ASME/ANS RA-Sb–2013

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

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ASME/ANS RA-Sb–2013

Following approval by the ASME/ANS RA-S Committee and ASME, and after public review, ASME/ANS RA-Sb–2013 was approved by the American National Standards Institute on July 1, 2013.

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 second 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, **(b)**, placed next to the affected area.

Page	*Location*	*Change*
iii, iii.1	Contents	Updated to reflect Addenda
iv	Foreword	Revised
v	Preparation on Technical Inquiries to the Committee on Nuclear Risk Management	ASME address updated
vi–vii.1	Roster	Updated
viii	Preface	Deleted
1	Part 1	*(1)* Sections 1-1, 1-2, 1-6, and 1-7, and Nonmandatory Appendix 1-A revised *(2)* Paragraph 1-4.2 corrected by errata *(3)* Last two paragraphs of paras. 1-1.3.3 and 1-3.6.2 inserted by errata
45	Part 2	Section 2-1 title, Sections 2-2 and 2-4, and para. 2-3.2 revised
123	Part 3	Revised in its entirety
143	Part 4	Sections 4-1, 4-2, and 4-4, and paras. 4-3.2 and 4-3.3 revised
228	Part 5	Revised in its entirety
273	Part 6	Revised in its entirety
287	Part 7	Part 7 title, Section 7-1 title, and Sections 7-2 and 7-3 revised
301	Part 8	Section 8-1 title and Sections 8-2 and 8-3 revised
316	Part 9	Revised in its entirety
330	Part 10	Paragraph 10-1.3 and Sections 10-2, 10-3, and 10-4 revised

SPECIAL NOTE:

The Interpretations to ASME/ANS RA-S, Volume 3, are included in this addenda beginning with page I-7 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), in which the following additional organizations participate: Brookhaven National Laboratory (BNL), Boiling Water Reactor Owners Group (BWROG), Electric Power Research Institute (EPRI), Institute of Electrical and Electronics Engineers (IEEE), National Aeronautics and Space Administration (NASA), Nuclear Energy Institute (NEI), Pressurized Water Reactor Owners Group (PWROG), U.S. Department of Energy (DoE), and U.S. Nuclear Regulatory Commission (USNRC). This committee was chartered to coordinate and harmonize standards activities related to probabilistic risk assessment (PRA) for nuclear power plants and other nuclear installations among all interested standards development organizations and other interested parties. Implementing a proposal by NRMCC, ASME and ANS formed a Joint Committee on Nuclear Risk Management (JCNRM) to develop and maintain PRA standards. The JCNRM operates under procedures accredited by the American National Standards Institute (ANSI) as meeting the criteria of consensus procedures for American National Standards.

In 2002, ASME issued an initial PRA standard whose scope was Level 1 and large early release frequency (LERF) for internal events at-power for light-water-reactor (LWR) nuclear power plants. In 2003 and 2007, the ANS issued two other PRA standards, whose scopes were external hazards and internal fires at-power for LWR nuclear power plants. In 2008, the three standards were combined into one standard, ASME/ANS RA-S–2008, under the joint auspices of ASME and ANS. A revision, ASME/ANS RA-Sa–2009 [Addenda (a)], was issued in 2009. The JCNRM came into existence after Addenda (a) was issued. This Addenda, ASME/ANS RA-Sb–2013 [Addenda (b)], is a second revision; it supersedes all previous revisions. JCNRM is responsible for ensuring that this Standard is maintained and revised, as necessary. This responsibility includes appropriate coordination with and linkage to other standards under development for related risk-informed applications.

Users of this Standard are invited to provide feedback to improve its usefulness for inclusion in the next edition of this Standard, which is currently under development and planned to be issued in 2015. The JCNRM holds two formal meetings per year and users are invited to participate. Additional information about the JCNRM can be found on its Committee Page at http:// cstools.asme.org/.

PREPARATION OF TECHNICAL INQUIRIES TO THE COMMITTEE ON NUCLEAR RISK MANAGEMENT

(b)

INTRODUCTION

The ASME Committee on Nuclear Risk Management will consider written requests for interpretations and revisions to risk management standards and development of new requirements as dictated by technological development. The Committee's activities in this latter regard are limited strictly to interpretations of the requirements, or to the consideration of revisions to the requirements on the basis of new data or technology. As a matter of published policy, ASME does not approve, certify, rate, or endorse any item, construction, proprietary device, or activity, and accordingly, inquiries requiring such consideration will be returned. Moreover, ASME does not act as a consultant on specific engineering problems or on the general application or understanding of the Standard requirements. If, based on the inquiry information submitted, it is the opinion of the Committee that the inquirer should seek assistance, the inquiry will be returned with the recommendation that such assistance be obtained. All inquiries that do not provide the information needed for the Committee's full understanding will be returned.

INQUIRY FORMAT

Inquiries shall be limited strictly to interpretations of the requirements, or to the consideration of revisions to the present requirements on the basis of new data or technology. Inquiries shall be submitted in the following format:

(a) Scope. The inquiry shall involve a single requirement or closely related requirements. An inquiry letter concerning unrelated subjects will be returned.

(b) Background. State the purpose of the inquiry, which would be either to obtain an interpretation of the Standard requirement or to propose consideration of a revision to the present requirements. Provide concisely the information needed for the Committee's understanding of the inquiry (with sketches as necessary), being sure to include references to the applicable standard edition, addenda, part, appendix, paragraph, figure, or table.

(c) Inquiry Structure. The inquiry shall be stated in a condensed and precise question format, omitting superfluous background information, and, where appropriate, composed in such a way that "yes" or "no" (perhaps with provisos) would be an acceptable reply. This inquiry statement should be technically and editorially correct.

(d) Proposed Reply. State what it is believed that the Standard requires. If, in the inquirer's opinion, a revision to the Standard is needed, recommended wording shall be provided.

(e) The inquiry shall be submitted in typewritten form; however, legible, handwritten inquiries will be considered.

(f) The inquiry shall include name, telephone number, and mailing address of the inquirer.

(g) The inquiry shall be submitted to the following address:

Secretary, Committee on Nuclear Risk Management
The American Society of Mechanical Engineers
Two Park Avenue
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ASME/ANS RA-S COMMITTEE
Standard for Level 1/LERF Probabilistic Risk Assessment for Nuclear Power Plant Applications

(The following is the roster of the Committee at the time of approval of this Standard.)

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PREFACE

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PART 1 (a)
GENERAL REQUIREMENTS FOR A LEVEL 1 PRA, INCLUDING LARGE EARLY RELEASE FREQUENCY (b)

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, terrorism). 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.

1-1.2.1 Treatment of Hazard Groups

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)
(g) Other Hazards (Part 9)

In addition to providing technical requirements for PRAs of other hazards, this Standard provides requirements for screening and conservative analyses of external hazards (Part 6), and technical requirements for seismic margin analysis (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 an 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.

1-1.2.2 Hazards and Initiating Events

In using this Standard, it is necessary to understand the relationship between hazard, hazard event, hazard group, and initiating event, which are defined in 1-2.2. "Hazard group" refers to a collection of hazards that are assessed in the PRA using a common approach, methods, and data, while a "hazard" is the specific phenomenon that puts the plant at risk. A hazard group may consist of a single hazard (e.g., internal fires or seismic events) such that the hazard group and hazard are synonymous,[1] or multiple hazards [e.g., an internal-events hazard group, which includes transients and loss of coolant accident (LOCA) hazards; or a high-wind hazard group, which includes hurricane, tornado, and straight-wind hazards]. In this context, the hazard is the phenomenon; the [hazard] event[2] is an occurrence of the phenomenon that can result in a plant trip and, in many cases, other damage. The initiating event is the specific plant perturbation that challenges plant control and safety systems.

In general, there is a range of hazard events associated with any given hazard, and, for analysis purposes, the range can be divided into bins characterized by their severity. Hazard events of different severity can result in different initiating events.

Consider the internal-events hazard group, since this group provides the fundamental understanding of plant response. As noted above, this hazard group includes several hazards, such as transients and LOCAs, which can be considered as generic types of hazards.

For transients, different transient events, such as reactor trip and loss of feedwater, can be identified in terms of the different demands they place on critical safety functions; these demands characterize the events' severity.

For LOCAs, the specific LOCA events are the large LOCA, medium LOCA, small LOCA, etc. The small LOCA leading to plant trip on low pressure or low level is the specific initiating event for the small LOCA event.

Because the internal-events hazard group serves as the fundamental basis for the plant model, the terms "[hazard] events" and "initiating events" are synonymous, and this structure forms the primary consideration for the remaining hazard groups.

For the remaining hazard groups, the terms "[hazard] event" and "initiating event" are not synonymous. Rather, a [hazard] event is identified as the cause of an initiating event by virtue of the effect it has on the plant. The assessment of the effect on the plant defines the reason for the plant trip as well as any additional failures, and provides the starting point for the analysis of the plant response. Therefore, in keeping with the definition of initiating event, for the occurrence of a given hazard event, the initiating event (or events, as more than one outcome may be possible) is (are) a perturbation to the steady-state operation of the plant that challenges plant control and safety systems whose failure could potentially lead to core damage. For example, consider the earthquake hazard group, which involves only one hazard, i.e., earthquakes are the hazard and also the hazard group. This hazard (earthquakes) can be defined in terms of a range of seismic events (e.g., 0.1g, 0.3g, 0.5g, >0.75g) and their associated spectral shapes and time histories.

(a) A manual scram may be an initiating event for the 0.1g earthquake.

(b) A loss of offsite power (LOOP) is often assumed as the initiating event for the 0.3g and 0.5g earthquakes.

(c) A LOCA may be the initiating event for very large (>0.75g) earthquakes.

These assessments would be made based on an assessment of their impact on the plant. For example, for a 0.1g seismic event, the likelihood of any physical damage resulting in an automatic trip is small; for 0.3g and 0.5g seismic events, the most likely effect may be damage to the switchyard or the transmission system; and for a >0.75g seismic event, in addition to a LOOP, there may be a significant likelihood of failure of vessel or piping anchorage. A [hazard] event can be associated with multiple initiating events (each with a conditional probability of occurrence), so that a 0.3g seismic event might result in a manual scram, a LOOP, a LOCA, or a combination of a LOOP and a LOCA, each with an associated conditional probability, which, when combined with the [hazard] event frequency, provides the corresponding initiating-event frequency.

It is even possible that a [hazard] event would not result in an initiating event (i.e., there would be no perturbation of the plant operation). For example, a plant may automatically trip (initiating event), may be manually tripped (initiating event), or may continue (no initiating event) to operate through a hurricane event. These examples highlight why the distinction between "[hazard] event" and "initiating event" is important and must be maintained.

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[1] If every individual hazard was analyzed using a different approach, method, or data, then there would be no rationale to have hazard groups. However, grouping hazards that are analyzed by using the same approach, methods, and data allows them to be analyzed in an integrated fashion and to meet each SR in a similar manner.

[2] "Hazard" is placed in brackets here since the term "hazard event" is generally not used. In practice, the word "hazard" would be replaced with the designation of the specific hazard. For example, one would generally refer to a transient event, internal flood event, seismic event, etc., to denote the specific hazards that are addressed in the PRA model.

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 Category. 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.

It is intended that, by meeting all the SRs under a given HLR, a PRA will meet that HLR. The Technical Requirements section of each respective Part of this Standard also specifies the required documentation to facilitate PRA applications, upgrades, and peer review.

The SRs specify what to do rather than how to do it, and, in that sense, specific methods for satisfying the requirements are not prescribed. Nevertheless, certain established methods were contemplated during the development of these requirements. Alternative methods and approaches for meeting the requirements of this

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 Floods	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 Hazards (internal or external)	See Note (1)	"X" Hazard Analysis (XHA) "X" Hazard Fragility Analysis (XFR) "X" Hazard Plant Response Analysis (XPR)

NOTE:
(1) For any other hazard group "X," the approach for performing a PRA for the hazard group shall meet Requirements HLR-XHA, HLR-XFR, and HLR-XPR in Part 9. Each hazard for which a unique approach is developed shall constitute its own hazard group. Hazards that share a common approach, methods, and data shall be treated as a single hazard group. Examples of such hazard groups include biological events and external fires.

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 4-2.1). 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

Standard may be used if they provide results that are equivalent or superior to the methods usually used and they meet the HLRs and SRs presented in this Standard. The use of any particular method for meeting an SR shall be documented and shall be subject to review by the peer review process described in Section 1-6. All Notes and Commentaries, which follow many SRs, are nonmandatory.

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 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

ACRS: Advisory Committee for Reactor Safeguards

ADS: automatic depressurization system

ANS: American Nuclear Society

AOPs: abnormal operating procedures

AOT: allowed outage time

ARI: alternate rod insertion

ASEP: accident sequence evaluation program

ATWS: anticipated transient without scram

BWR: boiling water reactor

CCDP: conditional core damage probability

CCF: common cause failure(s)

CCW: component cooling water

CDF: core damage frequency

CDFM: conservative deterministic failure margin

CEUS: central and eastern U.S.

CLERP: conditional large early release probability

DBE: design-basis earthquake

DC: direct current

DOE: U.S. Department of Energy

DW: drywell

ECCS: emergency core cooling system

EDG: emergency diesel generator

EOPs: emergency operating procedures

EPRI: Electric Power Research Institute

FHA: fire hazards analysis (or assessment)

FIVE: fire-induced vulnerability evaluation

FSAR: Final Safety Analysis Report

GIP: generic implementation procedure

HCLPF: high confidence of low probability of failure

HELB: high energy line break

HEP: human error probability

HFE: human failure event

HLR: High Level Requirement

HPCI: high pressure coolant injection

HRA: human reliability analysis

HVAC: heating, ventilation, and air conditioning

I&C: instrumentation and control

IE: initiating event

IPE: individual plant examination

IPEEE: individual plant examination of external events

ISLOCA: interfacing systems loss of coolant accident

LCO: limiting condition of operation

LERF: large early release frequency

LLNL: Lawrence Livermore National Laboratory

LOCA: loss of coolant accident

LOOP: loss of offsite power (also referred to as “LOSP”)

LWR: light water reactor

MCR: main control room

MMI: modified Mercalli intensity

MOV: motor operated valve

NEI: Nuclear Energy Institute

NFPA: National Fire Protection Association

NPP: nuclear power plant

NPSH: net positive suction head

NRC: Nuclear Regulatory Commission

NSSS: nuclear steam supply system

OBE: operating-basis earthquake

P&IDs: piping and instrumentation drawings (or diagrams)

PCS: power conversion system

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

SSEL: 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 [1-1])

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 nondeterministic (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: uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in offsite public health effects.

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.

design-basis hazard event: a particular hazard event having the characteristics of the hazard severity and type that are specified in the plant design basis, and against which the plant is designed. If no specific characteristics are specified in the plant design basis for a specific hazard, then there is no design-basis hazard event for that hazard. Examples include wind speed (for high winds and tornadoes); peak ground acceleration, spectral shape, and time history (for seismic); and maximum rate and duration of precipitation (for rainfall or snowfall).

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.

essential low-ruggedness relays: electromechanical relays on circuits of equipment that are modeled in the PRA accident sequences and for which relay malfunction causes an undesirable system actuation or termination.

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., <1E-6/reactor-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 [1-2]). (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 [1-3]).

fire compartment:[3] 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 EPRI TR-1011989–NUREG/CR-6850 [1-4]). 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 EPRI TR-1011989–NUREG/CR-6850 [1-4]).

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.

[3] 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 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 [1-2]).

fire-resistance rating: the time, in minutes or hours, that materials or assemblies have withstood a fire exposure as established in accordance with an approved test procedure appropriate for the structure, building material, or component under consideration (NFPA 805 [1-3]).

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.

flood area: an area within a plant that is defined for the purpose of performing an internal-flooding PRA. Flood areas are normally defined in terms of one or more of the following: building types; location within a building; and the physical barriers that delay, restrict, or prevent the propagation of floods to adjacent areas.

flood-induced accident sequence: an accident sequence that includes a flood-induced initiating event and the potential for undesired consequences, with a specified end state, e.g., core damage.

flood-induced failure mechanism: the failure mechanism of an SSC induced by a flood. Possible SSC failure mechanisms include shorting out of electrical connections,

[4] DELETED.

blockage of air intakes, and structural damage from flood loads.

flood-induced initiating event: an initiating event that is caused by a flood either directly (e.g., loss of system function caused by diversion of flow associated with the flood) or indirectly (e.g., an exigent plant shutdown caused by the loss of function of one or more flood-damaged SSCs).

flood propagation path: a physical pathway that would allow the progression of a flood and associated flood damage within and among different flood areas.

flood rate: the flow rate of water or steam across the breach or opening in the pressure boundary of the flood source during the flood event. Depending on the context, the flood rate may be a time-dependent rate, a maximum rate, or an average rate over the duration of the flood.

flood scenario: a description of an event that results in a flood-induced initiating event. The factors considered in the definition of a flood scenario include flood area; flood source; flood rate; flood propagation path; impact on plant SSCs; human actions considered in flood initiation, mitigation, and termination; and means of detection (sensors, alarms, indications, etc.).

flood source: an inventory of water or steam normally contained within a system, tank, component, reservoir, river, lake, or ocean that provides the potential for flooding-induced failure of SSCs in the event the flood source container or pressure or retention boundary is breached.

flood termination: as used in the definitions of flood scenario and flood volume, the cessation of the flood rate by isolation of the flood source or exhaustion of the flood source inventory.

flood volume: the total flood volume of water released from the source from flood initiation to termination or to a specific point in time during a flood scenario; unless specified as the localized volume in specific flood areas for scenarios that involve multiple flood areas, flood volume is normally used to calculate the nominal flood height, which is associated with the submergence failure cause. Water-spray volumes are generally different from flood volumes, but spray water may accumulate and contribute to flood volumes.

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 analysis: the process to determine an estimate of the expected frequency of exceedance (over some specified time interval) of various levels of some characteristic measure of the intensity of a hazard (e.g., peak ground acceleration to characterize ground shaking from an earthquake). The time period of interest is typically 1 yr, in which case the estimate is called the annual frequency of exceedance.

[hazard] event: an event brought about by the occurrence of the specified hazard. A hazard event is described in terms of the specific levels of severity of impact that a hazard can have on the plant. For example, an internal flood event would be expressed in terms of the specific flood source and its local impact, such as the resulting water levels in affected plant areas or the extent of the area subjected to spray; a seismic event would be expressed in terms of spectral acceleration and associated spectral shape; a transient event would be expressed in terms of the plant systems affected by the event.

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 *H*igh *C*onfidence of *L*ow *P*robability of *F*ailure 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 $A_m \exp[-1.65\beta_R + \beta_U)]$. 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 energy line: a pipe or piping system component is classified as high energy if it contains water or steam at maximum operating temperature exceeding 200°F or maximum operating pressure exceeding 275 psig.

high energy line break (HELB): a break or breach in a high energy line.

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.

human response action: a postinitiator operator action, following a cue or symptom of an event, taken to satisfy the procedural requirements for control of a function or system.

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 [1-2]).

initiating event: a perturbation to the steady-state operation of the plant that challenges plant control and safety systems whose failure could potentially lead to core damage. An initiating event is defined in terms of the change in plant status that results in a condition requiring a reactor trip (e.g., loss of main feedwater system, small LOCA), or a manual trip prompted by conditions other than those in the normal shutdown procedure when the plant is at power. An initiating event may result from human causes, equipment failure from causes internal to the plant (e.g., hardware faults, floods, or fires) or external to the plant (e.g., earthquakes or high winds), or combinations thereof.

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: a hazard group that encompasses events that result from or involve mechanical, electrical, structural, or human failures from causes originating within a nuclear power plant that directly or indirectly cause an initiating event and may cause safety system failures or operator errors that may lead to core damage. By historical convention, loss of offsite power, which may result from causes within or outside the plant, is considered an internal event (except when the loss is caused by another evaluated hazard group). Also by historical convention, internal flood and internal fire are separate hazard groups and thus not considered internal events.

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.

low-ruggedness relays: electromechanical relays that may chatter at low levels of earthquake excitation or on impact, causing malfunction of electrical circuits.

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.

passive SSC: an SSC that performs one or more safety functions either fully or partially via passive means (i.e., relying on natural physical processes such as natural convection, thermal conduction, radiation, gravity, or pressure differentials, or depending on the integrity of a pressure boundary or structural component). Examples include piping systems that are used to maintain an inventory of fluid and deliver flow along a fluid path, and structural supports for SSCs.

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 an internal flood or internal fire PRA is based. The physical analysis units are generally defined in terms of flood or fire areas and/or flood 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 response model: a logic model, including the event trees and fault trees and the various SSC and human failures, that is used to delineate and evaluate the CDF/LERF accident sequences conditional on the occurrence of a hazard event (or hazard group).

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 [1-2]).

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/reactor-yr).

reactor-operating-state-year: an equivalent calendar year of operation in a particular plant operating state. See Note (1) in Table 2-2.1-4.

reactor-year: a calendar year in the operating life of one reactor, regardless of power level. See Note (1) in Table 2-2.1-4.

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 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 0.3*g* PGA anchored to a median NUREG/CR-0098 spectrum [1-5].

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,[5] 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,[5] 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,[5] 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

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,[5] 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 1.174, 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.

[5] 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.

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 [1-2]).

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 [1-4]).

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 [1-2]). (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

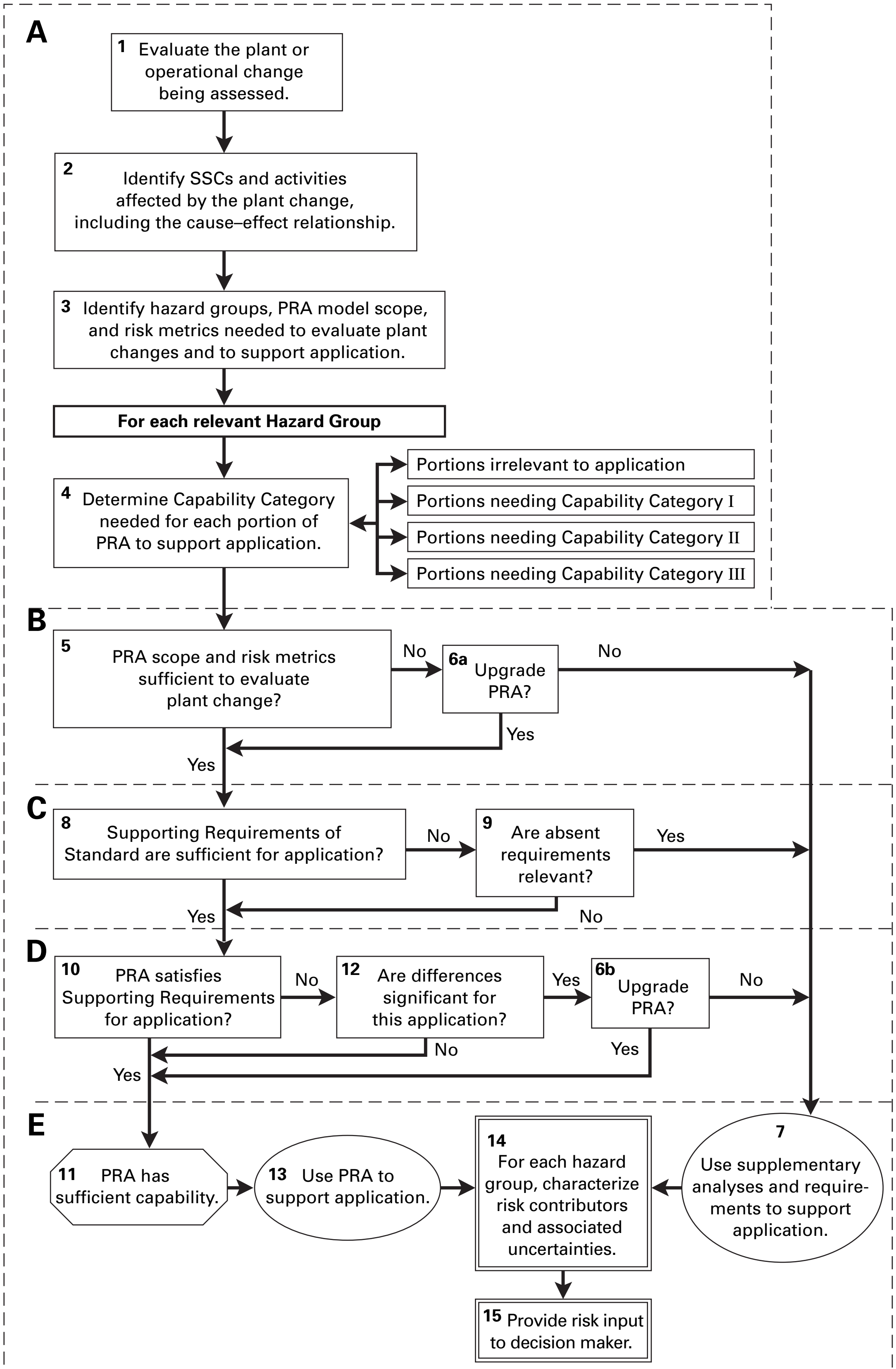
(a) evaluating the plant design or operational change being assessed (Box 1 of Fig. 1-3-1)

(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 [1-6] and NUREG-0800 [1-7] provide guidance for the above activities.

Fig. 1-3-1 Application Process Flowchart



EXAMPLE:[6] 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 2×10^{-5}/yr and 1×10^{-6}/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

[6] The examples in this version of the Standard are focused primarily on internal events. Additional examples will be added in a future edition.

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 ΔCDF 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 ASSESSING 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 [1-8], 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* [1-9]) 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.

For risk-informed applications, the terms "relevance" or "significance" can be evaluated from different perspectives. "Relevance" is related to the applicability of a hazard group. "Significance" of sequences, contributors, cutsets, etc., can be measured either by their contribution to a specific hazard group (e.g., fires within the plant) or by their contribution to the overall plant risk. When performing a baseline PRA using this Standard, addressing "significance" requires an assessment and characterization of the relative contribution of risk contributors within a given hazard group. For example, a supporting requirement in Part 4 of this Standard that identifies an action to be performed for "significant" fire zones is assessed within the context of the other fire-risk contributors only (i.e., within the fire PRA itself). However, when performing a risk-informed application, it is often more appropriate to evaluate "significance" across all relevant hazard groups. When the risk-informed application is implemented, it is necessary to determine whether it would alter baseline assumptions or plant conditions such that "significance" within a hazard group is now altered or more uncertain. The evaluation of "significance" at this level may or may not require further analysis within a hazard group.

In meeting the requirements of this Standard, those supporting requirements associated with assessing or identifying levels of significance are first performed for the baseline PRA models that are used to quantify average annual estimates of risk from all hazard groups. With regard to the applications process of this Standard (Section 1-3), the assessment or identification of significance (Box 12 of Fig. 1-3-1) is to be evaluated first across all risk contributors within the context of the change(s) being proposed by the risk-informed application, and then within each hazard group to determine if additional analysis is necessary.

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.

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 [1-10] and NUREG-1563 [1-11] 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 [1-12] and the Yucca Mountain project's study of volcanic hazards [1-13]. 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 [1-10].

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 [1-14] and the NEI-00-02 [1-15] 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).

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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 the PRA to be used in risk-informed decisions for commercial nuclear power plants. Those portions of PRAs used for applications applying this Standard shall be peer reviewed. The peer review shall assess the PRA to the extent necessary to determine if the methodology andits implementation meet the requirements of this Standard. Another purpose of the peer review is to determine the potential gaps in the PRA relative to this Standard's requirements. 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 the assessment of each applicable supporting requirement as well as on the methodologies and their implementation for each PRA Element.

1-6.1.1 Scope

Peer reviews shall be performed against the requirements in those Parts of this Standard that are applicable to the hazard groups of the PRA that are being used to support risk-informed decisions. It is permissible to conduct a separate and distinct peer review for each relevant hazard. This Standard does not require that a single peer review be integrated across all hazard groups of the PRA.

1-6.1.2 Frequency

The peer review specified in 1-6.1.1 is performed prior to using the PRA in risk-informed regulatory decisions. 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.

1-6.1.3 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

It is posssible that peer reviews will be conducted on PRAs that have adopted new methods, or for hazards that have never been (or have been infrequently) analyzed (e.g., external hazards that are frequently screened out). Peer reviews may also occur when a PRA applies new methods that have not been applied previously (e.g., the application of a newly developed HRA quantification method). If the new methods, hazards, or applications have not been separately peer reviewed, then the task of peer reviewing the technical adequacy and appropriateness of the method (rather than just its application) will fall to the PRA peer review team.

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) If a peer review team member has performed or directly supervised work on any PRA Element(s) for any hazard group evaluated in the overall PRA, they shall not participate in the peer review of that PRA Element(s) for any other hazard group in that PRA.

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 Specific Review Team Qualifications

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 relevant Part of this Standard under review and shall have demonstrated the capability to integrate these PRA Elements. When more than one Part is under review, a separate technical integrator may be used for each Part.

(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. Regardless of any such exceptions, the collective qualification of the review team shall be appropriate to the full scope of SRs within the scope of the hazard group PRA being peer reviewed.

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. These hazard-group–specific requirements are provided in the corresponding peer review section of each Part (e.g., 2-3.3, 3-3.3). The peer review team shall review the technical requirements of the hazard group 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. 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 appropriate hazard group 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).

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) a statement of the scope of the peer review

(c) the names of the peer review team members

(d) a brief resume on each team member describing the individual's employer, education, PRA training, and PRA and PRA Element experience and expertise

(e) the elements of the PRA reviewed by each team member

(f) a discussion of the extent to which each PRA Element was reviewed, including justification for any supporting requirements within the peer review scope that were not reviewed

(g) 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 and the methodology implemented, defined to a sufficient level of detail that will allow the resolution of the differences

(h) identification and significance of exceptions and gaps relative to the Standard's requirements, in sufficient

detail to allow the resolution of the gaps that the peer reviewers have determined to be material to the PRA

(i) an assessment of PRA assumptions that the peer reviewers have determined to be material to the PRA

(j) at the request of any peer reviewer, differences or dissenting views among peer reviewers

(k) recommended alternatives for resolution of any differences

(l) 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

References are cited here and in other Parts of this Standard as guides to the user. The user is cautioned that there may be more recent versions of the references, or alternative documents more pertinent to particular applications.

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[1-2] Regulatory Guide 1.189, Fire Protection for Nuclear Power Plants, Rev. 2; October 2009; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852

[1-3] NFPA Standard 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants; 2001; Publisher: National Fire Protection Association (NFPA), 1 Batterymarch Park, Quincy, MA 02169

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[1-10] 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

[1-11] NUREG/CR-1563, Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program; J. P. Kotra, M.P. Lee, N.A. Eisenberg, and A. R. DeWispelare; U.S. NRC Office of Nuclear Materials Safety and Safeguards, 1996; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852

[1-12] NRC 50-275, 50-323, Final Report of the Diablo Canyon Long Term Seismic Program, Pacific Gas and Electric Company; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852

[1-13] BA000-1717-2200-00082, Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada, U.S. Department of Energy Yucca Mountain Project, Geomatrix Consultants, Inc., 1996; Publisher: U.S. Department of Energy Yucca Mountain Project, P.O. Box 364629, North Las Vegas, NV 89036

[1-14] NUREG/CR-2300, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, 1983; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852; Publisher: U.S.

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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 if the method itself is not new and has been applied in other PRAs. 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. Section 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,[1] prudence may call for a peer review for such a case.[2] 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

[1] A change in the risk insights is considered significant when it changes the conclusions drawn from the PRA.

[2] 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 [1-17].

example, changing from a modest on-line preventative 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.

(e) When performing an internal review (i.e., a review by the PRA owner), the reviewer(s) should use as guidance those applicable requirements in the Standard.

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 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 43 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

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.

Classification. PRA maintenance.

Rationale. If the change does not have significant impact on risk insights, it would fall into the category

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
Fire	38, 39, 40, 41	42, 43

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.

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

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

Classification. PRA maintenance.

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

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

Classification. PRA maintenance.

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

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

Classification. PRA upgrade.

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

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

Classification. PRA upgrade.

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

Change. Logic errors in some system analyses are corrected.

Classification. PRA maintenance.

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

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

Change. Diesel dependence on HVAC added as a new dependency.

Classification. PRA maintenance

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

1-A.3.8 Example 8 (BWR Only)

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 in the same manner as was used for the prior calculations.

Classification. PRA maintenance.

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).

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

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

Classification. PRA maintenance.

Rationale. This change corrects an omission or reflects new knowledge.

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)

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 in the same manner as was used for the prior calculations.

Classification. PRA maintenance.

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

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

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

Classification. PRA maintenance.

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.

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

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).

Classification. PRA upgrade.

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

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

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.

Classification. PRA upgrade.

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

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

Classification. PRA maintenance.

Rationale. This classification assumes that the change results in a small change in risk insights.

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

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.

Classification. PRA maintenance.

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

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

Classification. PRA upgrade.

Rationale. This change represents an introduction of new methodology.

1-A.3.17 Example 17

Change. Times to core damage slightly changed based on new thermal-hydraulic calculations, using the same computer code in the same manner as was used for the prior calculations.

Classification. PRA maintenance.

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.

Discussion and/or Alternative Recommendation. This classification is predicated on the changes being small. If they have an impact on success criteria or if risk insights are changed, a focused peer review may be prudent.

1-A.3.18 Example 18

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.

Classification. PRA maintenance.

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.

Discussion and/or Alternative Recommendation. This classification is predicated on the changes being small. If they have an impact on success criteria or if risk insights are changed, a focused peer review may be prudent.

1-A.3.19 Example 19

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

Classification. PRA maintenance.

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.

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

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.

Classification. PRA maintenance.

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).

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

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.

Classification. PRA maintenance.

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

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.

Classification. PRA maintenance.

Rationale. Change due solely to plant procedure change. No new methods are incorporated.

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

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.

Classification. PRA maintenance.

Rationale. This change is due to plant changes and does not involve new methodology.

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

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

Classification. PRA upgrade

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

Change. Added common cause failure for several components by using the existing common cause failure methodology.

Classification. PRA maintenance.

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

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

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.

Classification. PRA maintenance.

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

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

Classification. PRA upgrade.

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

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).

Classification. PRA maintenance.

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).

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

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.

Classification. PRA maintenance.

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.

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

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.

Classification. PRA maintenance.

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

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.

Classification. PRA upgrade.

Rationale. This change is a fundamentally different approach to the development of fragility curves.

1-A.3.32 Example 32

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.

Classification. PRA maintenance.

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

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

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.

Classification. PRA upgrade.

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

Change. SPSA 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.

Classification. PRA maintenance.

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

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.

Classification. PRA maintenance.

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.

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

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.

Classification. PRA upgrade.

Rationale. This is a significant change in the scope of the PRA.

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

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.

Classification. PRA maintenance

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.3.38 Example 38

Change. Newly published industry fire frequencies are incorporated into the fire PRA. No new methodology is employed.

Classification. PRA maintenance.

Rationale. The analysis requirement to perform the change is very similar to that in Example 2.

Discussion and/or Alternative Recommendation. This change would impact essentially all of the fire-risk scenarios, including any previously found to be risk significant. However, the change reflects new information on plant performance (new data) without introducing new methods, changes in scope, or changes in category and thus conforms to the definition of PRA maintenance.

1-A.3.39 Example 39

Change An additional train of plant equipment is added to the fire PRA plant response model. That equipment was credited in the internal-events PRA but not in the original fire PRA.

Classification. PRA maintenance.

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

Discussion and/or Alternative Recommendation. This change would require updating various fire PRA analysis elements beyond the fire PRA plant response model (PRM). For example, the change might involve the identification and tracing of new plant equipment (ES) and cables (CS), new fire-induced circuit failure modes and effects analyses (CF), and reanalysis of existing fire scenarios and development of new fire scenarios (FSS). 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.3.40 Example 40

Change. The fire PRA incorporates detailed cable-routing information for cables that were previously treated using an exclusionary routing approach. (Note: The "exclusionary routing approach" establishes locations where a cable is not present and, hence, may be credited as operational for fire scenarios impacting those locations. The exclusionary approach is in contrast to detailed cable-routing information that would establish locations where a cable is present and, hence, must be considered a potential fire-damage target for fire scenarios impacting those locations.)

Classification. PRA maintenance.

Rationale. The change increases the level of realism in the analysis but does not involve the application of new methods of analysis.

Discussion and/or Alternative Recommendation. The development of more detailed cable-routing information can result from ongoing plant operation and maintenance activities. Incorporating such information will increase the fidelity and accuracy of the fire PRA. In general, this change would be considered PRA maintenance. However, if the new information leads to a significant impact on the fire-risk insights, then a focused-scope peer review of the changes may be appropriate.

1-A.3.41 Example 41

Change. A plant, which has transitioned to NFPA-805 [1-A-4] risk-informed, performance-based fire protection, implements a plant change that disables the automatic actuation feature for an existing fixed fire-suppression system. The change is analyzed by the NFPA-805 change analysis process and is also incorporated into the fire PRA. [Note that the terms "plant change" and "change analysis" have particular meaning in the context of NFPA-805. Once a plant completes the transition to the NFPA-805 risk-informed, performance-based fire protection approach, subsequent changes impacting the fire protection program (plant changes) must be supported by a risk-informed analysis to demonstrate that risk and performance goals are met given the change (a change analysis).]

Classification. PRA maintenance.

Rationale. The change impacts the assumptions used in the original fire PRA but does not require the use of new methodology.

Discussion and/or Alternative Recommendation. Many plant changes will likely be implemented via the risk-informed fire protection strategies of NFPA-805, and most of those changes will eventually be incorporated into the plant's fire PRA. In this case, the change would likely be reflected as a change to the actuation timing or the reliability of the fire-suppression system, or both. While not all NFPA-805 plant changes would be considered PRA maintenance, in this case the change is, in effect, updating plant-specific system performance data for a credited fire protection feature or system to reflect the as-built, as-operated plant conditions.

1-A.3.42 Example 42

Change. An incipient fire-detection system is installed in a physical analysis unit that was previously found to be a significant fire-risk contributor, and the new detection system is incorporated into the fire PRA. This is the first such system installed at the plant. (For reference, incipient fire-detection systems are designed to detect fire precursors during the earliest, or incipient, stage. This detection affords the opportunity for preignition intervention to disrupt or prevent fire development.)

Classification. PRA upgrade.

Rationale. Incipient fire-detection systems are analyzed using methods that are entirely unique from those applied to other types of detection systems.

Discussion and/or Alternative Recommendation. Incipient fire-detection systems operate on entirely unique principles as compared to more traditional fire-detection systems (e.g., smoke or heat detectors). The fire PRA update would also impact a risk-significant physical analysis unit. Hence, a focused-scope peer review of the analysis methods, assumptions, and results would be appropriate. If the plant had already credited other previously installed incipient detection systems in the fire PRA, and was applying the same methods of analysis to a newly installed system, then the change would be considered PRA maintenance.

1-A.3.43 Example 43

Change. The original fire scenario analyses for significant physical analysis units had relied on simple zone-type compartment fire model calculations and the analysis is upgraded by using a computational fluid dynamics (CFD) fire model.

Classification. PRA upgrade.

Rationale. The change involves new methods of analysis impacting risk-significant fire scenarios.

Discussion and/or Alternative Recommendation. In this case, a peer review of the modeling assumptions, methods, and results would be appropriate, in particular, to ensure that the new fire-modeling tools have been applied within their limits of applicability and that all relevant fire phenomena have been appropriately considered. If the original fire PRA had utilized the same CFD-based fire models, and those same methods of analysis were being extended to other fire scenarios or other physical analysis units, then the change would likely be classified as PRA maintenance.

