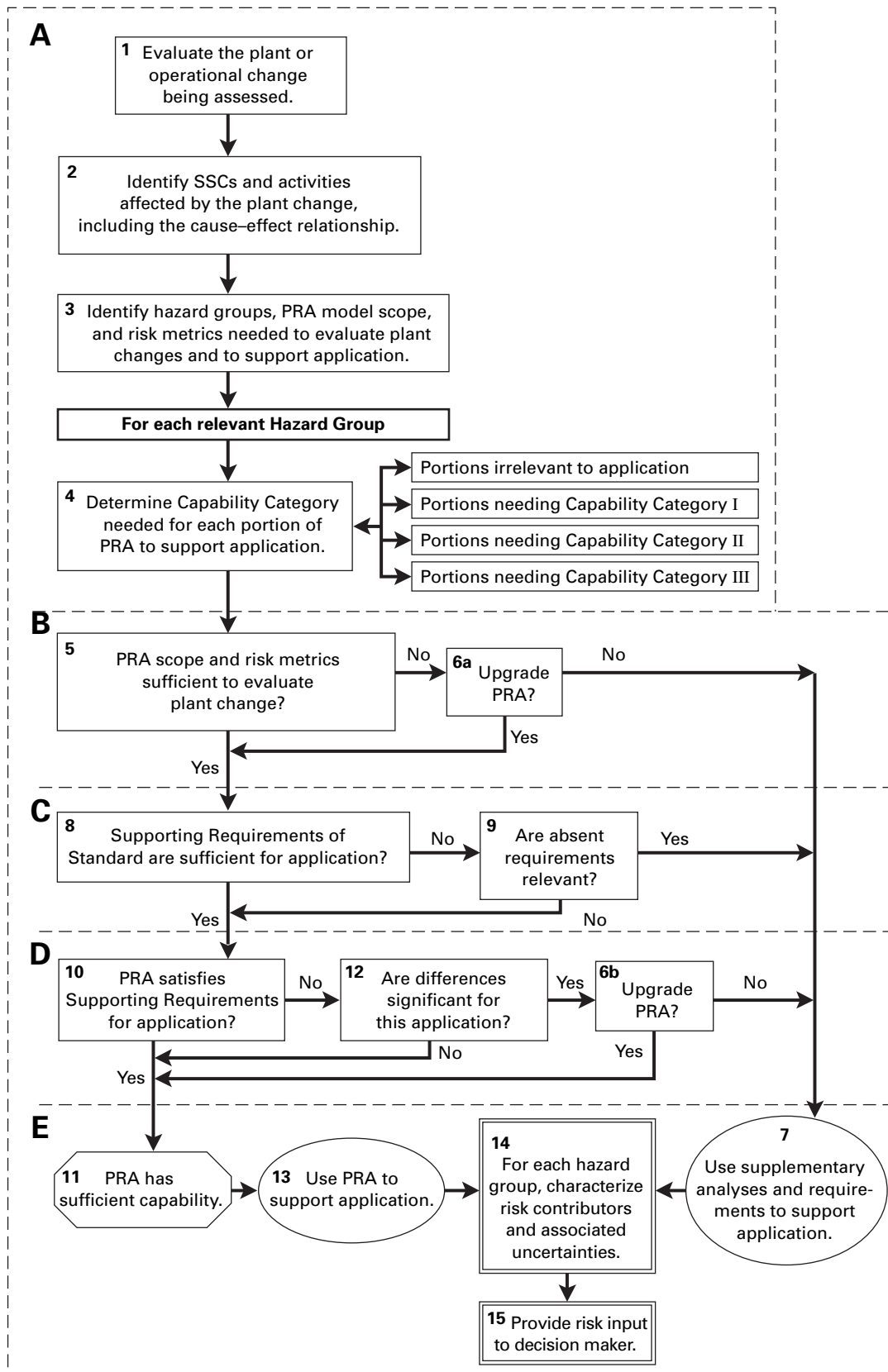
Define the application by

(a) evaluating the plant design or operational change being assessed (Box 1 of Fig. 1-3-1)

(b) identifying the SSCs and plant activities affected by the change including the cause-effect relationship between the plant design or operational change and the PRA model (Box 2 of Fig. 1-3-1)

(c) identifying the hazard groups, PRA model scope, and PRA risk metrics that are needed to assess the change (Box 3 of Fig. 1-3-1).

EPRI TR-105396 [1-8] and NUREG-0800 [1-9] provide guidance for the above activities.

Fig. 1-3-1 Application Process Flowchart

EXAMPLE:⁴ A change in technical specifications (TS) is proposed that redefines the requirements for an operable service water (SW) system. This change removes the TS requirement for an allowed outage time (AOT) from one of the three pumps in each SW loop. In addition, the AOT for other selected combinations of inoperable components is increased. The changes in TS and/or procedures that are involved need to be identified in detail.

To assess the impact of the proposed change in the TS, those SSCs, such as the SW system, affected by the proposed change need to be identified. The plant SW system has two redundant loops, each having two full capacity SW pumps that use the ocean as the ultimate heat sink, and a third SW pump that uses a cooling-tower (CT) and the atmosphere as the heat sink. The SW system is designed such that, in the event of a LOCA concurrent with a loss of offsite power, a single SW pump powered from its associated EDG will have sufficient capacity to meet the heat load. The existing TS require two operable SW loops with each loop having three operable pumps. This requirement exceeds single failure criteria since the second SW pump is required for neither normal conditions nor the design basis accident, and the CT SW pump provides the redundancy for the design basis LOCA. The proposed change redefines an operable SW loop as having one operable SW pump and one operable CT SW pump, removes the AOT requirements from two SW pumps, lengthens the AOT requirement for SW pumps in the same loop to bring it into line with the AOT for single SW train unavailability, and increases the standby CT SW pump AOT based on its lower risk importance.

The proposed change in the AOT impacts the core damage frequency (CDF) by increasing the likelihood that an SW pump would be unavailable due to planned or unplanned maintenance. This change is evaluated by considering the impact on system unavailability and on the frequency of sequences involving unavailability of a single train of SW.

1-3.2.2 Determination of Capability Categories

The Technical Requirements section of each respective Part of this Standard sets forth SRs for three PRA Capability Categories whose attributes are described in 1-1.3.

For many of the SRs, the distinction between Capability Categories is based on the treatment of significant contributors. Definitions in this Standard containing the word “significance” or “significant” are generally written from the perspective of a specific hazard group. It is important to recognize that, for applications whose risk stems from more than one hazard group, these definitions should be generalized to apply to the sum of risks from all contributing hazard groups.

“Significance” should also be treated differently for those SRs, which refer to SRs in other hazard groups. For example, SR-HR-G1 in Part 2 is incorporated by reference into the HRA requirements of Part 4. For an

application for which internal events and fire are relevant, a Capability Category II for SR-HR-G1 in Section 2 would require by reference in Part 4 that the significant HFEs be treated with a detailed assessment for each hazard group, with significance being measured with respect to the hazard groups individually. However, for the purposes of the application, it would be sufficient to measure significance with respect to the sum of the risk metrics for the two hazard groups. In this case, the intent of Capability Category II would have been met.

For the application, determine the relative importance of each portion of the PRA for each hazard group needed to support the application (Box 4 of Fig. 1-3-1). This determination dictates which Capability Category is needed for each SR for each portion of the PRA to support the application. To determine these capabilities, an evaluation shall be performed of the application to assess the role of the PRA in supporting that application including determining the relative importance of SRs to the application, identifying the portions of the hazard group PRA relevant to the application, and for each relevant portion, determining the Capability Category for each SR needed to support the application. When performing this evaluation, the following application attributes shall be considered:

- (a) role of the PRA in the application and extent of reliance of the decision on the PRA results
- (b) risk metrics to be used to support the application and associated decision criteria
- (c) the significance of the risk contribution from the hazard group to the decision
- (d) degree to which bounding or conservative methods for the PRA or in a given portion of the PRA would lead to inappropriately influencing the decisions made in the application, and approach(es) for accounting for this in the decision-making process
- (e) degree of accuracy and evaluation of uncertainties and sensitivities required of the PRA results
- (f) degree of confidence in the results that is required to support the decision
- (g) extent to which the decisions made in the application will impact the plant design basis.

The Capability Categories and the bases for their determination shall be documented.

EXAMPLE: Continuing with the SW pump AOT change example, the proposed change is a risk-informed application to justify a change to an operating license in accordance with Regulatory Guides 1.174 and 1.177. If the plant has CDF and LERF of $2 \times 10^{-5}/\text{yr}$ and $1 \times 10^{-6}/\text{yr}$, respectively, and it is expected that the changes in CDF can be shown to be small, then the portions of the PRA that are impacted by changes in SW pump unavailability due to maintenance are determined to require PRA Capability Category II, whereas the remaining portions of the PRA needed to determine CDF are determined to only require PRA Capability Category I. Hence the supporting requirements for initiating events, accident sequences, data parameters, system models, human actions, and quantification process for those sequences and cutsets impacted by the AOT changes are in PRA Capability

Category II, and the supporting requirements for the remaining portions of the PRA needed to evaluate CDF are in Capability Category I. The LERF is determined to be not needed for this application based on a qualitative evaluation and hence does not have to meet any of the Capability Categories.