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

[1-A-4] NFPA Standard 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants; 2001; Publisher: National Fire Protection Association (NFPA), 1 Batterymarch Park, Quincy, MA 02169

PART 2 (a)
REQUIREMENTS FOR INTERNAL-EVENTS AT-POWER PRA (b)

Section 2-1
Overview of Internal-Events At-Power PRA Requirements

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 technical 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 an SR 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:

Requirement 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 Requirement AS-A9.

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

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 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 by 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) Transients.* INCLUDE among the transients both equipment and human-induced events that disrupt the plant and leave the primary system pressure boundary intact. *(b) LOCAs.* 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. DELINEATE the LOCA initiators, using a defined rationale for the delineation. Examples of LOCA types include *(1) Small LOCAs.* Examples: reactor coolant pump seal LOCAs, small pipe breaks *(2) Medium LOCAs.* Examples: stuck open safety or relief valves *(3) Large LOCAs.* Examples: inadvertent ADS, component ruptures *(4) Excessive LOCAs* (LOCAs that cannot be mitigated by any combination of engineered systems). Example: reactor pressure vessel rupture *(5) LOCAs Outside Containment.* Example: primary system pipe breaks outside containment (BWRs) *(c) SGTRs.* INCLUDE spontaneous rupture of a steam generator tube (PWRs). *(d) HELBs.* Examples: steam line breaks inside and outside containment. *(e) ISLOCAs.* 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. *(f)* 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 Requirement 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.

Table 2-2.1-2 Supporting Requirements for HLR-IE-A (Cont'd)

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-A5	PERFORM a systematic evaluation of each system, including support systems, to assess the possibility of an initiating event occurring due to a failure of the system. PERFORM a qualitative review of system impacts to identify potentially system initiating events.	PERFORM a systematic evaluation of each system, including support systems, to assess the possibility of an initiating event occurring due to a failure of the system. USE a structured approach [such as a system-by-system review of initiating-event potential, or a failure modes and effects analysis (FMEA), or other systematic process] to assess and document the possibility of an initiating event resulting from individual systems or train failures.	PERFORM a systematic evaluation of each system, including support systems, to assess the possibility of an initiating event occurring due to a failure of the system. PERFORM a detailed analysis of system interfaces. PERFORM an FMEA (failure modes and effects analysis) or other systematic process to assess and document the possibility of an initiating event resulting from individual systems or train failures.
IE-A6	When performing the systematic evaluation required in Requirement IE-A5, INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause.	When performing the systematic evaluation required in Requirement IE-A5, INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause, and from routine system alignments.	When performing the systematic evaluation required in Requirement IE-A5, INCLUDE initiating events resulting from multiple failures, including equipment failures resulting from random and common causes, and from routine system alignments.
IE-A7	In the identification of the initiating events, INCLUDE *(a)* events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions), and for which it is determined that the event could also occur during at-power operation *(b)* events resulting in an unplanned controlled shutdown that includes a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to at-power operation		
IE-A8	No requirements for interviews.	INTERVIEW plant personnel (e.g., operations, maintenance, engineering, safety analysis) to determine if potential initiating events have been overlooked.	INTERVIEW plant operations, maintenance, engineering, and safety analysis personnel to determine if potential initiating events have been overlooked.

Table 2-2.1-2 Supporting Requirements for HLR-IE-A (Cont'd)

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-A9	No requirement for precursor review.	REVIEW plant-specific operating experience for initiating event precursors, for identifying additional initiating events. For example, plant-specific experience with intake structure clogging might indicate that loss of intake structures should be identified as a potential initiating event.	REVIEW plant-specific and industry operating experience for initiating-event precursors, for identifying additional initiating events.
IE-A10	For multi-unit sites with shared systems, INCLUDE multi-unit site initiators (e.g., multi-unit LOOP events or total loss of service water) that may impact the model.		

NOTE:
(1) These initiators may result in either a transient or a LOCA type of sequence.

Table 2-2.1-3 Supporting Requirements for 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-B).

Index No. IE-B	Capability Category I	Capability Category II	Capability Category III
IE-B1	GROUP initiating events to facilitate definition of accident sequences in the Accident Sequence Analysis (2-2.2) and to facilitate quantification in the Quantification (2-2.7).		
IE-B2	USE a structured, systematic process for grouping initiating events. For example, such a systematic approach may employ master logic diagrams, heat balance fault trees, or failure modes and effects analysis (FMEA).		
IE-B3	GROUP initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* events can be bounded by the worst case impacts within the group	GROUP initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* events can be bounded by the worst case impacts within the group and the grouping does not impact significant accident sequences	GROUP initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* events can be bounded by the worst-case impacts within the grouping of events does not impact significant accident sequences and impacts of each event are comparable to those of the remaining events in that group
IE-B4	GROUP separately from other initiating-event categories those categories with different plant response (i.e., those with different success criteria) impacts or those that could have more severe radionuclide release potential (e.g., LERF). This includes such initiators as excessive LOCA, interfacing systems LOCA, steam generator tube ruptures, and unisolated breaks outside containment.		
IE-B5	For multi-unit sites with shared systems, DO NOT SUBSUME multi-unit initiating events if they impact mitigation capability.		

Table 2-2.1-4 Supporting Requirements for HLR-IE-C

The initiating-event analysis shall estimate the annual frequency of each initiating event or initiating-event group (HLR-IE-C).

<table>
<tr><th>Index No. IE-C</th><th>Capability Category I</th><th>Capability Category II</th><th>Capability Category III</th></tr>
<tr><td>IE-C1</td><td colspan="3">ESTIMATE the initiating-event frequency accounting for relevant generic and plant-specific data unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty. (See also Requirement IE-C13 for requirements for rare and extremely rare events.)</td></tr>
<tr><td>IE-C2</td><td colspan="3">When using plant-specific data, USE the most recent applicable data to quantify the initiating-event frequencies. JUSTIFY excluded data that is not considered to be either recent or applicable (e.g., provide evidence via design or operational change that the data are no longer applicable).</td></tr>
<tr><td>IE-C3</td><td colspan="3">INCLUDE recovery actions [those implied in Requirement IE-C6(c), and those implied and discussed in Requirements IE-C8 through IE-C11] as appropriate. JUSTIFY each recovery action (e.g., as evidenced through procedures or training).</td></tr>
<tr><td>IE-C4</td><td colspan="3">When combining evidence from generic and plant-specific data, USE a Bayesian update process or equivalent statistical process. JUSTIFY the selection of any informative prior distribution used on the basis of industry experience (e.g., see reference [2-1]).</td></tr>
<tr><td>IE-C5</td><td colspan="2">ESTIMATE initiating-event frequencies on a reactor-year basis [Note (1)]. INCLUDE in the initiating-event analysis the plant availability, such that the frequencies are weighted by the fraction of time the plant is at-power.</td><td>ESTIMATE initiating-event frequencies on a reactor-year basis [Note (1)]. INCLUDE in the initiating-event analysis the plant availability, such that the frequencies are weighted by the fraction of time the plant is at-power. INCLUDE differences between historical plant availability over the period of event occurrences in the plant database and existing or expected future plant availability that could be different from historical values.</td></tr>
</table>

Table 2-2.1-4 Supporting Requirements for HLR-IE-C (Cont'd)

The initiating-event analysis shall estimate the annual frequency of each initiating event or initiating-event group (HLR-IE-C).

Index No. IE-C	Capability Category I	Capability Category II	Capability Category III
IE-C6	USE as screening criteria no higher than the following characteristics (or more stringent characteristics as devised by the analyst) to eliminate initiating events or groups from further evaluation: *(a)* the frequency of the event is less than 1E-7/reactor-yr, and the event does not involve either an ISLOCA, containment bypass, or reactor pressure vessel rupture *(b)* the frequency of the event is less than 1E-6/reactor-yr, and core damage could not occur unless at least two trains of mitigating systems are failed independent of the initiator, or *(c)* the resulting reactor shutdown is not an immediate occurrence. That is, the event does not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating-event conditions, with a high degree of certainty (based on supporting calculations), are detected and corrected before normal plant operation is curtailed (either administratively or automatically). If either criterion (a) or (b) above is used, ENSURE that the value specified in the criterion meets the applicable requirements in Data Analysis (2-2.6) and Quantification (2-2.7).		
IE-C7	No requirement for time trend analysis.		USE time trend analysis to account for established trends (e.g., decreasing reactor trip rates in recent years). JUSTIFY excluded data that are not considered to be either recent or applicable (e.g., provide evidence via design or operational change that the data are no longer applicable). Acceptable methodologies for time-trend analysis can be found in NUREG/CR-5750 [2-2] and NUREG/CR-6928 [2-20].
IE-C8	Some initiating events are amenable to fault-tree modeling as the appropriate way to quantify them. These initiating events, usually support system failure events, are highly dependent upon plant-specifc design features. If fault-tree modeling is used for initiating events, USE the applicable systems-analysis requirements for fault-tree modeling found in Systems Analysis (2-2.4).		
IE-C9	If fault tree modeling is used for initiating events, ESTIMATE the initiating-event frequency [as opposed to the probability of an initiating event over a specific time frame, which is the usual fault tree quantification model described in Systems Analysis (2-2.4)]. MODIFY, as necessary, the fault tree computational methods that are used so that the top event quantification produces a failure frequency rather than a top event probability as normally computed. USE the applicable requirements in Data Analysis (2-2.6) for the data used in the fault-tree quantification.		
IE-C10	If fault-tree modeling is used for initiating events, INCLUDE within the initiating-event fault tree models all relevant combinations of events involving the annual frequency of one component failure combined with the unavailability (or failure during the repair time of the first component) of other components.		

Table 2-2.1-4 Supporting Requirements for HLR-IE-C (Cont'd)

The initiating-event analysis shall estimate the annual frequency of each initiating event or initiating-event group (HLR-IE-C).

Index No. IE-C	Capability Category I	Capability Category II	Capability Category III
IE-C11	If fault-tree modeling is used for initiating events, USE plant-specific information in the assessment and quantification of recovery actions where available, in a manner consistent with the applicable requirements in Human Reliability Analysis (2-2.5).		
IE-C12	Where plant-specific data is used, COMPARE results with generic data sources and EXPLAIN differences in the initiating-event analysis to provide a reasonableness check of the results.		
IE-C13	For rare initiating events, USE industry generic data and INCLUDE plant-specific features to decide which generic data are most applicable. For extremely rare initiating events, engineering judgment may be used; if used, AUGMENT with applicable generic data sources. Refer to 1-4.3, Use of Expert Judgment, as appropriate.		For rare initiating events, USE industry generic data and AUGMENT with a plant-specific fault tree or other similar evaluation that accounts for plant-specific features. For extremely rare initiating events, engineering judgment may be used; if used, AUGMENT with applicable generic data sources. Refer to 1-4.3, Use of Expert Judgment, as appropriate. INCLUDE in the quantification the plant-specific features that could influence initiating events and recovery probabilities. Examples of plant-specific features that sometimes merit inclusion are the following: *(a)* plant geography, climate, and meteorology for LOOP and LOOP recovery *(b)* service water intake characteristics and plant experience *(c)* LOCA frequency calculation

Table 2-2.1-4 Supporting Requirements for HLR-IE-C (Cont'd)

The initiating-event analysis shall estimate the annual frequency of each initiating event or initiating-event group (HLR-IE-C).

Index No. IE-C	Capability Category I	Capability Category II	Capability Category III
IE-C14	In the ISLOCA frequency analysis, INCLUDE the following features of plant and procedures that influence the ISLOCA frequency: *(a)* configuration of potential pathways including numbers and types of valves and their relevant failure modes and the existence, size, and positioning of relief valves *(b)* provision of protective interlocks *(c)* relevant surveillance test procedures *(d)* the capability of secondary system piping *(e)* isolation capabilities given high flow/differential pressure conditions that might exist following breach of the secondary system		In the ISLOCA frequency analysis, INCLUDE the following features of the plant and procedures that influence the ISLOCA frequency: *(a)* configuration of potential pathways including numbers and types of valves and their relevant failure modes, existence, and positioning of relief valves *(b)* provision of protective interlocks *(c)* relevant surveillance test procedures Also, *(1)* EVALUATE surveillance procedure steps *(2)* INCLUDE surveillance test intervals explicitly *(3)* EVALUATE on-line surveillance testing quantitatively *(4)* ESTIMATE pipe rupture probability *(5)* ADDRESS explicitly valve design (e.g., air operated testable check valves) *(6)* INCLUDE quantitatively the valve isolation capability given the high-to-low-pressure differential

Table 2-2.1-4 Supporting Requirements for HLR-IE-C (Cont'd)

The initiating-event analysis shall estimate the annual frequency of each initiating event or initiating-event group (HLR-IE-C).

Index No. IE-C	Capability Category I	Capability Category II	Capability Category III
IE-C15	CALCULATE a point estimate for the initiating-event frequencies. CHARACTERIZE the uncertainty for those initiating-event frequencies associated with significant accident sequences. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	CALCULATE a mean value for the frequencies of the significant initiating events. PROVIDE the probabilistic representation of the uncertainty of the parameter estimates of significant initiating events. Acceptable methods include Bayesian updating or expert judgment. For the nonsignificant initiating events, CALCULATE point estimates and CHARACTERIZE the uncertainty for those initiating-event frequencies. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	CALCULATE a mean value for the frequencies of the initiating events. PROVIDE the probabilistic representation of the uncertainty for the parameter estimates of initiating events. Acceptable methods include Bayesian updating or expert judgment.

NOTE:
(1) For the computation of annual average core damage frequency/large early release frequency (i.e., for comparison to Reg. Guide 1.174 quantitative acceptance guidelines), the appropriate units for initiating-event frequency are events per calendar year, commonly expressed as events per reactor-year, where a reactor-year is one full calendar year of experience for one reactor. However, when determining total annual plant CDF (or LERF), which includes contributions from events occurring during power operation as well as during other plant operating states, the calculation of the contribution for each operating state must account for the fraction of the year that the plant is in that operating state. Two simple examples follow:

(a) Loss of Bus Initiating Event. A loss of bus initiating event can be computed by annualizing the hourly failure rate of the bus and associated breakers, relays, etc., that could lead to loss of power on the bus

Table 2-2.1.4 Supporting Requirements for HLR-IE-C (Cont'd)

The initiating-event analysis shall estimate the annual frequency of each initiating event or initiating-event group (HLR-IE-C).

NOTE: (Cont'd)

during the time the plant is at-power. For example, for the bus itself, the initiating-event frequency over a full year would be calculated as

$$f_{\text{bus-8,760}} = \lambda_{\text{bus}} * H_{\text{year}}$$

where

$f_{\text{bus-8,760}}$ = frequency of loss of bus over a full 8,760-hr year
H_{year} = hours in 1 calendar yr or reactor-yr, 8,760 hr/yr
λ_{bus} = failure rate of bus per hour, say 1×10^{-7}/hr

However, to calculate CDF (or LERF) for events at-power only (i.e., for the scope of PRA covered by this Standard), it is necessary to adjust for the fraction of time the plant is at-power. Thus, the result obtained from the above equation needs to be multiplied by an additional term, say $F_{\text{at-power}}$,
where
$F_{\text{at-power}}$ = fraction of year that, on average, the plant is at-power, for example 90%

Thus,

$$f_{\text{bus at-power}} = 1 \times 10^{-7}/\text{hr} * 8{,}760\ \text{hr/yr} * 0.90 = 7.9 \times 10^{-4}/\text{reactor-yr}$$

(b) Turbine Trip Initiating Event. Some initiating events, such as a turbine trip initiating event, may be computed based on plant-specific experience. In this case, the number of events classified as turbine trip events is in the numerator, and the number of applicable calendar years of operation is in the denominator. The fraction of time at-power is implicitly included in the numerator because the turbine trip experience is limited to at-power experience by the nature of the event.

Thus

$$f_{TT} = N_{TT}/Y_{OP}$$

where

f_{TT} = frequency of turbine trip events per reactor-year
N_{TT} = number of events classified as turbine trip events (for example, 27 events)
Y_{OP} = number of applicable calendar years of plant operation, regardless of operating mode (for example, 23 yr)

Therefore,

$$f_{TT} = 27\ \text{events}/23\ \text{yr} = 1.2/\text{reactor-yr}$$

The number of applicable calendar years should be based on the time period of the event data being used and may exclude unusual periods of non-operation (i.e., if the plant was in an extended forced shutdown).

For some applications, such as configuration risk management or analyses that compare specific risks during different modes of operation, it may be appropriate to use initiating-event frequencies that do not consider the fraction of time in the operating state. In these cases, the initiating-event frequency should simply be per unit of time (i.e., per hour or per year). For at-power operation, this basis is sometimes referred to as per reactor-critical-year (i.e., assuming that the reactor operated continuously for a year). On a more general basis, it could be considered to be per reactor-operating-state-year.

In the loss of bus initiating-event example above, the term $F_{\text{at-power}}$ would not be included in the computation of initiating-event frequency for these kinds of applications.

In the turbine trip initiating-event example above, the value must be adjusted by dividing f_{TT} by $F_{\text{at-power}}$.

Table 2-2.1-5 Supporting Requirements for HLR-IE-D

Documentation of the initiating-event analysis shall be consistent with the applicable supporting requirements (HLR-IE-D).

Index No. IE-D	Capability Category I	Capability Category II	Capability Category III
IE-D1	DOCUMENT the initiating-event analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
IE-D2	DOCUMENT the processes used to select, group, and screen the initiating events and to model and quantify the initiating-event frequencies, including the inputs, methods, and results. For example, this documentation typically includes *(a)* the functional categories considered and the specific initiating events included in each *(b)* the systematic search for plant-unique and plant-specific support system initiators *(c)* the systematic search for RCS pressure boundary failures and interfacing system LOCAs *(d)* the approach for assessing completeness and consistency of initiating events with plant-specific experience, industry experience, other comparable PRAs and FSAR initiating events *(e)* the basis for screening out initiating events *(f)* the basis for grouping and subsuming initiating events *(g)* the dismissal of any observed initiating events, including any credit for recovery *(h)* the derivation of the initiating-event frequencies and the recoveries used *(i)* the approach to quantification of each initiating-event frequency *(j)* the justification for exclusion of any data		
IE-D3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in Requirements QU-E1 and QU-E2) associated with the initiating-event analysis.		

2-2.2 ACCIDENT SEQUENCE ANALYSIS (AS)

2-2.2.1 Objectives

The objectives of the accident sequence element are to ensure that the response of the plant's systems and operators to an initiating event is reflected in the assessment of CDF in such a way that

(a) significant operator actions, mitigation systems, and phenomena that can alter sequences are appropriately included in the accident sequence model event tree structure and sequence definition

(b) plant-specific dependencies are reflected in the accident sequence structure

(c) success criteria are available to support the individual function successes, mission times, and time windows for operator actions for each critical safety function modeled in the accident sequences

(d) end states are clearly defined to be core damage or successful mitigation with capability to support the Level 1 to Level 2 interface

Table 2-2.2-1 High Level Requirements for Accident Sequence Analysis (AS)

Designator	Requirement
HLR-AS-A	The accident sequence analysis shall describe the plant-specific scenarios that can lead to core damage following each modeled initiating event. These scenarios shall address system responses and operator actions, including recovery actions that support the key safety functions necessary to prevent core damage.
HLR-AS-B	Dependencies that can impact the ability of the mitigating systems to operate and function shall be addressed.
HLR-AS-C	Documentation of the accident sequence analysis shall be consistent with the applicable supporting requirements.

Table 2-2.2-2 Supporting Requirements for HLR-AS-A

The accident sequence analysis shall describe the plant-specific scenarios that can lead to core damage following each modeled initiating event. These scenarios shall address system responses and operator actions, including recovery actions that support the key safety functions necessary to prevent core damage (HLR-AS-A).

Index No. AS-A	Capability Category I	Capability Category II	Capability Category III
AS-A1	USE a method for accident sequence analysis that *(a)* explicitly models the appropriate combinations of system responses and operator actions that affect the key safety functions for each modeled initiating event *(b)* includes a graphical representation of the accident sequences in an "event tree structure" or equivalent such that the accident sequence progression is displayed *(c)* provides a framework to support sequence quantification		
AS-A2	For each modeled initiating event, IDENTIFY the key safety functions that are necessary to reach a safe, stable state and prevent core damage. [See Note (1).]		
AS-A3	For each modeled initiating event, using the success criteria defined for each key safety function (in accordance with Requirement SC-A3), IDENTIFY the systems that can be used to mitigate the initiator. [See Note (1).]		
AS-A4	For each modeled initiating event, using the success criteria defined for each key safety function (in accordance with Requirement SC-A3), IDENTIFY the necessary operator actions to achieve the defined success criteria. [See Notes (1) and (2).]		
AS-A5	DEVELOP the accident sequences in a manner consistent with the plant-specific system design, EOPs, abnormal procedures, and plant transient response.		
AS-A6	Where practical, sequentially ORDER the events representing the response of the systems and operator actions according to the timing of the event as it occurs in the accident progression. Where not practical, PROVIDE the rationale used for the ordering.		
AS-A7	DELINEATE the possible accident sequences for each modeled initiating event, unless the sequences can be shown to be a noncontribution using qualitative arguments.		DELINEATE the possible accident sequences for each modeled initiating event.
AS-A8	DEFINE the end state of the accident progression as occurring when either a core damage state or a steady state condition has been reached.		
AS-A9	USE generic thermal-hydraulic analyses (e.g., as performed by a plant vendor for a class of similar plants) to determine the accident progression parameters (e.g., timing, temperature, pressure, steam) that could potentially affect the operability of the mitigating systems.	USE realistic, applicable (i.e., from similar plants) thermal-hydraulic analyses to determine the accident progression parameters (e.g., timing, temperature, pressure, steam) that could potentially affect the operability of the mitigating systems. (See Requirement SC-B4.)	USE realistic, plant-specific thermal-hydraulic analyses to determine the accident progression parameters (e.g., timing, temperature, pressure, steam) that could potentially affect the operability of the mitigating systems. (See Requirement SC-B4.)

Table 2-2.2-2 Supporting Requirements for HLR-AS-A (Cont'd)

The accident sequence analysis shall describe the plant-specific scenarios that can lead to core damage following each modeled initiating event. These scenarios shall address system responses and operator actions, including recovery actions that support the key safety functions necessary to prevent core damage (HLR-AS-A).

Index No. AS-A	Capability Category I	Capability Category II	Capability Category III
AS-A10	In constructing the accident sequence models, INCLUDE, for each modeled initiating event, individual events in the accident sequence sufficient to bound system operation, timing, and operator actions necessary for key safety functions.	In constructing the accident sequence models, INCLUDE, for each modeled initiating event, sufficient detail that differences in requirements on systems and required operator interactions (e.g., systems initiations or valve alignment) are captured. Where diverse systems and/or operator actions provide a similar function, if choosing one over another changes the requirements for operator intervention or the need for other systems, MODEL each separately.	In constructing the accident sequence models, explicitly INCLUDE, for each modeled initiating event, each system and operator action required for each key safety function.
AS-A11	Transfers between event trees may be used to reduce the size and complexity of individual event trees. DEFINE any transfers that are used and the method that is used to implement them in the qualitative definition of accident sequences and in their quantification. USE a method for implementing an event tree transfer that preserves the dependencies that are part of the transferred sequence. These include functional, system, initiating event, operator, and spatial or environmental dependencies.		

NOTES:

(1) Requirements AS-A2 through AS-A4 define the model in terms of how the plant works, but do not address what the model should include. Modeling details are addressed in Requirements AS-A5 through AS-11.

(2) The intent of this requirement is not to address specific procedures, but rather to identify, at a functional level, what is required of the operators for success.

Table 2-2.2-3 Supporting Requirements for HLR-AS-B

Dependencies that can impact the ability of the mitigating systems to operate and function shall be addressed (HLR-AS-B).

Index No. AS-B	Capability Category I	Capability Category II	Capability Category III
AS-B1	For each modeled initiating event, IDENTIFY mitigating systems impacted by the occurrence of the initiator and the extent of the impact. INCLUDE the impact of initiating events on mitigating systems in the accident progression either in the accident sequence models or in the system models.		
AS-B2	IDENTIFY the dependence of modeled mitigating systems on the success or failure of preceding systems, functions, and human actions. INCLUDE the impact on accident progression, either in the accident sequence models or in the system models. For example, *(a)* turbine-driven system dependency on SORV, depressurization, and containment heat removal (suppression pool cooling) *(b)* low-pressure system injection success dependent on need for RPV depressurization		
AS-B3	For each accident sequence, IDENTIFY the phenomenological conditions created by the accident progression. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc. that could impact the success of the system or function under consideration [e.g., loss of pump net positive suction head (NPSH), clogging of flow paths, pipe whip, jet impingement, and other high energy line break impacts such as flooding]. INCLUDE the impact of the accident progression phenomena, either in the accident sequence models or in the system models.		
AS-B4	When the event trees with conditional split fraction method is used, if the probability of Event B is dependent on the occurrence or non-occurrence of Event A, where practical, PLACE Event A to the left of Event B in the ordering of event tops. Where not practical, DESCRIBE the rationale used for the ordering.		
AS-B5	DEVELOP the accident sequence models to a level of detail sufficient to identify intersystem dependencies and train level interfaces, either in the event trees or through a combination of event tree and fault tree models and associated logic.		
AS-B6	If plant configurations and maintenance practices create dependencies among various system alignments, DEFINE and MODEL these configurations and alignments in a manner that reflects these dependencies, either in the accident sequence models or in the system models.		
AS-B7	MODEL time-phased dependencies (i.e., those that change as the accident progresses, due to such factors as depletion of resources, recovery of resources, and changes in loads) in the accident sequences. Examples are as follows: *(a)* For SBO/LOOP sequences, key time-phased events, such as *(1)* AC power recovery *(2)* DC battery adequacy (time-dependent discharge) *(3)* environmental conditions (e.g., room cooling) for operating equipment and the control room *(b)* For ATWS/failure to scram events (for BWRs), key time-dependent actions, such as *(1)* SLCS initiation *(2)* RPV level control *(3)* ADS inhibit *(c)* Other events that may be subject to explicit time-dependent characterization include *(1)* CRD as an adequate RPV injection source *(2)* long-term make-up to RWST		

Table 2-2.2-4 Supporting Requirements for HLR-AS-C

Documentation of the accident sequence analysis shall be consistent with the applicable supporting requirements (HLR-AS-C).

Index No. AS-C	Capability Category I	Capability Category II	Capability Category III
AS-C1	DOCUMENT the accident sequence analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
AS-C2	DOCUMENT the processes used to develop accident sequences and treat dependencies in accident sequences, including the inputs, methods, and results. For example, this documentation typically includes *(a)* the linkage between the modeled initiating event in Initiating-Event Analysis (2-2.1) and the accident sequence model *(b)* the success criteria established for each modeled initiating event including the bases for the criteria (i.e., the system capacities required to mitigate the accident and the necessary components required to achieve these capacities) *(c)* a description of the accident progression for each sequence or group of similar sequences (i.e., descriptions of the sequence timing, applicable procedural guidance, expected environmental or phenomenological impacts, dependencies between systems and operator actions, end states, and other pertinent information required to fully establish the sequence of events) *(d)* the operator actions reflected in the event trees, and the sequence-specific timing and dependencies that are traceable to the HRA for these actions *(e)* the interface of the accident sequence models with plant damage states *(f)* (when sequences are modeled using a single top event fault tree) the manner in which the requirements for accident sequence analysis have been satisfied		
AS-C3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in Requirements QU-E1 and QU-E2) associated with the accident sequence analysis.		

2-2.3 SUCCESS CRITERIA (SC)

2-2.3.1 Objectives

The objectives of the success criteria element are to define the plant-specific measures of success and failure that support the other technical elements of the PRA in such a way that

(a) overall success criteria are defined (i.e., core damage)

(b) success criteria are defined for critical safety functions, supporting SSCs, and operator actions necessary to support accident sequence development

(c) the methods and approaches have a firm technical basis

(d) the resulting success criteria are referenced to the specific deterministic calculations

Table 2-2.3-1 High Level Requirements for Success Criteria (SC)

Designator	Requirement
HLR-SC-A	The overall success criteria for the PRA, and the SSC and human action success criteria used in the PRA shall be defined and referenced, and shall be consistent with the features, procedures, and operating philosophy of the plant.
HLR-SC-B	The thermal-hydraulic, structural, and other supporting engineering bases shall be capable of providing success criteria and event timing sufficient for quantification of CDF, determination of the relative impact of success criteria on SSC and human actions, and the impact of uncertainty on this determination.
HLR-SC-C	Documentation of success criteria shall be consistent with the applicable supporting requirements.

Table 2-2.3-2 Supporting Requirements for HLR-SC-A

The overall success criteria for the PRA, and the SSC and human action success criteria used in the PRA shall be defined and referenced, and shall be consistent with the features, procedures, and operating philosophy of the plant (HLR-SC-A).

Index No. SC-A	Capability Category I	Capability Category II	Capability Category III
SC-A1	USE the definition of "core damage" provided in Section 1-2 of this Standard. If core damage has been defined differently than in Section 1-2 *(a)* IDENTIFY any substantial differences from the Section 1-2 definition *(b)* PROVIDE the bases for the selected definition		
SC-A2	SPECIFY the plant parameters (e.g., highest node temperature, core collapsed liquid level) and associated acceptance criteria (e.g., temperature limit) to be used in determining core damage. Examples of measures for core damage suitable for Capability Category I are provided in NUREG/CR-4550 [Note (1)].	SPECIFY the plant parameters (e.g., highest node temperature, core collapsed liquid level) and associated acceptance criteria (e.g., temperature limit) to be used in determining core damage. SELECT these parameters such that the determination of core damage is as realistic as practical, in a manner consistent with current best practice. SPECIFY computer code-predicted acceptance criteria with sufficient margin on the code-calculated values to allow for limitations of the code, sophistication of the models, and uncertainties in the results, in a manner consistent with the requirements specified under Requirement HLR-SC-B. Examples of measures for core damage for non-ATWS scenarios suitable for Capability Category II/III, that have been used in PRAs, include ***(a)*** collapsed liquid level less than one-third core height or code-predicted peak core temperature >2,500°F (BWR) ***(b)*** Collapsed liquid level below top of active fuel for a prolonged period; or code-predicted peak core temperature >1,800°F using a code with simplified (e.g., single-node core model, lumped parameter) core modeling; or code-predicted core exit temperature >1,200°F for 30 min using a code with simplified core modeling (PWR) The "peak core temperature" in this example refers to post-shutdown conditions.	
SC-A3	SPECIFY success criteria for each of the key safety functions identified per Requirement AS-A2 for each modeled initiating event [Note (2)].		
SC-A4	IDENTIFY mitigating systems that are shared between units and the manner in which the sharing is performed should both units experience a common initiating event (e.g., LOOP).		

Table 2-2.3-2 Supporting Requirements for HLR-SC-A (Cont'd)

The overall success criteria for the PRA, and the SSC and human action success criteria used in the PRA shall be defined and referenced, and shall be consistent with the features, procedures, and operating philosophy of the plant (HLR-SC-A).

Index No. SC-A	Capability Category I	Capability Category II	Capability Category III
SC-A5	SPECIFY an appropriate mission time for the modeled accident sequences. For sequences in which stable plant conditions have been achieved, USE a minimum mission time of 24 hr. Mission times for individual SSCs that function during the accident sequence may be less than 24 hr, as long as an appropriate set of SSCs and operator actions are modeled to support the full sequence mission time. For example, if following a LOCA, low-pressure injection is available for 1 hr, after which recirculation is required, the mission time for LPSI may be 1 hr and the mission time for recirculation may be 23 hr. For sequences in which stable plant conditions would not be achieved within 24 hr using the modeled plant equipment and human actions, ASSUME core damage.	SPECIFY an appropriate mission time for the modeled accident sequences. For sequences in which stable plant conditions have been achieved, USE a minimum mission time of 24 hr. Mission times for individual SSCs that function during the accident sequence may be less than 24 hr, as long as an appropriate set of SSCs and operator actions are modeled to support the full sequence mission time. For example, if following a LOCA, low pressure injection is available for 1 hr, after which recirculation is required, the mission time for LPSI may be 1 hr and the mission time for recirculation may be 23 hr. For sequences in which stable plant conditions would not be achieved within 24 hr using the modeled plant equipment and human actions, PERFORM additional evaluation or modeling by using an appropriate technique. Examples of appropriate techniques include *(a)* assigning an appropriate plant damage state for the sequence *(b)* extending the mission time and adjusting the affected analyses, to the point at which conditions can be shown to reach acceptable values; or *(c)* modeling additional system recovery or operator actions for the sequence, in accordance with requirements stated in Systems Analysis (2-2.4) and Human Reliability (2-2.5) to demonstrate that a successful outcome is achieved	
SC-A6	ENSURE that the bases for the success criteria are consistent with the features, procedures, and operating philosophy of the plant.		

NOTES:

(1) Pages 3 through 8 of reference [2-3] used the following simplified definitions of core damage to avoid the need for "detailed thermal-hydraulic calculations beyond the scope and resources of the work." For BWRs, "the core is considered to be in a damaged state when the reactor water level is less than 2 ft above the bottom of the active fuel." For PWRs, "the core is considered to be in a damaged state once the top of the active fuel assemblies is uncovered."

(2) Requirements for specification of success criteria appear under high level requirements for other elements as well (e.g., Requirements HLR-AS-A, HLR-SY-A). These requirements are intended to be complementary, not duplicative. For example, for accident sequences, Requirements AS-A2, SC-A3, SC-A4 (if applicable), AS-A3, and AS-A4 are intended to be used together to capture the specification of the set of systems and human actions necessary to meet the key safety function success criteria.

Table 2-2.3-3 Supporting Requirements for HLR-SC-B

The thermal-hydraulic, structural and other supporting engineering bases shall be capable of providing success criteria and event timing sufficient for quantification of CDF, determination of the relative impact of success criteria on the importance of the SSCs and human actions, and the impact of uncertainty on this determination (HLR-SC-B).

Index No. SC-B	Capability Category I	Capability Category II	Capability Category III
SC-B1	USE appropriate conservative, generic analyses/evaluations that are applicable to the plant.	USE appropriate realistic generic analyses/evaluations that are applicable to the plant for thermal-hydraulic, structural, and other supporting engineering bases in support of success criteria requiring detailed computer modeling. (See Requirement SC-B4.) Realistic models or analyses may be supplemented with plant-specific/generic FSAR or other conservative analysis applicable to the plant, but only if such supplemental analyses do not affect the determination of which combinations of systems and trains of systems are required to respond to an initiating event.	USE realistic plant-specific models for thermal-hydraulic, structural, and other supporting engineering bases in support of success criteria requiring detailed computer modeling. (See Requirement SC-B4.) DO NOT USE assumptions that could yield conservative or optimistic success criteria.
SC-B2	No restrictions regarding the use of expert judgment, but Requirement SC-C2 must be met.	DO NOT USE expert judgment except in those situations in which there is lack of available information regarding the condition or response of a modeled SSC, or a lack of analytical methods upon which to base a prediction of SSC condition or response. USE the requirements in 1-4.3 when implementing an expert judgment process.	
SC-B3	When defining success criteria, USE thermal-hydraulic, structural, or other analyses/evaluations appropriate to the event being analyzed, and accounting for a level of detail consistent with the initiating event grouping (Requirement HLR-IE-B) and accident sequence modeling (Requirements HLR-AS-A and HLR-AS-B).		
SC-B4	USE analysis models and computer codes that have sufficient capability to model the conditions of interest in the determination of success criteria for CDF, and that provide results representative of the plant. A qualitative evaluation of a relevant application of codes, models, or analyses that has been used for a similar class of plant (e.g., Owners Group generic studies) may be used. USE computer codes and models only within known limits of applicability.		
SC-B5	ENSURE the reasonableness and acceptability of the results of the thermal-hydraulic, structural, or other supporting engineering bases used to support the success criteria. Examples of methods to achieve this include *(a)* comparison with results of the same analyses performed for similar plants, accounting for differences in unique plant features *(b)* comparison with results of similar analyses performed with other plant-specific codes *(c)* check by other means appropriate to the particular analysis		

Table 2-2.3-4 Supporting Requirements for HLR-SC-C

Documentation of success criteria shall be consistent with the applicable supporting requirements (HLR-SC-C).

Index No. SC-C	Capability Category I	Capability Category II	Capability Category III
SC-C1	DOCUMENT the success criteria in a manner that facilitates PRA applications, upgrades, and peer review.		
SC-C2	DOCUMENT the processes used to develop overall PRA success criteria and the supporting engineering bases, including the inputs, methods, and results. For example, this documentation typically includes *(a)* the definition of core damage used in the PRA including the bases for any selected parameter value used in the definition (e.g., peak cladding temperature or reactor vessel level) *(b)* calculations (generic and plant-specific) or other references used to establish success criteria, and identification of cases for which they are used *(c)* identification of computer codes or other methods used to establish plant-specific success criteria *(d)* a description of the limitations (e.g., potential conservatisms or limitations that could challenge the applicability of computer models in certain cases) of the calculations or codes *(e)* the uses of expert judgment within the PRA, and rationale for such uses *(f)* a summary of success criteria for the available mitigating systems and human actions for each accident initiating group modeled in the PRA *(g)* the basis for establishing the time available for human actions *(h)* descriptions of processes used to define success criteria for grouped initiating events or accident sequences		
SC-C3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in Requirements QU-E1 and QU-E2) associated with the development of success criteria.		

2-2.4 SYSTEMS ANALYSIS (SY)

2-2.4.1 Objectives

The objectives of the systems analysis element are to identify and quantify the causes of failure for each plant system represented in the initiating-event analysis and accident sequence analysis in such a way that

(a) system-level success criteria, mission times, time windows for operator actions, and assumptions provide the basis for the system logic models as reflected in the model. A reasonably complete set of system failure and unavailability modes for each system is represented.

(b) human errors and operator actions that could influence the system unavailability or the system's contribution to accident sequences are identified for development as part of Human Reliability Analysis (2.2-5).

(c) different initial system alignments are evaluated to the extent needed for CDF determination.

(d) intersystem dependencies and intra-system dependencies including functional, human, phenomenological, and common-cause failures that could influence system unavailability or the system's contribution to accident-sequence frequencies are identified and accounted for.

Table 2-2.4-1 High Level Requirements for Systems Analysis (SY)

Designator	Requirement
HLR-SY-A	The systems analysis shall provide a reasonably complete treatment of the causes of system failure and unavailability modes represented in the initiating-events analysis and sequence definition.
HLR-SY-B	The systems analysis shall provide a reasonably complete treatment of common cause failures and intersystem and intra-system dependencies.
HLR-SY-C	Documentation of the systems analysis shall be consistent with the applicable supporting requirements.

Table 2-2.4-2 Supporting Requirements for HLR-SY-A

The systems analysis shall provide a reasonably complete treatment of the causes of system failure and unavailability modes represented in the initiating-events analysis and sequence definition (HLR-SY-A).