EXAMPLE VARIATION: If the above example application was being evaluated at a plant with core damage frequency that was greater than 1×10^{-4} or baseline LERF greater than 1×10^{-5} , or the changes in CDF or LERF were expected to be significant such that the degree of confidence in the risk evaluation needed to be much greater than with the previous example, it may be determined that those portions of the PRA impacting the change might need to be upgraded. In addition, in this example, it might be necessary to expand the application to include a determination of LERF to confirm that the impacts on LERF are acceptable. This need might mean expansion of the applicable SRs in the LERF PRA element in comparison with the previous example.

1-3.3 ASSESSMENT OF PRA FOR NECESSARY SCOPE, RESULTS, AND MODELS (STAGE B)

1-3.3.1 Necessary Scope and Risk Metrics

Determine if the PRA provides the results needed to assess the plant or operational change (Box 5 of Fig. 1-3-1). If some aspects of the PRA are insufficient to assess the change, then upgrade them in accordance with the SRs in the Technical Requirements section of each respective Part of this Standard for its corresponding Capability Category (Box 6a of Fig. 1-3-1), or generate supplementary analyses (see 1-3.6).

If it is determined that the PRA is sufficient, the bases for this determination shall be documented. Any upgrade of the PRA shall be done and documented in accordance with Section 1-5.

EXAMPLE: The proposed change in the SW AOT has been determined to affect the SW unavailability during at power operating conditions. For the plant in question, the SW provides cooling to the ECCS pumps, the diesel generators, the feedwater pumps, the CCW system, and the radwaste system. Therefore, for internal events, the scope of the internal initiating events at power analysis element of the PRA must include

- (a) LOCA initiators, since the change in SW unavailability will affect ECCS pump cooling in the recirculation phase
- (b) loss of offsite power initiators, since the SW change will affect the diesel generators
- (c) loss of feedwater initiators, since the feedwater pumps are SW cooled

Although the SW cools the CCW system, there is enough thermal inertia in the CCW system to allow it to function for several hours after the loss of SW, thereby enabling the plant to be placed in a safe stable state; a loss of CCW initiator would not be needed for this application. Also, since the radwaste system does not play a part in determining CDF, it need not be considered in the PRA. Any impact would be considered in Box 15 of Fig. 1-3.1-1, as needed. It is determined that the changes in maintenance unavailability are too small to consider impacts on the reliability of the SW pumps that could impact a wider range of sequences, including loss of service water initiating events and sequences with SW pump failures. These impacts are combined in the plant model to calculate the change in CDF. A determination is made that there are no unique contributions to LERF for this plant, and hence the changes

in LERF are proportional to the changes in CDF. Since only the Δ CDF is needed, only CDFs before and after the change in TS are needed.

1-3.3.2 Modeling of SSCs and Activities

Determine if the SSCs or plant activities affected by the plant design or operational change are modeled in the PRA (Box 5 of Fig. 1-3-1). If the affected SSCs or plant activities are not modeled, then either upgrade the PRA to include the SSCs in accordance with the SRs in the Technical Requirements section of each respective Part of this Standard for their corresponding Capability Category (Box 6a of Fig. 1-3-1), or generate supplementary analyses (see 1-3.6).

If it is determined that each portion of the PRA is sufficient, the bases for this determination shall be documented. Any upgrade of the PRA shall be done and documented in accordance with Section 1-5.

EXAMPLE: Continuing with the previous example, the SSCs and plant activities related to the systems impacted by the proposed change in the SW, and which contribute to the change in CDF (i.e., ECCS, DGs, Feedwater, and CCW), need to be modeled in the PRA. For example, if the loss of feedwater initiator is modeled as one global initiator, then either the PRA needs to be upgraded to include the relationship between SW and Feedwater, or the effect of SW on Feedwater must be resolved by using supplementary analyses outside of this Standard.

1-3.3.3 Peer Review

The portions of a PRA that are needed for an application shall have been reviewed pursuant to the requirements of the Peer Review Section of each respective Part of this Standard.

1-3.4 DETERMINATION OF THE STANDARD'S SCOPE AND LEVEL OF DETAIL (STAGE C)

Determine if the scope of coverage and level of detail of the SRs stated in the Technical Requirements section of each respective Part of this Standard, for the corresponding Capability Categories determined in 1-3.2.2, are sufficient to assess the application under consideration (Box 8 of Fig. 1-3-1).

If it is determined that the Standard lacks specific requirements, their relevance to the application shall be assessed (Box 9 of Fig. 1-3-1). If the absent requirements are not relevant, the requirements of the Standard are sufficient for the application. The bases for determining the sufficiency of this Standard shall be documented. If the absent requirements are relevant, supplementary requirements may be used (Box 7 of Fig. 1-3-1).

1-3.5 COMPARISON OF PRA MODEL TO STANDARD (STAGE D)

Determine if each portion of the PRA satisfies the SRs at the appropriate Capability Category needed to support the application (Box 10 of Fig. 1-3-1). The results

of the Peer Review may be used. If the PRA meets the SRs necessary for the application, the PRA is acceptable for the application being considered (Box 11 of Fig. 1-3-1). The bases for this determination shall be documented.

If the PRA does not satisfy a SR for the appropriate Capability Category, then determine whether the reason it is not being satisfied is relevant or significant (Box 12 of Fig. 1-3-1). Acceptable requirements for demonstrating the relevance or significance include either of the following:

(a) The reason for not meeting the SR at the appropriate Capability Category is not relevant if it is not applicable or does not affect quantification relative to the impact of the proposed application (for example, if an SR related to the treatment of Human Reliability has not been met because some of the HEPs for HFEs that are significant in the base case have not been evaluated using a detailed HRA method, but those particular HFEs play no role in the results needed for the application, then the failure to meet Capability Category II is not relevant to the decision).

(b) The difference is not significant if the modeled accident sequences accounting for at least 90% of CDF/LERF for the hazard group or hazard groups being evaluated, as applicable, are not affected by appropriate sensitivity studies or bounding evaluations. These studies or evaluations should measure the aggregate impact of the exceptions to the requirements in the Technical Requirements section of each respective Part of this Standard as applied to the application. The relevant hazard groups may be evaluated separately or in a combined fashion, as needed to determine the significance of the difference for the application.

This determination will depend on the particular application being considered and may involve determinations made by an expert panel.

If the difference is neither relevant nor significant, then the PRA is acceptable for the application. If the difference is relevant or significant, then either upgrade the PRA to address the corresponding SRs stated in the Technical Requirements section of each respective Part of this Standard (Box 6b of Fig. 1-3-1), or generate supplementary analyses (see 1-3.6). Any upgrade of the PRA shall be done and documented in accordance with Section 1-5.

1-3.6 ACCESSING THE RISK IMPLICATIONS (STAGE E)

1-3.6.1 Use of Supplementary Analyses

If the scope of either the PRA or the Standard is insufficient, supplementary analyses or requirements may be used (Box 7 of Fig. 1-3-1). These supplementary analyses will depend on the particular application being considered, but may involve deterministic methods such as

bounding or screening analyses, and determinations made by an expert panel. They shall be documented.

EXAMPLE OF SUPPLEMENTARY ANALYSIS: A change in testing frequency is desired for MOVs judged to be of low safety significance by using a risk-informed ranking method. Not all MOVs or MOV failure modes of interest within the program are represented in the PRA. Specifically, valves providing an isolation function between the reactor vessel and low-pressure piping may only be represented in the interfacing system LOCA initiator frequency. The inadequate PRA model representation can be supplemented by categorizing the group of high/low pressure interface MOVs in an appropriate LERF category. The categorization is based on PRA insights that indicate failure of MOVs to isolate reactor vessel pressure have the potential to lead to an LERF condition. This example illustrates a process of addressing SSC model adequacy by using general risk information to support the placement of MOVs into the appropriate risk category.

Supplementary requirements shall be drawn from other recognized codes or standards whose scopes complement that of this Standard and that are applicable to the application, but may be generated by an expert panel if no such recognized code or standard can be identified.

SUPPLEMENTARY REQUIREMENTS, EXAMPLE 1: A risk ranking/categorization for a plant's ISI program is being pursued. The current PRA model meets the requirements set forth in this Standard. However, the Standard does not provide requirements for modeling piping or pipe segments adequate to support a detailed quantitative ranking. The Standard can be supplemented with an expert panel to determine the safety significance of pipe segments. Considerations of deterministic and other traditional engineering analyses, defense-in-depth philosophy, or maintenance of safety margins could be used to categorize pipe segments. Use of published industry or NRC guidance documents on risk-informed ISI could also be used to supplement the Standard. The PRA model could also be used to supplement the Standard by estimating the impact of each pipe segment's failure on risk without modifying the PRA's logic. This estimate could be accomplished by identifying an initiating event, basic event, or group of events, already modeled in the PRA, whose failures capture the effects of the pipe segment failure.

SUPPLEMENTARY REQUIREMENTS, EXAMPLE 2: It is desired to rank the snubbers in a plant according to their risk significance for developing a graded approach to snubber testing. With the exception of snubbers on large primary system components, snubbers have been shown to have a small impact on CDF; therefore, the Standard does not require their failure to be addressed in determining CDF and LERF. However, snubbers are considered safety-related, and testing programs are required to demonstrate their capability to perform their dynamic support function. As shown in ASME Code Case OMN-10 [1-10], evaluation of failure mechanisms may show that the safety significance of snubbers can be approximated by the safety significance of the components that they support for the events in which the snubbers are safety significant, and this supplementary criterion could be used to rank the safety importance of the snubbers.

SUPPLEMENTARY REQUIREMENTS, EXAMPLE 3: It is desired to replace certain MOVs that are currently considered safety-grade with commercial-grade equipment when new valves are procured. The internal-events PRA shows that these valves have a minor role in important accident sequences, and that the only important failure mode is failure to open on demand. The failure rate of the commercial-grade valves for this mode is known through reliable

data to be identical to the failure rate for safety-grade valves. However, the question arises about whether the commercial-grade valves will perform as well as safety-grade valves during and after a large earthquake. To address it, supplementary requirements, found in an appropriate reference (e.g., *Nuclear Engineering and Design* [1-11]) may be used. By using this reference, the seismic capacity of the commercial-grade valves can be evaluated and can be compared to that of the safety-grade valves that they would replace.

1-3.6.2 Results of Supplementary Analyses

If it has been determined that the PRA has sufficient capability, its results can be used to support the

application (Box 13 of Fig. 1-3-1). If not, the results of supplementary analyses, some of which may respond to supplementary requirements, can also be used to support the application (Box 7 of Fig. 1-3-1). Such supplementary analyses/requirements are outside the scope of this Standard.

The risk contributors and associated uncertainties should be characterized for each hazard group (Box 14 of Fig. 1-3-1). Once all relevant hazard groups have been characterized, the risk input is provided to the decision maker (Box 15 of Fig. 1-3-1). The relevant hazard groups may be characterized separately or in a combined fashion, as needed to support the application.

Section 1-4

Risk Assessment Technical Requirements

1-4.1 PURPOSE

The purpose of this Section is to provide requirements by which adequate PRA capability can be identified when a PRA is used to support applications of risk-informed decision making. This Section also includes general requirements for process checking of analyses and calculations and for use of expert judgment.

1-4.2 PROCESS CHECK

Analyses and/or calculations used directly by the PRA (e.g., HRA, data analysis) or used to support the PRA (e.g., thermal-hydraulics calculations to support mission success definition) shall be reviewed by knowledgeable individuals who did not perform those analyses or calculations. Documentation of this review may take the form of hand-written comments, signatures, or initials on the analyses/calculations; formal sign-offs; or other equivalent methods.

This paragraph provides requirements for the use of expert judgment outside of the PRA analysis team to resolve a specific technical issue.

1-4.3 USE OF EXPERT JUDGMENT

This paragraph provides requirements for the use of expert judgment outside of the PRA analysis team to resolve a specific technical issue.

Guidance from NUREG/CR-6372 [1-12] and NUREG-1563 [1-13] may be used to meet the requirements in this paragraph. Other approaches, or a mix of these, may also be used.

EXAMPLES: Use of expert judgment to resolve difficult issues includes Pacific Gas and Electric's Diablo Canyon seismic study [1-14] and the Yucca Mountain project's study of volcanic hazards [1-15]. These reports provide useful insights into both the strengths and the potential pitfalls of using experts. A review of expert-aggregation methods, the different types of consensus, and issues with resolving disagreements among experts can be found in Appendix J of NUREG/CR-6372 [1-12].

1-4.3.1 Objective of Using Expert Judgment

The PRA analysis team shall explicitly and clearly define the objective of the information that is being sought through the use of outside expert judgment, and shall explain this objective and the intended use of the information to the expert(s).

1-4.3.2 Identification of the Technical Issue

The PRA analysis team shall explicitly and clearly define the specific technical issue to be addressed by the expert(s).

1-4.3.3 Determination of the Need for Outside Expert Judgment

The PRA analysis team may elect to resolve a technical issue using their own expert judgment, or the judgment of others within their organization.

The PRA analysis team shall use outside experts when the needed expertise on the given technical issue is not available within the analysis team or within the team's organization. The PRA analysis team should use outside experts, even when such expertise is available inside, if there is a need to obtain broader perspectives, for any of the following or related reasons:

(a) complex experimental data exist that the analysts know have been interpreted differently by different outside experts

(b) more than one conceptual model exists for interpreting the technical issue, and judgment is needed as to the applicability of the different models

(c) judgments are required to assess whether bounding assumptions or calculations are appropriately conservative

(d) uncertainties are large and significant, and judgments of outside technical experts are useful in illuminating the specific issue

1-4.3.4 Identification of Expert Judgment Process

The PRA analysis team shall determine

(a) the degree of importance and the level of complexity of the issue

(b) whether the process will use a single entity (individual, team, company, etc.) that will act as an evaluator and integrator and will be responsible for developing the community distribution, or will use a panel of expert evaluators and a facilitator/integrator

The facilitator/integrator shall be responsible for aggregating the judgments and community distributions of the panel of experts so as to develop the composite distribution of the informed technical community.

1-4.3.5 Identification and Selection of Evaluator Experts

The PRA analysis team shall identify one or more experts capable of evaluating the relative credibility of

multiple alternative hypotheses to explain the available information. These experts shall evaluate all potential hypotheses and bases of inputs from the literature, and from proponents and resource experts, and shall provide

- (a) their own input
- (b) their representation of the community distribution

1-4.3.6 Identification and Selection of Technical Issue Experts

If needed, the PRA analysis team shall also identify other technical issue experts such as

- (a) experts who advocate particular hypotheses or technical positions (e.g., an individual who evaluates data and develops a particular hypothesis to explain the data)
- (b) technical experts with knowledge of a particular technical area of relevance to the issue

1-4.3.7 Responsibility for the Expert Judgment

The PRA analysis team shall assign responsibility for the resulting judgments, either to an integrator or shared with the experts. Each individual expert shall accept responsibility for his individual judgments and interpretations.

1-4.4 DERIVATION OF PRA REQUIREMENTS

Objectives were established for each technical element used to characterize the respective scope of a PRA. The objectives reflect substantial experience accumulated with PRA development and usage, and are consistent with the PRA Procedures Guide [1-16] and the NEI-00-02 [1-17] Peer Review Process Guidance, where applicable. These objectives form the basis for development of the high-level requirements (HLRs) for each element that were used, in turn, to define the supporting requirements (SRs).

In setting the HLRs for each element, the goal was to derive, based on the objectives, an irreducible set of firm requirements, applicable to PRAs that support all levels of application, to guide the development of SRs. This

goal reflects the diversity of approaches that have been used to develop existing PRAs and the need to allow for technological innovations in the future. An additional goal was to derive a reasonably small set of HLRs that capture all the important technical issues that were identified in the efforts to develop this Standard and to implement the NEI-00-02 PRA Peer Review process guidance.

The HLRs generally address attributes of the PRA element such as

- (a) scope and level of detail
- (b) model fidelity and realism
- (c) output or quantitative results (if applicable)
- (d) documentation

Three sets of SRs were developed to support the HLRs in the form of action statements for the various capability categories in the Standard. Therefore, there is a complete set of SRs provided for each of the three PRA Capability Categories described in 1-1.3.

1-4.5 PRA REQUIREMENTS

Tables of HLRs and SRs for the technical elements are provided for each PRA scope. The SRs are numbered and labeled to identify the HLR that is supported. For each Capability Category, the SRs define the minimum requirements necessary to meet that Capability Category. In these tables, some action statements apply to only one Capability Category and some extend across two or three Capability Categories. When an action spans multiple Capability Categories, it applies equally to each Capability Category. When necessary, the differentiation between Capability Categories is made in other associated SRs. The interpretation of a Supporting Requirement whose action statement spans multiple Capability Categories is stated in Table 1-1.3-2. It should be noted that some action statements span Capability Categories II and III because the authors were unable to specify a distinguishing requirement for Capability Category III at this time. It is intended that, by meeting all the SRs under a given HLR, a PRA will meet that HLR.

Section 1-5

PRA Configuration Control

1-5.1 PURPOSE

This Section provides requirements for configuration control of a PRA to be used with this Standard to support risk-informed decisions for nuclear power plants.

1-5.2 PRA CONFIGURATION CONTROL PROGRAM

A PRA Configuration Control Program shall be in place. It shall contain the following key elements:

- (a) a process for monitoring PRA inputs and collecting new information
- (b) a process that maintains and upgrades the PRA to be consistent with the as-built, as operated plant
- (c) a process that ensures that the cumulative impact of pending changes is considered when applying the PRA
- (d) a process that maintains configuration control of computer codes used to support PRA quantification
- (e) documentation of the Program

1-5.3 MONITORING PRA INPUTS AND COLLECTING NEW INFORMATION

The PRA Configuration Control Program shall include a process to monitor changes in the design, operation, maintenance, and industry-wide operational history that could affect the PRA. These changes shall include inputs that impact operating procedures, design configuration, initiating event frequencies, system or subsystem unavailability, and component failure rates. The program should include monitoring of changes to the PRA technology and industry experience that could change the results of the PRA model.