Index No. SY-A	Capability Category I	Capability Category II	Capability Category III
SY-A1	DEVELOP system models for those systems needed to provide or support the safety functions contained in the accident sequence analyses.		
SY-A2	COLLECT pertinent information to ensure that the systems analysis appropriately reflects the as-built and as-operated systems. Examples of such information include system P&IDs, one-line diagrams, instrumentation and control drawings, spatial layout drawings, system operating procedures, abnormal operating procedures, emergency procedures, success criteria calculations, the FSAR, Technical Specifications, training information, system descriptions and related design documents, actual system operating experience, and interviews with system engineers and operators.		
SY-A3	REVIEW plant information sources to define or establish *(a)* system components and boundaries *(b)* dependencies on other systems *(c)* instrumentation and control requirements *(d)* testing and maintenance requirements and practices *(e)* operating limitations such as those imposed by Technical Specifications *(f)* component operability and design limits *(g)* procedures for the operation of the system during normal and accident conditions *(h)* system configuration during normal and accident conditions		
SY-A4	ENSURE that the system analysis correctly reflects the as-built, as-operated plant through discussions with knowledgeable plant personnel (e.g., engineering, plant operations, etc.).	PERFORM plant walkdowns and interviews with knowledgeable plant personnel (e.g., engineering, plant operations, etc.) to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	
SY-A5	INCLUDE the effects of both normal and alternate system alignments, to the extent needed for CDF determination.		
SY-A6	In defining the system model boundary (see Requirement SY-A3), INCLUDE within the boundary the components required for system operation, and the components providing the interfaces with support systems required for actuation and operation of the system components.		

Table 2-2.4-2 Supporting Requirements for HLR-SY-A (Cont'd)

The systems analysis shall provide a reasonably complete treatment of the causes of system failure and unavailability modes represented in the initiating-events analysis and sequence definition (HLR-SY-A).

Index No. SY-A	Capability Category I	Capability Category II	Capability Category III
SY-A7	DEVELOP detailed systems models, unless *(a)* sufficient system-level data are available to quantify the system failure probability, or *(b)* system failure is dominated by operator actions, and omitting the model does not mask contributions to the results of support systems or other dependent-failure modes. For case (a), USE a single data value only for systems with no equipment or human-action dependencies, and if data exist that sufficiently represent the unreliability or unavailability of the system and account for plant-specific factors that could influence unreliability and unavailability. Examples of systems that have sometimes not been modeled in detail include the scram system, the power-conversion system, instrument air, and the keep-fill systems. JUSTIFY the use of limited (i.e., reduced or single data value) modeling.		DEVELOP detailed system models.
SY-A8	DEFINE the boundaries of the components required for system operation. MATCH the definitions used to establish the component failure data. For example, a control circuit for a pump does not need to be included as a separate basic event (or events) in the system model if the pump failure data used in quantifying the system model include control circuit failures. MODEL, as separate basic events, those subcomponents (e.g., a valve limit switch associated with a permissive signal for another component) that are shared by another component or affect another component, to account for the dependent failure mechanism.		
SY-A9	If a system model is developed in which a single failure of a super component (or module) is used to represent the collective impact of failures of several components, PERFORM the modularization process in a manner that avoids grouping events with different recovery potential, events that are required by other systems, or events that have probabilities dependent on the scenario. Examples of such events include *(a)* hardware failures that are not recoverable versus actuation signals, which are recoverable *(b)* HFEs that can have different probabilities dependent on the context of different accident sequences *(c)* events that are mutually exclusive of other events not in the module *(d)* events that occur in other fault trees (especially common-cause events) *(e)* SSCs that are used by other systems		

Table 2-2.4-2 Supporting Requirements for HLR-SY-A (Cont'd)

The systems analysis shall provide a reasonably complete treatment of the causes of system failure and unavailability modes represented in the initiating-events analysis and sequence definition (HLR-SY-A).

Index No. SY-A	Capability Category I	Capability Category II	Capability Category III
SY-A10	INCORPORATE the effect of variable success criteria (i.e., success criteria that change as a function of plant status) into the system modeling. Example causes of variable system success criteria include *(a)* different accident scenarios. Different success criteria are required for some systems to mitigate different accident scenarios (e.g., the number of pumps required to operate in some systems is dependent upon the modeled initiating event). *(b)* dependence on other components. Success criteria for some systems are also dependent on the success of another component in the system (e.g., operation of additional pumps in some cooling water systems is required if noncritical loads are not isolated). *(c)* time dependence. Success criteria for some systems are time-dependent (e.g., two pumps are required to provide the needed flow early following an accident initiator, but only one is required for mitigation later following the accident). *(d)* sharing of a system between units. Success criteria may be affected when both units are challenged by the same initiating event (e.g., LOOP).		
SY-A11	INCLUDE in the system model those failures of the equipment and components that would affect system operability (as identified in the system success criteria), except when excluded using the criteria in Requirement SY-A15. This equipment includes both active components (e.g., pumps, valves, and air compressors) and passive components (e.g., piping, heat exchangers, and tanks) required for system operation.		
SY-A12	DO NOT INCLUDE in a system model component failures that would be beneficial to system operation, unless omission would distort the results. Example of a beneficial failure: A failure of an instrument in such a fashion as to generate a required actuation signal.		
SY-A13	INCLUDE those failures that can cause flow diversion pathways resulting in failure to meet the system success criteria.		
SY-A14	When identifying the failures in Requirement SY-A11, INCLUDE consideration of all failure modes, consistent with available data and model level of detail, except where excluded using the criteria in Requirement SY-A15. For example *(a)* active component fails to start *(b)* active component fails to continue to run *(c)* failure of a closed component to open *(d)* failure of a closed component to remain closed *(e)* failure of an open component to close *(f)* failure of an open component to remain open *(g)* active component spurious operation *(h)* plugging of an active or passive component *(i)* leakage of an active or passive component *(j)* rupture of an active or passive component *(k)* internal leakage of a component *(l)* internal rupture of a component *(m)* failure to provide signal/operate (e.g., instrumentation) *(n)* spurious signal/operation *(o)* pre-initiator human failure events (see Requirement SY-A16) *(p)* other failures of a component to perform its required function		

Table 2-2.4-2 Supporting Requirements for HLR-SY-A (Cont'd)

The systems analysis shall provide a reasonably complete treatment of the causes of system failure and unavailability modes represented in the initiating-events analysis and sequence definition (HLR-SY-A).

Index No. SY-A	Capability Category I	Capability Category II	Capability Category III
SY-A15	In meeting Requirements SY-A11 and SY-A14, contributors to system unavailability and unreliability (i.e., components and specific failure modes) may be excluded from the model if one of the following screening criteria is met: *(a)* A component may be excluded from the system model if the total failure probability of the component failure modes resulting in the same effect on system operation is at least two orders of magnitude lower than the highest failure probability of the other components in the same system train that results in the same effect on system operation. *(b)* One or more failure modes for a component may be excluded from the systems model if the contribution of them to the total failure rate or probability is less than 1% of the total failure rate or probability for that component, when their effects on system operation are the same.		
SY-A16	In the system model, INCLUDE HFEs that cause the system or component to be unavailable when demanded. These events are referred to as pre-initiator human events. (See also Human Reliability Analysis, 2-2.5.)		
SY-A17	In the system model, INCLUDE HFEs that are expected during the operation of the system or component or that are accounted for in the final quantification of accident sequences unless they are already included explicitly as events in the accident sequence models. These HFEs are referred to as post-initiator human actions. [See also Human Reliability Analysis (2-2.5) and Accident Sequence Analysis (2-2.2).]		
SY-A18	INCLUDE in either the system model or accident sequence modeling those conditions that cause the system to isolate or trip, or those conditions that once exceeded cause the system to fail, or DEMONSTRATE that their exclusion does not impact the results. For example, conditions that isolate or trip a system include *(a)* system-related parameters such as a high temperature within the system *(b)* external parameters used to protect the system from other failures [e.g., the high reactor pressure vessel (RPV) water level isolation signal used to prevent water intrusion into the turbines of the RCIC and HPCI pumps of a BWR] *(c)* adverse environmental conditions (see Requirement SY-A22)		

Table 2-2.4-2 Supporting Requirements for HLR-SY-A (Cont'd)

The systems analysis shall provide a reasonably complete treatment of the causes of system failure and unavailability modes represented in the initiating-events analysis and sequence definition (HLR-SY-A).

Index No. SY-A	Capability Category I	Capability Category II	Capability Category III
SY-A19	In the systems model, INCLUDE out-of-service unavailability for components in the system model, unless screened out, in a manner consistent with the actual practices and history of the plant for removing equipment from service. *(a)* INCLUDE *(1)* unavailability caused by testing when a component or system train is reconfigured from its required accident mitigating position such that the component cannot function as required *(2)* maintenance events at the train level when procedures require isolating the entire train for maintenance *(3)* maintenance events at a sub-train level (i.e., between tagout boundaries, such as a functional equipment group) when directed by procedures *(b)* Examples of out-of-service unavailability to be modeled are as follows: *(1)* train outages during a work window for preventive/corrective maintenance *(2)* a functional equipment group (FEG) removed from service for preventive/corrective maintenance *(3)* a relief valve taken out of service		
SY-A20	INCLUDE events representing the simultaneous unavailability of redundant equipment when this is a result of planned activity (see Requirement DA-C14).		
SY-A21	IDENTIFY system conditions that cause a loss of desired system function (e.g., excessive heat loads, excessive electrical loads, excessive humidity, etc.).		
SY-A22	DO NOT INCLUDE system or component operability when the potential exists for rated or design capabilities to be exceeded.	INCLUDE system or component operability only if an analysis exists to demonstrate that rated or design capabilities are not exceeded.	INCLUDE system or component operability, including functionality for beyond design or rated capabilities, if supported by an appropriate combination of *(a)* test or operational data *(b)* engineering analysis *(c)* expert judgment
SY-A23	DEFINE system model nomenclature in a consistent manner to allow model manipulation and to represent the same designator when a component failure mode is used in multiple systems or trains.		
SY-A24	DO NOT MODEL the repair of hardware faults, unless the probability of repair is justified through an adequate analysis or examination of data. (See Requirement DA-C15.)		

Table 2-2.4-3 Supporting Requirements for HLR-SY-B

The systems analysis shall provide a reasonably complete treatment of common cause failures and intersystem and intra-system dependencies (HLR-SY-B).

Index No. SY-B	Capability Category I	Capability Category II	Capability Category III
SY-B1	MODEL intra-system common-cause failures when supported by generic or plant-specific data (an acceptable model is the screening approach of NUREG/CR-5485 [2-4], which is consistent with Requirement DA-D5), or DEMONSTRATE that they do not impact the results.	MODEL intra-system common-cause failures when supported by generic or plant-specific data. An acceptable method is represented in NUREG/CR-5485 [2-4].	
SY-B2	No requirement to model inter-system common cause failures.		MODEL inter-system common-cause failures (i.e., across systems performing the same function) when supported by generic or plant-specific data, or DEMONSTRATE that they do not impact the results.
SY-B3	DEFINE common cause failure groups by using a logical, systematic process that considers similarity in *(a)* service conditions *(b)* environment *(c)* design or manufacturer *(d)* maintenance JUSTIFY the basis for selecting common cause component groups. Candidates for common-cause failures include, for example, *(a)* motor-operated valves *(b)* pumps *(c)* safety-relief valves *(d)* air-operated valves *(e)* solenoid-operated valves *(f)* check valves *(g)* diesel generators *(h)* batteries *(i)* inverters and battery charger *(j)* circuit breakers		
SY-B4	INCLUDE common cause failures into the system model in a manner consistent with the common cause model used for data analysis. (See Requirement DA-D6.)		
SY-B5	INCLUDE the modeled system's dependency on support systems or interfacing systems in the modeling process. This may be accomplished in one of the following ways: *(a)* for the fault tree linking approach by modeling the dependencies as a link to an appropriate event or gate in the support system fault tree *(b)* for the linked event tree approach, by using event tree logic rules, or calculating a probability for each split fraction conditional on the scenario definition		

Table 2-2.4-3 Supporting Requirements for HLR-SY-B (Cont'd)

The systems analysis shall provide a reasonably complete treatment of common cause failures and intersystem and intra-system dependencies (HLR-SY-B).

Index No. SY-B	Capability Category I	Capability Category II	Capability Category III
SY-B6	PERFORM engineering analyses to determine the need for support systems that are plant-specific and reflect the variability in the conditions present during the postulated accidents for which the system is required to function.		
SY-B7	In support system modeling, USE conservative success criteria and timing.	In support system modeling, USE realistic success criteria and timing, for significant contributors.	In support system modeling, USE realistic plant-specific success criteria and timing.
SY-B8	IDENTIFY spatial and environmental hazards that may impact multiple systems or redundant components in the same system, and INCLUDE them in the system fault tree or the accident sequence evaluation. Example: Use results of plant walkdowns as a source of information regarding spatial/environmental hazards, for resolution of spatial/environmental issues, or evaluation of the impacts of such hazards.		
SY-B9	When modeling a system, INCLUDE appropriate interfaces with the support systems required for successful operation of the system for a required mission time (see also Requirement SY-A6). Examples of support systems include *(a)* actuation logic *(b)* support systems required for control of components *(c)* component motive power *(d)* cooling of components *(e)* any other identified support function (e.g., heat tracing) necessary to meet the success criteria and associated systems		
SY-B10	IDENTIFY those systems that are required for initiation and actuation of a system. MODEL them unless a justification is provided (e.g., the initiation and actuation system can be argued to be highly reliable and is only used for that system, so that there are no inter-system dependencies arising from failure of the system). In the model quantification, INCLUDE the presence of the conditions needed for automatic actuation (e.g., low vessel water level). INCLUDE permissive and lockout signals that are required to complete actuation logic.	MODEL those systems that are required for initiation and actuation of a system. In the model quantification, INCLUDE the presence of the conditions needed for automatic actuation (e.g., low vessel water level). INCLUDE permissive and lockout signals that are required to complete actuation logic.	

Table 2-2.4-3 Supporting Requirements for HLR-SY-B (Cont'd)

The systems analysis shall provide a reasonably complete treatment of common cause failures and intersystem and intra-system dependencies (HLR-SY-B).

Index No. SY-B	Capability Category I	Capability Category II	Capability Category III
SY-B11	MODEL the capability of the available inventories of air, power, and cooling to support the mission time.		
SY-B12	DO NOT USE proceduralized recovery actions as the sole basis for eliminating a support system from the model; however, INCLUDE these recovery actions in the model quantification. For example, it is not acceptable to not model a system such as HVAC or CCW on the basis that there are procedures for dealing with losses of these systems.		
SY-B13	Some systems use components and equipment that are required for operation of other systems. INCLUDE components that, using the criteria in Requirement SY-A15, may be screened out from each system model individually, if their failure affects more than one system (e.g., a common suction pipe feeding two separate systems).		
SY-B14	IDENTIFY SSCs that may be required to operate in conditions beyond their environmental qualifications. INCLUDE dependent failures of multiple SSCs that result from operation in these adverse conditions. Examples of degraded environments include *(a)* LOCA inside containment with failure of containment heat removal *(b)* safety relief valve operability (small LOCA, drywell spray, severe accident) (for BWRs) *(c)* high energy line breaks in different locations, e.g., steam line breaks outside containment *(d)* debris that could plug screens/filters (both internal and external to the plant) *(e)* heating of the water supply (e.g., BWR suppression pool, PWR containment sump) that could affect pump operability *(f)* loss of NPSH for pumps *(g)* steam binding of pumps *(h)* harsh environments induced by containment venting, failure of the containment venting ducts, or failure of the containment boundary that may occur prior to the onset of core damage		
SY-B15	INCLUDE operator interface dependencies across systems or trains, where applicable.		

Table 2-2.4-4 Supporting Requirements for HLR-SY-C

Documentation of the systems analysis shall be consistent with the applicable supporting requirements (HLR-SY-C).

Index No. SY-C	Capability Category I	Capability Category II	Capability Category III
SY-C1	DOCUMENT the systems analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
SY-C2	DOCUMENT the system functions and boundary, the associated success criteria, the modeled components and failure modes including human actions, and a description of modeled dependencies including support system and common cause failures, including the inputs, methods, and results. For example, this documentation typically includes *(a)* system function and operation under normal and emergency operations *(b)* system model boundary *(c)* system schematic illustrating all equipment and components necessary for system operation *(d)* information and calculations to support equipment operability considerations and assumptions *(e)* actual operational history indicating any past problems in the system operation *(f)* system success criteria and relationship to accident sequence models *(g)* human actions necessary for operation of system *(h)* reference to system-related test and maintenance procedures *(i)* system dependencies and shared component interface *(j)* component spatial information *(k)* assumptions or simplifications made in development of the system models *(l)* the components and failure modes included in the model and justification for any exclusion of components and failure modes *(m)* a description of the modularization process (if used) *(n)* records of resolution of logic loops developed during fault tree linking (if used) *(o)* results of the system model evaluations *(p)* results of sensitivity studies (if used) *(q)* the sources of the above information (e.g., completed checklist from walkdowns, notes from discussions with plant personnel) *(r)* basic events in the system fault trees so that they are traceable to modules and to cutsets *(s)* the nomenclature used in the system models		
SY-C3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in Requirements QU-E1 and QU-E2) associated with the systems analysis.		

2-2.5 HUMAN RELIABILITY ANALYSIS (HR)[1]

2-2.5.1 Objectives

The objective of the human reliability element of the PRA is to ensure that the impacts of plant personnel actions are reflected in the assessment of risk in such a way that

(a) both pre–initiating-event and post–initiating-event activities, including those modeled in support system initiating-event fault trees, are addressed

(b) logic model elements are defined to represent the effect of such personnel actions on system availability/unavailability and on accident sequence development

(c) plant-specific and scenario-specific factors are accounted for, including those factors that influence either what activities are of interest or human performance

(d) human performance issues are addressed in an integral way so that issues of dependency are captured

[1] The following reference provides useful background information for Human Reliability Analysis:
D. T. Wakefield, G. W. Parry, G. W. Hannaman, A. J. Spurgin, "SHARP1 — Revised Systematic Human Action Reliability Procedure" EPRI Report TR-101711-T2, March 1993.

Table 2-2.5-1 High Level Requirements for Human Reliability Analysis (HR)

Designator	Requirement
Pre-Initiator HRA	
HLR-HR-A	A systematic process shall be used to identify those specific routine activities that, if not completed correctly, may impact the availability of equipment necessary to perform system function modeling in the PRA.
HLR-HR-B	Screening of activities that need not be addressed explicitly in the model shall be based on an assessment of how plant-specific operational practices limit the likelihood of errors in such activities.
HLR-HR-C	For each activity that is not screened out, an appropriate human failure event (HFE) shall be defined to characterize the impact of the failure as an unavailability of a component, system, or function modeled in the PRA.
HLR-HR-D	The assessment of the probabilities of the pre-initiator human failure events shall be performed by using a systematic process that addresses the plant-specific and activity-specific influences on human performance.
Post-Initiator HRA	
HLR-HR-E	A systematic review of the relevant procedures shall be used to identify the set of operator responses required for each of the accident sequences.
HLR-HR-F	Human failure events shall be defined that represent the impact of not properly performing the required responses, in a manner consistent with the structure and level of detail of the accident sequences.
HLR-HR-G	The assessment of the probabilities of the post-initiator HFEs shall be performed by using a well-defined and self-consistent process that addresses the plant-specific and scenario-specific influences on human performance, and addresses potential dependencies between human failure events in the same accident sequence.
HLR-HR-H	Recovery actions (at the cutset or scenario level) shall be modeled only if it has been demonstrated that the action is plausible and feasible for those scenarios to which they are applied. Estimates of probabilities of failure shall address dependency on prior human failures in the scenario.
Pre- and Post-Initiator HRA	
HLR-HR-I	Documentation of the human reliability analysis shall be consistent with the applicable supporting requirements.

Table 2-2.5-2 Supporting Requirements for HLR-HR-A

A systematic process shall be used to identify those specific routine activities that, if not completed correctly, may impact the availability of equipment necessary to perform system function modeling in the PRA (HLR-HR-A).

Index No. HR-A	Capability Category I	Capability Category II	Capability Category III
HR-A1	For equipment modeled in the PRA, IDENTIFY those test, inspection, and maintenance activities that require realignment of equipment outside its normal operational or standby status.		
HR-A2	IDENTIFY those calibration activities that if performed incorrectly can have an adverse impact on the automatic initiation of standby equipment.		
HR-A3	IDENTIFY the work practices identified in Requirements HR-A1 and HR-A2 that involve a mechanism that simultaneously affects equipment in either different trains of a redundant system or diverse systems [e.g., use of common calibration equipment by the same crew on the same shift, a maintenance or test activity that requires realignment of an entire system (e.g., SLCS)].		

Table 2-2.5-3 Supporting Requirements for HLR-HR-B

Screening of activities that need not be addressed explicitly in the model shall be based on an assessment of how plant-specific operational practices limit the likelihood of errors in such activities (HLR-HR-B).

Index No. HR-B	Capability Category I	Capability Category II	Capability Category III
HR-B1	If screening is performed, SPECIFY rules for screening classes of activities from further consideration. Example: Screen out maintenance and test activities from further consideration only if the plant practices are generally structured to include independent checking of restoration of equipment to standby or operational status on completion of the activity.	If screening is performed, SPECIFY rules for screening individual activities from further consideration. Example: Screen out maintenance and test activities from further consideration only if *(a)* equipment is automatically re-aligned on system demand *(b)* following maintenance activities, a post-maintenance functional test is performed that reveals misalignment *(c)* equipment position is indicated in the control room, status is routinely checked, and realignment can be affected from the control room, or *(d)* equipment status is required to be checked frequently (i.e., at least once a shift)	
HR-B2	DO NOT SCREEN OUT activities that could simultaneously have an impact on multiple trains of a redundant system or diverse systems (see Requirement HR-A3).		

Table 2-2.5-4 Supporting Requirements for HLR-HR-C

For each activity that is not screened out, an appropriate human failure event (HFE) shall be defined to characterize the impact of the failure as an unavailability of a component, system, or function modeled in the PRA (HLR-HR-C).

Index No. HR-C	Capability Category I	Capability Category II	Capability Category III
HR-C1	For each unscreened activity, DEFINE a human failure event (HFE) that represents the impact of the human failure at the appropriate level (i.e., function, system, train, or component affected).		
HR-C2	INCLUDE those modes of unavailability that, following completion of each unscreened activity, result from failure to restore *(a)* equipment to the desired standby or operational status *(b)* initiation signal or set point for equipment start-up or realignment *(c)* automatic realignment or power	INCLUDE those modes of unavailability that, following completion of each unscreened activity, result from failure to restore *(a)* equipment to the desired standby or operational status *(b)* initiation signal or set point for equipment start-up or realignment *(c)* automatic realignment or power INCLUDE failure modes identified during the collection of plant-specific or applicable generic operating experience that leave equipment unavailable for response in accident sequences.	
HR-C3	INCLUDE the impact of miscalibration as a mode of failure of initiation of standby systems.		

Table 2-2.5-5 Supporting Requirements for HLR-HR-D

The assessment of the probabilities of the pre-initiator human failure events shall be performed by using a systematic process that addresses the plant-specific and activity-specific influences on human performance (HLR-HR-D).

Index No. HR-D	Capability Category I	Capability Category II	Capability Category III
HR-D1	ESTIMATE the probabilities of human failure events using a systematic process. Acceptable methods include THERP [2-5] and ASEP [2-6].		
HR-D2	USE screening estimates in the quantification of the pre-initiator HEPs.	For significant HFEs, USE detailed assessments in the quantification of pre-initiator HEPs. USE screening values based on a simple model, such as ASEP in the quantification of the pre-initiator HEPs for nonsignificant human failure basic events. When bounding values are used, ENSURE that they are based on limiting cases from models such as ASEP [2-6].	USE detailed assessments in the quantification of pre-initiator HEPs for each system.
HR-D3	No requirement for evaluating the quality of written procedures, administrative controls, or human-machine interfaces.	For each detailed human error probability assessment, INCLUDE in the evaluation process the following plant-specific relevant information: *(a)* the quality (e.g., format, logical structure, ease of use, clarity, and comprehensiveness) of written procedures (for performing tasks) and administrative controls that support independent review of written procedures (e.g., configuration control process, technical review process, training processes, and management emphasis on adherence to procedures) *(b)* the quality of the human-machine interface, including both the equipment configuration and the instrumentation and control layout [e.g., adherence to human factors guidelines (see NUREG-0700 [2-22]) and results of any quantitative evaluations of performance per functional requirements]	
HR-D4	When taking into account self-recovery or recovery from other crew members in estimating HEPs for specific HFEs, USE pre-initiator recovery factors in a manner consistent with selected methodology. If recovery of pre-initiator errors is credited *(a)* SPECIFY the maximum credit that can be given for multiple recovery opportunities *(b)* USE the following information to assess the potential for recovery of pre-initiator errors: *(1)* post-maintenance or post-calibration tests required and performed by procedure *(2)* independent verification, using a written check-off list, that verifies component status following maintenance/testing *(3)* a separate check of component status made at a later time, using a written check-off list, by the original performer *(4)* work shift or daily checks of component status, using a written check-off list		

Table 2-2.5-5 Supporting Requirements for HLR-HR-D (Cont'd)

The assessment of the probabilities of the pre-initiator human failure events shall be performed by using a systematic process that addresses the plant-specific and activity-specific influences on human performance (HLR-HR-D).

Index No. HR-D	Capability Category I	Capability Category II	Capability Category III
HR-D5	EVALUATE the potential for dependencies of pre-initiator HFEs (i.e., whether the HFEs have some common elements in their causes, such as work performed by the same crew in the same timeframe), and ESTIMATE the joint probability of the dependencies identified.		
HR-D6	CALCULATE a point estimate and CHARACTERIZE the uncertainty for the HFEs. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	CALCULATE a mean value for the significant HFEs. PROVIDE the probabilistic representation of the uncertainty of the parameter estimates of significant HFEs. Acceptable methods include Bayesian updating or expert judgment. For the nonsignificant HFEs, CALCULATE point estimates and CHARACTERIZE the uncertainty. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	CALCULATE a mean value for the HFEs. PROVIDE the probabilistic representation of the uncertainty for the parameter estimates of the HFEs. Acceptable methods include Bayesian updating or expert judgment.
HR-D7	No requirement to check reasonableness of HEPs in light of the plant's experience.		ENSURE the reasonableness of the HEPs in light of the plant's experience.

Table 2-2.5-6 Supporting Requirements for HLR-HR-E

A systematic review of the relevant procedures shall be used to identify the set of operator responses required for each of the accident sequences (HLR-HR-E).

Index No. HR-E	Capability Category I	Capability Category II	Capability Category III
HR-E1	When identifying the key human response actions REVIEW *(a)* the plant-specific emergency operating procedures, and other relevant procedures (e.g., AOPs, annunciator response procedures) in the context of the accident scenarios *(b)* system operation such that an understanding of how the system(s) functions and the human interfaces with the system is obtained		
HR-E2	IDENTIFY those actions *(a)* required to initiate (for those systems not automatically initiated), operate, control, isolate, or terminate those systems and components used in preventing or mitigating core damage as defined by the success criteria (e.g., operator initiates RHR) *(b)* performed by the control room personnel either in response to procedural direction or as skill-of-the-craft to diagnose and then recover a failed function, system, or component that is used in the performance of a response action as identified in Requirement HR-H1		
HR-E3	REVIEW the interpretation of the procedures with plant operations or training personnel to confirm that interpretation is consistent with plant operational and training practices.	USE talk-throughs (i.e., review in detail) with plant operations and training personnel the procedures and sequence of events to confirm that interpretation of the procedures is consistent with plant observations and training procedures.	
HR-E4	No requirement for using simulator observations or talk-throughs with operators to confirm human response actions.	USE simulator observations or talk-throughs with operators to confirm the human response actions for scenarios modeled.	

Table 2-2.5-7 Supporting Requirements for HLR-HR-F

Human failure events shall be defined that represent the impact of not properly performing the required responses, consistent with the structure and level of detail of the accident sequences (HLR-HR-F).

Index No. HR-F	Capability Category I	Capability Category II	Capability Category III
HR-F1	DEFINE human failure events (HFEs) that represent the impact of the human failures at the function, system, train, or component level, as appropriate. Failures to correctly perform several responses may be grouped into one HFE if the impact of the failures is similar or can be conservatively bounded.		DEFINE human failure events (HFEs) that represent the impact of the human failures at the function, system, train, or component level, as appropriate.
HR-F2	COMPLETE the definition of the HFEs by specifying *(a)* accident sequence-specific timing of cues, and time window for successful completion *(b)* accident sequence-specific procedural guidance (e.g., AOPs, and EOPs) *(c)* the availability of cues and other indications for detection and evaluation errors *(d)* the complexity of the response (Task analysis is not required.)	COMPLETE the definition of the HFEs by specifying *(a)* accident sequence-specific timing of cues, and time window for successful completion *(b)* accident sequence-specific procedural guidance (e.g., AOPs, and EOPs) *(c)* the availability of cues and other indications for detection and evaluation errors *(d)* the specific high level tasks (e.g., train level) required to achieve the goal of the response	COMPLETE the definition of the HFEs by specifying *(a)* accident sequence-specific timing of cues, and time window for successful completion *(b)* accident sequence-specific procedural guidance (e.g., AOPs, and EOPs) *(c)* the availability of cues and other indications for detection and evaluation errors *(d)* the specific detailed tasks (e.g., at the level of individual components, such as pumps or valves) required to achieve the goal of the response

Table 2-2.5-8 Supporting Requirements for HLR-HR-G

The assessment of the probabilities of the post-initiator HFEs shall be performed using a well-defined and self-consistent process that addresses the plant-specific and scenario-specific influences on human performance, and addresses potential dependencies between human failure events in the same accident sequence (HLR-HR-G).

Index No. HR-G	Capability Category I	Capability Category II	Capability Category III
HR-G1	USE conservative estimates (e.g., screening values) for the HEPs of the HFEs in accident sequences that survive initial quantification.	PERFORM detailed analyses for the estimation of HEPs for significant HFEs. USE screening values for HEPs for nonsignificant human failure basic events.	PERFORM detailed analyses for the estimation of human failure basic events.
HR-G2	USE an approach to estimation of HEPs that addresses failure in cognition as well as failure to execute.		
HR-G3	USE an approach that takes the following into account: *(a)* the complexity of detection, diagnosis, decision making, and executing the required response *(b)* the time available and time required to complete the response *(c)* some measure of scenario-induced stress The ASEP Approach [2-6] is an acceptable approach.	When estimating HEPs, EVALUATE the impact of the following plant-specific and scenario-specific performance shaping factors: *(a)* quality [type (classroom or simulator) and frequency] of the operator training or experience *(b)* quality of the written procedures and administrative controls *(c)* availability of instrumentation needed to take corrective actions *(d)* degree of clarity of cues/indications *(e)* human-machine interface *(f)* time available and time required to complete the response *(g)* complexity of detection, diagnosis, decision making, and executing the required response *(h)* environment (e.g., lighting, heat, radiation) under which the operator is working *(i)* accessibility of the equipment requiring manipulation *(j)* necessity, adequacy, and availability of special tools, parts, clothing, etc.	
HR-G4	For the time available to complete actions, USE applicable generic studies (e.g., thermal-hydraulic analysis for similar plants). SPECIFY the point in time at which operators are expected to receive relevant indications.	For the time available to complete actions, USE appropriate realistic generic thermal-hydraulic analyses, or simulation from similar plants (e.g., plant of similar design and operation). SPECIFY the point in time at which operators are expected to receive relevant indications.	For the time available to complete actions, USE plant-specific thermal-hydraulic analysis, or simulations. SPECIFY the point in time at which operators are expected to receive relevant indications.

Table 2-2.5-8 Supporting Requirements for HLR-HR-G (Cont'd)

The assessment of the probabilities of the post-initiator HFEs shall be performed using a well-defined and self-consistent process that addresses the plant-specific and scenario-specific influences on human performance, and addresses potential dependencies between human failure events in the same accident sequence (HLR-HR-G).

Index No. HR-G	Capability Category I	Capability Category II	Capability Category III
HR-G5	When needed, ESTIMATE the time required to complete actions. The approach described in ASEP [2-6] is an acceptable approach.	When needed, for the required time to complete actions for significant HFEs, USE action-time measurements in either walk-throughs or talk-throughs of the procedures or simulator observations.	When needed, for the required time to complete actions, USE action-time measurements in either walk-throughs or talk-throughs of the procedures or simulator observations.
HR-G6	ENSURE the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to ensure their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.		
HR-G7	For multiple human actions in the same accident sequence or cutset, identified in accordance with Requirement QU-C1, ASSESS the degree of dependence, and calculate a joint human error probability that reflects the dependence. INCLUDE the influence of success or failure in preceding human actions and system performance on the human event under consideration, including *(a)* time required to complete all actions in relation to the time available to perform the actions *(b)* factors that could lead to dependence (e.g., common instrumentation, common procedures, increased stress, etc.) *(c)* availability of resources (e.g., personnel) [Note (1)]		
HR-G8	CALCULATE a point estimate and CHARACTERIZE the uncertainty for the HFEs. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	CALCULATE a mean value for the significant HFEs. PROVIDE the probabilistic representation of the uncertainty of the parameter estimates of significant HFEs. Acceptable methods include Bayesian updating or expert judgment. For the nonsignificant HFEs, CALCULATE point estimates and CHARACTERIZE the uncertainty. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	CALCULATE a mean value for the HFEs. PROVIDE the probabilistic representation of the uncertainty for the parameter estimates of the HFEs. Acceptable methods include Bayesian updating or expert judgment.

NOTE:
(1) The state of the art in HRA is such that the assessment of dependency is largely based on the analyst's judgment. While it should be expected that there will be a progressively more detailed treatment of dependency in going from CC I to CC III, the distinction is not made at the level of this SR. Instead, it is expected to follow from the increase in the level of detail in the analysis of HFEs in going from CC I to CC III.

Table 2-2.5-9 Supporting Requirements for HLR-HR-H

Recovery actions (at the cutset or scenario level) shall be modeled only if it has been demonstrated that the action is plausible and feasible for those scenarios to which they are applied. Estimates of probabilities of failure shall address dependency on prior human failures in the scenario (HLR-HR-H) [Note (1)].

Index No. HR-H	Capability Category I	Capability Category II	Capability Category III
HR-H1	INCLUDE operator recovery actions that can restore the functions, systems, or components on an as-needed basis to provide a more realistic evaluation of CDF.	INCLUDE operator recovery actions that can restore the functions, systems, or components on an as-needed basis to provide a more realistic evaluation of significant accident sequences.	INCLUDE operator recovery actions that can restore the functions, systems, or components to provide a realistic evaluation of modeled accident sequences.
HR-H2	INCLUDE operator recovery actions only if, on a plant-specific basis, the following occur: *(a)* A procedure is available and operator training has included the action as part of crew's training, or justification for the omission for one or both is provided. *(b)* "Cues" (e.g., alarms) that alert the operator to the recovery action provided procedure, training, or skill-of-the-craft exist. *(c)* Attention is given to the relevant performance shaping factors provided in Requirement HR-G3. *(d)* There is sufficient manpower to perform the action.		
HR-H3	INCLUDE any dependency between the HFE for operator recovery and any other HFEs in the sequence, scenario, or cutset to which the recovery is applied (see Requirement HR-G7).		

NOTE:
(1) Recovery actions are actions taken in addition to those normally identified in the review of emergency, abnormal, and system operating procedures, which would normally be addressed in Requirements HLR-HR-E through HLR-HR-G. They are included to allow credit for recovery from failures in cutsets or scenarios when failure to take credit would distort the insights from the risk analysis. The potential for recovery (e.g., manually opening a valve that had failed to open automatically) may well differ between scenarios or cutsets. In this context, recovery is associated with workarounds but does not include repair, which is addressed in Requirements SY-A24 and DA-C15.

Table 2-2.5-10 Supporting Requirements for HLR-HR-I

Documentation of the human reliability analysis shall be consistent with the applicable supporting requirements (HLR-HR-I).

Index No. HR-I	Capability Category I	Capability Category II	Capability Category III
HR-I1	DOCUMENT the human reliability analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
HR-I2	DOCUMENT the processes used to identify, characterize, and quantify the pre-initiator, post-initiator, and recovery actions modeled in the PRA, including the inputs, methods, and results. For example, this documentation typically includes *(a)* HRA methodology and process used to identify pre- and post-initiator HEPs *(b)* qualitative screening rules and results of screening *(c)* factors used in the quantification of the human action, how they were derived (their bases), and how they were incorporated into the quantification process *(d)* quantification of HEPs, including *(1)* screening values and their bases *(2)* detailed HEP analyses with uncertainties and their bases *(3)* the method and treatment of dependencies for post-initiator actions *(4)* tables of pre- and post-initiator human actions evaluated by model, system, initiating event, and function *(5)* HEPs for recovery actions and their dependency with other HEPs		
HR-I3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in Requirements QU-E1 and QU-E2) associated with the human reliability analysis.		

2-2.6 DATA ANALYSIS (DA)

2-2.6.1 Objectives

The objectives of the data analysis elements are to provide estimates of the parameters used to determine the probabilities of the basic events representing equipment failures and unavailabilities modeled in the PRA in such a way that

(a) parameters, whether estimated on the basis of plant-specific or generic data, appropriately reflect that configuration and operation of the plant

(b) component or system unavailabilities due to maintenance or repair are accounted for

(c) uncertainties in the data are understood and appropriately accounted for

A useful reference document for parameter estimation is NUREG/CR-6823 [2-1].

Table 2-2.6-1 High Level Requirements for Data Analysis (DA)

Designator	Requirement
HLR-DA-A	Each parameter shall be clearly defined in terms of the logic model, basic event boundary, and the model used to evaluate event probability.
HLR-DA-B	Grouping components into a homogeneous population for parameter estimation shall consider both the design, environmental, and service conditions of the components in the as-built and as-operated plant.
HLR-DA-C	Generic parameter estimates shall be chosen, and collection of plant-specific data shall be consistent with the parameter definitions of HLR-DA-A and the grouping rationale of HLR-DA-B.
HLR-DA-D	The parameter estimates shall be based on relevant generic industry or plant-specific evidence. Where feasible, generic and plant-specific evidence shall be integrated using acceptable methods to obtain plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty.
HLR-DA-E	Documentation of the data analysis shall be consistent with the applicable supporting requirements.

Table 2-2.6-2 Supporting Requirements for HLR-DA-A

Each parameter shall be clearly defined in terms of the logic model, basic event boundary, and the model used to evaluate event probability (HLR-DA-A).

Index No. DA-A	Capability Category I	Capability Category II	Capability Category III
DA-A1	IDENTIFY from the systems analysis the basic events for which probabilities are required. Examples of basic events include *(a)* independent or common cause failure of a component or system to start or change state on demand *(b)* independent or common cause failure of a component or system to continue operating or provide a required function for a defined time period *(c)* equipment unavailable to perform its required function due to being out of service for maintenance *(d)* equipment unavailable to perform its required function due to being in test mode *(e)* failure to recover a function or system (e.g., failure to recover offsite-power) *(f)* failure to repair a component, system, or function in a defined time period		
DA-A2	DEFINE SSC boundaries, failure modes, and success criteria in a manner consistent with corresponding basic event definitions in Requirements SY-A5, SY-A7 through SY-A14, and SY-B4 for failure rates and common cause failure parameters, and DEFINE boundaries of unavailability events in a manner consistent with corresponding definitions in Requirement SY-A19.		
DA-A3	USE an appropriate probability model for each basic event. Examples include *(a)* $1-e^{-\lambda T} \cong \lambda T$ when $\lambda T < 0.1$ for failure to continue running over a mission time, T, with a constant failure rate, λ *(b)* $(\lambda\tau)/2$ for a periodically tested standby component subject to a standby failure rate of λ and a testing interval of τ		
DA-A4	IDENTIFY the parameter to be estimated and the data required for estimation. Examples are as follows: *(a)* For failures on demand, the parameter is the probability of failure, and the data required are the number of failures given a number of demands. *(b)* For standby failures, operating failures, and initiating events, the parameter is the failure rate, and the data required are the number of failures in the total (standby or operating) time. *(c)* For unavailability due to test or maintenance, the parameter is the unavailability on demand, and the alternatives for the data required include *(1)* the total time of unavailability or a list of the maintenance events with their durations, together with the total time required to be available; or *(2)* the number of maintenance or test acts, their average duration, and the total time required to be available		

Table 2-2.6-3 Supporting Requirements for HLR-DA-B

The rationale for grouping components into a homogeneous population for parameter estimation shall consider the design, environmental, and service conditions of the components in the as-built and as-operated plant (HLR-DA-B).