1-5.4 PRA MAINTENANCE AND UPGRADES

The PRA shall be maintained and upgraded, such that its representation of the as-built, as-operated plant is sufficient to support the applications for which it is being used.

Changes in PRA inputs or discovery of new information identified pursuant to 1-5.3 shall be evaluated to determine whether such information warrants PRA maintenance or PRA upgrade. (See Section 1-2 for the distinction between PRA maintenance and PRA upgrade.) Changes that would impact risk-informed decisions should be incorporated as soon as practical. Changes that are relevant to a specific application shall meet the SRs pertinent to that application as determined through the process described in 1-3.5.

Changes to a PRA due to PRA maintenance and PRA upgrade shall meet the requirements of the Technical

Requirements Section of each respective Part of this Standard. Upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the Peer Review Section of each respective Part of this Standard, but limited to aspects of the PRA that have been upgraded.

1-5.5 PENDING CHANGES

This Standard recognizes that immediately following a plant change [e.g., modifications, procedure changes, plant performance (data)], or upon identification of a subject for model improvement (e.g., new human error analysis methodology, new data update methods), a PRA may not represent the plant until the subject plant change or model improvement is incorporated into the PRA. Therefore, the PRA configuration control process shall consider the cumulative impact of pending plant changes or model improvements on the application being performed. The impact of these plant changes or model improvements on the results of the PRA and the decision under consideration in the application shall be evaluated in a fashion similar to the approach used in Section 1-3.

1-5.6 USE OF COMPUTER CODES

The computer codes used to support and to perform PRA analyses shall be controlled to ensure consistent, reproducible results.

1-5.7 DOCUMENTATION

Documentation of the Configuration Control Program and of the performance of the above elements shall be adequate to demonstrate that the PRA is being maintained consistently with the as-built, as-operated plant.

The documentation typically includes

- (a) a description of the process used to monitor PRA inputs and collect new information
- (b) evidence that the aforementioned process is active
- (c) descriptions of proposed changes
- (d) description of changes in a PRA due to each PRA upgrade or PRA maintenance
- (e) record of the performance and results of the appropriate PRA reviews (consistent with the requirements of 1-6.6)
- (f) record of the process and results used to address the cumulative impact of pending changes
- (g) a description of the process used to maintain software configuration control

Section 1-6

Peer Review

1-6.1 PURPOSE

This Section provides requirements for peer review of a PRA to be used in risk-informed decisions for commercial nuclear power plants. PRAs used for applications applying this Standard shall be peer reviewed. Peer reviews for this purpose shall be performed against the requirements in those Sections of this Standard applicable to the portions of the PRA that are being used to support such applications. The peer review shall assess the PRA to the extent necessary to determine if the methodology and its implementation meet the requirements of this Standard. Another purpose of the peer review is to determine strengths and weaknesses in the PRA. The peer review need not assess all aspects of the PRA against all requirements in the Technical Requirements Section of each respective Part of this Standard; however, enough aspects of the PRA shall be reviewed for the reviewers to achieve consensus on the adequacy of methodologies and their implementation for each PRA Element.

1-6.1.1 Frequency

Only a single complete peer review is necessary prior to using a PRA. In addition, Section 1-5 of this Standard requires peer review for upgrades of a PRA. When peer reviews are conducted on PRA upgrades, the latest review shall be considered the review of record. The scope of an additional peer review may be confined to changes to the PRA that have occurred since the previous review.

1-6.1.2 Methodology

The review shall be performed using a written methodology that assesses the requirements of the Technical Requirements section of each respective Part of this Standard and addresses the requirements of the Peer Review Section of each respective Part of this Standard.

The peer review methodology shall consist of the following elements:

- (a) process for selection of the peer review team
- (b) training in the peer review process
- (c) an approach to be used by the peer review team for assessing if the PRA meets the supporting requirements of the Technical Requirements section of each respective Part of this Standard
- (d) a process by which differing professional opinions are to be addressed and resolved

(e) an approach for reviewing the PRA configuration control

(f) a method for documenting the results of the review

1-6.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATIONS

1-6.2.1 Collective Team

The peer review team shall consist of personnel whose collective qualifications include

(a) the ability to assess all the PRA Elements of the Technical Requirements section of each respective Part of this Standard, as applicable, and the interfaces between those elements

(b) the collective knowledge of the plant NSSS design, containment design, and plant operation

1-6.2.2 Individual Team Members

(a) The peer review team members individually shall be

(1) knowledgeable of the requirements in this Standard for their area of review

(2) experienced in performing the activities related to the PRA Elements for which the reviewer is assigned

(b) To avoid any perception of a technical conflict of interest, the peer review team members shall have neither performed nor directly supervised any work on the portions of the PRA being reviewed.

1-6.2.3 Review Team Members for PRA Upgrades

When a peer review is being performed on a PRA upgrade, reviewers shall have knowledge and experience appropriate for the specific PRA Elements being reviewed. However, the other requirements of this Section shall also apply.

1-6.2.4 Knowledge of Specific Aspects of the PRA Elements

The peer reviewer shall also be knowledgeable (by direct experience) of the specific methodology, code, tool, or approach (e.g., accident sequence support state approach, MAAP code, THERP method) that was used in the PRA Element assigned for review. Understanding and competence in the assigned area shall be demonstrated by the range of the individual's experience in the number of different, independent activities performed in the assigned area, as well as the different levels of complexity of these activities.

(a) One member of the peer review team (the technical integrator) shall be familiar with all the PRA Elements identified in the Section of this Standard under review and shall have demonstrated the capability to integrate these PRA Elements. When more than one Section is under review, a separate technical integrator may be used for each Section.

(b) The peer review team shall have a team leader to lead the team in the performance of the review. The team leader need not be the technical integrator.

(c) The peer review should be conducted by a team with a minimum of five members, and shall be performed over a minimum period of one week. If the review is focused on a particular PRA Element, such as a review of an upgrade of a PRA Element, then the peer review should be conducted by a team with a minimum of two members, performed over a time necessary to address the specific PRA Element.

(d) Exceptions to the requirements of this paragraph may be taken based on the availability of appropriate personnel to develop a team. A single-person peer review shall only be justified when the review involves an upgrade of a single element and the reviewer has acceptable qualifications for the technologies involved in the upgrade. All such exceptions shall be documented in accordance with 1-6.6 of this Standard.

1-6.3 REVIEW OF PRA ELEMENTS TO CONFIRM THE METHODOLOGY

The peer review team shall use the requirements of the Peer Review Section of each respective Part of this Standard for the PRA Elements being reviewed to determine if the methodology and the implementation of the methodology for each PRA Element meet the requirements of this Standard. Additional material for those Elements may be reviewed depending on the results obtained. These suggestions are not intended to be a minimum or comprehensive list of requirements. The judgment of the reviewer shall be used to determine the specific scope and depth of the review in each PRA Element.

The results of the overall PRA, including models and assumptions, and the results of each PRA Element shall be reviewed to determine their reasonableness given the design and operation of the plant (e.g., investigation of cutset or sequence combinations for reasonableness).

The HLRs and the composite of the SRs of the Technical Requirements Section of each respective Part of this Standard shall be used by the peer review team to assess the completeness of a PRA Element.

1-6.4 EXPERT JUDGMENT

The use of expert judgment to implement requirements in this Standard shall be reviewed using the considerations in 1-4.3.

1-6.5 PRA CONFIGURATION CONTROL

The peer review team shall review the process, including implementation, for maintaining or upgrading the PRA against the configuration control requirements of this Standard.

1-6.6 DOCUMENTATION

1-6.6.1 Peer Review Team Documentation

The peer review team's documentation shall demonstrate that the review process appropriately implemented the review requirements.

Specifically, the peer review documentation shall include the following:

- (a) identification of the version of the PRA reviewed
- (b) the names of the peer review team members
- (c) a brief resume on each team member describing the individual's employer, education, PRA training, and PRA and PRA Element experience and expertise
- (d) the elements of the PRA reviewed by each team member
- (e) a discussion of the extent to which each PRA Element was reviewed
- (f) results of the review identifying any differences between the requirements in the Technical Requirements section of each respective Part of this Standard and Section 1-5 of this Section and the methodology implemented, defined to a sufficient level of detail that will allow the resolution of the differences
- (g) identification and significance of exceptions and deficiencies with respect to SRs, including an assessment of PRA assumptions that the reviewers have determined to be relevant
- (h) at the request of any peer reviewer, differences or dissenting views among peer reviewers
- (i) recommended alternatives for resolution of any differences
- (j) identification of the strengths and weaknesses that have a significant impact on the PRA
- (k) an assessment of the Capability Category of the SRs (i.e., identification of what Capability Category is met for the SRs)

1-6.6.2 Resolution of Peer Review Team Comments

Resolution of Peer Review Team comments shall be documented. Exceptions to the alternatives recommended by the Peer Review team shall be justified.

Section 1-7

References

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NONMANDATORY APPENDIX 1-A

PRA MAINTENANCE, PRA UPGRADE, AND THE ADVISABILITY OF PEER REVIEW

1-A.1 PURPOSE

The purpose of this Appendix is to provide guidance in determining when a change to a nuclear power plant PRA is *PRA maintenance* and when it is a *PRA upgrade*, and when peer review is advisable. PRA maintenance and PRA upgrade are defined in Section 1-2 of this Standard and are restated below. Within the context of Section 1-5, PRA Configuration Control, 1-5.4 requires such a determination and further requires that a PRA upgrade be peer reviewed pursuant to the requirements of the Peer Review Section of each respective Part of this Standard. There is no requirement for PRA maintenance to be peer reviewed.

PRA maintenance: the update of the PRA models to reflect plant changes such as modifications, procedure changes, or plant performance (data).

PRA upgrade: the incorporation into a PRA model of a new methodology or changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. This could include items such as new human error analysis methodology, new data update methods, new approaches to quantification or truncation, or new treatment of common cause failure.

In the definition of “PRA upgrade,” “new” should be interpreted as new to the subject PRA even though the methodology in question has been applied in other PRAs. It is not intended to imply a newly developed method. This interpretation has been used in the criteria, and examples are provided in this Guideline. Also in this definition and elsewhere in the Guideline, “a significant change...in capability” does not necessarily mean a change in Capability Category, which term is described in 1-1.3 of the Standard.

The following section provides guidance on when additional peer review might be advisable even for those changes that are classified as PRA maintenance, and on when a change, nominally classified as an upgrade, may be regarded as PRA maintenance and not subject to peer review. Subsection 1-A.3 provides several examples to illustrate these exceptions.

1-A.2 Nonmandatory Guidance for ASME PRA Standard Regarding Determination of Need for Additional PRA Peer Review

Criterion: The criterion for deciding which PRA changes should be subject to peer review is provided

in Section 1-5 of this Standard. The general requirement is to require such review for PRA upgrades but not for PRA maintenance.

The rationale for this criterion is that *PRA upgrades* represent more extensive changes to the PRA (relative to *PRA maintenance*) and are likely to involve methodologies or scope that were not covered in previous peer reviews. PRA maintenance generally involves changes within the framework of an existing model structure and PRA configuration control program, and involves methodologies that have been applied in the PRA, and been previously peer reviewed.

The following paragraphs are intended to provide guidance to users of the Standard regarding the intended interpretation of the requirement for additional peer reviews. A set of examples, which should be viewed as representative only, and not comprehensive, is also provided to indicate the intended interpretation.

(a) The definition of “PRA maintenance” has a single criterion that it reflects a plant change of which three examples are given, viz. modifications, procedure changes, plant performance (data). The change to the PRA model to reflect the plant change should involve neither new methodology nor changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences; such model changes characterize a PRA upgrade. Under this criterion, a substantial and complex plant design change using the existing PRA model and methodologies would be classified as PRA maintenance. However, if there is significant change in risk insights, prudence may call for a peer review for such a case.⁵ The recommendation is that such a peer review need not be scheduled only on the basis of that change. Instead, an internal review should be performed, thoroughly documented and, when a peer review is scheduled for another reason, its scope should include the complex design change at issue. Alternatively, a focused peer review [1-6.2.4(d)] could be performed, limited to the changes. As a second example, changing from a modest on-line preventative

⁵ Note that footnote 3 under 1-1.1 of the NEI peer review guidance suggests the need for additional (beyond the initial) peer review if “substantial changes are made to the model” independent of the reasons for the change. Refer to Probabilistic Risk Assessment Peer Review Guidance, NEI-00-02, Revision A3, March 20, 2000 [1-16].

maintenance program to a relatively aggressive (extensive) on-line maintenance may not involve new methodology and therefore might be classified as PRA maintenance. However, if many new cutsets are introduced, the aforementioned recommendations may be prudent.

(b) Consideration should be given to the scope or number of PRA maintenances performed. Although individual changes to a PRA model may be considered PRA maintenances, the integrated nature of several changes may make a peer review desirable. Multiple PRA maintenances can, over time, lead to considerable change in the insights (e.g., importance rankings, relative risk significance of SSCs). Multiple parties might perform maintenance activities over an extended period, and a periodic peer review could serve as a process to assess PRA maintenance consistency and integration of the changes to the model. Thus, a peer review might be prudent.

(c) The definition of “PRA upgrade” satisfies one of three criteria:

- (1) new methodology
- (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences
- (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences

A change made to correct a model error or to enhance completeness may or may not be a PRA upgrade. If the correction of an omission leads to a change in scope or capability that impacts the significant accident sequences or the significant accident progression sequences, it would qualify as an upgrade. However, it is not, for example, a PRA upgrade if an error or omission is addressed by using the existing methodology, and the change does not result in a significant change in risk-estimation capability. It is expected that such changes would generally be classified as PRA maintenance because, in most cases, the method of correction would be similar to that used for typical PRA maintenance where some new plant feature or change in operation is incorporated using the existing model structure and methods. In such cases, however, performance of a focused peer review might be advisable if the changes to the model were significant to an application even if they did not lead to significant changes to the base PRA risk insights.

(d) In the context of the above guidance, consideration should be given to the number of model errors being corrected. If they demonstrate a lack of understanding of the methodology being used, the change should be classified as a PRA upgrade. A focused peer review would be warranted.

These and other changes that may be difficult to classify should be treated on a case-by-case basis recognizing the basic purpose of a peer review (1-6.1), the

Table 1-A.3-1 Example Numbers for PRA Change Topics

PRA Change Topic	Example Numbers	
	PRA Maintenance	PRA Upgrade
Initiating events	1, 2, 3	4, 5
Model logic	6, 7, 8, 9, 10, 11	12, 13, 14
LOCA	15	16
Success criteria	9, 17, 18	...
System model	6, 9, 11, 19	5
Software	11	12
Human error	20, 21, 22, 23	24, 33
Common cause	25, 26	27
Data	2, 3, 26	4, 27
Quantification	11, 28	5, 12
LERF	15	...
Seismic	29, 30, 32, 34, 35	31, 33
High winds	37	36

rationale behind the criteria for classification, the existence and validity of internal reviews, and the option of deferring the peer review in certain instances until a peer review is scheduled for other reasons. The next section will list examples of PRA model changes, give a recommended classification with respect to need for peer review, and present some discussion on the choice.

1-A.3 CLASSIFICATION OF EXAMPLE PRA CHANGES

This Section contains 37 typical PRA changes and classifies each change as a PRA maintenance or PRA upgrade. The examples are realistic and most represent past changes made by specific utilities, but the list is not complete. For each PRA change, the following information is given:

(a) *Change*: brief description of the PRA change and its basis

(b) *Classification*: definition of the change as either a PRA maintenance or PRA upgrade

(c) *Rationale*: brief description of the basis for the classification and the advisability of peer review

(d) *Discussion and/or Alternative Recommendation (Optional)*: further discussion and/or an alternate recommendation of whether a peer review is appropriate

In the examples, when the classification is clearly *PRA maintenance*, the implication regarding peer review is that it is not required solely as a result of the changes in the example. When the classification is clearly *PRA upgrade*, the implication regarding peer review is that it is required as a result of the changes in the example. When a peer review is required, it may be a focused peer review [pursuant to 1-6.2.4(d)] depending on the extent of the change. When the classification involves interpretation of the criteria in the definitions of *PRA maintenance* or *PRA upgrade*, reference is made to one of

the Guidelines in 1-A.2 to support the rationale for the classification and the recommendation regarding whether or not to perform a peer review. Table 1-A.3-1 relates a PRA change topic to an example number. Note that the same example may be cited for more than one topic.

1-A.3.1 Example 1

1-A.3.1.1 Change. A few initiating events are added to the model as a result of initial peer review comments. No new methodology is required to implement them.

1-A.3.1.2 Classification. PRA maintenance.

1-A.3.1.3 Rationale. If the change does not have significant impact on risk insights, it would fall into the category of completeness, discussed in 1-A.2(c). The increased capability gained by this change would not be considered significant, since the new initiators represent only a modest increase in the total number of initiators, and the impact on the risk insights is not significant. The determination for this example is further reinforced by the fact that the change was recommended by the initial peer review so that the initiator completeness issue was apparently covered in that review.

1-A.3.1.4 Discussion and/or Alternative Recommendation. If the change has a significant impact on risk insights, a focused peer review would be appropriate.

1-A.3.2 Example 2

1-A.3.2.1 Change. A change of initiating event frequencies caused by incorporating plant data by using Bayesian update method that had been previously used.

1-A.3.2.2 Classification. PRA maintenance.

1-A.3.2.3 Rationale. This change reflects new information on plant performance (new data) and thus conforms to the definition of PRA maintenance.

1-A.3.3 Example 3

1-A.3.3.1 Change. A change of initiating event frequencies caused by the use of a more relevant generic database. No new methodology is employed.

1-A.3.3.2 Classification. PRA maintenance.

1-A.3.3.3 Rationale. The analysis requirement to perform the change is very similar to Example 2; the principal difference is the need to select the data set.

1-A.3.4 Example 4

1-A.3.4.1 Change. A change of initiating event frequencies caused by using a Bayesian update method for the first time.

1-A.3.4.2 Classification. PRA upgrade.

1-A.3.4.3 Rationale. This change involves introduction of a new methodology, so it meets criterion (a) in the guidance of 1-A.2(c).