Index No. DA-B	Capability Category I	Capability Category II	Capability Category III
DA-B1	For parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve).	For parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve) and according to the characteristics of their usage to the extent supported by data: *(a)* mission type (e.g., standby, operating) *(b)* service condition (e.g., clean vs. untreated water, air)	For parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve) and according to the detailed characteristics of their usage to the extent supported by data: *(a)* design/size *(b)* system characteristics *(1)* mission type (e.g., standby, operating) *(2)* service condition (e.g., clean vs. untreated water, air) *(3)* maintenance practices *(4)* frequency of demands *(c)* environmental conditions *(d)* other appropriate characteristics
DA-B2	DO NOT INCLUDE outliers in the definition of a group (e.g., do not group valves that are never tested and unlikely to be operated with those that are tested or otherwise manipulated frequently).		DO NOT INCLUDE outliers in the definition of a group (e.g., do not group values that are never tested and unlikely to be operated with those that are tested or otherwise manipulated frequently). When warranted by sufficient data, USE appropriate hypothesis tests to ensure that data from grouped components are from compatible populations.

Table 2-2.6-4 Supporting Requirements for HLR-DA-C

Generic parameter estimates shall be chosen, and collection of plant-specific data shall be consistent with the parameter definitions of HLR-DA-A and the grouping rationale of HLR-DA-B.

<table>
<tr><th>Index No. DA-C</th><th>Capability Category I</th><th>Capability Category II</th><th>Capability Category III</th></tr>
<tr><td>DA-C1</td><td colspan="3">USE generic parameter estimates from recognized sources. ENSURE that the parameter definitions and boundary conditions are consistent with those established in response to Requirements DA-A1 to DA-A4. (Example: some sources include the breaker within the pump boundary, whereas others do not.) DO NOT INCLUDE generic data for unavailability due to test, maintenance, and repair unless it can be established that the data are consistent with the test and maintenance philosophies for the subject plant.
Examples of parameter estimates and associated sources include
(a) component failure rates and probabilities: NUREG/CR-4639 [2-7], NUREG/CR-4550 [2-3], NUREG-1715 [2-21], NUREG/CR-6928 [2-20]
(b) common cause failures: NUREG/CR-5497 [2-8], NUREG/CR-6268 [2-9]
(c) AC off-site power recovery: NUREG/CR-5496 [2-10], NUREG/CR-5032 [2-11]
(d) component recovery
See NUREG/CR-6823 [2-1] for a listing of additional data sources.</td></tr>
<tr><td>DA-C2</td><td colspan="3">COLLECT plant-specific data for the basic event/parameter grouping corresponding to that defined by Requirements DA-A1, DA-A3, DA-A4, DA-B1, and DA-B2.</td></tr>
<tr><td>DA-C3</td><td colspan="3">COLLECT plant-specific data, in a manner consistent with uniformity in design, operational practices, and experience. JUSTIFY the rationale for screening out or disregarding plant-specific data (e.g., plant design modifications, changes in operating practices).</td></tr>
<tr><td>DA-C4</td><td colspan="2">When evaluating maintenance or other relevant records to extract plant-specific component failure event data, SPECIFY a clear basis for the identification of events as failures.
DELINEATE between those degraded states for which a failure, as modeled in the PRA, would have occurred during the mission and those for which a failure would not have occurred (e.g., slow pickup to rated speed).
INCLUDE all failures that would have resulted in failure to perform the mission as defined in the PRA.</td><td>When evaluating maintenance or other relevant records to extract plant-specific component failure event data, SPECIFY a clear basis for the identification of events as failures.
DELINEATE between those degraded states for which a failure, as modeled in the PRA, would have occurred during the mission and those for which a failure would not have occurred (e.g., slow pickup to rated speed).
INCLUDE all failures that would have resulted in failure to perform the mission as defined in the PRA, excluding events captured in the pre-initiator-HEP calculation.</td></tr>
</table>

Table 2-2.6-4 Supporting Requirements for HLR-DA-C (Cont'd)

Generic parameter estimates shall be chosen, and collection of plant-specific data shall be consistent with the parameter definitions of HLR-DA-A and the grouping rationale of HLR-DA-B.

Index No. DA-C	Capability Category I	Capability Category II	Capability Category III
DA-C5	COUNT repeated plant-specific component failures occurring within a short time interval as a single failure if there is a single, repetitive problem that causes the failures. In addition, COUNT only one demand.		
DA-C6	ESTIMATE the number of plant-specific demands on standby components on the basis of the number of *(a)* surveillance tests *(b)* maintenance acts *(c)* surveillance tests or maintenance on other components *(d)* operational demands DO NOT COUNT additional demands from post-maintenance testing; that is part of the successful renewal.		
DA-C7	ESTIMATE the number of surveillance tests and planned maintenance activities on plant requirements.	BASE the number of surveillance tests on plant surveillance requirements and actual practice. BASE the number of planned maintenance activities on plant maintenance plans and actual practice. BASE the number of unplanned maintenance acts on actual plant experience.	
DA-C8	When required, ESTIMATE the time that components were configured in their standby status.	When required, USE plant-specific operational records to determine the time that components were configured in their standby status.	
DA-C9	ESTIMATE operational time from surveillance test practices for standby components, and from actual operational data.		ESTIMATE operational time from surveillance test records for standby components, and from actual operational data.

Table 2-2.6-4 Supporting Requirements for HLR-DA-C (Cont'd)

Generic parameter estimates shall be chosen, and collection of plant-specific data shall be consistent with the parameter definitions of HLR-DA-A and the grouping rationale of HLR-DA-B.

Index No. DA-C	Capability Category I	Capability Category II	Capability Category III
DA-C10	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. INCLUDE only completed tests or unplanned operational demands as success for component operations.	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. INCLUDE only completed tests or unplanned operational demands as success for component operation. If the component failure mode is decomposed into subelements (or causes) that are fully tested, then USE tests that exercise specific subelements in their evaluation. Thus, one subelement sometimes has many more successes than another. [Example: a diesel generator is tested more frequently than the load sequencer. If the sequencer were to be included in the diesel generator boundary, the number of valid test would be significantly decreased.]	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. INCLUDE only completed tests or unplanned operational demands as success for component operation. DECOMPOSE the component failure mode into subelements (or causes) that are fully tested, and USE tests that exercise specific subelements in their evaluation. Thus, one subelement sometimes has many more successes than another.
DA-C11	When using data on maintenance and testing durations to estimate unavailabilities at the component, train, or system level, as required by the system model, only INCLUDE those maintenance or test activities that could leave the component, train, or system unable to perform its function when demanded.		
DA-C12	When an unavailability of a front-line system component is caused by an unavailability of a support system, INCLUDE support system unavailability independent of front-line system unavailability, to avoid double counting unavailabilities and to include dependency on support system correctly.		

Table 2-2.6-4 Supporting Requirements for HLR-DA-C (Cont'd)

Generic parameter estimates shall be chosen, and collection of plant-specific data shall be consistent with the parameter definitions of HLR-DA-A and the grouping rationale of HLR-DA-B.

Index No. DA-C	Capability Category I	Capability Category II	Capability Category III
DA-C13	ESTIMATE the duration of the actual time that the equipment was unavailable for each contributing activity. Since maintenance outages are a function of the plant status, INCLUDE only outages occurring during plant at-power. INCLUDE consideration of the unavailability of shared systems at a multi-plant site, when the Technical Specifications (TS) requirements can be different depending on the status of both plants. Accurate modeling generally leads to a particular allocation of outage data among basic events to take this mode dependence into account. In the case that reliable estimates of the start and finish times of periods of unavailability are not available, USE conservative estimates.	ESTIMATE the duration of the actual time that the equipment was unavailable for each contributing activity. Since maintenance outages are a function of the plant status, INCLUDE only outages occurring during plant at-power. INCLUDE consideration of the unavailability of shared systems at a multiplant site, when the Technical Specifications (TS) requirements can be different depending on the status of both plants. Accurate modeling generally leads to a particular allocation of outage data among basic events to take this mode dependence into account. In the case that reliable estimates or the start and finish times are not available, INTERVIEW the knowledgeable plant personnel (e.g., engineering, plant operations, etc.) to generate estimates of ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are significant basic events.	
DA-C14	EVALUATE coincident unavailability due to maintenance for redundant equipment (both intrasystem and intersystem) that is a result of a planned, repetitive activity based on actual plant experience. CALCULATE coincident maintenance unavailabilities that are a result of a planned, repetitive activity that reflect actual plant experience. Such coincident maintenance unavailability can arise, for example, for plant systems that have "installed spares" (i.e., plant systems that have more redundancy than is addressed by Technical Specifications). For example (intrasystem case), the charging system in some plants has a third train that may be out of service for extended periods of time coincident with one of the other trains and yet is in compliance with Technical Specifications. Examples of intersystem unavailability include plants that routinely take out multiple components on a "train schedule" (such as AFW train A and HPI train A at a PWR, or RHR train A and LPCS train A at a BWR).		
DA-C15	For each SSC for which repair is to be modeled (see Requirement SY-A24), IDENTIFY instances of plant-specific or applicable industry experience and for each repair, COLLECT the associated repair time with the repair time being the period from identification of the component failure until the component is returned to service.		
DA-C16	Data on recovery from loss of offsite power, loss of service water, etc. are rare on a plant-specific basis. If available, for each recovery, COLLECT the associated recovery time with the recovery time being the period from identification of the system or function failure until the system or function is returned to service.		

Table 2-2.6-5 Supporting Requirements for HLR-DA-D

The parameter estimates shall be based on relevant generic industry and plant-specific evidence. Where feasible, generic and plant-specific evidence shall be integrated using acceptable methods to obtain plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty (HLR-DA-D).

Index No. DA-D	Capability Category I	Capability Category II	Capability Category III
DA-D1	USE plant-specific parameter estimates for events modeling the unique design or operational features if available, or use generic information modified as discussed in Requirement DA-D2; USE generic information for the remaining events.	ESTIMATE realistic parameters for significant basic events based on relevant generic and plant-specific evidence unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty. When it is necessary to combine evidence from generic and plant-specific data, USE a Bayes update process or equivalent statistical process that assigns appropriate weight to the statistical significance of the generic and plant-specific evidence and provides an appropriate characterization of uncertainty. SELECT prior distributions as either noninformative, or representative of variability in industry data. ESTIMATE parameters for the remaining events by using generic industry data.	ESTIMATE realistic parameters based on relevant generic and plant-specific evidence unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty. When it is necessary to combine evidence from generic and plant-specific data, USE a Bayes update process or equivalent statistical process that assigns appropriate weight to the statistical significance of the generic and plant-specific evidence and provides an appropriate characterization of uncertainty. SELECT prior distributions as either noninformative, or representative of variability in industry data.
DA-D2	If neither plant-specific data nor generic parameter estimates are available for the parameter associated with a specific basic event, USE data or estimates for the most similar equipment available, adjusting if necessary to account for differences. Alternatively, USE expert judgment and document the rationale behind the choice of parameter values.		

Table 2-2.6-5 Supporting Requirements for HLR-DA-D (Cont'd)

The parameter estimates shall be based on relevant generic industry and plant-specific evidence. Where feasible, generic and plant-specific evidence shall be integrated using acceptable methods to obtain plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty (HLR-DA-D).

Index No. DA-D	Capability Category I	Capability Category II	Capability Category III
DA-D3	CALCULATE a point estimate and CHARACTERIZE the uncertainty for the basic event probabilities. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	CALCULATE a mean value for the parameters used to calculate the probabilities of the significant basic events. PROVIDE the probabilistic representation of the uncertainty of the parameter estimates of the significant basic events. Acceptable methods include Bayesian updating or expert judgment. For the nonsignificant basic events, CALCULATE a point estimate and CHARACTERIZE the uncertainty for those basic event probabilities. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	CALCULATE a mean value for the parameters used to calculate the probabilities of the basic events. PROVIDE the probabilistic representation of the uncertainty of the parameter estimates of the basic events. Acceptable methods include Bayesian updating or expert judgment.
DA-D4	No requirement for use of Bayesian approach.	When the Bayesian approach is used to derive a distribution and mean value of a parameter, ENSURE that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: *(a)* confirmation that the Bayesian updating does not produce a posterior distribution with a single bin histogram *(b)* examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes *(c)* examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate *(d)* confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being considered *(e)* confirmation of the reasonableness of the posterior distribution mean value	

Table 2-2.6-5 Supporting Requirements for HLR-DA-D (Cont'd)

The parameter estimates shall be based on relevant generic industry and plant-specific evidence. Where feasible, generic and plant-specific evidence shall be integrated using acceptable methods to obtain plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty (HLR-DA-D).

Index No. DA-D	Capability Category I	Capability Category II	Capability Category III
DA-D5	USE the Beta-factor approach (i.e., the screening approach in NUREG/CR-5485 [4]) or an equivalent for estimating common cause failure (CCF) parameters.	USE one of the following models for estimating CCF parameters for significant CCF basic events: *(a)* Alpha Factor Model *(b)* Basic Parameter Model *(c)* Multiple Greek Letter Model *(d)* Binomial Failure Rate Model JUSTIFY the use of alternative methods (i.e., provide evidence of peer review or verification of the method that demonstrates its acceptability).	USE one of the following models for estimating CCF parameters: *(a)* Alpha Factor Model *(b)* Basic Parameter Model *(c)* Multiple Greek Letter Model *(d)* Binomial Failure Rate Model JUSTIFY the use of alternative methods (i.e., provide evidence of peer review or verification of the method that demonstrates its acceptability).
DA-D6	USE generic CCF beta factors or equivalent. ENSURE that the beta factors are evaluated in a manner consistent with the component boundaries.	USE CCF failure probabilities consistent with available plant experience. ESTIMATE the CCF probabilities in a manner consistent with the component boundaries.	USE realistic CCF probabilities consistent with available plant-specific data, supported by plant-specific screening and mapping of industry-wide data for significant common-cause events. An example approach is provided in NUREG/CR-5485 [2-4]. ESTIMATE the CCF probabilities in a manner consistent with the component boundaries.
DA-D7	If screening of generic event data is performed for plant-specific estimation, ENSURE that screening is performed on both the CCF events and the independent failure events in the database used to generate the CCF parameters.		

Table 2-2.6-5 Supporting Requirements for HLR-DA-D (Cont'd)

The parameter estimates shall be based on relevant generic industry and plant-specific evidence. Where feasible, generic and plant-specific evidence shall be integrated using acceptable methods to obtain plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty (HLR-DA-D).

Index No. DA-D	Capability Category I	Capability Category II	Capability Category III
DA-D8	If modifications to plant design or operating practice lead to a condition where past data are no longer representative of current performance, LIMIT the use of old data: *(a)* If the modification involves new equipment or a practice where generic parameter estimates are available, USE the generic parameter estimates updated with plant-specific data as it becomes available for unique design or operational features; or *(b)* If the modification is unique to the extent that generic parameter estimates are not available and only limited experience is available following the change, then ANALYZE the impact of the change and ASSESS the hypothetical effect on the historical data to determine to what extent the data can be used.	If modifications to plant design or operating practice lead to a condition where past data are no longer representative of current performance, LIMIT the use of old data: *(a)* If the modification involves new equipment or a practice where generic parameter estimates are available, USE the generic parameter estimates updated with plant-specific data as it becomes available for significant basic events; or *(b)* If the modification is unique to the extent that generic parameter estimates are not available and only limited experience is available following the change, then ANALYZE the impact of the change and ASSESS the hypothetical effect on the historical data to determine to what extent the data can be used.	If modifications to plant design or operating practice lead to a condition where past data are no longer representative of current performance, LIMIT the use of old data: *(a)* If the modification involves new equipment or a practice where generic parameter estimates are available, USE the generic parameter estimates updated with plant-specific data as it becomes available; or *(b)* If the modification is unique to the extent that generic parameter estimates are not available and only limited experience is available following the change, then ANALYZE the impact of the change and ASSESS the hypothetical effect on the historical data to determine to what extent the data can be used.

Table 2-2.6-6 Supporting Requirements for HLR-DA-E

Documentation of the data analysis shall be consistent with the applicable supporting requirements (HLR-DA-E).

Index No. DA-E	Capability Category I	Capability Category II	Capability Category III
DA-E1	DOCUMENT the data analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
DA-E2	DOCUMENT the processes used for data parameter definition, grouping, and collection including parameter selection and estimation, including the inputs, methods, and results. For example, this documentation typically includes *(a)* system and component boundaries used to establish component failure probabilities *(b)* the model used to evaluate each basic event probability *(c)* sources for generic parameter estimates *(d)* the plant-specific sources of data *(e)* the time periods for which plant-specific data were gathered *(f)* justification for exclusion of any data *(g)* the basis for the estimates of common cause failure probabilities, including justification for screening or mapping of generic and plant-specific data *(h)* the rationale for any distributions used as priors for Bayesian updates, where applicable *(i)* parameter estimate including the characterization of uncertainty, as appropriate		
DA-E3	DOCUMENT the sources of model uncertainty and related assumptions (as identified in Requirements QU-E1 and QU-E2) associated with the data analysis.		

2-2.7 QUANTIFICATION (QU)

2-2.7.1 Objectives

The objectives of the quantification element are to provide an estimate of CDF based upon the plant-specific core damage scenarios, in such a way that

(a) the results reflect the design, operation, and maintenance of the plant

(b) significant contributors to CDF are identified such as initiating events, accident sequences, and basic events (equipment unavailability and human failure events)

(c) dependencies are accounted for

(d) uncertainties are understood

Table 2-2.7-1 High Level Requirements for Quantification (QU)

Designator	Requirement
HLR-QU-A	The level 1 quantification shall quantify core damage frequency and shall support the quantification of LERF.
HLR-QU-B	The quantification shall use appropriate models and codes, and shall account for method-specific limitations and features.
HLR-QU-C	Model quantification shall determine that all identified dependencies are addressed appropriately.
HLR-QU-D	The quantification results shall be reviewed, and significant contributors to CDF, such as initiating events, accident sequences, and basic events (equipment unavailabilities and human failure events), shall be identified. The results shall be traceable to the inputs and assumptions made in the PRA.
HLR-QU-E	Uncertainties in the PRA results shall be characterized. Sources of model uncertainty and related assumptions shall be identified, and their potential impact on the results understood.
HLR-QU-F	Documentation of the quantification shall be consistent with the applicable supporting requirements.

Table 2-2.7-2 Supporting Requirements for HLR-QU-A

The level 1 quantification shall quantify core damage frequency and shall support the quantification of LERF (HLR-QU-A).

Index No. QU-A	Capability Category I	Capability Category II	Capability Category III
QU-A1	INTEGRATE the accident sequences, system models, data, and HRA in the quantification process for each initiating-event group, accounting for system dependencies, to arrive at accident sequence frequencies.		
QU-A2	ESTIMATE the frequencies of the individual sequences in a manner consistent with the estimation of total CDF to identify significant accident sequences/cutsets and confirm that the logic is appropriately reflected. The estimates may be accomplished by using either fault tree linking or event trees with conditional split fractions.		
QU-A3	CALCULATE a point estimate CDF using the point estimate values for the initiating-event frequencies, HEPs, and basic event probabilities.	ESTIMATE the mean CDF based on the mean values of the significant input parameters, and using the point estimates for nonsignificant parameters. ENSURE that the state-of-knowledge correlation between event frequencies or probabilities is taken into account when the state-of-knowledge correlation is significant [Note (1)].	CALCULATE the mean CDF by propagating the uncertainty distributions of the input parameters. ENSURE that the state-of-knowledge correlation between event frequencies or probabilities is taken into account.
QU-A4	SELECT a method that is capable of discriminating the contributors to the CDF commensurate with the level of detail in the model.		
QU-A5	INCLUDE recovery actions in the quantification process in applicable sequences and cutsets (see Requirements HR-H1, HR-H2, and HR-H3).		

NOTE:
(1) When the probabilities of a number of basic events are estimated by using the same data, the probabilities of the events will be identical. When an uncertainty analysis is performed by using a Monte Carlo sampling approach, the same sample value should be used for each basic event probability, since the state of knowledge about the parameter value is the same for each event. This is called the state of knowledge correlation, and it results in a mean value for the joint probability that is larger than the product of the mean values of the event probabilities. This result is most important for cutsets that contain multiple basic events whose probabilities are based on the same data, and in particular when the uncertainty on the parameter value is large. It has been found to be significant in cutsets contributing to ISLOCA frequency that involve rupture of multiple valves; for example, see reference [2-12].

Table 2-2.7-3 Supporting Requirements for HLR-QU-B

The quantification shall use appropriate models and codes, and shall account for method-specific limitations and features (HLR-QU-B).

Index No. QU-B	Capability Category I	Capability Category II	Capability Category III
QU-B1	PERFORM quantification using computer codes that have been demonstrated to generate appropriate results when compared to those from accepted algorithms. IDENTIFY method-specific limitations and features that could impact the results.		
QU-B2	TRUNCATE accident sequences and associated system models at a sufficiently low cutoff value that dependencies associated with significant cutsets or accident sequences are not eliminated. NOTE: Truncation should be carefully assessed in cases where cutsets are merged to create a solution (e.g., where system level cutsets are merged to create sequence level cutsets).		
QU-B3	ESTABLISH truncation limits by an iterative process of demonstrating that the overall model results converge and that no significant accident sequences are inadvertently eliminated. For example, convergence can be considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in CDF, and the final change is less than 5%.		
QU-B4	Where cutsets are the means used in quantification, USE the minimal cutset upper bound or an exact solution. The rare event approximation may be used when basic event probabilities are below 0.1.		
QU-B5	Fault tree linking and some other modeling approaches may result in circular logic that must be broken before the model is solved. BREAK the circular logic appropriately. Guidance for breaking logic loops is provided in NUREG/CR-2728 [2-13]. When resolving circular logic, DO NOT INTRODUCE unnecessary conservatisms or nonconservatisms.		
QU-B6	INCLUDE system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF. This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation and includes the treatment of transfers among event trees where the "successes" may not be transferred between event trees.		
QU-B7	IDENTIFY cutsets (or sequences) containing mutually exclusive events in the results.		
QU-B8	CORRECT cutsets containing mutually exclusive events by either *(a)* developing logic to eliminate mutually exclusive situations, or *(b)* deleting cutsets containing mutually exclusive events		
QU-B9	When using logic flags, SET logic flag events to either TRUE or FALSE (instead of setting the event probabilities to 1.0 or 0.0), as appropriate for each accident sequence, prior to the generation of cutsets.		
QU-B10	If modules, subtrees, or split fractions are used to facilitate the quantification, USE a process that allows *(a)* identification of shared events *(b)* correct formation of modules that are truly independent *(c)* results interpretation based on individual events within modules (e.g., risk significance)		

Table 2-2.7-4 Supporting Requirements for HLR-QU-C

Model quantification shall determine that all identified dependencies are addressed appropriately (HLR-QU-C).

Index No. QU-C	Capability Category I	Capability Category II	Capability Category III
QU-C1	IDENTIFY cutsets with multiple HFEs that potentially impact significant accident sequences/cutsets by requantifying the PRA model with HEP values set to values that are sufficiently high that the cutsets are not truncated. The final quantification of these post-initiator HFEs may be done at the cutset level or saved sequence level.		
QU-C2	ASSESS the degree of dependency between the HFEs in the cutset or sequence in accordance with Requirements HR-D5 and HR-G7.		
QU-C3	When linking event trees, TRANSFER the sequence characteristics (e.g., failed equipment, flag settings) that impact the logic or quantification of the subsequent accident development, as well as the sequence frequency. For example, sequence characteristics can be transferred to another event tree by using the appropriate cutsets.		

Table 2-2.7-5 Supporting Requirements for HLR-QU-D

The quantification results shall be reviewed, and significant contributors to CDF (and LERF), such as initiating events, accident sequences, and basic events (equipment unavailabilities and human failure events) shall be identified. The results shall be traceable to the inputs and assumptions made in the PRA (HLR-QU-D).

Index No. QU-D	Capability Category I	Capability Category II	Capability Category III
QU-D1	REVIEW a sample of the significant accident sequences/cutsets sufficient to determine that the logic of the cutset or sequence is correct.		
QU-D2	REVIEW the results of the PRA for modeling consistency (e.g., event sequence model's consistency with systems models and success criteria) and operational consistency (e.g., plant configuration, procedures, and plant-specific and industry experience).		
QU-D3	REVIEW results to determine that the flag event settings, mutually exclusive event rules, and recovery rules yield logical results.		
QU-D4	No requirements to compare results to those from similar plants.	COMPARE results to those from similar plants and IDENTIFY causes for significant differences. For example: Why is LOCA a large contributor for one plant and not another?	
QU-D5	REVIEW a sampling of nonsignificant accident cutsets or sequences to determine they are reasonable and have physical meaning.		
QU-D6	IDENTIFY significant contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors.	IDENTIFY significant contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors. When evaluating the significance of contributors, INCLUDE contributors to the occurrence of both initiating events and event mitigation failures.	
QU-D7	REVIEW the importance of components and basic events to determine that they make logical sense.		

Table 2-2.7-6 Supporting Requirements for HLR-QU-E

Uncertainties in the PRA results shall be characterized. Sources of model uncertainty and related assumptions shall be identified, and their potential impact on the results understood (HLR-QU-E).

Index No. QU-E	Capability Category I	Capability Category II	Capability Category III
QU-E1	IDENTIFY sources of model uncertainty.		
QU-E2	IDENTIFY assumptions made in the development of the PRA model.		
QU-E3	CHARACTERIZE the uncertainty interval of the CDF results by specifying or discussing the range of the uncertainty, consistent with the characterization of parameter uncertainties (see Requirements IE-C15, HR-D6, HR-G8, and DA-D3).	ESTIMATE the uncertainty interval of the CDF results, taking into account those model uncertainties explicitly characterized by a probability distribution. PROPAGATE uncertainties in such a way that the state-of-knowledge correlation between event frequencies or probabilities is taken into account when the state-of-knowledge correlation is significant.	For the CDF results, PROPAGATE the parameter uncertainties (see Requirements IE-C15, HR-D6, HR-G8, and DA-D3) and those model uncertainties explicitly characterized by a probability distribution, using the Monte Carlo approach or other comparable means. PROPAGATE uncertainties in such a way that the state-of-knowledge correlation between event frequencies or probabilities is taken into account.
QU-E4	For each source of model uncertainty and related assumption identified in Requirements QU-E1 and QU-E2, respectively, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).		

Table 2-2.7-7 Supporting Requirements for HLR-QU-F

The documentation of model quantification shall be consistent with the applicable supporting requirements (HLR-QU-F).

Index No. QU-F	Capability Category I	Capability Category II	Capability Category III
QU-F1	DOCUMENT the model quantification in a manner that facilitates PRA applications, upgrades, and peer review.		
QU-F2	DOCUMENT the model integration process including any recovery analysis, and the results of the quantification including uncertainty analyses. For example, documentation typically includes *(a)* records of the process/results when adding non-recovery terms as part of the final quantification *(b)* records of the cutset review process *(c)* a general description of the quantification process including accounting for systems successes, the truncation values used, how recovery and post-initiator HFEs are applied *(d)* the process and results for establishing the truncation screening values for final quantification demonstrating that convergence towards a stable result was achieved *(e)* the total plant CDF and contributions from the different initiating events and accident classes *(f)* the accident sequences and their contributing cutsets *(g)* equipment or human actions that are the key factors in causing the accident sequences to be nonsignificant *(h)* the uncertainty distribution (as specified for each Capability Category in Requirement QU-E3) for the total CDF *(i)* importance measure results *(j)* a list of mutually exclusive events eliminated from the resulting cutsets and their bases for elimination *(k)* asymmetries in quantitative modeling to provide application users the necessary understanding of the reasons such asymmetries are present in the model *(l)* the process used to illustrate the computer code(s) used to perform the quantification will yield correct results process		
QU-F3	DOCUMENT the significant contributors (such as initiating events, accident sequences, basic events) to CDF in the PRA results summary.	DOCUMENT the significant contributors (such as initiating events, accident sequences, basic events) to CDF in the PRA results summary. DESCRIBE significant accident sequences or functional failure groups.	
QU-F4	DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in Requirement QU-E4).		
QU-F5	DOCUMENT limitations in the quantification process that would impact applications.		
QU-F6	DOCUMENT the quantitative definitions used for significant basic event, significant cutset, and significant accident sequence. If definitions other than those in 1-2.2 are used, JUSTIFY the alternatives.		

2-2.8 LERF ANALYSIS (LE)

2-2.8.1 Objectives

The objectives of the LERF analysis element are to identify and quantify the contributors to large early releases, based upon the plant-specific core damage scenarios, in such a way that

(a) the methodology is clear and consistent with the Level 1 evaluation, and creates an adequate transition from Level 1

(b) operator actions, mitigation systems, and phenomena that can alter sequences are appropriately included in the LERF event tree structure and sequence definition

(c) dependencies are reflected in the accident sequence model structure, if necessary

(d) success criteria are available to support the individual function successes, mission times, and time windows for operator actions and equipment recovery for each critical safety function modeled in the accident sequences

(e) end states are clearly defined to be large early release or non–large early release

NOTE: In a number of cases, the LERF supporting requirements include references to applicable supporting requirements in other sections of the Standard (e.g., technical elements AS, SC, SY, HR, DA, and QU). The requirements in other sections of this Standard were primarily written in the context of CDF. Where applicable to LERF, these requirements should be interpreted in the context of LERF. New requirements that are only applicable to LERF are identified in this section.

Table 2-2.8-1 High Level Requirements for LERF Analysis (LE)

Designator	Requirement
HLR-LE-A	Core damage sequences shall be grouped into plant damage states based on their accident progression attributes.
HLR-LE-B	The accident progression analyses shall include an evaluation of contributors (e.g., phenomena, equipment failures, and human actions) to a large early release.
HLR-LE-C	The accident progression analysis shall include identification of those sequences that would result in a large early release.
HLR-LE-D	The accident progression analyses shall include an evaluation of the containment structural capability for those containment challenges that would result in a large early release.
HLR-LE-E	The frequency of different containment failure modes leading to a large early release shall be quantified and aggregated.
HLR-LE-F	The quantification results shall be reviewed, and significant contributors to LERF, such as plant damage states, containment challenges, and failure modes, shall be identified. Sources of model uncertainty and related assumptions shall be identified, and their potential impact on the results understood.
HLR-LE-G	The documentation of LERF analysis shall be consistent with the applicable supporting requirements.

Table 2-2.8-2 Supporting Requirements for HLR-LE-A

Core damage sequences with similar accident progression attributes shall be grouped into plant damage states based on their accident progression attributes (HLR-LE-A).

Index No. LE-A	Capability Category I	Capability Category II	Capability Category III
LE-A1	IDENTIFY those physical characteristics at the time of core damage that can influence LERF. Examples include *(a)* RCS pressure (high RCS pressure can result in high pressure melt ejection) *(b)* status of emergency core coolant systems (failure in injection can result in a dry cavity and extensive core concrete interaction) *(c)* status of containment isolation (failure of isolation can result in an unscrubbed release) *(d)* status of containment heat removal *(e)* containment integrity (e.g., vented, bypassed, or failed) *(f)* steam generator pressure and water level (PWRs) *(g)* status of containment inerting (BWRs)		
LE-A2	IDENTIFY the accident sequence characteristics that lead to the physical characteristics identified in Requirement LE-A1. Examples include *(a)* type of initiator *(1)* transients can result in high RCS pressure *(2)* LOCAs usually result in lower RCS pressure *(3)* ISLOCAs, SGTRs can result in containment bypass. *(b)* status of electric power: loss of electric power can result in loss of ECC injection *(c)* status of containment safety systems such as sprays, fan coolers, igniters, or venting systems: operability of containment safety systems determines status of containment heat removal References [2-14] and [2-15] provide example lists of typical characteristics.		
LE-A3	IDENTIFY how the physical characteristics identified in Requirement LE-A1 and the accident sequence characteristics identified in Requirement LE-A2 are addressed in the LERF analysis. For example, *(a)* which characteristics are addressed in the level 1 event trees *(b)* which characteristics, if any, are addressed in bridge trees *(c)* which characteristics, if any, are addressed in the containment event trees JUSTIFY any characteristics identified in Requirement LE-A1 or LE-A2 that are excluded from the LERF analysis.		
LE-A4	PROVIDE a method to explicitly account for the characteristics of Requirements LE-A1 and LE-A2 and ensure that dependencies between the Level 1 and Level 2 models are properly treated. Examples include treatment in Level 2, expanding Level 1, construction of a bridge tree, transfer of the information via PDS, or a combination of these.		
LE-A5	DEFINE plant damage states in a manner consistent with Requirements LE-A1, LE-A2, LE-A3, and LE-A4.		

Table 2-2.8-3 Supporting Requirements for HLR-LE-B

The accident progression analysis shall include an evaluation of contributors (e.g., phenomena, equipment failures, and human actions) to a large early release.

Index No. LE-B	Capability Category I	Capability Category II	Capability Category III
LE-B1	IDENTIFY LERF contributors from the set identified in Table 2-2.8-9. An acceptable approach for identifying contributors that could influence LERF for the various containment types is contained in NUREG/CR-6595 [2-16]. INCLUDE, as appropriate, unique plant issues as determined by expert judgment and/or engineering analyses.	IDENTIFY LERF contributors from the set identified in Table 2-2.8-9. INCLUDE, as appropriate, unique plant issues as determined by expert judgment and/or engineering analyses.	INCLUDE LERF contributors sufficient to support development of realistic accident progression sequences. ADDRESS those contributors identified by IDCOR [2-14] and NUREG-1150 [2-15]. INCLUDE, as appropriate, unique plant issues as determined by expert judgment and/or engineering analyses.
LE-B2	ESTIMATE the containment challenges (e.g., temperature, pressure loads, debris impingement) resulting from contributors identified in Requirement LE-B1 using applicable generic analyses. Where applicable generic analyses are not available, conservative plant-specific analyses may be used. An acceptable alternative is the approach in NUREG/CR-6595 [2-16].	ESTIMATE the containment challenges (e.g., temperature, pressure loads, debris impingement) resulting from contributors identified in Requirement LE-B1 using applicable generic or plant-specific analyses for significant containment challenges. USE conservative treatment or a combination of conservative and realistic treatment for nonsignificant containment challenges. If generic calculations are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	CALCULATE the containment challenges (e.g., temperature, pressure loads, debris impingement) resulting from contributors identified in Requirement LE-B1 in a realistic manner. REVIEW differential pressure loadings on the RCS and support vessel capabilities during vessel failure and blowdown, in order to address whether RCS motions may impact containment integrity.
LE-B3	USE supporting engineering analyses in accordance with the applicable requirements of Table 2-2.3-2.		

Table 2-2.8-4 Supporting Requirements for HLR-LE-C

The accident progression analysis shall include identification of those sequences that would result in a large early release.

Index No. LE-C	Capability Category I	Capability Category II	Capability Category III
LE-C1	DEVELOP accident sequences to a level of detail to account for the potential contributors identified in Requirement LE-B1 and analyzed in Requirement LE-B2. Containment event trees developed in NUREG/CR-6595 [2-16] (with plant-specific modifications, if needed) are acceptable.	DEVELOP accident sequences to a level of detail to account for the potential contributors identified in Requirement LE-B1 and analyzed in Requirement LE-B2. COMPARE the containment challenges analyzed in Requirement HLR-LE-B with the containment structural capability analyzed in Requirement HLR-LE-D and identify accident progressions that have the potential for a large early release. JUSTIFY any generic or plant-specific calculations or references used to categorize releases as non-LERF contributors based on release magnitude or timing. NUREG/CR-6595, App. A [2-16] provides a discussion and examples of LERF source terms.	DEVELOP accident sequences to a level of detail to account for the potential contributors identified in Requirement LE-B1 and analyzed in Requirement LE-B2. COMPARE the containment challenges analyzed in Requirement HLR-LE-B with the containment structural capability analyzed in Requirement HLR-LE-D and identify accident progressions that have the potential for a large early release. CALCULATE source terms for accident progressions that have the potential for large early releases.
LE-C2	INCLUDE conservative treatment of feasible operator actions following the onset of core damage. An acceptable conservative treatment of operator actions is provided in the event trees of NUREG/CR-6595 [2-16].	INCLUDE realistic treatment of feasible operator actions following the onset of core damage consistent with applicable procedures, e.g., EOPs/SAMGs, proceduralized actions, or Technical Support Center guidance.	
LE-C3	No requirement to address repair.	REVIEW significant accident progression sequences resulting in a large early release to determine if repair of equipment can be credited. JUSTIFY credit given for repair [i.e., ensure that plant conditions do not preclude repair and actuarial data exists from which to estimate the repair failure probability (see Requirements SY-A24 and DA-C15)]. AC power recovery based on generic data applicable to the plant is acceptable.	

Table 2-2.8-4 Supporting Requirements for HLR-LE-C (Cont'd)

The accident progression analysis shall include identification of those sequences that would result in a large early release.

Index No. LE-C	Capability Category I	Capability Category II	Capability Category III
LE-C4	INCLUDE model logic necessary to provide accident progression sequences resulting in a large early release. Containment event trees developed in NUREG/CR-6595 [2-16] (with plant-specific modifications, if needed) are acceptable.	INCLUDE model logic necessary to provide a realistic estimation of the significant accident progression sequences resulting in a large early release. INCLUDE mitigating actions by operating personnel, effect of fission product scrubbing on radionuclide release, and expected beneficial failures in significant accident progression sequences. PROVIDE technical justification (by plant-specific or applicable generic calculations demonstrating the feasibility of the actions, scrubbing mechanisms, or beneficial failures) supporting the inclusion of any of these features.	INCLUDE model logic necessary to provide a realistic estimation of the accident progression sequences resulting in a large early release. INCLUDE mitigating actions by operating personnel, effect of fission product scrubbing on radionuclide release, and expected beneficial failures. PROVIDE technical justification (by plant-specific or applicable generic calculations demonstrating the feasibility of the actions, scrubbing mechanisms, or beneficial failures) for the inclusion of any of these features.
LE-C5	USE appropriate conservative, generic analyses/evaluations of system success criteria that are applicable to the plant.	USE appropriate realistic generic or plant-specific analyses for system success criteria for the significant accident progression sequences. USE conservative or a combination of conservative and realistic system success criteria for nonrisk significant accident progression sequences.	USE appropriate realistic plant-specific system success criteria.
LE-C6	DEVELOP system models that support the accident progression analysis in a manner consistent with the applicable requirements for 2-2.4, as appropriate for the level of detail of the analysis.		
LE-C7	In crediting HFEs that support the accident progression analysis, USE the applicable requirements of 2-2.5 as appropriate for the level of detail of the analysis.		
LE-C8	INCLUDE accident sequence dependencies in the accident progression sequences in a manner consistent with the applicable requirements of 2-2.2, as appropriate for the level of detail of the analysis.		
LE-C9	DO NOT TAKE CREDIT for continued equipment operation or operator actions in adverse environments (i.e., beyond equipment qualification limits). An acceptable approach is NUREG/CR-6595 [2-16].	JUSTIFY any credit given for equipment survivability or human actions under adverse environments.	

Table 2-2.8-4 Supporting Requirements for HLR-LE-C (Cont'd)

The accident progression analysis shall include identification of those sequences that would result in a large early release.