1-A.3.5 Example 5

1-A.3.5.1 Change. Plant-specific fault trees are developed to model support system initiators and to quantify their frequency, replacing previous point estimate values.

1-A.3.5.2 Classification. PRA upgrade.

1-A.3.5.3 Rationale. This change is judged to constitute a change in capability that impacts the significant accident sequences or the significant accident progression sequences, since the model now captures explicit impact of support system SSCs on initiating events and introduces a new approach to quantification.

1-A.3.6 Example 6

1-A.3.6.1 Change. Logic errors in some system analyses are corrected.

1-A.3.6.2 Classification. PRA maintenance.

1-A.3.6.3 Rationale. This change is due to the correction of a model error, discussed in 1-A.2(c).

1-A.3.6.4 Discussion and/or Alternative Recommendation. If the changes were so large and/or numerous that they resulted in significant changes in the risk insights, the change should be considered a significant change in capability that impacts the significant accident sequences or the significant accident progression sequences and classified as a PRA upgrade due to the need for review of potential impacts throughout the model such as new cutsets of significance [see 1-A.2(c) and 1-A.2(d)].

1-A.3.7 Example 7

1-A.3.7.1 Change. Diesel dependence on HVAC added as a new dependency.

1-A.3.7.2 Classification. PRA maintenance

1-A.3.7.3 Rationale. This change involves correction a model error or omission, discussed in 1-A.2(c).

1-A.3.8 Example 8 (BWR Only)

1-A.3.8.1 Change. Credit for Control Rod Drive hydraulics as an injection source is added to the model based on new thermal-hydraulic calculations, using the same computer code.

1-A.3.8.2 Classification. PRA maintenance.

1-A.3.8.3 Rationale. Assuming that the same modeling techniques are used as for other injection sources, this change falls into the category of completeness, discussed in 1-A.2(c).

1-A.3.8.4 Discussion and/or Alternative Recommendation. If different modeling techniques are used from those of other injection sources, the change should be classified as PRA upgrade. Similarly, if a different computer code with significant changes in capability is used, this change should also be classified as an upgrade.

1-A.3.9 Example 9

1-A.3.9.1 Change. Added RHR strainer (BWR) or sump strainer (PWR) plugging event as potential LOCA consequence.

1-A.3.9.2 Classification. PRA maintenance.

1-A.3.9.3 Rationale. This change corrects an omission or reflects new knowledge.

1-A.3.9.4 Discussion and/or Alternative Recommendation. However, due to the common cause aspect of this new failure mechanism, the documentation should include evidence of a thorough internal review of the expected sequences involving loss of multiple injection sources and their quantitative impact.

1-A.3.10 Example 10 (BWR Only)

1-A.3.10.1 Change. Model logic is revised to take injection credit for control rod drive hydraulics early in the event as well as for the later low decay heat times. Justification for the change is based on new thermal-hydraulic calculations, using the same computer code that was used for the prior calculations.

1-A.3.10.2 Classification. PRA maintenance.

1-A.3.10.3 Rationale. The change falls under the category of completeness, discussed in 1-A.2(c).

1-A.3.10.4 Discussion and/or Alternative Recommendation. Documentation of the aspects of the thermal-hydraulic calculations that allowed this implied change in success criteria should be provided. If a different computer code with significant changes in capability is used, this change should be considered to be an upgrade.

1-A.3.11 Example 11

1-A.3.11.1 Change. Changed from one fault tree linking code to another (e.g., SETS code to CAFTA or WinNUPRA) for quantification of sequences.

1-A.3.11.2 Classification. PRA maintenance.

1-A.3.11.3 Rationale. Since the PRA methodology is essentially the same, this change would not be an upgrade providing the following stipulations are met:

(a) Both old and new codes use same model (e.g., linked fault tree).

(b) The new code is well documented and is generally accepted by the PRA community.

(c) The change documentation includes meaningful results comparisons and disposition of differences between the old and new codes.

1-A.3.11.4 Discussion and/or Alternative Recommendation. This issue involves a significant effort on transformation and transmitting data and models between the two computer codes. Since there is a high potential for introducing mistakes, the documentation should provide evidence of a thorough internal review.

1-A.3.12 Example 12

1-A.3.12.1 Change. Event tree with boundary conditions (linked event tree) model (e.g., using RISKMAN software) is replaced by linked fault tree model (e.g., using CAFTA software).

1-A.3.12.2 Classification. PRA upgrade.

1-A.3.12.3 Rationale. This change would involve a major modification to model logic and constitutes a new approach to quantification, a specific example for PRA upgrade.

1-A.3.12.4 Discussion and/or Alternative Recommendation. Contrast this example to Example 11, in which software but not model logic is changed.

1-A.3.13 Example 13

1-A.3.13.1 Change. Revised modeling of Station Blackout. The Loss of Offsite Power event tree is now incorporated into the Transient Event Tree, and recoveries are now handled by fault tree logic rather than by post quantification techniques.

1-A.3.13.2 Classification. PRA upgrade.

1-A.3.13.3 Rationale. This change represents a fairly extensive model structure/logic change, which falls into the spirit of changes in capability that impact the significant accident sequences or the significant accident progression sequences and new approaches to quantification. These changes can be complex and merit suitable scrutiny.

1-A.3.14 Example 14

1-A.3.14.1 Change. Model logic is modified to accommodate a loss of offsite power induced by the scram that follows a typical initiating event.

1-A.3.14.2 Classification. PRA maintenance.

1-A.3.14.3 Rationale. This classification assumes that the change results in a small change in risk insights.

1-A.3.14.4 Discussion and/or Alternative Recommendation. If the change results in significant changes in risk insights, it may be prudent to perform a peer review prior to use of the changed model for a risk-informed submittal, pursuant to 1-A.2(a).

1-A.3.15 Example 15

1-A.3.15.1 Change. A model of losses of coolant outside of containment using a single initiating event

to represent the sum of all contributors and assuming the most conservative consequences is replaced by several initiating events with individualized consequences.

1-A.3.15.2 Classification. PRA maintenance.

1-A.3.15.3 Rationale. While this change is fairly extensive in terms of the number of initiator locations involved, the modeling is straightforward and does not involve any new methodology beyond the separation of initiators described above. As long as the quantified impact on CDF is small, it falls into the category of model error correction, discussed in 1-A.2(c).

1-A.3.16 Example 16

1-A.3.16.1 Change. Replacement of generic LOCA initiating event frequencies with plant-specific LOCA frequencies assigned by using the EPRI pipe segment approach.

1-A.3.16.2 Classification. PRA upgrade.

1-A.3.16.3 Rationale. This change represents an introduction of new methodology.

1-A.3.17 Example 17

1-A.3.17.1 Change. Times to core damage slightly changed based on new thermal-hydraulic calculations, using the same computer code that was used for the prior calculations.

1-A.3.17.2 Classification. PRA maintenance.

1-A.3.17.3 Rationale. While not falling specifically within the definition of “PRA maintenance,” this change is simple in concept and constitutes neither new methodology nor significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences.

1-A.3.17.4 Discussion and/or Alternative Recommendation. This classification is predicated on the changes being small. If they have an impact on system success criteria and accident sequences are changed, a focused peer review may be prudent.

1-A.3.18 Example 18

1-A.3.18.1 Change. Definition of core damage used to support success criteria is changed from one to another accepted definition (e.g. 2,200°F instead of two-thirds of core height) without changing the thermal-hydraulic methodology.

1-A.3.18.2 Classification. PRA maintenance.

1-A.3.18.3 Rationale. While not falling specifically within the definition of PRA maintenance, the change is simple in concept and involves the choice of one from several accepted core damage criteria and thus does not constitute a new methodology.

1-A.3.18.4 Discussion and/or Alternative Recommendation. If this change leads to a significant change in risk insights, a focused peer review would be prudent.

1-A.3.19 Example 19

1-A.3.19.1 Change. Unavailability values for a number of mitigation systems are significantly increased due to the introduction of an aggressive on-line preventative maintenance program.

1-A.3.19.2 Classification. PRA maintenance.

1-A.3.19.3 Rationale. This change is clearly due to a plant change and does not involve new methodology. Documentation should include examination of the validity and accuracy of any significant new cutsets (sequences) that may emerge due to the increase in CDF.

1-A.3.19.4 Discussion and/or Alternative Recommendation. The increased unavailabilities could result in significant changes in frequencies of some cutsets and importance measures. It may be prudent to perform a peer review for such a case prior to use of the changed model for a risk-informed submittal, if there were specific SSCs important to the submittal whose risk importances are thus affected [see discussion in 1-A.2(a)].

1-A.3.20 Example 20

1-A.3.20.1 Change. To improve the modeling of operator/system interactions, several new human failure events have been added to the model and several others combined or eliminated. The HRA methodology already employed in the model is used.

1-A.3.20.2 Classification. PRA maintenance.

1-A.3.20.3 Rationale. If there is no significant impact on risk insights, this change falls into the category of enhancing completeness and thus should be treated as PRA maintenance, as discussed in 1-A.2(c).

1-A.3.20.4 Discussion and/or Alternative Recommendation. If there is a significant impact on risk insights, a focused peer review is appropriate.

1-A.3.21 Example 21

1-A.3.21.1 Change. All human actions are now processed by the ASEP method. Previously, only the important ones utilized ASEP while the remainder used conservative screening values.

1-A.3.21.2 Classification. PRA maintenance.

1-A.3.21.3 Rationale. If there is no significant impact on risk insights, this change falls into the category of enhancing completeness and thus should be treated as PRA maintenance, as discussed in 1-A.2(c).

1-A.3.22 Example 22

1-A.3.22.1 Change. Model change reflects extensive changes of the plant procedures dealing with shared

diesel response to loss of off-site power initiators for a multi-unit site. Corresponding extensive changes made to human error analyses using the methodology already employed in the model.

1-A.3.22.2 Classification. PRA maintenance.

1-A.3.22.3 Rationale. Change due solely to plant procedure change. No new methods are incorporated.

1-A.3.22.4 Discussion and/or Alternative Recommendation. Though no new methods involved, changes are extensive and could result in significant impact on component importance. A user may want to include this change in a subsequent peer review scheduled for another reason [see 1-A.2(a)].

1-A.3.23 Example 23

1-A.3.23.1 Change. Human error probabilities are modified because a reactor power uprate impacts sequence timing. The same HRA method is used to develop the new probabilities.

1-A.3.23.2 Classification. PRA maintenance.

1-A.3.23.3 Rationale. This change is due to plant changes and does not involve new methodology.

1-A.3.23.4 Discussion and/or Alternative Recommendation. If there is a significant impact on the risk insights, a focused peer review is advisable. The documentation should include the relevant information that leads to the new timing as well as its impact on human error probabilities.

1-A.3.24 Example 24

1-A.3.24.1 Change. A different HRA approach to human error analysis is employed.

1-A.3.24.2 Classification. PRA upgrade

1-A.3.24.3 Rationale. This change is a cited example in the definition of PRA upgrade. The classification applies whether the different HRA approach is applied to all human failure events or a subset thereof.

1-A.3.25 Example 25

1-A.3.25.1 Change. Added common cause failure for several components by using the existing common cause failure methodology.

1-A.3.25.2 Classification. PRA maintenance.

1-A.3.25.3 Rationale. This change enhances completeness discussed in 1-A.2(c).

1-A.3.25.4 Discussion and/or Alternative Recommendation. If new common cause failure methodology is employed, the change would be classified as a PRA upgrade.

1-A.3.26 Example 26

1-A.3.26.1 Change. Common cause MGL data is changed to use NUREG/CR-5497 [1-A-1] as a result of a recommendation from a peer review.

1-A.3.26.2 Classification. PRA maintenance.

1-A.3.26.3 Rationale. This change does not involve new data update methods, which is neither an example of new methodology, nor a change in scope or capability, which are criteria for a PRA upgrade. Moreover, the need for the change was identified by the peer review.

1-A.3.27 Example 27

1-A.3.27.1 Change. The beta-factor common cause method has been replaced by the alpha-factor technique.

1-A.3.27.2 Classification. PRA upgrade.

1-A.3.27.3 Rationale. This is a fairly extensive model change, involving a number of manipulations and logic revisions as well as a new data set, and constitutes a new treatment of common cause failure, which is a specific example in the definition of PRA upgrade.

1-A.3.28 Example 28

1-A.3.28.1 Change. As a result of concerns raised by the peer review, truncation limit is lowered by an order of magnitude (or equivalent for sequence cutoff value for linked event tree models).

1-A.3.28.2 Classification. PRA maintenance.

1-A.3.28.3 Rationale. While the definition of PRA upgrade speaks of new approaches to truncation, changing the truncation limit for a given and accepted truncation process is simple and does not constitute a change to the process and thus does not require peer review. However, the documentation should include evidence of the adequacy of the limit chosen and the results of the new cutsets (sequences).

1-A.3.28.4 Discussion and/or Alternative Recommendation. The discussion given in 1-A.2(a) may indicate a peer review if the results of the change appear significant, introducing many new important cutsets (sequences) and significantly affecting importance measures at issue for a pending risk-informed application.

1-A.3.29 Example 29

1-A.3.29.1 Change. A new seismic source zone is discovered in the vicinity of the plant, requiring a revision to the seismic hazard at the plant site.

1-A.3.29.2 Classification. PRA maintenance.

1-A.3.29.3 Rationale. The method of determining the seismic hazard is not changing, nor is the definition of the seismically induced initiating event or its use in the plant model. The incorporation of a new piece of

information into the seismic hazard model does not constitute an upgrade.

1-A.3.29.4 Discussion and/or Alternative Recommendation. Because this change is PRA maintenance, a peer review in accordance with this Standard and NEI 05-04 [1-A-2] is not required. However, seismic hazard analysis, even to simply add a new seismic source zone, requires the use of an expert elicitation process. Such processes generally comply with the Standard by following the guidance of the Senior Seismic Hazard Advisory Committee (SSHAC; NUREG/CR-6372 [1-A-3]). SSHAC emphasizes the importance of peer review in the seismic hazard development process (NUREG/CR-6372 [1-A-3], Section 3.4), and a change to the seismic hazard without a peer review could be considered to not comply with this Standard. Again, in this case, the peer review is required because the seismic hazard methodology uses expert elicitation, requiring interpretation of the meaning and significance of the new information rather than just a straightforward mathematical incorporation of the new information. So, while this PRA maintenance activity will not require a peer review that meets the peer review requirements of this Standard, it will require a peer review that needs the guidance in the SSHAC report. Such a peer review can be focused solely on the process of interpretation of the new seismic source zone, and need not consider any other parts of the PRA.

1-A.3.30 Example 30

1-A.3.30.1 Change. A new demineralized storage tank is installed to supply a back-up water supply to the AFW pumps. A fragility analysis is required for incorporation of this tank into the SPRA model.

1-A.3.30.2 Classification. PRA maintenance.

1-A.3.30.3 Rationale. The method for calculating the fragility of tanks has already been peer reviewed and has been applied to other tanks at the plant. The addition of a new tank to be analyzed by using the same method does not require a peer review.

1-A.3.31 Example 31

1-A.3.31.1 Change. The original PRA determined fragilities by calculating CDFM HCLPF levels and by applying a generic composite uncertainty factor to establish the fragility curves. New fragilities will be calculated by using median failure accelerations and specific separate factors for aleatory and epistemic uncertainties.

1-A.3.31.2 Classification. PRA upgrade.

1-A.3.31.3 Rationale. This change is a fundamentally different approach to the development of fragility curves.

1-A.3.32 Example 32

1-A.3.32.1 Change. The original PRA used two methods for calculating component fragilities. For some components, they were calculated from CDFM HCLPF levels and by applying a generic composite uncertainty factor. For other components, fragilities were calculated by using median failure accelerations and specific separate factors for aleatory and epistemic uncertainties. For some of the former components, it is proposed to change the fragility calculation to the other method.

1-A.3.32.2 Classification. PRA maintenance.

1-A.3.32.3 Rationale. The application of both methods has been peer reviewed; thus, a new methodology is not being incorporated.

1-A.3.32.4 Discussion and/or Alternative Recommendation. If the type of components, whose fragility calculation is being changed, is radically different from those to which the method has been previously applied, a focused peer review may be prudent.

1-A.3.33 Example 33

1-A.3.33.1 Change. The original SPRA adjusted HEP values by applying a single performance-shaping-factor multiplier to all HFEs for actions taken in the first 30 min after the earthquake, to account for operator confusion. These adjusted HEPs were used for all earthquakes that exceeded the OBE. This approach is to be changed to one that develops HEP “fragility curves” for the actions in the first 30 min that relates the performance shaping factor (and hence the HEP) to the size of the earthquake, to account for lower levels of operator confusion for smaller earthquakes and higher levels for larger earthquakes.

1-A.3.33.2 Classification. PRA upgrade.

1-A.3.33.3 Rationale. This is a significant change to the HRA methodology and pushes the approach to the edge of current practice.

1-A.3.34 Example 34

1-A.3.34.1 Change. SPSPA event trees are changed to move building failures from the individual system fault trees to be their own top events at the beginning of each event tree.

1-A.3.34.2 Classification. PRA maintenance.

1-A.3.34.3 Rationale. This change neither affects the Boolean logic of the overall model nor the seismic risk profile; it simply serves to highlight building failure at the event sequence level.

1-A.3.35 Example 35

1-A.3.35.1 Change. A seismic walkdown is to be conducted at the plant to confirm that there have been

no significant changes in SSC capacity since the previous walkdown 10 yr earlier. Any findings of change will be incorporated into the fragility analyses.

1-A.3.35.2 Classification. PRA maintenance.

1-A.3.35.3 Rationale. There are no changes in the approach to the walkdown, the screening level used in the walkdown, or the use of the findings from the walkdown. The addition of any new findings into the analysis does not require a peer review.

1-A.3.35.4 Discussion and/or Alternative Recommendation. Although unlikely, it is possible that the addition of previously screened SSCs to the model or significant degradation of SSCs already incorporated in the model could cause the seismic risk profile to also experience a significant change. In this case, the conduct of a peer review would be prudent.

1-A.3.36 Example 36

1-A.3.36.1 Change. Hurricane events and associated accident sequences and structure/equipment basic events are to be added to a PRA that currently does not have them. A hurricane-specific risk profile will be calculated.