Index No. LE-C	Capability Category I	Capability Category II	Capability Category III
LE-C10	No requirement; credit for equipment survivability or human actions in adverse environments is precluded by Requirement LE-C9.	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions during accident progression that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for nonsignificant accident progression sequences.	INCLUDE containment environmental impacts on continued operation of equipment and operator actions in a realistic manner based on engineering analyses.
LE-C11	DO NOT TAKE CREDIT for continued operation of equipment and operator actions that could be impacted by containment failure. An acceptable alternative is the approach in NUREG/CR-6595 [2-16].	JUSTIFY any credit given for equipment survivability or human actions that could be impacted by containment failure.	
LE-C12	No requirement; credit for post-containment failure operability of equipment or operator actions is precluded by Requirement LE-C11.	REVIEW significant accident progression sequences resulting in a large early release to determine if engineering analyses can support continued equipment operation or operator actions after containment failure that could reduce LERF. USE conservative or a combination of conservative and realistic treatment for non-significant accident progression sequences.	INCLUDE containment failure impacts on continued operation of equipment and operator actions in a realistic manner based on engineering analyses.
LE-C13	EVALUATE containment bypass events in a conservative manner. DO NOT TAKE CREDIT for scrubbing. An acceptable alternative is the approach in NUREG/CR-6595 [2-16].	PERFORM a containment bypass analysis in a realistic manner. JUSTIFY any credit taken for scrubbing (i.e., provide an engineering basis for the decontamination factor used).	

Table 2-2.8-5 Supporting Requirements for HLR-LE-D

The accident progression analysis shall include an evaluation of the containment structural capability for those containment challenges that would result in a large early release.

Index No. LE-D	Capability Category I	Capability Category II	Capability Category III
LE-D1	ESTIMATE the containment ultimate capacity for the containment challenges that result in a large early release. USE a conservative containment capacity analysis for the significant containment challenges. If generic assessments formulated for similar plants are used, JUSTIFY applicability to the plant being evaluated. Analyses may consider use of similar containment designs or estimating containment capacity based on design pressure and a conservative multiplier relating containment design pressure and median ultimate failure pressure. Quasi-static containment capability evaluations are acceptable unless hydrogen concentrations are expected to result in potential detonations. INCLUDE such considerations for small volume containments, e.g., ice-condenser type. An acceptable alternative is the approach in NUREG/CR-6595 [2-16].	ESTIMATE the containment ultimate capacity for the containment challenges that result in a large early release. PERFORM a realistic containment capacity analysis for the significant containment challenges. USE a conservative or a combination of conservative and realistic evaluation of containment capacity for nonsignificant containment challenges. If generic calculations are used in support of the assessment, JUSTIFY applicability to the plant being evaluated. Analyses may consider use of similar containment designs or estimating containment capacity based on design pressure and a realistic multiplier relating containment design pressure and median ultimate failure pressure. Quasi-static containment capability evaluations are acceptable unless hydrogen concentrations are expected to result in potential detonations. INCLUDE such considerations for small volume containments, e.g., ice-condenser type.	CALCULATE the containment ultimate capacity for the containment challenges that result in a large early release. PERFORM a realistic containment capacity analysis for containment challenges by using plant-specific input. ESTIMATE static and dynamic failure capabilities, as appropriate.
LE-D2	EVALUATE the impact of containment seals, penetrations, hatches, drywell heads (BWRs), and vent piping bellows and INCLUDE as potential containment failures, as required. An acceptable alternative is the approach in NUREG/CR-6595 [2-16].	EVALUATE the impact of containment seals, penetrations, hatches, drywell heads (BWRs), and vent pipe bellows and INCLUDE as potential containment failures, as required. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	EVALUATE the impact of the capacity of containment seals, penetrations, hatches, drywell heads (BWRs), and vent piping bellows using plant-specific analyses to determine if their capacities limit or reduce the containment ultimate capacity evaluated in Requirement LE-D1.

Table 2-2.8-5 Supporting Requirements for HLR-LE-D (Cont'd)

The accident progression analysis shall include an evaluation of the containment structural capability for those containment challenges that would result in a large early release.

Index No. LE-D	Capability Category I	Capability Category II	Capability Category III
LE-D3	When containment failure location [Note (1)] affects the classification of the accident progression as a large early release, SPECIFY failure location based on a conservative containment assessment that accounts for plant-specific features. JUSTIFY applicability of generic and other analyses. Analyses may consider comparison with similar failure locations in similar containment designs. An acceptable alternative is the approach in NUREG/CR-6595 [2-16].	When containment failure location [Note (1)] affects the event classification of the accident progression as a large early release, SPECIFY failure location based on a realistic containment assessment that accounts for plant-specific features. If generic analyses are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.	When containment failure location [Note (1)] affects the event classification of the accident progression as a large early release, SPECIFY failure location based on a realistic plant-specific containment assessment.
LE-D4	USE a conservative evaluation of interfacing system failure probability for significant accident progression sequences resulting in a large early release. If generic analyses generated for similar plants are used, JUSTIFY applicability to the plant being evaluated. Analyses may consider comparison with similar interfacing systems in similar containment designs.	PERFORM a realistic interfacing system failure probability analysis for the significant accident progression sequences resulting in a large early release. USE a conservative or a combination of conservative and realistic evaluation of interfacing system failure probability for nonsignificant accident progression sequences resulting in a large early release. INCLUDE behavior of piping relief valves, pump seals, and heat exchangers at applicable temperature and pressure conditions.	PERFORM a realistic interfacing system failure probability analysis for the accident progression sequences resulting in a large early release. USE plant-specific input. INCLUDE behavior of piping, relief valves, pump seals, and heat exchangers at applicable temperature and pressure conditions. ESTIMATE static and dynamic failure capabilities, as appropriate.

Table 2-2.8-5 Supporting Requirements for HLR-LE-D (Cont'd)

The accident progression analysis shall include an evaluation of the containment structural capability for those containment challenges that would result in a large early release.

Index No. LE-D	Capability Category I	Capability Category II	Capability Category III
LE-D5	USE a conservative evaluation of secondary side isolation capability for significant accident progression sequences caused by steam generator tube failure resulting in a large early release. If generic analyses generated for similar plants are used, JUSTIFY applicability to the plant being evaluated. Analyses may consider comparison with similar isolation capability in similar containment designs.	PERFORM a realistic secondary side isolation capability analysis for the significant accident progression sequences caused by steam generator tube failure resulting in a large early release. USE a conservative or a combination of conservative and realistic evaluation of secondary side isolation capability for nonsignificant accident progression sequences resulting in a large early release. JUSTIFY applicability to the plant being evaluated. Analyses may consider realistic comparison with similar isolation capability in similar containment designs.	PERFORM a realistic secondary side isolation capability analysis for the accident progression sequences caused by steam generator tube failure resulting in a large early release. INCLUDE behavior of relief and isolation valves at applicable temperatures and pressure conditions.

Table 2-2.8-5 Supporting Requirements for HLR-LE-D (Cont'd)

The accident progression analysis shall include an evaluation of the containment structural capability for those containment challenges that would result in a large early release.

Index No. LE-D	Capability Category I	Capability Category II	Capability Category III
LE-D6	PERFORM a conservative analysis of thermally induced SGTR that includes plant-specific procedures. An acceptable alternative is the approach in NUREG/CR-6595 [2-16].	PERFORM an analysis of thermally induced SGTR that includes plant-specific procedures and design features and conditions that could impact tube failure. An acceptable approach is one that arrives at plant-specific split fractions by selecting the steam generator tube conditional failure probabilities based on NUREG-1570 [2-17] or similar evaluation for induced steam generator failure of a similarly designed steam generators and loop piping. SELECT failure probabilities based on *(a)* RCS and steam generator post-accident conditions to sufficient to describe the important risk outcomes *(b)* secondary side conditions including plant-specific treatment of main steam safety valve and atmospheric dump valve failures JUSTIFY assumptions and selection of key inputs. An acceptable justification can be obtained by the extrapolation of the information in NUREG-1570 [2-17] to obtain plant-specific models, use of reasonably bounding assumptions, or performance of sensitivity studies indicating low sensitivity to changes in the range in question.	PERFORM a realistic analysis of thermally induced SGTR that includes plant-specific procedures and key design features. USE appropriate computer codes to calculate the plant-specific conditions.

Table 2-2.8-5 Supporting Requirements for HLR-LE-D (Cont'd)

The accident progression analysis shall include an evaluation of the containment structural capability for those containment challenges that would result in a large early release.

Index No. LE-D	Capability Category I	Capability Category II	Capability Category III
LE-D7	PERFORM containment isolation analysis in a conservative manner. INCLUDE consideration of both the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions.	PERFORM containment isolation analysis in a realistic manner for the significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative or realistic treatment for the nonsignificant accident progression sequences resulting in a large early release. INCLUDE consideration of both the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions.	PERFORM containment isolation analysis in a realistic manner. INCLUDE consideration of both the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions.

NOTE:
(1) Containment failures below ground level may not be a large early release even if the timing is early. Such failures may arise as a result of failures in the basemat region.

Table 2-2.8-6 Supporting Requirements for HLR-LE-E

The frequency of different containment failure modes leading to a large early release shall be quantified and aggregated (HLR-LE-E).

Index No. LE-E	Capability Category I	Capability Category II	Capability Category III
LE-E1	SELECT parameter values for equipment and operator response in the accident progression analysis in a manner consistent with the applicable requirements of 2-2.5 and 2-2.6 including consideration of the severe accident plant conditions, as appropriate for the level of detail of the analysis.		
LE-E2	USE conservative parameter estimates to characterize accident progression phenomena. A conservative data set for some key parameters is included in NUREG/CR-6595 [2-16].	USE realistic parameter estimates to characterize accident progression phenomena for significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative and realistic estimates for nonsignificant accident progression sequences resulting in a large early release.	USE realistic parameter estimates to characterize accident progression phenomena.
LE-E3	INCLUDE as LERF contributors potential large early release sequences in a conservative manner; i.e., designate early containment failures, bypass sequences, and isolation failures as LERF contributors. The large early release sequences identified in NUREG/CR-6595 [2-16] provide an acceptable alternative.	INCLUDE as LERF contributors potential large early release sequences identified from the results of the accident progression analysis of Requirement HLR-LE-C except those large early release sequences justified as non-LERF contributors in Requirement LE-C1.	INCLUDE as LERF contributors potential large early release sequences from the results of the accident progression analysis by carrying out the appropriate source term calculations.
LE-E4	QUANTIFY LERF in a manner consistent with the applicable requirements of Tables 2-2.7-2 and 2-2.7-3 [Note (1)].		

NOTE:
(1) The supporting requirements referenced in these tables are written in CDF language. Under this requirement, the applicable quantification requirements in Table 2-2.7-2 should be interpreted based on the approach taken for the LERF model. For example, Requirement QU-A2 addresses the calculation of point estimate/mean CDF. Under this requirement, the application of Requirement QU-A2 would apply to the quantification of point estimate/mean LERF.

Table 2-2.8-7 Supporting Requirements for HLR-LE-F

The quantification results shall be reviewed, and significant contributors to LERF, such as plant damage states, containment challenges and failure modes, shall be identified. Sources of uncertainty shall be identified and their potential impact on the results characterized.

Index No. LE-F	Capability Category I	Capability Category II	Capability Category III
LE-F1	IDENTIFY the significant contributors to large early releases (e.g., plant damage states, containment failure modes).	PERFORM a quantitative evaluation of the relative contribution to LERF from plant damage states and significant LERF contributors from Table 2-2.8-9.	
LE-F2	REVIEW contributors for reasonableness (e.g., to ensure excessive conservatisms have not skewed the results, level of plant-specificity is appropriate for significant contributors, etc.).		
LE-F3	CHARACTERIZE the LERF sources of model uncertainty and related assumptions, in a manner consistent with the applicable requirements of Table 2-2.7-6 [Note (1)].		

NOTE:
(1) The supporting requirements referenced in this table are written in CDF language. The applicable requirements of Table 2-2.7-6 should be interpreted based on LERF, including characterizing the sources of model uncertainty and related assumptions associated with the applicable contributors from Table 2-2.8-9. For example, Requirement QU-D6 addresses the significant contributors to CDF. Under this requirement, the contributors would be identified based on their contribution to LERF.

Table 2-2.8-8 Supporting Requirements for HLR-LE-G

Documentation of the LERF analysis shall be consistent with the applicable supporting requirements (HLR-LE-G).

Index No. LE-G	Capability Category I	Capability Category II	Capability Category III
LE-G1	DOCUMENT the LERF analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
LE-G2	DOCUMENT the process used to identify plant damage states and accident progression contributors, define accident progression sequences, evaluate accident progression analyses of containment capability, and quantify and review the LERF results. For example, this documentation typically includes *(a)* the plant damage states and their attributes, as used in the analysis *(b)* the method used to bin the accident sequences into plant damage states *(c)* the containment failure modes, phenomena, equipment failures, and human actions considered in the development of the accident progression sequences and the justification for their inclusion or exclusion from the accident progression analysis *(d)* the treatment of factors influencing containment challenges and containment capability, as appropriate for the level of detail of the analysis *(e)* the basis for the containment capacity analysis including the identification of containment failure location(s), if applicable *(f)* the accident progression analysis sequences considered in the containment event trees *(g)* the basis for parameter estimates *(h)* the model integration process including the results of the quantification *(i)* the uncertainty distribution (as specified for each Capability Category in Requirement QU-E3) for the total LERF		
LE-G3	DOCUMENT the significant contributors to LERF.	DOCUMENT the relative contribution of contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, containment failure modes) to LERF.	
LE-G4	DOCUMENT the sources of model uncertainty and related assumptions (as identified in Requirement LE-F3) associated with the LERF analysis, including results and important insights from the characterization of uncertainties.		
LE-G5	DOCUMENT limitations in the LERF analysis that would impact applications.		
LE-G6	DOCUMENT the quantitative definition used for significant accident progression sequence. If a definition other than that in 1-2.2 is used, JUSTIFY the alternative.		

Table 2-2.8-9 LERF Contributors to Be Considered

LERF Contributor	Containment Design				
	Large Dry Subatmospheric	Ice Condenser	BWR Mark I	BWR Mark II	BWR Mark III
Containment isolation failure	X	X	X	X	X [Note (1)]
Containment Bypass					
(a) ISLOCA	X	X	X	X	X
(b) SGTR	X	X	. . .	. . .	. . .
(c) Induced SGTR	X	X	. . .	. . .	. . .
Energetic containment failures					
(a) HPME	X	X	X	X	X
(b) Hydrogen combustion	. . .	X	X [Note (3)]	X [Note (3)]	X
(c) Core debris impingement	[Note (2)]	X	X	X	. . .
Steam explosion [Note (4)]	. . .	. . .	X	X	X
Shell melt-through	. . .	. . .	X (if applicable)	X (if applicable)	. . .
Pressure suppression bypass [Note (5)]	. . .	X	X	X	X
RPV and/or containment venting	X (if applicable)	X (if applicable)	X	X	X
Isolation condenser tube rupture	. . .	X (if applicable)	. . .	. . .	
Vacuum breaker failure	. . .	. . .	X	X	X
Hydrodynamic loads under severe accident conditions	. . .	. . .	X	X	X
Containment flooding	. . .	. . .	X	X	. . .
In-vessel recovery	X	X	X	X	X
ATWS-induced failure	. . .	. . .	X	X	X

GENERAL NOTE: Combinations of contributors should also be considered where appropriate. For example, in a BWR Mark I or II, the combination of containment flooding and containment venting should be considered.

NOTES:
(1) drywell (DW) isolation failure
(2) applicable to steel shell designs only
(3) during de-inerted operation only
(4) steam explosion challenges are of low probability for PWRs
(5) ice bed bypass for ice condensers and suppression pool bypass for BWR

Section 2-3
Peer Review for Internal Events At-Power

2-3.1 PURPOSE

This Section provides requirements for peer review of a Level 1 and LERF Internal Events At-Power PRA.

NEI-00-02 [2-18] provides an example of an acceptable review methodology; however, the differences between the supporting requirements in Part 2 of this Standard and the supporting requirements of Appendix B of NEI-00-02 shall be evaluated. This evaluation shall be documented.

NEI-05-04 [2-19] provides another example of an acceptable review methodology. NEI-05-04 references the Technical Requirements of Part 2 of this Standard.

2-3.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATIONS

In addition to the general requirements in Section 1-6, the peer review team shall have knowledge and collective experience in the areas of systems engineering, plant operations, fault and event tree modeling, thermal-hydraulic analysis, data analysis, HRA, and severe accident phenomenology as applicable to the scope of the review. The team members assigned to review the HRA and LERF Analysis shall have experience specific to these areas and be capable of recognizing the impact of plant-specific features on the analysis.

2-3.3 REVIEW OF PRA ELEMENTS TO CONFIRM THE METHODOLOGY

2-3.3.1 Initiating-Event Analysis (IE)

The entire initiating-event analysis shall be reviewed.

2-3.3.2 Accident Sequence Analysis (AS)

A review shall be performed on selected accident sequences. The portion of the accident sequences selected for review typically includes

(a) accident sequence model for a balance-of-plant transient

(b) the accident sequence model containing LOOP/SBO considerations

(c) accident sequence model for a loss of a support system initiating event

(d) LOCA accident sequence model

(e) ISLOCA accident sequence model

(f) the SGTR accident sequence model (for PWRs only)

(g) ATWS accident sequence model

2-3.3.3 Success Criteria (SC)

A review shall be performed on success criteria definitions and evaluations.

The portion of the success criteria selected for review typically includes

(a) the definition of core damage used in the success criteria evaluations and the supporting bases

(b) the conditions corresponding to a safe, stable state

(c) the core and containment response conditions used in defining LERF and supporting bases

(d) the core and containment system success criteria used in the PRA for mitigating each modeled initiating event

(e) the generic bases (including assumptions) used to establish the success criteria of systems credited in the PRA and the applicability to the modeled plant

(f) the plant-specific bases (including assumptions) used to establish the system success criteria of systems credited in the PRA

(g) calculations performed specifically for the PRA, for each computer code used to establish core cooling or decay heat removal success criteria and accident sequence timing

(h) calculations performed specifically for the PRA, for each computer code used to establish support system success criteria (e.g., a room heat-up calculation used to establish room cooling requirements or a load shedding evaluation used to determine battery life during an SBO)

(i) expert judgments used in establishing success criteria used in the PRA

2-3.3.4 Systems Analysis (SA)

A review shall be performed on the systems analysis.

The portion of system models selected for review typically includes a sample of the systems where failure contributes to significant sequences (CDF or LERF), including

(a) different models reflecting different levels of detail

(b) front-line system for each mitigating function (e.g., reactivity control, coolant injection, and decay heat removal)

(c) each major type of support system (e.g., electrical power, cooling water, instrument air, and HVAC)

(d) complex system with variable success criteria (e.g., a cooling water system requiring different numbers of pumps for success dependent upon whether nonsafety loads are isolated)

2-3.3.5 Human Reliability Analysis (HR)

A review shall be performed on the human reliability analysis.

The portion of the HRA selected for review typically includes a sample of the human failure events whose failure contributes to significant sequences (CDF or LERF), including

(a) the selection and implementation of any screening HEPs used in the PRA

(b) post-accident HFEs and associated HEPs

(c) pre-initiator HFEs and associated HEPs for both instrumentation miscalibration and failure of equipment

(d) HEPs for the same function but under the influence of different PSFs

(e) HEPs for dependent human actions, including dependencies of multiple HEPs in the same sequence

(f) HEPs less than 1E-4

(g) HFEs and associated HEPs involving remote actions in harsh environments

(h) the selection and identification of the HFEs associated with the HEPs for the above review topics

2-3.3.6 Data Analysis (DA)

A review shall be performed on the data analysis.

The portion of the data analysis selected for review typically includes

(a) data values and associated component boundary definitions for component failure modes (including those with high importance values) contributing to the CDF or LERF calculated in the PRA

(b) common cause failure values

(c) the numerator and denominator for one data value for each major failure mode (e.g., failure to start, failure to run, and test and maintenance unavailabilities)

(d) equipment repair and recovery data

2-3.3.7 Quantification (QU)

Level 1 quantification results shall be reviewed.

The portion of Level 1 quantification process selected for review typically includes

(a) appropriateness of the computer codes used in the quantification

(b) the truncation values and process

(c) the recovery analysis

(d) model asymmetries and sensitivity studies

(e) the process for generating modules (if used)

(f) logic flags (if used)

(g) the solution of logic loops (if appropriate)

(h) the summary and interpretation of results

2-3.3.8 LERF Analysis (LE)

The LERF analysis and the Level 1/LERF interface process shall be reviewed.

2-3.3.8.1 The portion of Level 1 and LERF interface process selected for a detailed review typically includes

(a) accident characteristics chosen for carryover to LERF analysis (and for binning of PDSs if PDS methods were used)

(b) interface mechanism used

(c) CDF carryover

2-3.3.8.2 The portion of the LERF analysis selected for review typically includes

(a) the LERF analysis method

(b) demonstration that the phenomena that impact radionuclide release characterization of LERF have been appropriately considered

(c) human action and system success considering adverse conditions that would exist following core damage

(d) the sequence mapping

(e) evaluation of containment performance under severe accident conditions

(f) the definition and bases for LERF

(g) inclusion in the containment event tree of the function events; necessary to achieve a safe stable containment end state

(h) sensitivity analysis

(i) the containment responses calculations, performed specifically for the PRA, for the significant plant damage states

Section 2-4
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PART 3 (a)
REQUIREMENTS FOR INTERNAL FLOOD AT-POWER PRA (b)

Section 3-1
Overview of Internal Flood At-Power PRA Requirements

3-1.1 PRA SCOPE

This Part establishes technical requirements for a Level 1 and large early release frequency (LERF) analysis of the internal flood hazard group while at-power.

3-1.2 COORDINATION WITH OTHER PARTS OF THIS STANDARD

This Part is intended to be used together with Parts 1 and 2 of this Standard. An internal-events at-power PRA developed in accordance with Part 2 is the starting point for the development of the flood-induced accident sequence model.

3-1.3 INTERNAL FLOOD EVENTS SCOPE

The scope of the flooding events covered in this Part includes all flood scenarios originating within the plant boundary. It does not include floods resulting from external events (e.g., weather, or offsite events such as upstream dam rupture).

The overall objective of the internal flood PRA is to ensure that the impact of internal flood as the cause of either an accident or a system failure is evaluated in such a way that

(a) the flood sources within the plant that could flood plant locations or create adverse conditions (e.g., spray, elevated temperature, humidity, pressure, pipe whip, jet impingement) that could damage mitigative plant equipment are identified

(b) the flood-induced accident sequences that contribute to the core damage frequency and large early release frequency are identified and quantified[1]

[1] In this Part of the Standard, "internal flood" is used as a modifier (e.g., "internal flood-induced") in several high-level and supporting requirements as a shorthand way of indicating that in meeting the requirement, consideration should be given to applicable flood-induced causes of SSC failure (e.g., submersion, spray, elevated temperature, humidity, pressure, pipe whip, jet impingement) and resulting flood-induced failure mechanisms. Applicability of the various flood-induced causes of SSC failure and resulting failure mechanisms to a particular requirement may need to be determined based on consideration of related supporting requirements.

Section 3-2
Internal Flood PRA Technical Elements and Requirements

A separate set of technical elements and associated requirements is provided for this initiating hazard group in this Standard because there are many different sources of flooding throughout the plant, with different potential impact on SSCs. Thus, there is the potential for a relatively large number of individual flood scenarios and flood-induced accident sequences with unique spatial dependencies. Although it is optional, as stated in Note (1) of Table 3-2.3-2, some degree of screening out of flood-induced scenarios and accident sequences is typically employed in analyzing risk from internal floods, so that, although the high level and supporting requirements are written in a discrete manner, the requirements are not necessarily presented in sequential order of application and, in some cases, must be considered jointly, so that screening out is performed appropriately. Thus, in determining the degree to which a particular supporting requirement is to be met, it is necessary to consider the degree to which other, related requirements (some of which may be under other high level requirements) are being addressed. Screening out is typically employed at the flood area, flood source, or flood scenario level with the understanding that screening out of areas and sources considers the relevant flood scenarios associated with the area or source.

An internal flood PRA need not be performed at a uniform level of detail. The analyses performed to support the screening out of physical analysis units may be performed at a less rigorous completeness level than analyses performed for flood areas, flood sources, and/or flood scenarios that are not screened out and hence require further analysis. An iterative process is also common in internal flood PRAs. Those physical analysis units that represent the higher risk contributors may be analyzed repeatedly, each time incorporating additional detail for specific aspects of the analysis [e.g., flood source and propagation modeling, credit for drains or mitigation, refinements to the internal flood PRA plant response model, and the human reliability analysis (HRA)]. At any stage, the additional detail may allow for the screening out of a physical analysis unit. It is intended that this Standard allow for analysis flexibility in this regard. As such, the level of detail and resolution for lower risk and/or screened-out physical analysis units may be lower than for higher risk and unscreened physical analysis units without affecting the capability of the internal flood PRA to identify significant flood-induced accident sequences. For example, a service building containing numerous flood sources may be treated as a single physical analysis unit [see subpara. (a), Internal Flood Plant Partitioning, below] and analyzed for screening purposes. If the building can be screened out (e.g., it contains no equipment modeled in the other portions of the PRA and there are no propagation paths to other buildings), then the overall categorization of the internal flood PRA is unaffected. Similarly, the requirements for developing specific internal flood scenarios, detailed HRA, etc., are not needed for screened-out physical analysis units and may not be needed for lower-risk unscreened physical analysis units as long as the overall validity of the final results is unaffected.

In accordance with the application process described in Section 1-3, the Capability Categories required for various aspects of the internal flood PRA are determined by the intended PRA application and may not be uniform across all aspects of the internal flood PRA.

The following is a short description of each technical PRA element included in the internal flood PRA process.

(a) Internal Flood Plant Partitioning (IFPP). This element defines the physical boundaries of the analysis (i.e., the locations within the plant where flood scenarios are postulated), and divides the various volumes within that boundary into physical analysis units referred to as flood areas.

(b) Internal Flood Source Identification and Characterization (IFSO). The various potential sources of floods and equipment spray within the plant are identified, along with the mechanisms resulting in flood or spray from these sources, and a characterization of the flood/spray sources (e.g., amount of liquid, flow rates, etc.) is made.

(c) Internal Flood Scenarios (IFSN). A set of flood scenarios is developed, relating flood source, propagation path(s), and affected equipment.

(d) Internal Flood-Induced Initiating Events (IFEV). The expected plant response(s) to the selected set of flood scenarios is determined, and an accident sequence, from the internal-events at-power PRA that is reasonably representative of this response, is selected for each scenario.

(e) Internal Flood Accident Sequences and Quantification (IFQU). The CDF and LERF results for the internal flood plant response model sequences are quantified.

Example approaches to performing each of the above elements of an internal flood PRA may be found in EPRI 1019194 [3-1].

3-2.1 INTERNAL FLOOD PLANT PARTITIONING

3-2.1.1 Objectives

The objective of internal flood plant partitioning is to identify plant areas where internal floods can be initiated in such a way that plant-specific physical layouts and separations are accounted for. The plant partitioning provides a framework for the definition of flood areas, flood scenarios, and flood-induced accident sequences.

Table 3-2.1-1 High Level Requirements for Internal Flood Plant Partitioning (IFPP)

Designator	Requirement
HLR-IFPP-A	A reasonably complete set of flood areas of the plant shall be identified.
HLR-IFPP-B	Documentation of the internal flood plant partitioning shall be consistent with the applicable supporting requirements.

Table 3-2.1-2 Supporting Requirements for HLR-IFPP-A

A reasonably complete set of flood areas of the plant shall be identified (HLR-IFPP-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFPP-A1	DEFINE flood areas by dividing the plant into physically separate areas where a flood area is viewed as a portion of a building or plant that is separated from other areas by barriers that delay, restrict, or prevent the propagation of floods to adjacent areas.		
IFPP-A2	*[This requirement has been deleted.]*		
IFPP-A3	For multi-unit sites with shared systems or structures, INCLUDE multi-unit areas, if applicable.		
IFPP-A4	USE plant information sources that reflects the as-built as-operated plant to support development of flood areas.		
IFPP-A5 [Note (1)]	CONDUCT plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify *(a)* spatial information needed for the development of flood areas *(b)* plant design features credited in defining flood areas		
IFPP-A6	IDENTIFY assumptions and sources of uncertainty associated with plant partitioning.		

NOTE:
(1) Walkdown(s) may be done in conjunction with Requirements IFSO-A6, IFSN-A17, and IFQU-A11.

Table 3-2.1-3 Supporting Requirements for HLR-IFPP-B

Documentation of the internal flood plant partitioning shall be consistent with the applicable supporting requirements (HLR-IFPP-B).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFPP-B1	DOCUMENT the internal flood plant partitioning in a manner that facilitates PRA applications, upgrades, and peer review.		
IFPP-B2	DOCUMENT the process used to identify flood areas. For example, this documentation typically includes *(a)* flood areas used in the analysis and the reason for eliminating areas from further analysis *(b)* any walkdowns performed in support of the plant partitioning		
IFPP-B3	DOCUMENT sources of model uncertainty and related assumptions (as identified in Requirement IFPP-A6) associated with the internal flood plant partitioning.		

3-2.2 INTERNAL FLOOD SOURCE IDENTIFICATION AND CHARACTERIZATION

3-2.2.1 Objectives

The objective of internal flood source identification and characterization is to identify and characterize the plant-specific sources of internal floods that could lead to core damage. Flood source characterization, which includes identification of sources of flooding, equipment failure modes, and associated flood mechanisms, is a necessary prerequisite to the definition of flood scenarios.

Table 3-2.2-1 High Level Requirements for Internal Flood Source Identification and Characterization (IFSO)

Designator	Requirement
HLR-IFSO-A	The potential flood sources in the flood areas, and their associated flood mechanisms, shall be identified and characterized in a manner sufficient to define flood scenarios.
HLR-IFSO-B	Documentation, identification, and characterization of flood sources shall be consistent with the applicable supporting requirements.

Table 3-2.2-2 Supporting Requirements for HLR-IFSO-A

The potential flood sources in the flood areas and their associated flood mechanisms shall be identified and characterized in a manner sufficient to define flood scenarios (HLR-IFSO-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFSO-A1	For each flood area, IDENTIFY the potential sources of flooding [Note (1)]. INCLUDE *(a)* equipment (e.g., piping, valves, pumps) located in the area that is connected to fluid systems (e.g., circulating water system, service water system, component cooling water system, fire protection system, feedwater system, condensate and steam systems, reactor coolant system, and other high energy lines) *(b)* plant internal sources of flooding (e.g., tanks or pools) located in the flood area *(c)* plant external sources of flooding (e.g., reservoirs or rivers) that are connected through some system or structure within the plant boundary		
IFSO-A2	For multi-unit sites with shared systems or structures, INCLUDE any sources with potential multi-unit or cross-unit impacts.		
IFSO-A3	When choosing to screen, SCREEN OUT flood areas with none of the potential sources of flooding listed in Requirements IFSO-A1 and IFSO-A2 unless the area is part of a flood propagation path from such sources.		
IFSO-A4	For each potential flood source, IDENTIFY the flooding mechanisms that would result in a release of water or steam from the flood source. INCLUDE *(a)* failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, and seals *(b)* human-induced mechanisms that could lead to overfilling tanks or the diversion of flow-through openings created to perform maintenance *(c)* inadvertent actuation of a fire suppression system *(d)* other events resulting in a release into the flood area		
IFSO-A5	For each source and its identified failure mechanism, IDENTIFY the characteristic of release and the capacity of the source. INCLUDE *(a)* a characterization of the breach, including flood type (e.g., leak, rupture, spray) *(b)* applicable range of flow rates *(c)* capacity of source (e.g., gallons of water) *(d)* the pressure and temperature of the source		
IFSO-A6 [Note (2)]	CONDUCT plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to determine or verify the location of flood sources and in-leakage pathways.		
IFSO-A7	IDENTIFY assumptions and sources of uncertainty associated with flood source identification and characterization.		

NOTES:
(1) Sources of flooding are typically expected to be water, and the requirements are generally written in terms of sources of water, but other fluid sources should also be considered as part of this Standard.
(2) Walkdown(s) may be done in conjunction with Requirements IFPP-A5, IFSN-A17, and IFQU-A11, if applicable.

Table 3-2.2-3 Supporting Requirements for HLR-IFSO-B

Documentation, identification, and characterization of flood sources shall be consistent with the applicable supporting requirements (HLR-IFSO-B).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFSO-B1	DOCUMENT the internal flood sources in a manner that facilitates PRA applications, upgrades, and peer review.		
IFSO-B2	DOCUMENT the process used to identify flood sources. For example, this documentation typically includes *(a)* flood sources identified in the analysis, rules used to screen out these sources, and the resulting list of sources to be further examined *(b)* screening criteria used in the analysis *(c)* calculations or other analyses used to support or refine the flooding evaluation *(d)* any walkdowns performed in support of the identification or screening of flood sources		
IFSO-B3	DOCUMENT sources of model uncertainty and related assumptions (as identified in Requirement IFSO-A7) associated with the flood sources.		

3-2.3 INTERNAL FLOOD SCENARIO DEVELOPMENT

3-2.3.1 Objectives

The objective of internal flood scenario development is to identify the plant-specific internal flood scenarios that could lead to core damage. It is important that the enumeration of flood scenarios be performed in a systematic matter so that potentially significant flood scenarios are not overlooked. Important elements of the flood scenario development include identification of the flood area and source; flood rate; flood volume; flood barriers; flood propagation paths; response of operators to terminate the flood and mitigate its consequences; impact of the flood on plant performance and damage to plant equipment; and other flood characteristics needed to determine risk impacts.

Table 3-2.3-1 High Level Requirements for Internal Flood Scenario Development (IFSN)

Designator	Requirement
HLR-IFSN-A	The flood scenarios shall be developed for each flood source by identifying the propagation path(s) of the source and the affected SSCs.
HLR-IFSN-B	Documentation of the flood scenarios shall be consistent with the applicable supporting requirements.

Table 3-2.3-2 Supporting Requirements for HLR-IFSN-A

The flood scenarios shall be developed for each flood source by identifying the propagation path(s) of the source and the affected SSCs (HLR-IFSN-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFSN-A1 [Note (1)]	For each defined flood area and each flood source, IDENTIFY the flood propagation path from the flood source to its area(s) of accumulation.		
IFSN-A2 [Note (1)]	For each defined flood area and each flood source, IDENTIFY plant design features that have the ability to terminate or contain the flood propagation. INCLUDE the presence of *(a)* flood alarms *(b)* flood dikes, curbs, sumps (i.e., physical structures that allow for the accumulation and retention of water) *(c)* drains (i.e., physical structures that can function as drains) *(d)* sump pumps, spray shields, water-tight doors *(e)* blowout panels or dampers with automatic or manual operation capability		
IFSN-A3 [Note (1)]	For each defined flood area and each flood source, IDENTIFY those automatic actuations or operator responses that have the ability to terminate or contain the flood propagation.		
IFSN-A4	ESTIMATE the capacity of the drains and the amount of water retained by sumps, berms, dikes, and curbs. INCLUDE these factors in estimating flood volumes and evaluating SSC impacts from flooding.		
IFSN-A5	For each flood area not screened out by using other internal flood requirements (e.g., Requirements IFSO-A3 and IFSN-A12), IDENTIFY the SSCs located in each defined flood area and along flood propagation paths that are modeled in the internal-events at-power PRA model as being required to respond to an initiating event or whose failure would challenge normal plant operation, and are susceptible to flood. For each identified SSC, IDENTIFY, for the purpose of determining its susceptibility per Requirement IFSN-A6, spatial location in the area and any flooding mitigative features (e.g., shielding, flood or spray capability ratings).		
IFSN-A6	For the SSCs identified in Requirement IFSN-A5, IDENTIFY the susceptibility of each SSC in a flood area to submergence and spray failure mechanisms.	For the SSCs identified in Requirement IFSN-A5, IDENTIFY the susceptibility of each SSC in a flood area to submergence, spray, humidity, and condensation failure mechanisms. For flood scenarios involving a high energy line break (HELB), IDENTIFY the susceptibility of each SSC identified in Requirement IFSN-A5 to jet impingement, pipe whip, temperature, and pressure failure mechanisms. JUSTIFY any determination that SSCs as identified in Requirement IFSN-A5 within the flood area are not susceptible to flood-induced failure mechanisms.	
IFSN-A7	When determining susceptibility of SSCs to flood-induced failure mechanisms (see Requirement IFSN-A6), INCLUDE the operability of SSCs identified in Requirement IFSN-A5 with respect to internal flood impacts only if the SSC functionality is supported by one or an appropriate combination of the following: *(a)* test or operational data *(b)* engineering analysis *(c)* expert judgment		

Table 3-2.3-2 Supporting Requirements for HLR-IFSN-A (Cont'd)

The flood scenarios shall be developed for each flood source by identifying the propagation path(s) of the source and the affected SSCs (HLR-IFSN-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFSN-A8	No requirement for inter-area propagation given that flood areas are independent (see Requirement IFPP-A1).	IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via backflow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads.	IDENTIFY inter-area propagation through the normal flow path from one area to another via drain lines; and areas connected via backflow through drain lines involving failed check valves, pipe and cable penetrations (including cable trays), doors, stairwells, hatchways, and HVAC ducts. INCLUDE potential for structural failure (e.g., of doors or walls) due to flooding loads, and the potential for barrier unavailability, including maintenance activities.
IFSN-A9	PERFORM necessary engineering calculations for flood rate, time for water to reach susceptible equipment, and the structural capacity of SSCs in accordance with the applicable requirements described in 2-2.3, particularly those associated with Requirements HLR-SC-B and HLR-SC-C in Part 2.		
IFSN-A10	DEVELOP flood scenarios by examining the equipment and relevant plant features in the flood area and areas in potential propagation paths, giving credit for appropriate flood mitigation systems or operator actions, and identifying susceptible SSCs. INCLUDE, in the development of scenarios, the flood area, flood source, flood rate, flood propagation path, flood impact on plant SSCs, and human actions considered in flood initiation, mitigation, and termination.		
IFSN-A11	For multi-unit sites with shared systems or structures, INCLUDE multi-unit flood scenarios.		
IFSN-A12 [Note (2)]	When choosing to screen, SCREEN OUT flood areas qualitatively where flooding of the flood area does not cause a flood-induced initiating event or a need for immediate plant shutdown, *and* any of the following applies: *(a)* The flood area (including adjacent areas where flood sources can propagate) contains no equipment modeled in the PRA or is required to function to permit operability of modeled equipment. *(b)* The flood area has no flood sources sufficient to cause failure (e.g., through spray, immersion, or other applicable cause) of the equipment identified in Requirement IFSN-A5 (including equipment in adjacent areas where floods may propagate). *(c)* Requirement IFSN-A13 or IFSN-A14 is met for all flood sources within the flood area. The screening out of a flood area does not eliminate the need to consider that flood area as a part of a flood propagation path between or among other flood areas. DO NOT USE failure of a barrier against inter-area propagation to justify screening (i.e., do not credit such failures as a means of beneficially draining the area). JUSTIFY any other qualitative screening criteria.		

Table 3-2.3-2 Supporting Requirements for HLR-IFSN-A (Cont'd)

The flood scenarios shall be developed for each flood source by identifying the propagation path(s) of the source and the affected SSCs (HLR-IFSN-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFSN-A13 [Notes (2) and (3)]	When choosing to screen, for those flood areas not screened out under Requirement IFSN-A12, SCREEN OUT flood sources qualitatively where flooding of the flood area, based on the limiting flood defined for that source, does not cause an initiating event nor a need for immediate plant shutdown due to loss of function of one or more SSCs due to the flood, *and* each of the following applies: *(a)* The flood area contains flood mitigation systems (e.g., drains or sump pumps) capable of preventing unacceptable flood levels. *(b)* The nature of the limiting flood does not cause failure of the flood mitigation systems or SSCs that are needed to prevent core damage or large early release due to a flood-induced failure mechanism. *(c)* There is no propagation to another flood area. If these criteria are met for all the sources within a given flood area, then the flood area may be screened out. DO NOT INCLUDE mitigation systems for screening out flood sources unless there is a basis for crediting the capability and reliability of the flood mitigation system(s).		

Table 3-2.3-2 Supporting Requirements for HLR-IFSN-A (Cont'd)

The flood scenarios shall be developed for each flood source by identifying the propagation path(s) of the source and the affected SSCs (HLR-IFSN-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFSN-A14 [Notes (2) and (3)]	When choosing to screen, for flood areas and sources not screened out under Requirement IFSN-A12 or IFSN-A13, SCREEN OUT flood sources qualitatively by using potential human response actions if all the following can be shown: *(a)* Flood indication is available in the control room. *(b)* The flood source can be isolated. *(c)* The amount of time until safe-shutdown equipment is damaged is significantly greater than the expected time available for human response actions to be performed for the most challenging flood (i.e., most challenging to human response actions defined for that source). If these criteria are met for all the sources within a given flood area, then the flood area may be screened out.	When choosing to screen, for flood areas and sources not screened out under Requirement IFSN-A12 or IFSN-A13, SCREEN OUT flood sources qualitatively by using potential human response actions if all the following can be shown: *(a)* Flood indication is available in the control room. *(b)* The flood source can be isolated. *(c)* The human response action can be performed with high reliability for the most challenging flood (i.e., most challenging to human response actions defined for that source). High reliability is established by demonstrating, for example, that the actions are procedurally directed; that an adequate time interval is available for response; that the area is accessible; and that there is sufficient manpower available to perform the actions for the most challenging flood. If these criteria are met for all the sources within a given flood area, then the flood area may be screened out.	DO NOT SCREEN OUT flood sources or areas qualitatively based on reliance on operator action to prevent challenges to normal plant operations.
IFSN-A15	*[This requirement has been deleted.]*		

Table 3-2.3-2 Supporting Requirements for HLR-IFSN-A (Cont'd)

The flood scenarios shall be developed for each flood source by identifying the propagation path(s) of the source and the affected SSCs (HLR-IFSN-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFSN-A16	*[This requirement has been deleted.]*		
IFSN-A17 [Note (4)]	CONDUCT plant walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify *(a)* SSCs located within each defined flood area *(b)* flood/spray/other applicable mitigative features of the SSCs located within each defined flood area (e.g., drains, shields) *(c)* flood propagation paths		
IFSN-A18	IDENTIFY assumptions and sources of uncertainty associated with flood scenario development.		

NOTE:
(1) The process of defining flood propagation paths is iterative; hence, Requirements IFSN-A1, IFSN-A2, and IFSN-A3 would normally be applied in parallel and not necessarily sequentially.
(2) The use and extent of screening out of flood areas and sources is optional. To facilitate an efficient qualitative screening process, conservative representations of the flood impact may be used for screening purposes. Examples of conservative representations include bounding assumptions on flood rate, flood volume, barrier effectiveness, mitigation, and SSC susceptibility to flood-induced failure mechanisms. The qualitative screening criteria for flood sources and areas in Requirements IFSN-A13 and IFSN-A14 may be used in conjunction with conservative representations of scenarios. For areas and sources not screened out under Requirement IFSN-A12, IFSN-A13, or IFSN-A14, flood scenarios need to be defined. The SRs of Requirements HLR-IFEV-A and HLR-IFQU-A that apply to flood scenarios require a realistic representation and enumeration of flood scenarios unless otherwise noted.
(3) The wording of this requirement recognizes that, to facilitate an efficient screening process, flood sources and flood areas may be screened out prior to the task of enumerating all relevant flood scenarios for each source and area. When defining the limiting flood for Requirement IFSN-A13 and the most challenging flood for Requirement IFSN-A14, it is necessary to ensure that all the parameters relevant to flood scenario definition are considered and bounded.
(4) Walkdown(s) may be done in conjunction with Requirements IFPP-A5, IFSO-A6, and IFQU-A11.

Table 3-2.3-3 Supporting Requirements for HLR-IFSN-B

Documentation of the flood scenarios shall be consistent with the applicable supporting requirements (HLR-IFSN-B).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFSN-B1	DOCUMENT the flood scenario development in a manner that facilitates PRA applications, upgrades, and peer review.		
IFSN-B2	DOCUMENT the process used to identify flood scenarios. For example, this documentation typically includes *(a)* flood propagation paths and assumptions, calculations, or other bases for identifying and eliminating them *(b)* accident-mitigating features and barriers credited in the analysis, and associated justification *(c)* flooding scenarios considered, screened out, and retained *(d)* screening criteria used in the analysis *(e)* assumptions, justifications, and calculations used in the determination of flood-induced failure mechanisms (e.g., justification for the nonsusceptibility of SSCs to flood-induced failure mechanisms for modeled flood scenarios) *(f)* description of how the internal event analysis models were modified to model these remaining internal flood scenarios *(g)* calculations or other analyses used to support or refine the flooding evaluation *(h)* any walkdowns performed in support of the identification or screening of flood scenarios		
IFSN-B3	DOCUMENT sources of model uncertainty and related assumptions (as identified in Requirement IFSN-A18) associated with the flood scenarios.		

3-2.4 INTERNAL FLOOD-INDUCED INITIATING-EVENT ANALYSIS

3-2.4.1 Objectives

The objective of internal flood-induced event analysis is to identify the applicable flood-induced initiating event for each flood scenario that could lead to core damage, and to quantify the frequency of the flood-induced intiating event.

The requirements for flood-induced initiating-event analysis assume that flood scenarios that have not been screened out according to the requirements in 3-2.3 will be retained for incorporation into the PRA model to quantify flood-induced accident sequences. Those aspects of the flood scenario development that are necessary and sufficient to define the flood-induced initiating event are subject to the requirements of 3-2.4. Such aspects include the occurrence of the flood, the flood-induced failure mechanisms of SSCs damaged by the flood, and operator actions that may have been considered to prevent the SSC damage. If there are other aspects of the flood scenario development that were considered as a means of mitigating the consequences of the flood-induced initiating event, those aspects are included in the flood-induced accident sequence quantification and are subject to the requirements of 3-2.5.

Table 3-2.4-1 High Level Requirements for Flood-Induced Initiating-Event Analysis (IFEV)

Designator	Requirement
HLR-IFEV-A	Flood-induced initiating events shall be identified, and the frequencies of the flood scenarios resulting in those initiating events shall be estimated.
HLR-IFEV-B	Documentation of the internal flood-induced initiating-event analysis shall be consistent with the applicable supporting requirements.

Table 3-2.4-2 Supporting Requirements for HLR-IFEV-A

Flood-induced initiating events shall be identified, and the frequencies of the flood scenarios resulting in those initiating events shall be estimated.

Index No.	Capability Category I	Capability Category II	Capability Category III
IFEV-A1 *(formerly IFEV-A2)*	GROUP flood scenarios identified in Requirement IFSN-A10 only when *(a)* flood scenarios can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* flood scenarios are bounded by the worst-case impacts within the group	GROUP flood scenarios identified in Requirement IFSN-A10 only when *(a)* flood scenarios can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* flood scenarios are bounded by the worst-case impacts within the group *and* the grouping does not impact identification of significant accident sequences	GROUP flood scenarios identified in Requirement IFSN-A10 only when *(a)* flood scenarios can be considered similar in terms of plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* the grouping of flood scenarios does not impact identification of significant accident sequences, *and* impacts of each event are comparable to those of the remaining events in that group
IFEV-A2 *(formerly IFEV-A1)*	For each flood scenario or flood-scenario group defined according to Requirement IFEV-A1, IDENTIFY the corresponding initiating-event group per Requirements HLR-IE-A and HLR-IE-D in Part 2 and the flood-induced failure mechanisms of SSCs required to respond to the initiating event. INCLUDE the potential for a flood-induced transient or LOCA. If an appropriate initiating-event group does not exist, CREATE a new initiating-event group in accordance with Requirements HLR-IE-A and HLR-IE-D in Part 2.		
IFEV-A3	*[This requirement has been deleted.]*		
IFEV-A4	For multi-unit sites with shared systems or structures, INCLUDE multi-unit impacts on SSCs in the definition and grouping of flood-induced initiating events.		
IFEV-A5	ESTIMATE the frequency for each flood-induced initiating event or initiating-event group by using the applicable requirements in 2-2.1 while taking into account the probability of any mitigating features that have been used to define the flood-induced initiating event.		

Table 3-2.4-2 Supporting Requirements for HLR-IFEV-A (Cont'd)

Flood-induced initiating events shall be identified, and the frequencies of the flood scenarios resulting in those initiating events shall be estimated.

Index No.	Capability Category I	Capability Category II	Capability Category III
IFEV-A6 [Note (1)]	In estimating the flood-induced initiating-event frequencies, USE one or a combination of the following: *(a)* generic operating experience *(b)* pipe, component, and tank rupture failure rates from generic data sources *(c)* engineering judgment	COLLECT plant-specific information on plant design, operating practices, and conditions that may impact flood-induced initiating-event frequency (e.g., material condition of fluid systems, experience with water hammer, and maintenance-induced floods). In estimating the flood-induced initiating-event frequencies, USE one or a combination of the following: *(a)* generic and plant-specific operating experience *(b)* pipe, component, and tank rupture failure rates from generic data sources and plant-specific experience *(c)* engineering judgment for consideration of the plant-specific information collected	
IFEV-A7	ESTIMATE the frequency of human-induced floods during maintenance through application of available generic or plant-specific data or engineering judgment.		ESTIMATE the frequency of human-induced floods by using human reliability analysis techniques in evaluating plant-specific maintenance activities consistent with the applicable requirements for human reliability analysis in 2-2.5.
IFEV-A8	When choosing to screen, SCREEN OUT flood-induced initiating events or initiating-event groups if *(a)* the quantitative screening criteria in Requirement IE-C6 in Part 2, as applied to the flood-induced initiating-event groups, are met, or *(b)* the flood-induced initiating event affects only components in a single system, AND it can be shown that the product of the frequency of the flood-induced initiating event and the probability of SSC failure given the flood is 2 orders of magnitude lower than the product of the non-flooding frequency for the corresponding initiating event in the PRA and the random (non–flood-induced) failure probability of the same SSCs that are assumed failed by the flood.		
IFEV-A9	IDENTIFY assumptions and sources of uncertainty associated with flood-induced initiating-events analysis.		

NOTE:
(1) Generic examples of piping system failure rates for use in estimating flood-induced initiating-event frequencies may be found in EPRI 1021086 [3-2].

Table 3-2.4-3 Supporting Requirements for HLR-IFEV-B

Documentation of the internal flood-induced initiating-events analysis shall be consistent with the applicable supporting requirements (HLR-IFEV-B).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFEV-B1	DOCUMENT the internal flood-induced initiating-events analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
IFEV-B2	DOCUMENT the process used to identify applicable flood-induced initiating events. For example, this documentation typically includes *(a)* flood frequencies, component unreliabilities/unavailabilities, and human error probabilities used in the analysis (i.e., the data values unique to the internal flooding analysis) *(b)* calculations or other analyses used to support or refine the flooding evaluation *(c)* screening criteria used in the analysis		
IFEV-B3	Document sources of model uncertainty and related assumptions (as identified in Requirement IFEV-A9) associated with the internal flood-induced initiating events.		

3-2.5 INTERNAL FLOOD ACCIDENT SEQUENCES AND QUANTIFICATION

3-2.5.1 Objectives

The objective of internal flood accident sequences and quantification is to identify the internal flood-induced accident sequences and quantify the core damage and large early release frequencies. Flood scenarios developed according to the requirements of 3-2.3 are generally identified, grouped, and screened prior to their incorporation into a PRA model. Those flood scenarios that are not screened out need to be incorporated into the PRA model. In 3-2.4, these unscreened flood scenarios are modeled in terms of the flood-induced initiating event. The purpose of this section is to state the requirements for completing the PRA modeling of the flood scenarios in terms of flood-induced accident sequences.

Table 3-2.5-1 High Level Requirements for Internal Flood Accident Sequences and Quantification (IFQU)

Designator	Requirement
HLR-IFQU-A	Core damage frequency and large early release frequency due to flood-induced accident sequences shall be quantified.
HLR-IFQU-B	Documentation of the internal flood accident sequences and quantification shall be consistent with the applicable supporting requirements.

Table 3-2.5-2 Supporting Requirements for HLR-IFQU-A

Core damage frequency and large early release frequency due to flood-induced accident sequences shall be quantified (HLR-IFQU-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFQU-A1	For each flood scenario, REVIEW the accident sequences for the associated initiating-event group, identified per the requirements in 2-2.2, to confirm applicability of the accident sequence model. If appropriate accident sequences do not exist, MODIFY sequences as necessary to account for any unique accident sequences that could result from the flood scenario and associated flood-induced failure mechanisms or phenomena in accordance with the applicable requirements described in 2-2.2.		
IFQU-A2	MODIFY the systems analysis results obtained by using Requirements HLR-SY-B and HLR-SY-C in Part 2 to include flood-induced failure mechanisms identified according to Requirement IFSN-A6.		
IFQU-A3	When choosing to screen, SCREEN OUT a flood-induced accident sequence if the product of the flood-induced initiating-event or initiating-event-group frequency and a conservative estimate of the conditional core damage probability (CCDP) for the sequence, including any flood mitigation events not included in the flood-induced initiating-event frequency, is less than 10^{-8}/reactor-yr. DO NOT SCREEN OUT individual sequences if the frequency of a group of sequences with similar characteristics (e.g., similar flood scenario, initiating event, and accident sequence) exceeds this screening criterion.		When choosing to screen, SCREEN OUT a flood-induced accident sequence if the product of the flood-induced initiating-event or initiating-event-group frequency and the conditional core damage probability (CCDP) for the sequence, including any flood mitigation events not included in the flood-induced initiating-event frequency, is less than 10^{-9}/reactor-yr. DO NOT SCREEN OUT individual sequences if the frequency of a group of sequences with similar characteristics (e.g., similar flood scenario, initiating event, and accident sequence) exceeds this screening criterion.
IFQU-A4	If additional analysis of SSC data is required to support quantification of flood-induced accident sequences, PERFORM the analysis in accordance with the applicable requirements in 2-2.6.		
IFQU-A5	If additional human failure events (HFEs) are required to support quantification of flood-induced accident sequences, PERFORM the associated HRA in accordance with the applicable requirements described in 2-2.5.		
IFQU-A6	For all HFEs in the flood-induced accident sequences, INCLUDE the following flood scenario-specific impacts on performance-shaping factors for control room and ex-control room actions, as appropriate to the HRA methodology being used: *(a)* additional workload and stress (above that for similar sequences not caused by internal floods) *(b)* cue availability *(c)* effect of flood on accident sequence mitigation, required response, timing, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) *(d)* flood-specific job aids and training (e.g., procedures, training exercises)		

Table 3-2.5-2 Supporting Requirements for HLR-IFQU-A (Cont'd)

Core damage frequency and large early release frequency due to flood-induced accident sequences shall be quantified (HLR-IFQU-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFQU-A7	PERFORM internal flood-induced accident sequence quantification in accordance with the requirements described in 2-2.7 as applicable to flood-induced accident sequences.		
IFQU-A8	INCLUDE, in the quantification, event sequences comprised of failures caused by the flood and those due to independent causes, including equipment failures, unavailability due to maintenance, common cause failures, and other credible causes that may reduce the plant capabilities to mitigate the flood-induced initiating event.		
IFQU-A9	INCLUDE, in the quantification, both the direct effects of the flood (e.g., loss of cooling from a service water train due to an associated pipe rupture) and indirect effects such as submergence, jet impingement, and pipe whip, as applicable.		
IFQU-A10	For each flood scenario, REVIEW the LERF analysis to confirm applicability of the LERF sequences in accordance with the applicable requirements described in 2-2.8. If appropriate LERF sequences do not exist, MODIFY the LERF analysis as necessary to account for any unique flood-induced failure mechanisms or phenomena in accordance with the applicable requirements described in 2-2.8.		
IFQU-A11 [Note (1)]	CONDUCT walkdown(s) to verify the accuracy of information obtained from plant information sources and to obtain or verify inputs to the following quantifications of flood-induced accident sequences: *(a)* engineering analyses *(b)* human reliability analyses *(c)* spray or other applicable impact assessments *(d)* screening decisions		
IFQU-A12 [Note (2)]	IDENTIFY assumptions and sources of model and parameter uncertainty associated with flood-induced accident sequence quantification, including those assumptions and uncertainties identified in Requirements IFPP-A6, IFSO-A7, IFSN-A18, and IFEV-A9, as well as those identified in Requirements QU-E1 and QU-E2 in Part 2 for those parts of the accident sequences derived from the internal-events PRA model.		
IFQU-A13 [Note (2)]	CHARACTERIZE the uncertainty interval of the CDF results for flood-induced accident sequences by specifying or discussing the range of the uncertainty, consistent with the characterization of parameter uncertainties (see Requirements IE-C15, HR-D6, HR-G8, and DA-D3 in Part 2).	ESTIMATE the uncertainty interval of the CDF results for flood-induced accident sequences, taking into account those model uncertainties explicitly characterized by a probability distribution. PROPAGATE uncertainties in such a way that the state-of-knowledge correlation between event frequencies or probabilities is taken into account when the state-of-knowledge correlation is significant.	For the CDF results for flood-induced accident sequences, PROPAGATE the parameter uncertainties (see Requirements IE-C15, HR-D6, HR-G8, and DA-D3 in Part 2) and those model uncertainties explicitly characterized by a probability distribution, using the Monte Carlo approach or other comparable means. PROPAGATE uncertainties in such a way that the state-of-knowledge correlation between event frequencies or probabilities is taken into account.

Table 3-2.5-2 Supporting Requirements for HLR-IFQU-A (Cont'd)

Core damage frequency and large early release frequency due to flood-induced accident sequences shall be quantified (HLR-IFQU-A).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFQU-A14 [Note (2)]	For each source of model uncertainty and related assumption identified in Requirement IFQU-A12, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, introduction of a new initiating event).		

NOTES:
(1) Walkdown(s) may be done in conjunction with Requirements IFPP-A5, IFSO-A6, and IFSN-A17.
(2) In general, flood-induced accident sequences will be comprised of a combination of initiating events and basic events associated with
(a) flood-induced initiating events
(b) portions of the accident sequences derived from the internal-events PRA model (i.e., basic events that are independent of the flood scenarios but otherwise contribute to the accident sequence)
Hence, the sources of uncertainty that impact quantification include a combination of uncertainties associated with the flood scenarios and flood-induced initiating events as well as those that are carried over from the internal-events PRA model. These requirements include all sources of uncertainty that impact the flood-induced accident sequence analysis.

Table 3-2.5-3 Supporting Requirements for HLR-IFQU-B

Documentation of the internal flood accident sequences and quantification shall be consistent with the applicable supporting requirements (HLR-IFQU-B).

Index No.	Capability Category I	Capability Category II	Capability Category III
IFQU-B1	DOCUMENT the internal flood accident sequences and quantification in a manner that facilitates PRA applications, upgrades, and peer review.		
IFQU-B2	DOCUMENT the process used to define the flood-induced accident sequences and their associated quantification. For example, this documentation typically includes *(a)* calculations or other analyses used to support or refine the evaluation *(b)* screening criteria used in the analysis *(c)* flood-induced initiating events, flood-induced accident sequences, and flood scenarios that have been considered, grouped, screened out, and retained for quantification *(d)* results of the internal flood analysis, consistent with the quantification requirements provided in Requirements HLR-QU-D and HLR-LE-F in Part 2 *(e)* walkdowns performed in support of flood-induced accident sequence quantification		
IFQU-B3	DOCUMENT sources of model uncertainty and related assumptions (as identified in Requirement IFQU-A12 for those aspects of the accident sequences associated with internal flood, as well as those identified in Requirements QU-E1 and QU-E2 in Part 2 for those parts of the accident sequences derived from the internal-events PRA model) associated with the internal flood accident sequences and quantification.		

Section 3-3
Peer Review for Internal Flood At-Power PRA

3-3.1 PURPOSE

This Section provides requirements for peer review of an internal flood PRA.

3-3.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATION

In addition to the general requirements in Section 1-6, the peer review team shall have knowledge and collective experience, as discussed in 1-6.2, in the technical elements of internal flood analysis.

3-3.3 REVIEW OF INTERNAL FLOOD PRA ELEMENTS TO CONFIRM THE METHODOLOGY

A review shall be performed on the internal flood analysis. The portion of the internal flood analysis selected for review typically includes a sample of the screening out of flood areas and the flooding scenarios contributing to significant sequences (CDF or LERF), including

(a) flood-induced initating-event frequencies

(b) flood scenarios involving each identified flood source for each flood areas

(c) flood scenarios involving flood propagation to adjacent flood areas

(d) flood scenario that involves each of the flood-induced failure causes (e.g., spray, submergence)

(e) one flood scenario involving each type of intiating event (e.g., transient and LOCA)

Section 3-4
References

[3-1] EPRI 1019194, Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment, 2009; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

[3-2] EPRI 1021086, Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments, Revision 2, 2010; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

PART 4 REQUIREMENTS FOR INTERNAL FIRES AT-POWER PRA

(a)
(b)

Section 4-1 Risk Assessment Technical Requirements for Internal Fires At-Power

4-1.1 PRA SCOPE

This Part establishes technical requirements for a Level 1 and large early release frequency (LERF) analysis of internal fires[1] while at-power.

4-1.2 COORDINATION WITH OTHER PARTS OF THIS STANDARD

This Part is intended to be used together with Parts 1 and 2 of this Standard.

This Part assumes as an entry point for the fire PRA that an internal-events PRA for initiators other than fire has been completed and that the PRA has been weighed against the requirements of Part 2. Therefore, many of the fire PRA requirements stated here build upon the foundations established by a preexisting internal-event PRA.[2]

Similarly, this Part is intended to be used with Parts of this Standard dealing with low-power/shutdown operation (to be provided at a later date). However, additions and modifications to the technical requirements of this Standard will be necessary and are anticipated in a future revision, to cover fire PRAs for accidents initiated by fires during low-power/shutdown operation.

Accident sequences associated with external fires (i.e., fires occurring outside the global analysis boundary as defined by Part 4) are covered by Parts 6 and 9. If the analyzed initiator is a result of an internal fire, such as a fire-induced loss of off-site power (LOOP) or a fire-induced reactor trip, or if the event is associated with a consequential internal fire that complicates plant response (e.g., a turbine blade ejection event or an earthquake that results in a consequential fire), it is intended that the requirements of this Part be followed. Accidents initiated by LOOP are explicitly included in Part 2 unless the LOOP is due to a fire event, in which case the LOOP is within the scope of this Part. If the fire is initiated outside the plant boundaries (e.g., a forest fire or nearby industrial fire), the event would be considered an external fire and is covered in Parts 6 and 9 of this Standard. Although this Standard is intended ultimately to be used with the shutdown PRA requirements when completed, accidents initiated by fire events occurring during low-power/shutdown conditions are explicitly not covered by the requirements herein.

4-1.3 FIRE PRA SCOPE

The scope of a fire PRA covered by this Part is associated with internal fires that might occur during nuclear power plant mode 1 (power) operations (i.e., accident sequences associated with internal fires that might occur while a nuclear power plant is at low power or shutdown

[1] Note that the term "internal fires" as used in this Standard is defined as any fire originating within the global analysis boundary as defined per the Plant Partitioning technical element (see 4-2.1).

[2] Examples of fire PRA requirements that build on internal-events PRA results can be found in various technical elements including, in particular, equipment selection, the fire PRA plant response model, risk quantification, human reliability analysis, and uncertainty analysis.

conditions are not covered in this Part).[3] It is further limited to requirements for

(a) a Level 1 PRA that estimates the core damage frequency (CDF)

(b) a large early release frequency (LERF) analysis consistent with corresponding sections of Part 2.

Part 9 of this Standard covers external fires.

4-1.3.1 Scope: The LERF Endpoint

As discussed above in 4-1.3, the requirements herein include the analysis of LERF as an output of the fire PRA in a manner consistent with corresponding sections of Part 2.

The approach to any Level 1 fire PRA typically uses as its starting point the internal-events PRA Level 1 model, to which must be added a number of systems, structures, and components (SSCs) and human actions not included in that model but that play a unique role in the postfire safe shutdown plant response or that could fail (or fail in a unique way) due to the fact that the accident initiator is a fire. Similarly, the fire PRA typically uses the internal-events PRA LERF models as the starting point for the fire PRA LERF analysis.

4-1.3.2 Scope: Other Types of Nuclear Power Reactors

This Part was written based on certain pre-existing conditions that reflect the status of all of the currently operating U.S. LWRs. In particular, all of the current U.S. LWRs (and some non-U.S. plants) have performed a post-fire safe shutdown analysis to meet regulatory requirements for fire protection (e.g., 10CFR50 Appendix R or the equivalent). The availability of a post-fire safe shutdown analysis whose scope and content are consistent with current U.S. LWR practice is taken as an entry point for the fire PRA (e.g., plant partitioning, fire-induced spurious operation analysis, equipment selection, human actions, and certain fire protection strategies). Hence, this Part is applicable to fire PRA methodologies and applications for evaluating the current generation of U.S. LWRs. It may also be useful, with appropriate adaptations, to other types of nuclear power reactors, including advanced LWRs, and to reactors outside the United States for which an equivalent post-fire safe shutdown analysis is available.

[3] The fire PRA scope includes accident sequences initiated as a result of fire-induced damage (such as a fire in nonvital equipment that damages electrical cables causing a plant transient). The fire PRA scope also includes plant accident sequences initiated by general plant equipment failures where a concurrent fire might complicate plant safe shutdown efforts (such as a turbine blade ejection event that causes both a plant transient and a concurrent turbine lube-oil fire).

[4] DELETED.

4-1.4 PRA CAPABILITY CATEGORIES

The capability categories, as defined in Part 1, are *not* based on the level of conservatism in a particular aspect of the analysis. In many cases, the level of conservatism decreases 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 fire-induced spurious operations. As the capability category increases, the depth of the analysis required also increases. Hence, for a system train that is analyzed with fewer fire-induced 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 fire-induced 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.

4-1.5 RISK-ASSESSMENT APPLICATION PROCESS

The risk-assessment application process shall be performed according to the requirements found in Section 1-3. In the context of fire PRA, wherever Part 1 uses "PRA," "fire PRA" is substituted.

In "Identification of Application" (1-3.2), in the context of fire PRA, "plant design" shall be interpreted to include the provisions of the plant fire protection program; and "plant activities" shall be interpreted to include any and all activities associated with the maintenance of fire protection systems and features, compliance with administrative aspects of the fire protection program, fire-specific compensatory measures, the training of plant personnel specific to fire, and the actual response of plant personnel to a fire event, as well as those activities related to the maintenance and operation of the SSCs required for safe shutdown.

In "Modeling of SSCs and Activities" (1-3.3.2), in addition to SSCs and activities, the assessment of fire PRA model requirements and acceptability SHALL include a like treatment of fire protection systems and features impacted by the plant design or operational change.

A fire PRA need not be performed at a uniform level of detail. The analyses performed for screened physical analysis units may be performed at a lower completeness level than analyses performed for fire areas, fire compartments, and/or fire scenarios which are not screened out. An iterative process is also common in fire PRA. Those physical analysis units that represent the higher risk contributors may be analyzed repeatedly, each time incorporating additional detail for specific aspects of the analysis (e.g., fire modeling, suppression credit, refinements to the fire PRA plant response model, the HRA, circuit fault analysis, etc.). At any stage the additional detail may allow for the screening of a physical analysis unit. It is intended that this Standard allow for analysis flexibility in this regard. As such, the level of detail and resolution for lower risk and/or screened physical analysis units may be lower than for higher risk and unscreened physical analysis units without affecting the overall capability level of the fire PRA. For example, a service building containing numerous fire areas may be treated as a single physical analysis unit (see plant partitioning below) and analyzed for screening purposes. If the building screens in either qualitative screening or quantitative screening using conservative estimates, then the overall categorization of the fire PRA is unaffected. Similarly, the requirements for developing specific fire scenarios, detailed HRA, etc., are not needed for screened physical analysis units and may not be needed for lower risk unscreened physical analysis units as long as the overall validity of the final results is unaffected.

The capability category required for various aspects of the fire PRA may also be determined by the intended fire PRA application and may not be uniform across all aspects of the fire PRA. For example, a fire PRA that generally meets Capability Category II, with focused enhancements to meet Capability Category III in specific areas, may be required to support a given application.

4-1.6 FIRE PRA PROCESS CHECK

Analyses and/or calculations used directly by the fire PRA (e.g., HRA, data analysis, ignition frequency calculations or updates, fire modeling calculations) or used to support the fire 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. The fire PRA process check is an entirely distinct task from the peer review that is described in Section 4-3. Documentation of this review may take the form of handwritten comments, signatures or initials on the analyses/calculations, formal sign-offs, or other equivalent methods.

Section 4-2
Fire PRA Technical Elements and Requirements

The requirements of this Part are organized by 13 fire PRA elements that compose a Level 1/CDF and LERF fire PRA for at-power plant states. These elements are derived from commonly applied fire PRA processes. Figure 4-1-1 provides a general overview of the fire PRA process as envisioned in this Standard. While a fire PRA is iterative (i.e., certain elements may be refined using information developed in one or more of the subsequent elements), for clarity the flowchart shown in Fig. 4-1-1 does not attempt to incorporate potential feedback paths.

The process flowchart presented in Fig. 4-1-1 reflects the structure of this Standard and its technical elements. This structure is not unique, and it is not intended that following this particular process flow be interpreted as a requirement of this Standard. Other process structures may be, and have in the past been, employed successfully in the conduct of a fire PRA. The application of an alternate process structure would not preclude a fire PRA from being weighed against the elements of this Standard.

The following is a short description of each element included in the fire PRA process as described in this Standard and its relationship to other elements. Additional detail is provided for each element in this Part.

(a) Plant Boundary Definition and Partitioning (PP). This element defines the physical boundaries of the internal fire analysis (i.e., the locations within a plant where internal fire scenarios are postulated) and divides the various volumes within that boundary into physical analysis units generally referred to as "fire areas" or "fire compartments." Fire is a highly spatial phenomenon; hence, fire PRA quantification and reporting are generally organized in accordance with the physical divisions (the physical analysis units) defined during plant partitioning.

(b) Fire PRA Equipment Selection (ES). This element identifies the set of plant equipment that will be included in the fire PRA. This includes

(1) equipment that if damaged as a result of a fire will lead to a plant trip (or other initiating event) either directly or as a result of operator action in response to a fire

(2) equipment (including alarms, indicators, and controls) required to respond to each of the initiating events identified

(3) equipment whose fire-induced spurious operation will adversely affect the response of systems or functions (including operator actions) required to respond to a fire

Equipment selection must occur in close coordination with the fire PRA plant response model (PRM) element because the PRM reflects the selected equipment within the accident sequences to be considered in the fire PRA. Selected equipment is also mapped to the fire physical units defined in the PP element. This mapping information is needed to complete the qualitative screening (QLS) and fire scenario selection and analysis (FSS) elements.

(c) Fire PRA Cable Selection (CS). This element identifies (and locates)

(1) cables (and the equipment to which the cables are connected) that are required to support the operation of fire PRA equipment selected (see element ES)

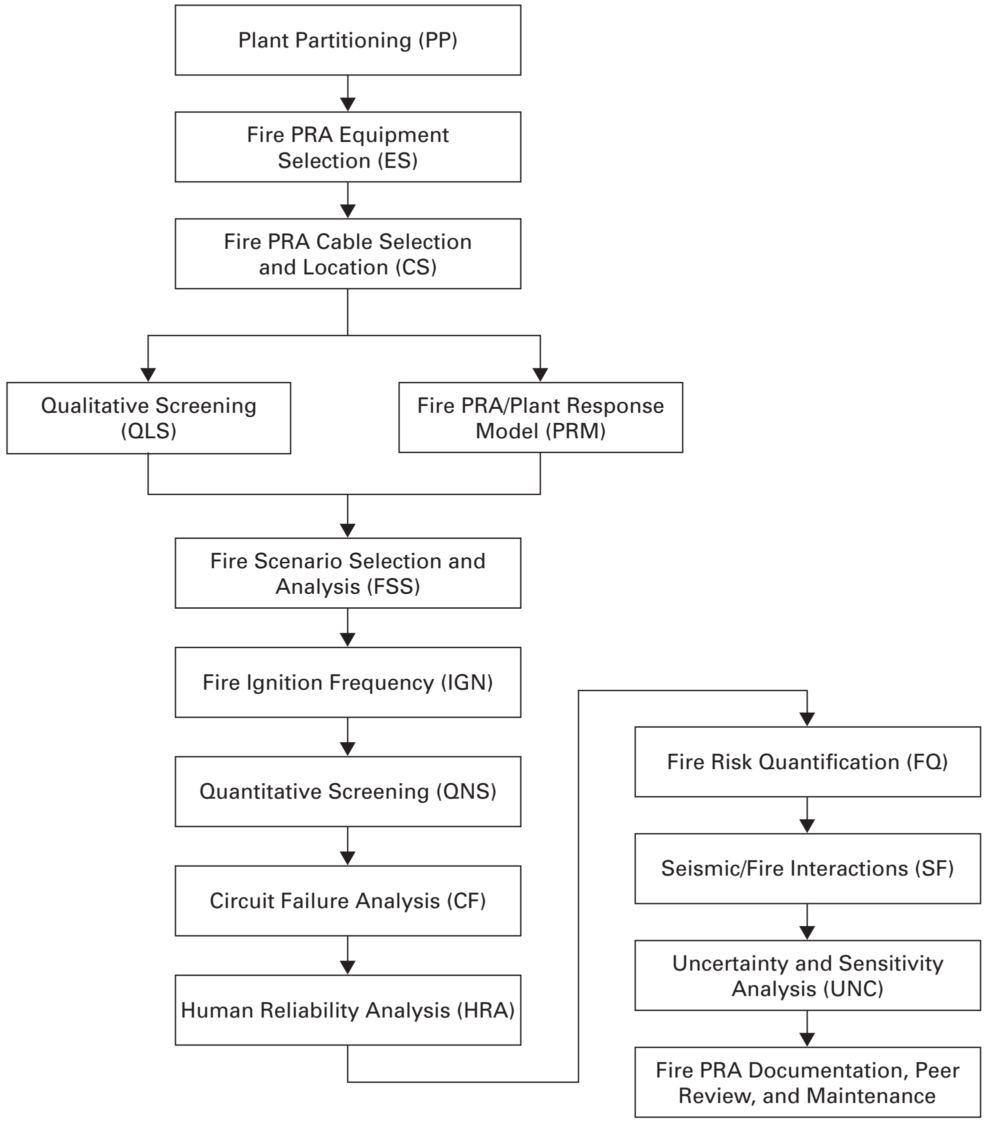
(2) cables whose failure could adversely affect credited systems and functions

This element includes an assessment of cable failure modes and effects including consideration of fire-induced spurious operations. Equipment failure mode information is used in the plant response model (PRM) element to ensure that all potentially risk-relevant equipment failure modes are included in the PRM (e.g., loss of function failures versus spurious operation). Selected cables are also mapped to the fire physical analysis units defined in the PP element. This mapping information is needed to complete the qualitative screening (QLS) and fire scenario selection and analysis (FSS) elements.

(d) Qualitative Screening (QLS). This element identifies fire physical analysis units that can be assumed to have little or no risk significance without quantitative analysis. [QLS only considers physical analysis units as individual contributors. All physical analysis units are reconsidered as a part of the multiphysical analysis units fire scenario analysis (see Requirement HLR-FSS-E).] Qualitative screening is based on the fire physical analysis units defined in element PP and on the equipment and cable location information provided by elements ES and CS. Any fire physical analysis unit that fails to satisfy the qualitative screening criteria is retained for further analysis.

(e) Fire PRA Plant Response Model (PRM). This element involves the *development* of a logic model that reflects the plant response following a fire. The fire PRA

Fig. 4-1-1 Fire PRA Process Flowchart



PRM is *central* to the quantification of fire risk and is *exercised* in the fire risk quantification (FQ) element to quantify conditional core damage probability (CCDP) and conditional large early release probability (CLERP) values for selected fire scenarios. This model is expected to be constructed based on an internal-events PRA model that is then modified to include only those initiating events that can result from a fire, to include unique additional equipment and/or failure modes, such as spurious operation, not addressed in an internal-events PRA model, and to reflect fire-specific plant procedures and operator actions (e.g., alternate and remote shutdown actions).

(f) Fire Scenario Selection and Analysis (FSS). In this element, fire scenarios are selected, defined, and analyzed to represent the collection of fire events that might contribute to plant fire risk. The purpose of the fire scenario analysis is to quantify the likelihood that given ignition of a fire, fire-induced damage to selected equipment and cables (as defined in the ES and CS elements) occurs. The result is expressed for each fire scenario as

(1) a set of cable and equipment failures, including specification of the failure modes, reflecting the loss of a specific set of damage targets

(2) a conditional probability that given the fire, the postulated cable and equipment failures are realized (potentially including both a severity factor and a non-suppression probability)

These results are fed forward to the FQ element for incorporation into the final risk calculations.

(g) Fire Ignition Frequency (IGN). This element estimates the frequency of fires (expressed as fire ignitions per reactor-year). Fire frequencies are ultimately estimated for each selected fire scenario (from the FSS element) and can be developed for a physical analysis unit as a whole, for a group of ignition sources, or for a specific individual ignition source depending on the nature of the fire scenario. The ignition frequency values are fed forward to the FQ element for incorporation into the final risk calculations.

(h) Quantitative Screening (QNS). This element involves the screening of fire compartments based on their quantitative contribution to fire risk. [As with QLS, element QNS only considers physical analysis units as individual risk contributors. All physical analysis units are reconsidered as a part of the multicompartment fire scenario analysis (see Requirement HLR-FSS-E).] Physical analysis units whose contribution to fire risk is shown to meet the quantitative screening criteria need not be analyzed in additional detail.

(i) Circuit Failure Analysis (CF). This element refines that treatment of fire-induced cable failures and their impact on the plant equipment, systems, and functions included in the fire PRA plant response model. This element also estimates the relative likelihood of various circuit failure modes such as loss of function failures versus spurious operation failures. Quantified circuit failure mode likelihood estimates are incorporated into the fire PRA plant response model (developed under element PRM) as a part of CCDP and CLERP quantification in element FQ.

(j) Postfire Human Reliability Analysis (HRA). This element considers operator actions as needed for safe shutdown including those called out in the relevant plant fire response procedures. It also includes the identification of human failure events (HFEs) for inclusion in the fire PRA plant response model. The HRA element also includes the quantification of human error probabilities (HEPs) for the modeled actions that are fed forward to element FQ in support of the CCDP and CLERP calculations for each selected fire scenario from element FSS.

(k) Fire Risk Quantification (FQ). This element involves the quantification and presentation of fire risk results. In this element the fire PRA plant response model (developed under element PRM), including HFEs as identified in the HRA, is exercised for each fire scenario (as defined in element FSS). CCDP and CLERP values are calculated based on translation of the cable and equipment failures for each scenario, including specification of the failure modes, into PRM basic events, quantitative equipment failure mode values (from element CF), and HEP values (from element HRA). Final quantification mathematically combines the calculated CCDP/CLERP values with the corresponding fire frequency (IGN) and the conditional probability of fire damage [potentially including both a severity factor and nonsuppression probability (FSS)] to yield estimates of fire risk in the form of CDF and LERF.