1-A.3.36.2 Classification. PRA upgrade.

1-A.3.36.3 Rationale. This is a significant change in the scope of the PRA.

1-A.3.36.4 Discussion and/or Alternative Recommendation. This example refers to the addition of hurricane events into a PRA, including consideration of wind damage to systems and structures, as well as consideration of damage caused by missiles. This addition would require development of a hurricane hazard curve, missile distribution and velocity, capacity screening for both wind and missile forces, and fragility analysis for both wind and missile forces. Since none of these activities was

done in the existing PRA, a peer review would be required.

1-A.3.37 Example 37

1-A.3.37.1 Change. In an existing PRA, the LOOP analysis used plant specific-data for the initiating event frequency, but a generic recovery curve. This curve is to be updated for long-duration outages by incorporating plant-specific data on hurricane-induced LOOP.

1-A.3.37.2 Classification. PRA maintenance

1-A.3.37.3 Rationale. Plant-specific data analysis is an integral part of the PRA and was used for many issues (e.g., initiating event frequencies, failure data, and maintenance unavailability). This analysis included hurricane-induced LOOP for the initiating event frequency, which has previously been peer reviewed. The expansion of this approach to update the LOOP recovery curve by using plant-specific experience is a minor change that does not require a peer review.

1-A.4 REFERENCES

[1-A-1] NUREG/CR-5497, Common-Cause Failure Parameter Estimations; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852

[1-A-2] NEI 05-04, Process for Performing Follow-on PRA Peer Reviews Using the ASME PRA Standard (Internal Events), Revision 1, 2007

[1-A-3] NUREG/CR-6372, Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on the Use of Experts; R. J. Budnitz, G. Apostolakis, D. M. Boore, L. S. Cluff, K. J. Coppersmith, C. A. Cornell, and P. A. Morris; U.S. NRC and Lawrence Livermore National Laboratory, 1997; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852

(a)

PART 2

REQUIREMENTS FOR INTERNAL EVENTS AT-POWER PRA

Section 2-1

Overview of Internal Events At-Power PRA Requirements At-Power

2-1.1 PRA SCOPE

This Part establishes technical requirements for a Level 1 and large early release frequency (LERF) analysis of internal events (excluding floods and fires within the plant) while at-power. Consistent with the definitions in 1-1.2, internal floods and internal fires are considered separately, as described in Parts 3 and 4, respectively.

2-1.2 COORDINATION WITH OTHER PARTS OF THIS STANDARD

This Part is intended to be used together with Part 1 of this Standard. In addition, many of the technical requirements in Part 2 are fundamental requirements for performing a PRA for any hazard group, and are therefore relevant to Parts 3 through 9 of this Standard. They are incorporated by reference in those requirements that address the development of the plant response to the damage states created by the hazard groups addressed in Parts 3 through 9. Their specific allocation to Part 2 is partially a historical artifact of the way this PRA Standard was developed, with the

at-power internal events (including internal floods) requirements being developed first, and those of the remaining hazard groups being developed later. However, it is also a reflection of the fact that a fundamental understanding of the plant response to a reasonably complete set of initiating events (as defined in 1-2.2) provides the foundation for modeling the impact of various hazards described in Part 3 (Internal Flood), Part 4 (Internal Fire), Part 5 (Seismic Events), Part 7 (High Winds), Part 8 (External Floods), and Part 9 (Other External Hazards). Hence, even though Part 2 is given a title associated with the internal events hazard group it is understood that the requirements in this Part are applicable to all the hazard groups within the scope of the PRA.

2-1.3 INTERNAL EVENTS SCOPE

The scope of internal events covered in this Part includes those events originating within the plant boundary. However, internal floods are covered in Part 3, and fires within the plant in Part 4, and loss of offsite power, by convention, is considered an internal event.

Section 2-2

Internal Events PRA Technical Elements and Requirements

The requirements of this Part, which are organized by eight technical elements that compose a Level 1/ LERF PRA for internal events (excluding internal fire) at-power (and their abbreviations), are as follows:

- (a) Initiating Event Analysis (IE)
- (b) Accident Sequence Analysis (AS)
- (c) Success Criteria (SC)
- (d) Systems Analysis (SY)
- (e) Human Reliability Analysis (HR)
- (f) Data Analysis (DA)
- (g) Quantification (QU)
- (h) LERF Analysis (LE)

Tables of HLRs and SRs for the eight PRA elements are provided in 2-2.1 through 2-2.8. The SRs are numbered and labeled to identify the HLR that is supported. For each Capability Category, the SRs define the minimum requirements necessary to meet that Capability Category. In these tables, some action statements apply to only one Capability Category and some extend across two or three Capability Categories. When an action spans multiple Capability Categories, it applies equally to each Capability Category. When necessary, the differentiation between Capability Categories is made in other associated SRs; two examples are stated below. The interpretation of a Supporting Requirement whose action statement spans multiple Capability Categories is stated in Table 1-1.3-2. It should be noted that some action statements span Capability Categories II and III because the authors were unable to specify a distinguishing requirement for Capability Category III

at this time. It is intended that, by meeting all the SRs under a given HLR, a PRA will meet that HLR.

Examples of how the requirements for Capability Categories are differentiated:

IE-A2 requires initiating events and event categories to be identified that can challenge the plant. The scope of identifying the events should be the same for all Capability Categories. However, the treatment of the identified events does vary in scope and detail between Capability Categories as seen in AS-A9.

HR-F1 is a general action statement about the way a human failure event is included in the PRA model, while HR-F2 distinguishes different levels of analysis for the subsequent quantification.

Boldface is used to highlight the differences among the requirements in the three Capability Categories.

2-2.1 INITIATING EVENT ANALYSIS (IE)

2-2.1.1 Objectives

The objectives of the initiating event analysis are to identify and quantify events that could lead to core damage in such a way that

(a) events that challenge normal plant operation and that require successful mitigation to prevent core damage are included

(b) initiating events are grouped according to the mitigation requirements to facilitate the efficient modeling of plant response

(c) frequencies of the initiating event groups are quantified

Table 2-2.1-1 High Level Requirements for Internal Initiating Event Analysis (IE)

Designator	Requirement
HLR-IE-A	The initiating event analysis shall provide a reasonably complete identification of initiating events.
HLR-IE-B	The initiating event analysis shall group the initiating events so that events in the same group have similar mitigation requirements (i.e., the requirements for most events in the group are less restrictive than the limiting mitigation requirements for the group) to facilitate an efficient but realistic estimation of CDF.
HLR-IE-C	The initiating event analysis shall estimate the annual frequency of each initiating event or initiating event group.
HLR-IE-D	Documentation of the initiating event analysis shall be consistent with the applicable supporting requirements.

Table 2-2.1-2(a) Supporting Requirements for HLR-IE-A

The initiating event analysis shall provide a reasonably complete identification of initiating events (HLR-IE-A).

Index No. IE-A	Capability Category I	Capability Category II	Capability Category III
IE-A1	IDENTIFY those initiating events that challenge normal plant operation and that require successful mitigation to prevent core damage using a structured, systematic process for identifying initiating events that accounts for plant-specific features. For example, such a systematic approach may employ master logic diagrams, heat balance fault trees, or failure modes and effects analysis (FMEA). Existing lists of known initiators are also commonly employed as a starting point.		
IE-A2	INCLUDE in the spectrum of internal-event challenges considered at least the following general categories: (a) <i>Transients</i>. INCLUDE among the transients both equipment and human-induced events that disrupt the plant and leave the primary system pressure boundary intact. (b) <i>LOCAs</i>. INCLUDE in the LOCA category both equipment and human-induced events that disrupt the plant by causing a breach in the core coolant system with a resulting loss of core coolant inventory. DIFFERENTIATE the LOCA initiators, using a defined rationale for the differentiation. Examples of LOCA types include (1) <i>Small LOCAs</i>. Examples: reactor coolant pump seal LOCAs, small pipe breaks (2) <i>Medium LOCAs</i>. Examples: stuck open safety or relief valves (3) <i>Large LOCAs</i>. Examples: inadvertent ADS, component ruptures (4) <i>Excessive LOCAs</i> (LOCAs that cannot be mitigated by any combination of engineered systems). Example: reactor pressure vessel rupture (5) <i>LOCAs Outside Containment</i>. Example: primary system pipe breaks outside containment (BWRs). (c) <i>SGTRs</i>. INCLUDE spontaneous rupture of a steam generator tube (PWRs). (d) <i>ISLOCAs</i>. INCLUDE postulated events in systems interfacing with the reactor coolant system that could fail or be operated in such a manner as to result in an uncontrolled loss of core coolant outside the containment [e.g., interfacing systems LOCAs (ISLOCAs)]. (e) Special initiators (e.g., support systems failures, instrument line breaks) [Note (1)].		
IE-A3	REVIEW the plant-specific initiating event experience of all initiators to ensure that the list of challenges accounts for plant experience. See also IE-A7.		
IE-A4	REVIEW generic analyses of similar plants to assess whether the list of challenges included in the model accounts for industry experience.	REVIEW generic analyses and operating experience of similar plants to assess whether the list of challenges included in the model accounts for industry experience.	