(l) Seismic/Fire Interactions (SF). This element involves a qualitative review of potential interactions between an earthquake and fire that might contribute to plant risk. This element *does not* include quantitative estimates of the risk associated with such interactions but, rather, seeks to ensure that such interactions have been considered and that steps are taken to ensure that the potential risk contributions are not significant.

(m) Uncertainty and Sensitivity Analyses (UNC). This element involves the identification and treatment of uncertainties throughout the fire PRA process.

Tables of HLRs and SRs for the 13 fire PRA elements are provided in 4-2.1 through 4-2.13. The SRs are numbered and labeled to identify the HLR that is supported. Section 4-2 describes a general discussion of SRs and the assigned Capability Category. 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.

4-2.1 PLANT PARTITIONING

4-2.1.1 Objectives

The objectives of the plant partitioning (PP) element are to define

(a) the global analysis boundary of the fire PRA; that is, to define the physical extent of the plant to be encompassed by the internal fire analysis

(b) the physical analysis units (spatial units) upon which the analysis will be based

Fire PRA is driven largely by spatial considerations; hence, the basic fire PRA physical analysis units are defined in terms of physical regions (or volumes) of the plant. In practice, these physical analysis units are typically called "fire areas" and/or "fire compartments" but may also include (with justification) physical analysis units based on features such as spatial separation. Note (2) from Table 1-1.3-2 states the following:

"*(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 4-2.1). 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 supporting requirements for the PP element make no distinctions based on Capability Category with regard to defining the global analysis boundary and the base set of physical analysis units. The distinctions with respect to the Capability Categories are derived from those HLRs and SRs in other elements that rely on these definitions. That is, many of the HLRs and SRs associated with other technical elements (i.e., CS, QLS, IGN, QNS, FSS, SF, and FQ) require specific levels of treatment for a specific topic as applied to each physical analysis unit. The PP requirements establish a fundamental definition of what constitutes a valid physical analysis unit and for documenting this aspect of the analysis.

A typical nuclear power plant is made up of several fire areas. The term "fire area" is defined by NRC regulatory requirements, and the same meaning is intended here. Fire areas, as identified in the licensee's fire protection program, are generally defined (bounded on all sides) by fire barriers with an established fire-resistance rating. (Exceptions may be made for outdoor locations such as an exterior switchyard.) Use of the predefined fire areas as the basic fire PRA physical analysis units is considered acceptable practice for *all* capability categories (see Requirement PP-B1). However, it may be advantageous to define smaller and more localized physical analysis units, especially for larger fire areas. That is, a fire area may be subdivided into two or more physical analysis units (see Requirements PP-B1 through PP-B5).

The fire PRA HLRs and SRs are predicated on an analysis structure wherein most fire scenarios will involve damage confined to a single physical analysis unit. [A multicompartment analysis ensures the completeness of the fire PRA by considering the potential for fire damage to more than one physical analysis unit (see Requirement HLR-FSS-E).] Consistent with this structure, the primary intent of the PP requirements is to ensure that the boundaries that define each physical analysis unit will substantially contain the damaging fire behaviors. In general terms, "substantially contain damaging fire behaviors" is interpreted in the context of fire plume development, the development of a hot gas layer, direct radiant heating by the fire, and the actual spread of fire between contiguous or noncontiguous fuel elements.[5] Smoke spread behavior is not a required consideration in the partitioning analysis (any potential for damage due to smoke spread beyond a fire compartment is captured in the multicompartment fire scenarios; see Requirement HLR-FSS-E and its corresponding SRs). However, features other than complete and permanent physical boundaries may, with justification, be credited in defining the fire PRA physical analysis units (e.g., see Requirement PP-B2).

The PP requirements do not preclude the analyst from subdividing physical analysis units into more localized segments when that practice is intended to support the efficient collection and organization of fire PRA information. That is, under some approaches, an analyst may define further subdivisions of the physical analysis units where those subdivisions do not meet the PP requirements. For purposes of discussion, call these "administrative partitions." In effect, administrative partitions would be purely a matter of bookkeeping convenience.

4-2.1.2 Acceptability

The acceptability of a plant partitioning analysis relies on the following three factors:

(a) the acceptability of the global physical boundaries defined for the fire PRA (see Requirement HLR-PP-A)

(b) the credibility of the credited partitioning elements as being capable of substantially confining damaging fire behaviors (see Requirement HLR-PP-B)

(c) a complete corresponding analysis of the risk contribution of multicompartment fire scenarios (see Requirement HLR-FSS-E)

This Standard presumes that the fire PRA will analyze one entire plant unit, and the global analysis boundary is established accordingly (see Requirement HLR-PP-A). It is recognized that some applications may only require analysis of part of the plant. In such cases, adjustments to the global analysis boundary to suit the intended application would be expected.

[5] The definition of "fire compartment" purposely relaxes the criteria relative to the degree of fire confinement below those used in defining fire areas. For fire compartments, open leakage paths to other fire compartments are allowable. The phrase "substantially confined" means that

(a) the direct spread of fire between fire compartments is unlikely even under the most severe fire conditions possible

(b) fire-induced damage to potential damage targets will be confined to a single fire compartment except under the most severe possible fire conditions

The potential for fire-induced damage to targets in multiple fire compartments is treated per Requirement HLR-FSS-E.

Table 4-2.1-1 High Level Requirements for Plant Partitioning (PP)

Designator	Requirement
HLR-PP-A	The fire PRA shall define the global boundaries of the analysis so as to include all plant locations relevant to the plant-wide fire PRA.
HLR-PP-B	The fire PRA shall perform a plant partitioning analysis to identify and define the physical analysis units to be considered in the fire PRA.
HLR-PP-C	The fire PRA shall document the results of the plant partitioning analysis in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.1-2 Supporting Requirements (SR) for HLR-PP-A

The fire PRA shall define the global boundaries of the analysis so as to include all plant locations relevant to the plant-wide fire PRA (HLR-PP-A).

Index No. PP-A	Capability Category I	Capability Category II	Capability Category III
PP-A1 [Notes (1) and (2)]	INCLUDE within the global analysis boundary all fire areas, fire compartments, or locations within the licensee-controlled area where a fire could adversely affect any equipment or cable item to be credited in the fire PRA plant response model including those locations of a sister unit that contain shared equipment credited in the fire PRA.		

NOTES:
(1) The intent of this requirement is to include sister unit locations that meet the selection criteria as stated.
(2) The intent of this requirement is that the global analysis boundary will include locations that may contain fire sources that could threaten credited equipment or cable items by virtue of a multicompartment fire scenario but that may not themselves contain credited equipment or cable items.

Table 4-2.1-3 Supporting Requirements (SR) for HLR-PP-B

The fire PRA shall perform a plant partitioning analysis to identify and define the physical analysis units to be considered in the fire PRA (HLR-PP-B).

Index No. PP-B	Capability Category I	Capability Category II	Capability Category III
PP-B1	DEFINE a set of fire PRA physical analysis units that reflect the physical characteristics of the plant, the nature of the fire hazards present in each plant location, and the potential extent of fire damage that could reasonably result from fires involving those fire sources.		
PP-B2 [Notes (1) and (2)]	If any physical plant feature that lacks a specific fire-endurance rating has been credited as a partitioning element in defining the boundaries of the physical analysis units (see Requirement PP-B1), JUSTIFY the judgment that the nonrated partitioning element will substantially contain the damaging effects of fires given the nature of the fire sources present in each physical analysis unit, separated by the nonrated partitioning element.		
PP-B3 *[formerly PP-B4]*	DO NOT INCLUDE raceway fire barriers, thermal wraps, fire-retardant coatings, radiant energy shields, or any other localized cable or equipment protection feature as partitioning elements in defining physical analysis units.		
PP-B4 *[formerly PP-B6]* [Note (3)]	ENSURE *(a)* that collectively, the defined physical analysis units encompass all locations within the global analysis boundary (see Requirement PP-A1) *(b)* that defined physical analysis units do not overlap		
PP-B5	CONDUCT a confirmatory walkdown of credited barriers that are not maintained as a part of the fire protection program to confirm the conditions and characteristics of credited partitioning elements.		

NOTES:

(1) The intent of Requirement PP-B2 is to allow an analysis to credit partitioning features that have a specific fire-endurance rating in the plant partitioning analysis without further justification, subject only to the restriction imposed by Requirement PP-B4. However, plant partitioning may also, with justification, credit partitioning features that lack a specific fire-endurance rating (nonrated elements), such as spatial separation or nonrated structural elements.

(2) Volume 2, Chapter 1 of reference [4-1] discusses criteria that may be applied in justifying decisions related to spatial separation, active fire barrier elements, and partitioning features that lack a fire-resistance rating.

(3) Either a single physical analysis unit or a collection of two or more physical analysis units will generally correspond to a plant fire area as defined in the plant's fire protection program. If Requirement PP-B4 is met, then no two physical analysis units would share the same space.

Table 4-2.1-4 Supporting Requirements (SR) for HLR-PP-C

The fire PRA shall document the results of the plant partitioning analysis in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-PP-C).

Index No. PP-C	Capability Category I	Capability Category II	Capability Category III
PP-C1	DOCUMENT the global analysis boundaries of the fire PRA or fire PRA application in a manner that facilitates fire PRA applications, upgrades, and peer review.		
PP-C2	JUSTIFY the exclusion of any locations within the licensee-controlled area that are not included in the global analysis boundary by demonstrating that they do not satisfy the selection criteria as defined per Requirement PP-A1.		
PP-C3	DOCUMENT the general nature and key or unique features of the partitioning elements that define each physical analysis unit defined in plant partitioning in a manner that facilitates fire PRA applications, upgrades, and peer review.		
PP-C4 [Note (1)]	DOCUMENT a consistent scheme for naming and identifying the fire PRA physical analysis units in a manner that facilitates fire PRA applications, upgrades, and peer review.		

NOTE:
(1) It is to the advantage of both the analyst and reviewers that when fire areas are partitioned into two or more physical analysis units, the analysis documentation map the resulting fire compartments back to the original fire areas as defined in the plant's fire protection program.

4-2.2 EQUIPMENT SELECTION

The objective of the equipment selection (ES) element is to select the plant equipment that will be included in the fire PRA plant response model.

Note that the selection of fire PRA equipment serves as the foundation for identifying corresponding cables that will need to be selected and located under the cable selection and location technical element (nonelectrical equipment will not need cable information but may still be in the fire PRA). The ES element needs to include the following major categories of equipment:

(a) equipment whose fire-induced failure including spurious operation will contribute to or otherwise cause an initiating event to be modeled in the fire PRA (Requirement HLR-ES-A)

(b) equipment to support the success of mitigating safety functions to be included in the fire PRA, including equipment implicitly included in recovery models, and therefore whose failure including spurious operation would adversely affect the success of the mitigating safety functions included in the fire PRA (Requirement HLR-ES-B)

(c) equipment to support the success of operator actions for achieving and maintaining safe shutdown to be credited in the fire PRA and, therefore, whose failure including spurious operation would likely induce inappropriate or otherwise unsafe actions (or prevent appropriate or otherwise safe actions) by the plant operators during a fire damage sequence (Requirement HLR-ES-C)

The requirements of this element complement the PRM element in which the fire PRA plant response model is developed. The requirements are written in anticipation that analysts will not be performing this element in a vacuum but will instead conduct this element with full knowledge of what equipment is included for safe shutdown in the plant's current Fire Safe Shutdown/Appendix R analysis and what equipment is included in the plant's internal-events PRA that has been assessed against the requirements of Part 2.

Table 4-2.2-1 High Level Requirements for Equipment Selection (ES)

Designator	Requirement
HLR-ES-A	The fire PRA shall identify equipment whose failure, including fire-induced spurious operation, would contribute to or otherwise cause an initiating event.
HLR-ES-B	The fire PRA shall identify equipment whose failure, including spurious operation, would adversely affect the operability/functionality of that portion of the plant design to be credited in the fire PRA.
HLR-ES-C	The fire PRA shall identify instrumentation whose failure, including spurious operation, would impact the reliability of operator actions associated with that portion of the plant design to be credited in the fire PRA.
HLR-ES-D	The fire PRA shall document the fire PRA equipment selection, including that information about the equipment necessary to support the other fire PRA tasks (e.g., equipment identification; equipment type; normal, desired, failed states of equipment) in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.2-2 Supporting Requirements (SR) for HLR-ES-A

The fire PRA shall identify equipment whose failure, including fire-induced spurious operation, would contribute to or otherwise cause an initiating event (HLR-ES-A).

Index No. ES-A	Capability Category I	Capability Category II	Capability Category III
ES-A1 [Notes (1)–(3)]	INCLUDE equipment whose failure, including fire-induced spurious operation (see Requirement ES-A4), would contribute to or otherwise cause an automatic trip, a manual trip per procedure direction or would invoke a limiting condition of operation (LCO) that would necessitate a shutdown where *(a)* shutdown is likely to be required before the fire is extinguished *(b)* a potentially significant effect on safe shutdown capability is caused by the affected equipment, or *(c)* the shutdown will be modeled as a plant trip rather than a slow, controlled shutdown of the plant based on the current modeling practice in the internal-events PRA		
ES-A2 [Note (4)]	REVIEW power supply, interlock circuits, instrumentation, and support system dependencies, and IDENTIFY additional equipment whose fire-induced failure, including fire-induced spurious operation, could adversely affect any of the equipment identified per Requirement ES-A1.		
ES-A3	INCLUDE equipment whose fire-induced failure, not including fire-induced spurious operation, contributes to or causes *(a)* fire-induced initiating events included in the Fire Safe Shutdown/Appendix R analysis *(b)* internal-events PRA initiators as modified per the PRM technical element (see 4-2.5) *(c)* unique fire-induced initiating events that either were screened out in the internal-events analysis [(b) above] or were not included in the fire safe shutdown analysis [(a) above] if Requirement IE-C6 in Part 2 cannot be met		

Table 4-2.2-2 Supporting Requirements (SR) for HLR-ES-A (Cont'd)

The fire PRA shall identify equipment whose failure, including fire-induced spurious operation, would contribute to or otherwise cause an initiating event (HLR-ES-A).

Index No. ES-A	Capability Category I	Capability Category II	Capability Category III
ES-A4 [Note (5)]	INCLUDE additional equipment based on the consideration of cases where any single fire-induced spurious operation of equipment alone or in combination with other fire-induced loss of function failures could cause an initiating event considering *(a)* fire-induced initiating events treated in the Fire Safe Shutdown/Appendix R analysis *(b)* internal-events PRA initiators as modified per the PRM technical element (see 4-2.5) *(c)* unique fire-induced initiating events that either were screened out in the internal-events analysis [(b) above] or were not included in the fire safe shutdown analysis [(a) above] if Requirement IE-C6 in Part 2 cannot be met		INCLUDE additional equipment based on the consideration of cases where up to two fire-induced spurious operations of equipment alone or in combination with other fire-induced loss of function failures could cause an initiating event considering *(a)* fire-induced initiating events treated in the Fire Safe Shutdown/Appendix R analysis *(b)* internal-events PRA initiators as modified per the PRM technical element (see 4-2.5) *(c)* unique fire-induced initiating events that either were screened out in the internal-events analysis [(b) above] or were not included in the fire safe shutdown analysis [(a) above] if Requirement IE-C6 in Part 2 cannot be met
ES-A5 [Notes (6) and (7)]	INCLUDE any single fire-induced spurious operation of equipment alone or in combination with other fire-induced loss of function failures for the special case where fire-induced failures could contribute not only to an initiating event but also simultaneously *(a)* affect the operability/functionality of that portion of the plant design to be credited in response to the initiating event in the fire PRA *(b)* result in an initiating event where the mitigating function is not addressed in the Fire Safe Shutdown/Appendix R Analysis, or *(c)* result in a loss of reactor coolant system integrity.	INCLUDE up to two fire-induced spurious operations of equipment alone or in combination with other fire-induced loss of function failures for the special case where fire-induced failures could contribute not only to an initiating event but also simultaneously *(a)* affect the operability/functionality of that portion of the plant design to be credited in response to the initiating event in the fire PRA *(b)* result in an initiating event where the mitigating function is not addressed in the Fire Safe Shutdown/Appendix R Analysis, or *(c)* result in a loss of reactor coolant system integrity	INCLUDE up to three fire-induced spurious operations of equipment alone or in combination with other fire-induced loss of function failures for the special case where fire-induced failures could contribute not only to an initiating event but also simultaneously *(a)* affect the operability/functionality of that portion of the plant design to be credited in response to the initiating event in the fire PRA *(b)* result in an initiating event where the mitigating function is not addressed in the Fire Safe Shutdown/Appendix R Analysis or *(c)* result in a loss of reactor coolant system integrity

Table 4-2.2-2 Supporting Requirements (SR) for HLR-ES-A (Cont'd)

The fire PRA shall identify equipment whose failure, including fire-induced spurious operation, would contribute to or otherwise cause an initiating event (HLR-ES-A).

Index No. ES-A	Capability Category I	Capability Category II	Capability Category III
ES-A6 [Notes (8)–(10)]	INCLUDE up to two fire-induced spurious operations of equipment alone or in combination with other fire-induced loss of function failures for the special case where fire-induced failures could contribute to an initiating event that in turn leads to core damage and a large early release.	INCLUDE up to three fire-induced spurious operations of equipment alone or in combination with other fire-induced loss of function failures for the special case where fire-induced failures could contribute to an initiating event that in turn leads to core damage and a large early release.	INCLUDE up to four fire-induced spurious operations of equipment alone or in combination with other fire-induced loss of function failures for the special case where fire-induced failures could contribute to an initiating event that in turn leads to core damage and a large early release.

NOTES:

(1) At this stage it is not necessary to explicitly consider fire scenarios potentially leading to equipment failure. Rather, the intent is to simply identify equipment that might be failed by any fire and whose failure could cause an initiating event. The QNS and FSS elements will assess the actual fire-induced failure likelihoods.

(2) This Requirement covers the same portion of equipment as is addressed in Requirements HLR-IE-A and HLR-SY-A in Part 2 (including any gradation therein across capability categories) as applied to defining initiating events, unless a different level of definition can be justified as sufficient. This level of definition typically involves the primary equipment item (called a component in Part 2) that directly performs the operation/function of interest such as a valve that needs to remain open to allow flow or a pump that provides injection flow. Because of the spatial nature of a fire PRA, when addressing other requirements associated with cable identification (see Requirements CS-A1, CS-A2, and CS-A3), it is understood that the primary equipment item is extended to mean itself and any supportive equipment (e.g., power supply, associated actuating instrumentation, and interlocks) needed to perform the intended operation/function of the primary equipment item. In recognition that it is impractical to explicitly identify and locate all equipment and their cables that could contribute to or cause an initiating event such as, for instance, all the balance-of-plant equipment, the intent of this requirement is to allow the analyst to use other levels of equipment definition (e.g., rather than identifying and locating individual equipment items in the main feedwater system such as the pumps and regulator valves, the analyst chooses to identify the equipment more globally as "main feedwater"). This can be done as long as its failure in terms of an initiating event is treated conservatively (for instance, treating any failure of main feedwater as causing an unrecoverable total loss of main feedwater initiating event even though some individual equipment item failures may not actually cause a total unrecoverable loss of the entire system).

(3) The action verb "INCLUDE" implies that the effect of the failure of the equipment item will be included as a contributing factor to the resulting modeled initiating event in the fire PRA plant response model in the same manner that initiating events are modeled under Requirement HLR-IE-A in Part 2 (see Requirements HLR-PRM-A and HLR-PRM-B for more detail). "INCLUDE" also implies that specific equipment items will be included in the plant response model. As a result, initiators more challenging to the plant than a reactor trip would be included in the model as appropriate. A reactor trip may be assumed as the initiator if no more challenging initiator could occur as a result of the particular fire being analyzed.

(4) Requirement ES-A2 ensures that not only the primary components but also any supporting equipment for the primary components (interlock circuits, instrumentation, etc.) are also identified as potentially contributing to possible initiating events because of their possible effects on the primary components [just as is intended per Note (1)].

Table 4-2.2-2 Supporting Requirements (SR) for HLR-ES-A (Cont'd)

NOTES: (Cont'd)

(5) This requirement is included for the following two reasons:

(a) to ensure that analysts do not consider only loss of equipment operation as a fire-induced failure but instead also consider spurious operation of equipment as a fire-induced failure contributing to an initiating event

(b) to limit the number of spurious events to be treated considering current state-of-the-art and associated practicalities for performing such investigative searches

(6) An example for item (a) would be loss of service water equipment that contributes to or causes a loss of service water initiating event and simultaneously reduces the redundancy or causes complete failure of the service water system credited in the fire PRA as needed to provide cooling of other mitigating equipment. An example for items (b) and (c) would be a PWR LOCA into containment, when containment sump recirculation is not a credited flow path for safe shutdown. Another example would be reactor coolant system depressurization in a PWR when a turbine-driven pump is credited for feedwater.

(7) For plants adopting NFPA 805 [4-2], the Nuclear Safety Analysis is used in lieu of Fire Safe Shutdown/Appendix R Analysis in the context of Requirements ES-A3, ES-A4, and ES-A5.

(8) Fire-induced failures leading to interfacing system loss-of-coolant accident (ISLOCA) or containment bypass are examples of cases where fire-induced failures could contribute to an initiating event that in turn leads to core damage and large early release.

(9) Random failures do not need to be included in the analyses for this requirement.

(10) This requirement also covers a part of Requirement HLR-ES-B in addressing operability/functionality of portions of the plant design that may be credited in the fire PRA.

Table 4-2.2-3 Supporting Requirements (SR) for HLR-ES-B

The fire PRA shall identify equipment whose failure, including spurious operation, would adversely affect the operability/functionality of that portion of the plant design to be credited in the fire PRA (HLR-ES-B).

Index No. ES-B	Capability Category I	Capability Category II	Capability Category III
ES-B1 [Notes (1)–(7)]	IDENTIFY Fire Safe Shutdown/Appendix R equipment to be credited in the fire PRA.	IDENTIFY Fire Safe Shutdown/Appendix R equipment to be credited in the fire PRA, and INCLUDE risk-significant equipment from the internal-events PRA.	IDENTIFY Fire Safe Shutdown/Appendix R equipment to be credited in the fire PRA, and INCLUDE all equipment from the internal-events PRA.
ES-B2 [Notes (8)–(10)]	For every train of equipment that is to be credited in the fire PRA, IDENTIFY equipment whose fire-induced failures including any single spurious operation will contribute to failure to meet the success criteria in the fire PRA.	For every train of equipment that is to be credited in the fire PRA, IDENTIFY equipment whose fire-induced failures up to and including two spurious operations will contribute to failure to meet the success criteria in the fire PRA.	For every train of equipment that is to be credited in the fire PRA, IDENTIFY equipment whose fire-induced failures up to and including three spurious operations will contribute to failure to meet the success criteria in the fire PRA.
ES-B3 [Note (11)]	INCLUDE additional equipment if that equipment is associated with new initiating events or different accident sequences that go beyond that treated within the scope of either or both the Fire Safe Shutdown/Appendix R work or the internal-events PRA with a potential for being a significant contributor to the CDF/LERF in the fire PRA.		
ES-B4 [Note (12)]	REVIEW power supply, interlock circuits, instrumentation, and support system dependencies, and IDENTIFY additional equipment whose fire-induced failure, including fire-induced spurious operation, could adversely affect any of the equipment identified per Requirements ES-B1 through ES-B3.		

Table 4-2.2-3 Supporting Requirements (SR) for HLR-ES-B (Cont'd)

The fire PRA shall identify equipment whose failure, including spurious operation, would adversely affect the operability/functionality of that portion of the plant design to be credited in the fire PRA (HLR-ES-B).

Index No. ES-B	Capability Category I	Capability Category II	Capability Category III
ES-B5 [Note (13)]	EXCLUDE, if desired, equipment or failure modes from identification and inclusion in the fire PRA based on the following: *(a)* A fire-induced spurious operation of a component may be excluded from a system model if the conditional probability of occurrence given fire-induced damage to the component and/or associated cables is at least two orders of magnitude lower than the non-fire-induced random failure probability of the other components in the same system train that results in the same effect on system operation. The justification for exclusion must include the consideration of the scope of potential fire-induced failures to the system/train under consideration that may reasonably occur. *(b)* One or more fire-induced spurious operations of components may be excluded from the systems model if the contribution of their conditional probability of occurrence given fire-induced damage to them and/or their associated cables is <1% of the total failure rate or probability for that component or group of components, when their effects on system operation are the same. The justification for exclusion must include the consideration of the scope of potential fire-induced failures to the system/train under consideration that may reasonably occur.		

NOTES:

(1) The intent of Requirement ES-B1 is to ensure that certain portions of the equipment treated in the internal-events PRA will be carried forward to the fire PRA. The graded approach acknowledges the implications not only for analysis realism and accuracy but also for the level of effort that adding equipment to the fire PRA brings relative to, in particular, technical elements CS, PRM, and CF.

(2) It is anticipated that as a matter of good practice at all capability categories, the fire PRA will pursue an iterative approach to identifying additional plant equipment that, if credited, would significantly impact fire risk estimates. Ultimately, the selected equipment and the resulting fire PRA plant response model must be sufficiently complete so that the objectives with respect to level of detail, realism, and accuracy as stated in Table 1-1.3-2 of this Standard are met consistent with the intended capability category.

(3) For all equipment identified for inclusion in the fire PRA plant response model, the CS element will require that the associated cables be identified and traced to specific plant locations. Special provisions are made for cases where cable routing is not known in detail (see Requirement CS-A10).

(4) This requirement is intended to encompass equipment whose failure, including fire-induced spurious operation, could have an adverse effect on fire risk estimates. For Capability Category II, the SR is intended to allow for the analyst to choose what mitigating equipment to include. Requirement PRM-B9 establishes requirements for treatment of equipment that will not be included in the fire PRA plant response model. Per Requirement PRM-B9, equipment from the internal-events PRA that is not credited in the fire PRA will be failed in the most conservative mode for risk quantification.

(5) This SR covers the same portion of equipment credited to mitigate the initiating event as is addressed in the AS, SC, SY, QU, and LE requirements in Part 2 (including any gradation therein across capability categories) unless a different level of definition can be justified as sufficient. This level of definition typically involves the primary equipment item (called "a component" in Part 2) that directly performs an operation or function of interest, such as a valve that needs to remain open to allow flow or a pump that provides injection flow. Because of the spatial nature of a fire PRA and when addressing other requirements associated with cable identification (see Requirements CS-A1, CS-A2, and CS-A3), it is understood that the primary equipment item is extended to mean itself and any supportive equipment (e.g., power supply, interlocks, instrumentation) needed to perform the intended operation/function of the primary equipment item. Requirement HLR-ES-B purposely does not cover that portion of equipment involving instrumentation for operator actions, which is covered under Requirement HLR-ES-C.

(6) The action verb "IDENTIFY" implies that the failure of the equipment item will be included as a contributing factor in the fire PRA plant response model in the same manner that equipment failures are modeled in the internal-events PRA that has been assessed against Part 2.

Table 4-2.2-3 Supporting Requirements (SR) for HLR-ES-B (Cont'd)

NOTES: (Cont'd)

(7) The gradation across capability categories is intended to address the anticipated major scope differences when selecting equipment and the extent of realism achieved. To meet Capability Category I, only Fire Safe Shutdown/Appendix R equipment as modified by subsequent SRs need to be modeled in the fire PRA (other equipment can be assumed failed in the worst possible failure mode, including spurious operation). This will tend to limit the resources needed to perform the fire PRA, but because there is no credit for other mitigating features available in the plant, generally, though not necessarily, this will lead to a higher CDF/LERF than for the other capability categories. For Capability Category II, as modified by subsequent SRs, the analyst is expected to credit some equipment in the plant beyond that credited in the Fire Safe Shutdown Analysis/Appendix R, but credited in the internal-events PRA, to achieve a generally more realistic CDF/LERF based on anticipated significance considerations (some equipment may still be assumed to be failed in the worst possible failure mode). For Capability Category III, as modified by subsequent SRs, all equipment in the internal-events PRA as well as the Fire Safe Shutdown/ Appendix R equipment is addressed. This will generally lead to the most realistic CDF/LERF results but requires the most resources to perform the fire PRA.

(8) The term "train" is used to describe a series set of equipment that is associated with or otherwise affects a common operation or function, such as delivering flow from a water source through one pump and valves in series to a desired delivery point. For example, an auxiliary feedwater system may have three trains. Similarly, two pressurizer pilot operated relief valves (PORVs) would be viewed as consisting of two PORV trains, both with two functions: to remain closed when desirable and to open when needed such as for feed-and-bleed cooling. Either PORV train could fail to operate as a result of fire effects or spuriously operate because of a fire. It is anticipated that the "train" distinction will need to be modified such as when three trains merge into a shared header forming two delivery flow paths or when there is one suction path that then supplies two separate pump flow paths. "Train" is expected to be similarly reinterpreted when necessary, such as when addressing multiple electrical flow paths involving buses, breakers, etc.

(9) The expectation is that equipment associated with the operability/functionality of each train will be identified including consideration of one, two, or three spurious operations depending on the capability category. Thus, for instance, for Capability Category I, if consideration of any one spurious event at a time could affect the operability/functionality of a train (e.g., the train has two normally open valves in the main flow path that need to remain open but that each one, by itself, could receive a single fire-induced spurious closure signal failing that flow path), then each valve would be included in the equipment selection process. On the other hand, for Capability Category I, if there is a diversion flow path that would require two concurrent spurious events to open two normally closed valves, these valves do not need to be included in the equipment selection process. The same principles apply to Capability Categories II and III except that the number of concurrent spurious events that must be considered is increased. In the diversion flow path example above, the two valves would be included in the equipment selection process for Capability Categories II and III.

(10) Spurious operations may also impact the available time to achieve the defined success criteria. For example, a set of spurious operations may decrease operator response time from 20 min to 10 min affecting the HEP.

Table 4-2.2-3 Supporting Requirements (SR) for HLR-ES-B (Cont'd)

NOTES: (Cont'd)

(11) The intent of this requirement is to ensure that the equipment selection process is not performed simply on the basis of what has already been done in the work defined under Requirement ES-B1 (i.e., the Fire Safe Shutdown/Appendix R for Capability Category I, and the addition of the internal-events PRA in Capability Categories II and III). It is expected that a systematic search will be conducted for additional equipment to be included in the fire PRA even if that equipment was not considered or otherwise screened from the prior Fire Safe Shutdown/Appendix R analysis or internal-events PRA. For example, an equipment item may have not been included in either former analysis on the basis of having a very low probability of random failure (e.g., random spurious opening of a valve). The fire-induced spurious opening of that valve could be much more likely, and that valve would be included under equipment selection. As another example, an equipment item and associated scenario may not have been included on the basis of being involved only in a scenario that could be screened out in the prior analyses. For example, a previously screened-out scenario may have involved a demand and possible sticking open of the pressurizer safety relief valves following a transient on the basis that successful pressurizer PORV operation is highly likely thus precluding the need for safety relief valve operation. Such a scenario may be much more likely considering possible fire-induced effects of keeping the PORVs closed or spuriously closing their associated block valves thus putting a demand on the safety relief valves. Finally, if any assumptions or justifications provided in the former analyses precluded the identification of equipment in those analyses that could affect the equipment being credited for the fire PRA, and those justifications or assumptions are inconsistent with Requirement HLR-ES-B, it is expected that a further search and equipment identification will be conducted to meet Requirement HLR-ES-B. For example, a Fire Safe Shutdown/Appendix R analysis may have limited its search for diversion paths that could affect credited safe shutdown trains within the scope of that analysis to a single spurious event. Thus, a diversion path requiring two spurious operations to open the diversion path would not be included in the original Fire Safe Shutdown/Appendix R analysis. If one is intending to meet Capability Category II for equipment selection, this diversion path as well as additional equipment not in the original analysis would be expected to be included under equipment selection (ES).

(12) If the prior SRs are performed as intended, inclusion of support equipment should already have been done including the number of spurious operations to be treated [see Note (1) under Requirement ES-B1]. Nevertheless, to avoid focusing on only the primary equipment item that performs the operation or function of interest, this requirement is included to ensure these additional supporting equipment items are not missed, but also identified.

(13) Requirement ES-B5 is a modification of Requirements SY-A15 and SY-B13 in Part 2. Exclusion of equipment or failure modes such as multiple spurious operations during the equipment selection phase can be performed given sufficient justification that the impact of this exclusion meets the above criteria. For example, if cable-to-cable interactions are considered unlikely, and it can be shown that multiple cable-to-cable spurious operations will not impact the model per the above exclusion criteria, then the failure mode and the possible associated accident sequences can be excluded from the model (with supporting justification).

Table 4-2.2-4 Supporting Requirements (SR) for HLR-ES-C

The fire PRA shall identify instrumentation whose failure, including spurious operation, would impact the reliability of operator actions associated with that portion of the plant design to be credited in the fire PRA (HLR-ES-C).

Index No. ES-C	Capability Category I	Capability Category II	Capability Category III
ES-C1 [Notes (1)–(3)]	IDENTIFY instrumentation that is relevant to the operator actions for which HFEs are defined or modified to account for the context of fire scenarios in the fire PRA, per Requirements HRA-B1 and HRA-B2.		
ES-C2 [Notes (4)–(6)]	IDENTIFY instrumentation associated with each operator action to be addressed, based on the following: *(a)* fire-induced failure of any single instrument whereby one of the modes of failure to be considered is spurious operation of the instrument *(b)* instances in which the potential consequence of the instrumentation failure is different from the consequences of other selected equipment whose failures, including spurious operation, will be included in the fire PRA plant response model	IDENTIFY instrumentation associated with each operator action to be addressed, based on the following: *(a)* fire-induced failure of any single instrument whereby one of the modes of failure to be considered is spurious operation of the instrument *(b)* fire-induced failure, including a spurious indication, even if it is not relevant to the HFEs for which instrumentation is identified within the scope defined by Requirement ES-C1, if the failure could cause an undesired operator action related to that portion of the plant design credited in the analysis	IDENTIFY instrumentation associated with each operator action to be addressed, based on the following: *(a)* fire-induced failures of up to and including two instruments at a time whereby one of the modes of failure to be considered is spurious operation of the instruments *(b)* fire-induced failures, including spurious indications, even if they are not relevant to the HFEs for which instrumentation is identified within the scope defined by Requirement ES-C1, if the failure could cause an undesired operator action related to that portion of the plant design credited in the analysis

NOTES:

(1) The scope of the fire PRA as determined by how the prior HLRs in this Part have been met (i.e., what of the plant design is being credited in the fire PRA per the capability categories) implies a scope of operator actions that are relevant to the fire PRA. For example, if the reactor core isolation cooling system in a BWR is not being credited in the fire PRA, then operator actions associated with its use are not relevant (to the extent that the system and its associated actions do not affect other systems/equipment and associated actions that are being credited in the fire PRA). Instrumentation needed to perform such actions as starting, stopping, isolating, recovering from spurious events, or otherwise controlling the reactor core isolation cooling system do not need to be identified. For the operator actions to be addressed in the fire PRA per Requirements HLR-HRA-A through HLR-HRA-D and its incorporation of appropriate Part 2 requirements for human actions, instrumentation associated with the relevant actions is expected to be identified. Fire-induced failure, including spurious operation of this instrumentation, may prevent or delay a desirable action (e.g., fire causes the indications for the need to start feed and bleed to not be available) or cause an inappropriate action (e.g., a spurious pump high-temperature alarm causes the operator to immediately shut down a pump per procedures even though the pump is not really experiencing high temperature, thereby reducing mitigation capability). The gradation of the amount of instrumentation to be identified across capability categories is inherently based on the above considerations.

(2) Instrumentation needs to be considered because of the higher probability for fire-induced indication failure including spurious indications as compared to the potential for random indication failure. Hence, while random failures of instrumentation may often be ignored in an internal-events PRA, fire-induced instrumentation failure needs to be included in a fire PRA.

Table 4-2.2-4 Supporting Requirements (SR) for HLR-ES-C (Cont'd)

NOTES: (Cont'd)

(3) The intent of Requirement ES-C1 is to provide limits on the scope of instruments to be identified in accordance with the risk importance of credited operator actions. For example, if the use of a conservative screening HEP shows that an operator action is not a significant contributor, then the analyst may choose not to identify instrumentation and, by implication of Requirement CS-A1, not to complete cable tracing for such instruments. However, it is intended that this SR will require that the instruments that are relied on for credited operator actions will be identified and verified as available to a level of detail commensurate with the risk importance and quantification of the HEPs.

(4) Random instrumentation failures during the fire do not need to be addressed.

(5) Consideration of just one fire-induced spurious indication relevant to each operator action being addressed for Capability Categories I and II is indicative of balancing (a) the current state of the art and the resources required to consider almost innumerable combinations of two or more spurious indications against (b) the desire to capture in the fire PRA the associated risk caused by such spurious indications.

(6) Capability Category III includes consideration of other instrumentation not needed to directly affect the modeled actions (e.g., other "nuisance" alarms or indications) but that may still cause undesired operator effects that are relevant to the fire PRA.

Table 4-2.2-5 Supporting Requirements (SR) for HLR-ES-D

The fire PRA shall document the fire PRA equipment selection, including that information about the equipment necessary to support the other fire PRA tasks (e.g., equipment identification; equipment type; normal, desired, failed states of equipment) in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-ES-D).

Index No. ES-D	Capability Category I	Capability Category II	Capability Category III
ES-D1 [Note (1)]	DOCUMENT the identified equipment in a manner that facilitates fire PRA applications, upgrades, and peer review and is sufficient to support the other fire PRA tasks so that *(a)* it is clear which equipment will be associated with determining initiating events in the fire PRA plant response model for the postulated fires *(b)* the equipment and its failures including spurious operation or indication can be modeled appropriately *(c)* cables associated with the equipment can be identified *(d)* failure modes of interest for the equipment are clear so as to support circuit analyses if required Justifications are provided with regard to equipment considered but screened out of the fire PRA including when meeting Requirement ES-A3 relevant to meeting Requirement IE-C4 in Part 2 for initiating events, meeting Requirement ES-B5 for the mitigating equipment to be credited in the fire PRA, and using the "exception" under Requirement ES-C2 for instrumentation considerations.		

NOTE:

(1) Documentation does not necessarily imply a separate/unique list of equipment, although this may prove useful. For instance, inclusion in the fire PRA plant response model can be a part of "documenting" the equipment included and its failure modes. The ability to create such a list should exist especially for peer review efficiency as well as for conducting the fire PRA itself.

4-2.3 CABLE SELECTION AND LOCATION

The objectives of the cable selection and location element (CS) are to ensure that

(a) all cables needed to support proper operation of equipment selected per technical element ES (see 4-2.2) are identified and assessed for relevance to the fire PRA plant response model

(b) the plant location information for selected cables is sufficient to support the fire PRA and its intended applications

The development of a fire PRA requires detailed spatial location information for credited plant equipment, cables, features, and systems. The extent and level of resolution of these data have a material impact on the validity of the resulting risk assessments. The consequences of postulated fires include the failure of plant equipment and cables. The failure of cables could cause plant equipment to become unavailable to perform their credited function or could cause them to operate in an undesired manner (i.e., spurious operation). These failures include pump motors failing to operate, valves failing to open or close, breakers failing to trip or close, and instrument control and system logic signals failing to be generated or being generated spuriously. Spurious operation events include the unintended operation of the equipment mentioned above. The treatment of spurious signals includes the occurrence of erroneous instrument indications. The consequences of such events are treated in the fire PRA plant response model.

The level of spatial resolution for the cable location data has a direct effect on the precision of the resulting risk assessment. An important attribute of a fire PRA is the ability to correlate cable spatial location information to physical analysis units, to specific locations within a physical analysis unit, and/or to specific raceways, as applicable, to allow the treatment of fire consequences for the fire scenario under consideration. The level of detail to which cable spatial location information is available may impact the ability to analyze fire scenarios in which cable damage is shown to be localized.

Table 4-2.3-1 High Level Requirement for Cable Selection and Location (CS)

Designator	Requirement
HLR-CS-A	The fire PRA shall identify and locate the plant cables whose failure could adversely affect credited equipment or functions included in the fire PRA plant response model, as determined by the equipment selection process (Requirements HLR-ES-A, HLR-ES-B, and HLR-ES-C).
HLR-CS-B	The fire PRA shall *(a)* perform a review for additional circuits that are required to support a credited circuit (i.e., per Requirement HLR-CS-A) or whose failure could adversely affect a credited circuit *(b)* identify any additional equipment and cables related to these additional circuits in a manner consistent with the other equipment and cable selection requirements of this Standard
HLR-CS-C	The fire PRA shall document the cable selection and location process and results in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.3-2 Supporting Requirements (SR) for HLR-CS-A

The fire PRA shall identify and locate the plant cables whose failure could adversely affect credited equipment or functions included in the fire PRA plant response model, as determined by the equipment selection process (Requirements HLR-ES-A, HLR-ES-B, and HLR-ES-C) (HLR-CS-A).

Index No. CS-A	Capability Category I	Capability Category II	Capability Category III
CS-A1 [Notes (1) and (2)]	IDENTIFY cables whose fire-induced failure could adversely affect selected equipment and/or credited functions in the fire PRA plant response model.		
CS-A2 [Notes (3)–(4)]	IDENTIFY those circuits whose fire-induced failure due to hot shorts (intracable and intercable), by themselves, would adversely affect selected equipment due to spurious operation. IDENTIFY the cables supporting any identified circuits where hot shorts impacting any one cable (including both intracable and intercable hot shorts) could lead to spurious operation of selected equipment.	IDENTIFY those circuits whose fire-induced failure due to hot shorts (intracable and intercable), by themselves, would adversely affect selected equipment due to spurious operation. IDENTIFY the cables supporting any identified circuits where hot shorts impacting up to and including two cables (including both intracable and intercable hot shorts) could lead to spurious operation of selected equipment.	IDENTIFY those circuits whose fire-induced failure due to hot shorts (intracable and intercable), by themselves, would adversely affect selected equipment due to spurious operation. IDENTIFY the cables supporting any identified circuit where hot shorts impacting more than two cables (including both intracable and intercable hot shorts) could lead to spurious operation of selected equipment. JUSTIFY the acceptability of the number of concurrent hot shorts assumed feasible if relied upon for limiting the scope of this task by demonstrating that higher-order combinations are a negligible contributor to overall risk.
CS-A3 [Note (5)]	IDENTIFY any additional cables required to support the proper operation of, or whose failure could adversely affect, credited equipment or functions due to power supply and support system equipment, and IDENTIFY the related equipment per Requirement HLR-ES-A, HLR-ES-B, or HLR-ES-C, as applicable.		
CS-A4	If additional cables are selected based on Requirement CS-A3, ENSURE that the adverse effects due to failure of the selected cables are included in the fire PRA plant response model.		
CS-A5	INCLUDE cable conductor-to-ground and conductor-to-conductor shorts (both intracable and intercable) as potential cable and circuit failure modes.		
CS-A6	INCLUDE circuit failure modes associated with the effects of circuits de-energizing as a result of the design operation of overcurrent protective devices responding to fire-induced cable short circuits.		

Table 4-2.3-2 Supporting Requirements (SR) for HLR-CS-A (Cont'd)

The fire PRA shall identify and locate the plant cables whose failure could adversely affect credited equipment or functions included in the fire PRA plant response model, as determined by the equipment selection process (Requirements HLR-ES-A, HLR-ES-B, and HLR-ES-C) (HLR-CS-A).

Index No. CS-A	Capability Category I	Capability Category II	Capability Category III
CS-A7 [Note (6)]	For ungrounded power distribution systems for three-phase-powered equipment that could spuriously operate due to proper polarity intercable hot shorts, INCLUDE these cable and circuit failure modes in the fire PRA plant response model to the extent that a spurious operation of a single piece of equipment might lead to an interfacing system LOCA or containment bypass that results in core damage and large early release.		
CS-A8 [Note (7)]	IDENTIFY instances where thermoplastic insulated power supply circuits are applied. INCLUDE the treatment of cable failures involving three-phase-powered equipment that could spuriously operate and lead to an interfacing system LOCA or containment bypass that results in core damage and large early release due to a proper polarity three-phase hot short.		
CS-A9	INCLUDE consideration of proper polarity hot shorts on ungrounded DC circuits; requiring up to and including two independent faults could result in adverse consequences.		
CS-A10	IDENTIFY the physical analysis units, consistent with the plant partitioning analysis, through which each cable associated with a credited fire PRA function passes. ENSURE that the information includes treatment of cable terminal end locations.		IDENTIFY the physical analysis units, consistent with the plant partitioning analysis, and electrical raceways through which each cable associated with a credited fire PRA function passes. ENSURE that the information includes treatment of cable terminal end locations.
CS-A11 [Note (11)]	If assumed cable routing is used in the fire PRA, IDENTIFY the scope and extent, and SPECIFY a basis for the assumed cable routing.		

NOTES:

(1) The scope of equipment included in the fire PRA is identified in Requirements HLR-ES-A, HLR-ES-B, and HLR-ES-C. The treatment of their specific credited function(s) or postulated failures of concern is addressed in the fire PRA PRM element requirements.

(2) The explicit identification of individual cables is not necessary in those instances where the provision of Requirement CS-A11 is used.

(3) This SR limits consideration to hot shorts that might be imposed upon the specified number of target cables (i.e., one target cable for Category I, two target cables for Category II, and more than two for Category III). However, the analysis must include the possibility that the energizing source might be introduced through an intercable short to any second cable.

(4) The treatment of hot shorts leading to fire-induced spurious operation in Requirement CS-A2 is intended to be applied on a per component basis. That is, it is not necessary, or intended, that at this stage the analyst would have any knowledge of the specific fire scenarios that might ultimately be defined (i.e., based on the FSS element). Rather, cable hot shorts and fire-induced spurious operations are considered strictly in the context of how such failures might impact each piece of plant equipment that was selected per the ES element.

Table 4-2.3-2 Supporting Requirements (SR) for HLR-CS-A (Cont'd)

NOTES: (Cont'd)

(5) The process of identifying credited equipment in Requirements HLR-ES-A, HLR-ES-B, and HLR-ES-C is necessarily limited by the scope of drawing and documents that are reviewed to perform that task. During the process of identifying required cables, control circuit elements may be identified that require power supplies or support systems not otherwise identified in Requirement HLR-ES-A, HLR-ES-B, or HLR-ES-C.

(6) Ungrounded power distribution systems are those designed to continue to function without automatic tripping (isolation) of the affected circuit in the event of a single line-to-ground fault.

(7) This SR is based on the interpretation of existing experimental evidence that indicates that the conditional probability of intercable hot shorts between thermoplastic insulated cables is high enough that proper polarity three-phase hot shorts cannot be dismissed based on their likelihood alone. Hence, some consideration of potential consequences is appropriate. For interfacing system LOCAs and CDF leading to LERF scenarios, the consideration of this cable failure mode is required. In contrast, for thermoset insulated cables the conditional probability of a three-phase proper polarity intercable short is considered of such low likelihood that they need not be considered as a plausible failure mode. The intent of Requirement CS-A8 is to ensure treatment consistent with these insights.

(8) The fire PRA should strive for completeness in its cable routing information. It is acknowledged, however, that practicality may limit the completeness of cable routing information. If full cable routing information is not developed, the routing of cables on an exclusionary basis is acceptable. That is, if it can be established (based on the physical features and layout of the plant) that a particular cable (or group of cables) is not routed through a given physical analysis unit (or specific location within a physical analysis unit), then the fire PRA may assume that the excluded cable(s) will not fail for fire scenarios where fire-induced damage is limited to that physical analysis unit (or to a specific location within a physical analysis unit).

(9) A cable terminal end location refers to the location where each end of the cable is terminated at some piece of plant equipment. In some cases, the cable might enter this equipment from the floor below. In these cases, the cable routing information must reflect the presence of the cable in the fire area or fire compartment where it is actually terminated.

(10) The resolution of the cable location information (areas versus compartments versus rooms versus tray nodes) has an influence on the capability category determination for Requirement FSS-A5.

(11) The fire PRA may make conservative assumptions regarding cable locations. That is, if the exact routing of a cable (or group of cables) has not been established, the fire PRA should assume that those cables fail for any fire scenario that has a damaging effect on any raceway or location where the subject cable might reasonably exist. The resulting capability category if this option is taken is to be based on the general guidance provided in Table 1-1.3-2 for both resolution and realism. The determination of where cables might reasonably exist should consider factors that include the physical layout of the plant equipment and the routing of cables treated explicitly using Requirement CS-A10 from nearby or identical locations. The intent is to allow for the application of conservative assumptions in cases where the specific routing of a cable is not known. For example, if the analyst can provide reasonable assurance that a cable is *not* located in a particular physical analysis unit, the intent is to allow the fire PRA to assume that cable would not fail for fire scenarios whose effects remain confined to that physical analysis unit. The intent of the related Requirements CS-A11 and FSS-A3 is to impose a burden on the analyst to identify all such cases and to justify those assumptions in the context of the fire scenario selection and analysis (FSS) technical element.

Table 4-2.3-3 Supporting Requirements (SR) for HLR-CS-B

The fire PRA shall
(a) perform a review for additional circuits that are required to support a credited circuit (i.e., per Requirement HLR-CS-A) or whose failure could adversely affect a credited circuit
(b) identify any additional equipment and cables related to these additional circuits in a manner consistent with the other equipment and cable selection requirements of this Standard (HLR-CS-B)

Index No. CS-B	Capability Category I	Capability Category II	Capability Category III
CS-B1	REVIEW the existing electrical overcurrent coordination and protection analysis. IDENTIFY any additional circuits and cables whose failure could challenge power supply availability due to inadequate or unanalyzed electrical overcurrent protective device coordination.	ANALYZE all electrical distribution buses credited in the fire PRA plant response model for proper overcurrent coordination and protection. IDENTIFY any additional circuits and cables whose failure could challenge power supply availability due to inadequate electrical overcurrent protective device coordination.	

Table 4-2.3-4 Supporting Requirements (SR) for HLR-CS-C

The fire PRA shall document the cable selection and location process and results in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-CS-C).

Index No. CS-C	Capability Category I	Capability Category II	Capability Category III
CS-C1	DOCUMENT the cable selection and location methodology applied in the fire PRA in a manner that facilitates fire PRA applications, upgrades, and peer review.		
CS-C2	DOCUMENT cable selection and location results such that those results are traceable to plant source documents in a manner that facilitates fire PRA applications, upgrades, and peer review.		
CS-C3	If the provision of Requirement CS-A11 is used, DOCUMENT the assumed cable routing and the basis for concluding that the routing is reasonable in a manner that facilitates fire PRA applications, upgrades, and peer review.		
CS-C4	DOCUMENT the review of the electrical distribution system overcurrent coordination and protection analysis in a manner that facilitates fire PRA applications, upgrades, and peer review.		

4-2.4 QUALITATIVE SCREENING

(a) The objective of the qualitative screening (QLS) element is to identify physical analysis units whose potential fire risk contribution can be judged negligible without quantitative analysis.[6]

(b) In this element, physical analysis units are examined only in the context of their individual contribution to fire risk. The potential risk contribution of all physical analysis units is reexamined in the multicompartment fire scenario analysis regardless of the physical analysis unit's disposition during qualitative screening.[7]

The QLS element is not an absolute necessity of a fire PRA. Under some circumstances, an analyst may choose to bypass the QLS element and simply retain all physical analysis units for quantitative analysis. However, if any one (or more) physical analysis unit(s) defined as within the global analysis boundary is (are) not analyzed quantitatively, then a qualitative screening analysis is implied, and the QLS element requirements would apply.

[6] Quantitative screening considers physical analysis units consistent with the results of the plant partitioning analysis as discussed per Requirement HLR-PP-B and its supporting requirements as specified in 4-2.1.

[7] See 4-2.6 for further discussion of the identification and evaluation of multicompartment fire scenarios.

The SRs for QLS are nominally the same for all capability categories. However, an inherent distinction exists due to the intimate relationship between QLS and the prior elements PP (4-2.1), ES (4-2.2), and CS (4-2.3). These prior elements define the predominant factors assessed in the qualitative screening criteria, namely, the physical analysis units being examined, the list of relevant equipment, the list of relevant cables, and the mapping of cables (including cable end points) to physical analysis units and/or to electrical raceways. Hence, the scope defined by these prior elements will largely define the scope and level of rigor associated with qualitative screening. The intent is to ensure that the QLS element is performed to a scope and level of rigor in a manner consistent with these three prior and related elements.

Table 4-2.4-1 High Level Requirement for Qualitative Screening (QLS)

Designator	Requirement
HLR-QLS-A	The fire PRA shall identify those physical analysis units that screen out as individual risk contributors without quantitative analysis.
HLR-QLS-B	The fire PRA shall document the results of the qualitative screening analysis in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.4-2 Supporting Requirements (SR) for HLR-QLS-A

The fire PRA shall identify those physical analysis units that screen out as individual risk contributors without quantitative analysis (HLR-QLS-A).

Index No. QLS-A	Capability Category I	Capability Category II	Capability Category III
QLS-A1	RETAIN for quantitative analysis those physical analysis units that contain equipment or cables required to ensure as-designed circuit operation, or whose failure could cause spurious operation, of any equipment, system, function, or operator action credited in the fire PRA plant response model.		
QLS-A2 [Note (1)]	RETAIN for quantitative analysis those physical analysis units where a fire might require a manual or automatic plant trip or a controlled manual shutdown based on plant Technical Specifications. If a time limit is established for a required Technical Specifications required shutdown, SPECIFY a basis for the applied time window.		
QLS-A3 [Note (2)]	APPLY the screening criteria to each physical analysis unit defined in the plant partitioning analysis.		
QLS-A4 [Note (3)]	If additional qualitative screening criteria are applied, SPECIFY the applied criteria and SPECIFY a basis that shows the applied criteria provide reasonable assurance that the screened-out physical analysis units are negligible contributors to fire risk in a manner consistent, at a minimum, with Requirements QLS-A1, QLS-A2, and QLS-A3.		

NOTES:
(1) Fire PRA practice may involve screening out physical analysis units if the time available before a required shutdown due to a Technical Specification violation is long. This Standard does not establish a specific time limit but acknowledges the potential validity of this approach. It is expected that analysts will define and provide a basis for their approach if an upper-bound time limit is applied beyond which a shutdown required by the Technical Specifications will not be considered as an initiating event.
(2) It is acceptable for the qualitative screening analysis to retain any physical analysis unit for quantitative analysis without a rigorous application of the defined qualitative screening criteria.
(3) Requirements QLS-A1, QLS-A2, and QLS-A3 represent minimum criteria. The intent of Requirement QLS-A4 is to allow for the application of additional screening criteria. However, if additional criteria are applied, then they must be defined, and a basis for their acceptability must be established.

Table 4-2.4-3 Supporting Requirements (SR) for HLR-QLS-B

The fire PRA shall document the results of the qualitative screening analysis in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-QLS-B).

Index No. QLS-B	Capability Category I	Capability Category II	Capability Category III
QLS-B1	DOCUMENT the qualitative screening criteria applied.		
QLS-B2	DOCUMENT the disposition of each physical analysis unit defined by the plant partitioning analysis as either "screened out" or "retained for quantitative analysis" and in a manner that facilitates fire PRA applications, upgrades, and peer review.		
QLS-B3	DOCUMENT the exclusion basis for each physical analysis unit defined in the plant partitioning analysis that has been screened out in a manner that facilitates fire PRA applications, upgrades, and peer review.		

4-2.5 FIRE PRA PLANT RESPONSE MODEL

The objectives of the fire PRA plant response model (PRM) element are

(a) to identify the initiating events that can be caused by a fire event and develop a related accident sequence model

(b) to depict the logical relationships among equipment failures (both random and fire induced) and human failure events (HFEs) for CDF and LERF assessment when combined with the initiating event frequencies

The fire PRA PRM requires the use and integration of the results of meeting many other parts in this Standard as is iterated in the requirements in this Part. The fire PRA PRM must ultimately be consistent with the results of the equipment and cable selection elements ES and CS of this Standard and will include all selected plant equipment from ES and the associated cable failures from CS but will not include (or will fail) plant equipment that was not selected in the ES element.

The requirements are written in anticipation that analysts will not be performing this element in a vacuum but will instead conduct this element starting with an internal-events PRA that has been assessed against Part 2. Appropriately, many of the requirements in this Part call upon or otherwise parallel requirements found in Part 2 with clarifications as noted herein to produce the fire PRA PRM.

This Part establishes expectations of the fire PRA plant response model as well as overall scope considerations for the model. Subsections 4-2.2 and 4-2.3 provide the majority of the overall scope by defining the equipment and corresponding cables as well as the locations of both the equipment and cables that are to be treated in the fire PRA plant response model (i.e., the impacts of both equipment and cable failures are modeled). This treatment is to include modeling of the equipment failure modes attributable to fire-induced damage to either or both the equipment and cables depending on the location of the fire. The remaining HLRs and SRs of this Part provide the detailed requirements for constructing and documenting the model, calling upon other parts of this standard where necessary, and paralleling Part 2 for internal-events PRAs as appropriate. The level of modeling detail is expected to be consistent with that allowed by the quantitative screening per 4-2.8 if quantitative screening is performed. The capability categories of these other parts and those of the referenced Part 2 provide the possible gradations in meeting the requirements of this Part.

It is anticipated that substantial changes may be needed to the internal-events PRA model (i.e., the accident sequences) to meet the needs of the fire PRA. It is expected that the fire PRA PRM will be constructed by modifying the corresponding internal-events PRA models, and the PRM requirements are written from this perspective. Elements of the fire PRA plant response model that are carried over directly from the internal-events PRA are assumed to meet the same capability category as assigned for the internal-events PRA unless that factor requires modification or reanalysis given the specific context of a fire event. In such cases, the assessment of the capability category met by the fire PRA may be unique.

Table 4-2.5-1 High Level Requirement for Fire PRA Plant Response Model (PRM)

Designator	Requirement
HLR-PRM-A	The fire PRA shall include the fire PRA plant response model capable of supporting Requirements HLR-FQ-A through HLR-FQ-F.
HLR-PRM-B	The fire PRA plant response model shall include fire-induced initiating events, both fire-induced and random failures of equipment, fire-specific as well as non–fire-related human failures associated with safe shutdown, accident progression events (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the SRs provided under this HLR that parallel, as appropriate, Part 2 of this Standard, for internal-events PRA.
HLR-PRM-C	The fire PRA shall document the fire PRA plant response model in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.5-2 Supporting Requirements (SR) for HLR-PRM-A

The fire PRA shall include the fire PRA plant response model capable of supporting the HLR requirements of FQ.

Index No. PRM-A	Capability Category I	Capability Category II	Capability Category III
PRM-A1	CONSTRUCT the fire PRA plant response model so that it is capable of determining fire-initiated conditional core damage probabilities (CCDPs) and conditional large early release probabilities (CLERPs) for various fire scenarios.		
PRM-A2	CONSTRUCT the fire PRA plant response model so that it is capable of determining fire-initiated core damage frequencies (CDFs) and fire-initiated large early release frequencies (LERFs) once the fire frequencies (see Requirements HLR-IGN-A and HLR-IGN-B, 4-2.7) are also applied to the quantification.		
PRM-A3	CONSTRUCT the fire PRA plant response model so that it is capable of determining the significant contributors to the fire-induced risk consistent with the FQ technical element (see 4-2.12).		
PRM-A4	CONSTRUCT the fire PRA plant response model in a manner consistent with the scope and location of equipment and cables (accounting for cable damage effects on the equipment of interest) per the ES and CS technical elements (see 4-2.2 and 4-2.3).		

Table 4-2.5-3 Supporting Requirements (SR) for HLR-PRM-B

The fire PRA plant response model shall include fire-induced initiating events, both fire-induced and random failures of equipment, fire-specific as well as non–fire-related human failures associated with safe shutdown, accident progression events (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the supporting requirements provided under this HLR that parallel, as appropriate, Part 2 for internal-events PRA (HLR-PRM-B).

Index No. PRM-B	Capability Category I	Capability Category II	Capability Category III
PRM-B1 [Note (1)]	USE the internal-events PRA initiating events and accident sequences for both CDF and LERF as the basis for development of the fire PRA PRM.		
PRM-B2	ENSURE that the peer review exceptions and deficiencies for the internal-events PRA are dispositioned, and that the disposition does not adversely affect the development of the fire PRA plant response model.		
PRM-B3 [Note (2)]	IDENTIFY any new initiating events arising from the considerations of the ES and CS technical elements (see 4-2.2 and 4-2.3) that might result from a fire event that were not included in the internal-events PRA, including those arising from the consideration of fire-induced spurious operation.		
PRM-B4 [Note (3)]	MODEL any new initiating events identified per Requirement PRM-B3 in accordance with Requirements HLR-IE-A, HLR-IE-B, and HLR-IE-C and their SRs in Part 2, with the following clarification: all SRs under Requirements HLR-IE-A and HLR-IE-B, and Requirements IE-C4, IE-C6, IE-C7, IE-C8, IE-C9, and IE-C12 in Part 2 are to be addressed in the context of a fire inducing the initiating events, excluding initiating events that cannot be induced by a fire. SPECIFY a defined basis to support the claim of nonapplicability of any of these requirements in Part 2.		
PRM-B5	For those fire-induced initiating events included in the internal-events PRA model, REVIEW the corresponding accident sequence models and *(a)* IDENTIFY any existing accident sequences that will require modification based on unique aspects of the plant fire response procedures in accordance with Requirements HLR-AS-A and HLR-AS-B of Part 2 and their supporting requirements *(b)* IDENTIFY any new accident sequences that might result from a fire event that were not included in the internal-events PRA in accordance with Requirements HLR-AS-A and HLR-AS-B in Part 2 and their supporting requirements		
PRM-B6	MODEL accident sequences for any new initiating events identified per Requirement PRM-B3 and any accident sequences identified per Requirement PRM-B5 reflective of the possible plant responses to the fire-induced initiating events in accordance with Requirements HLR-AS-A and HLR-AS-B and their SRs in Part 2 with the following clarifications, and SPECIFY a defined basis to support the claim of nonapplicability of any of the following requirements in Part 2: *(a)* All the SRs under Requirements HLR-AS-A and HLR-AS-B in Part 2 are to be addressed in the context of fire scenarios, including effects on equipment, associated cabling, operator actions, and accident progression and timing. *(b)* When applying Requirement AS-A5 in Part 2 to fire PRA, INCLUDE consideration of fire response procedures as well as emergency operating procedures and abnormal procedures.		
PRM-B7	IDENTIFY any cases where new or modified success criteria will be needed to support the fire PRA consistently with Requirements HLR-SC-A and HLR-SC-B in Part 2 and their supporting requirements.		

Table 4-2.5-3 Supporting Requirements (SR) for HLR-PRM-B (Cont'd)

The fire PRA plant response model shall include fire-induced initiating events, both fire-induced and random failures of equipment, fire-specific as well as non–fire-related human failures associated with safe shutdown, accident progression events (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the supporting requirements provided under this HLR that parallel, as appropriate, Part 2 for internal-events PRA (HLR-PRM-B).

Index No. PRM-B	Capability Category I	Capability Category II	Capability Category III
PRM-B8	For any cases identified per Requirement PRM-B7, CONSTRUCT the fire PRA plant response model using success criteria that are defined in accordance with Requirements HLR-SC-A and HLR-SC-B and their SRs in Part 2. SPECIFY a defined basis to support the claim of nonapplicability of any of these requirements in Part 2.		
PRM-B9	For any cases where new system models or split fractions are needed, or existing models or split fractions need to be modified to include fire-induced equipment failures, fire-specific operator actions, and/or fire-induced spurious operations, PERFORM the systems analysis portion of the fire PRA model in accordance with Requirements HLR-SY-A and HLR-SY-B and their SRs in Part 2 with the following clarification, and SPECIFY a defined basis to support the claim of nonapplicability of any of these requirements in Part 2: All the SRs under Requirements HLR-SY-A and HLR-SY-B in Part 2 are to be addressed in the context of fire scenarios, including effects on system operability/functionality accounting for fire damage to equipment and associated cabling.		
PRM-B10 [Notes (4) and (5)]	MODIFY the fire PRA plant response model so that systems and equipment that were included in the internal-events PRA but were not selected in the ES technical element (see 4-2.2), and that are potentially vulnerable to fire-induced failure, are failed in the worst possible failure mode, including fire-induced spurious operation.		
PRM-B11	MODEL all operator actions and operator influences consistent with the HRA technical element (see 4-2.10).		
PRM-B12	IDENTIFY any fire PRA PRM probability input values that either require reanalysis given the fire context or that were not included in the internal-events PRA.		
PRM-B13 [Notes (6) and (7)]	For any item identified per Requirement PRM-B12, PERFORM the data analysis portion of the fire PRA plant response model in accordance with Requirements HLR-DA-A through HLR-DA-D and their SRs in Part 2 with the following clarification: all the SRs under Requirements HLR-DA-A through HLR-DA-D in Part 2 are to be addressed in the context of both random events as well as fire events causing damage to equipment and associated cabling. SPECIFY a defined basis to support the claim of nonapplicability of any of these requirements in Part 2.		
PRM-B14 [Notes (8)–(10)]	IDENTIFY any new accident progressions beyond the onset of core damage that would be applicable to the fire PRA that were not addressed for LERF estimation in the internal-events PRA.		

Table 4-2.5-3 Supporting Requirements (SR) for HLR-PRM-B (Cont'd)

The fire PRA plant response model shall include fire-induced initiating events, both fire-induced and random failures of equipment, fire-specific as well as non–fire-related human failures associated with safe shutdown, accident progression events (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the supporting requirements provided under this HLR that parallel, as appropriate, Part 2 for internal-events PRA (HLR-PRM-B).

Index No. PRM-B	Capability Category I	Capability Category II	Capability Category III
PRM-B15	MODEL any new accident progressions beyond the onset of core damage identified per Requirement PRM-B13 to determine the fire-induced LERF in accordance with Requirements HLR-LE-A through HLR-LE-D and their SRs in Part 2 with the following clarifications: *(a)* All the SRs under Requirements HLR-LE-A through HLR-LE-D in Part 2 are to be addressed in the context of fire scenarios, including effects on system operability/functionality, operator actions, accident progression, and possible containment failures, including fire damage to equipment and associated cabling. *(b)* Requirements LE-C2 and LE-C6 in Part 2 are to be met in a manner consistent with Requirements HLR-HRA-A through HLR-HRA-D (4-2.10). *(c)* Requirement LE-C6 in Part 2 is to be met in a manner consistent with Requirement PRM-B9. *(d)* Requirement LE-C8 in Part 2 is to be met in a manner consistent with Requirement PRM-B6. SPECIFY a defined basis to support the claim of nonapplicability of any of these requirements in Part 2.		

NOTES:
(1) If the available analysis has not been assessed against Part 2, then the fire PRA faces an additional burden to demonstrate that the entire fire PRA plant response model meets the applicable requirements of Part 2.
(2) Requirements HLR-ES-A addresses identification of equipment associated with initiating events.
(3) The modeling of initiating events will need to support the analysis of fire scenarios and will therefore need to be able to incorporate the corresponding fire-induced equipment and cable failures as defined by the CS, CF, and FSS technical elements. When complete, the PRM will encompass all of the initiating events needed to quantify fire risk.
(4) This SR ensures proper treatment of equipment credited in the internal-events PRA that has not been selected per the ES element and that has therefore not been traced to specific plant locations. Similar assumptions are made with respect to cables that have been selected per the CS element but were not fully traced to specific plant locations (see Requirement CS-A10).
(5) Analytical iteration on the ES element may result in changes to equipment selection, which may in turn require iteration on this SR as well.
(6) This requirement does not apply to data specific to technical elements FSS, IGN, and CF.
(7) It is expected that the following are included in meeting this SR:
(a) Recognize that some failure probabilities are 1.0 for certain physical analysis units (e.g., the target is expected to fail given the fire or associated cables of a component are not traced because of insufficient information).
(b) Data values should account for any data required per 4-2.9 to the extent that subsection is applied and it subsequently affects the data analysis.
(8) This requirement is intended to provide greater assurance than that obtained by meeting the SRs of Requirements HLR-ES-A through HLR-ES-D, HLR-PRM-A, and HLR-PRM-B that the fire PRA results capture the most risk-significant contributors. Such contributors include spurious operation–type failures that may have been limited in number in the model (e.g., the two spurious operation requirement under Capability Category II of Requirement ES-B2).
(9) It is acknowledged that this is an evolving technical area. It is expected that a generally accepted practice would evolve to address this SR.

Table 4-2.5-3 Supporting Requirements (SR) for HLR-PRM-B (Cont'd)

NOTES: (Cont'd)
(10) An example of a new initiator to be considered would be a PWR boron dilution event that was initially not modeled since it required three spurious operations to occur. An example of new basic events considered would be where a significant contributing sequence involving spurious operation of two valves (two spurious operations) results in a review of failures involving three spurious operations. If a combination of three spurious operations could lead to the same sequence, and if this could result in new significant contributing sequences, it may be appropriate to include the new basic events in the model. New basic events may also be added if a significant contributing sequence did not include consideration for spurious operation (due to limitations in Requirement ES-B2), but the same sequence can occur when additional spurious operations are considered. An example of a new accident sequence might include adding a spurious sump valve opening during a spurious Safety Injection, where new systems may be needed to provide sump water for injection.

Table 4-2.5-4 Supporting Requirements (SR) for HLR-PRM-C

The fire PRA shall document the fire PRA plant response model in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-PRM-C).

Index No. PRM-C	Capability Category I	Capability Category II	Capability Category III
PRM-C1	DOCUMENT the fire PRA plant response model in a manner consistent with Requirements HLR-IE-D, HLR-AS-C, HLR-SC-C, HLR-SY-C, and HLR-DA-E and their SRs in Part 2 as well as 4-2.10, with the following clarifications: *(a)* Requirement HLR-IE-D in Part 2 is to be met in a manner consistent with Requirement HLR-IGN-B of this Standard. *(b)* DOCUMENT any defined bases to support the claim of nonapplicability of any of the referenced requirements in Part 2 beyond that already covered by the clarifications in 4-2.5.		

4-2.6 FIRE SCENARIO SELECTION AND ANALYSIS

"Fire scenario" in this Standard is defined broadly to include the set of elements that describes a fire event. The elements usually include a fire location (i.e., a physical analysis unit or location within a physical analysis unit), the characteristics of the source fire (i.e., ignition source, flames, hot gas production, etc.), detection and suppression features to be considered, targets (i.e., damage targets), and intervening combustibles to which the fire might spread. Fire scenarios considered in a fire PRA may range from very simplistic (e.g., any fire within a physical analysis unit damages all damage targets present) to realistic (e.g., fire initiates at a specific ignition source, grows, and damages nearby damage targets while detection and suppression are delayed considerably).

The objectives of the fire scenario selection and analysis (FSS) element are to

(a) select a set of fire scenarios for each physical analysis unit that has not been screened out and upon which fire risk estimates will be based

(b) characterize the selected fire scenarios

(c) determine the likelihood and extent of risk-relevant fire damage for each selected fire scenario including

(1) an evaluation of the fire-generated conditions at the target location including fire spread to secondary combustibles

(2) an evaluation of the thermal response of damage targets to such exposure

(3) an evaluation of fire detection and suppression activities

(d) examine multicompartment fire scenarios

The total fire risk associated with a physical analysis unit is an aggregate of risk contributions from one or more individual fire scenarios postulated in that physical analysis unit. Paragraph 4-2.6 states the requirements associated with the fire scenario selection and analysis efforts including the application of fire modeling tools and performance assessments for fire protection systems and features. An additional area of analysis is the potential for severe fire-induced damage, including collapse of exposed structural steel. This Standard includes requirements for the treatment of such scenarios. The potential relevance of such scenarios would be dependent on the intended fire PRA application.

Requirements listed for fire scenario selection and analysis assume that the physical analysis units to be analyzed have been identified (e.g., through qualitative and/or quantitative screening). Requirements are listed for the selection and analysis of fire scenarios for single physical analysis units, multicompartment configurations, and the main control room (MCR).

Table 4-2.6-1 High Level Requirement for Fire Scenario Selection and Analysis (FSS)

Designator	Requirement
HLR-FSS-A	The fire PRA shall select one or more combinations of an ignition source and damage target sets to represent the fire scenarios for each physical analysis unit that has not been screened out and upon which an estimation of the risk contribution (CDF and LERF) will be based.
HLR-FSS-B	The fire PRA shall include an analysis of potential fire scenarios leading to the MCR abandonment.
HLR-FSS-C	The fire PRA shall characterize the factors that will influence the timing and extent of fire damage for each combination of an ignition source and damage target sets selected per Requirement HLR-FSS-A.
HLR-FSS-D	The fire PRA shall quantify the likelihood of risk-relevant consequences for each combination of an ignition source and damage target sets selected per Requirement HLR-FSS-A.
HLR-FSS-E	The parameter estimates used in fire modeling shall be based on relevant generic industry and plant-specific information. Where feasible, generic and plant-specific evidence shall be integrated by using acceptable methods to obtain plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty.
HLR-FSS-F	The fire PRA shall search for and analyze risk-relevant scenarios with the potential for causing fire-induced failure of exposed structural steel.
HLR-FSS-G	The fire PRA shall evaluate the risk contribution of multicompartment fire scenarios.
HLR-FSS-H	The fire PRA shall document the results of the fire scenario and fire modeling analyses including supporting information for scenario selection, underlying assumptions, scenario descriptions, and the conclusions of the quantitative analysis, in a manner that facilitates fire PRA applications, upgrades, and peer review.

GENERAL NOTE: Requirements HLR-FSS-A, HLR-FSS-B, and HLR-FSS-C are associated with those fire scenarios where the fire and fire-induced damage are both limited to a single physical analysis unit. Requirement HLR-FSS-G is associated with the analysis of fire scenarios where the fire and/or fire-induced damage impacts two or more physical analysis units (the multicompartment fire scenarios).

Table 4-2.6-2 Supporting Requirements (SR) for HLR-FSS-A

The fire PRA shall select one or more combinations of an ignition source and damage target set to represent the fire scenarios for each physical analysis unit that has not been screened out and upon which an estimation of the risk contribution (CDF and LERF) will be based (HLR-FSS-A).

Index No. FSS-A	Capability Category I	Capability Category II	Capability Category III
FSS-A1 [Note (1)]	IDENTIFY all risk-relevant ignition sources, both fixed and transient, in each physical analysis unit that has not been screened out within the global analysis boundary.		
FSS-A2 [Note (2)]	GROUP all risk-relevant damage targets in each physical analysis unit that has not been screened out within the global analysis boundary into one or more damage target sets. For each target set, SPECIFY the equipment and cable failures, and the failure modes, including those leading to spurious action.		
FSS-A3	If the exact routing of a cable (or group of cables) has not been established (see Requirements CS-A10 and CS-A11), ASSUME that those cables fail for any fire scenario that has a damaging effect on any raceway or conduit where the subject cable might reasonably exist.		
FSS-A4 [Note (3)]	IDENTIFY sufficient combinations of target sets for each physical analysis unit that has not been screened out, such that the range of system and function impacts has been represented.		
FSS-A5 [Notes (4)–(6)]	For each physical analysis unit that has not been screened out within the global analysis boundary, SELECT sufficient combinations of a fire ignition source (or group of ignition sources) as defined in Requirement FSS-A1 and a target set (or group of target sets) as defined in Requirement FSS-A4 as characteristics of the selected fire scenarios that will provide reasonable assurance that the fire risk contribution of each physical analysis unit that has not been screened out can be characterized commensurate with its risk significance.		For each physical analysis unit that has not been screened out within the global analysis boundary, SELECT sufficient combinations of a fire ignition source (or group of ignition sources) as defined in Requirement FSS-A1 and a target set (or group of target sets) as defined in Requirement FSS-A4 as characteristics of the selected fire scenarios that will provide reasonable assurance that the fire risk contribution of each physical analysis unit that has not been screened out can be characterized and such that the risk contributions can be correlated to specific ignition sources and locations within the physical analysis unit.
FSS-A6 [Note (7)]	When analyzing MCR fires, SELECT at least one scenario involving a fire in the main control board damaging more than one function that has been included in the fire PRA plant response model.		SELECT fire scenario(s) in the main control board that account for fire spread, and its timing, including those involving loss of more than one function that has been included in the fire PRA plant response model.

Table 4-2.6-2 Supporting Requirements (SR) for HLR-FSS-A (Cont'd)

NOTES:
(1) In this context, a risk-relevant ignition source would be any ignition source capable of creating a fire-induced environmental condition (perhaps through fire spread) that can cause the failure of at least one fire PRA equipment item or cable (i.e., a risk-relevant target). Note that an ignition source and first damaged target might be the same if the ignition source is also a fire PRA equipment item or cable.
(2) Note that Requirements FSS-A2, FSS-A3, and FSS-A4 are closely linked. The intent of Requirement FSS-A2 is to ensure that all of the risk-relevant damage targets present within each physical analysis unit that has not been screened out within the global analysis boundary as defined by plant partitioning are identified and that these targets are grouped into appropriate target sets. Per the definition (see Section 2-2), each target set will be treated based on one damage criterion and one damage threshold. Each fire scenario will lead to the failure of one or more target sets.
(3) The intent of Requirement FSS-A4 is to ensure that scenario-specific groups of target sets (which might collectively represent a subset of the damage targets present) are identified and that the identified target set groups appropriately represent the range of plant functional impacts that might arise in a physical analysis unit given the risk-relevant fire damage target present. Under Requirement FSS-A4, each selected fire scenario is tied to one or more target sets as defined in Requirements FSS-A2 and FSS-A3 (i.e., each fire scenario will lead to the loss of at least one target set).
(4) As used in this Standard, once a fire scenario has been "selected," it implies that the scenario will eventually be evaluated and/or quantified at a level of detail commensurate with the risk significance of the scenario.
(5) It is expected that the number of individual fire scenarios and the level of detail included in the analysis of each scenario will be commensurate with the relative risk importance, for fire, of the physical analysis unit under analysis (see Table 1-1.3-2). Physical analysis units with small risk contribution may, for example, be characterized based on the conservative analysis of a single bounding fire scenario. The more risk-important physical analysis units will likely be characterized by detailed analysis of multiple and/or more specific fire scenarios. In particular, those physical analysis units that are identified as the significant fire risk contributors should be characterized by the detailed quantification (see Requirement HLR-FSS-C) of one or more fire scenarios that combine specific ignition sources and specific target sets.
(6) In fire PRA practice, multiple ignition sources can be treated using a single fire scenario (e.g., a bank of several similar electrical panels might be grouped and treated with a single fire scenario), provided that the assumed fire ignition frequency and fire characteristics bound the cumulative contribution of all of the individual ignition sources included under the selected fire scenario.
(7) The fire scenarios affecting the main control board may or may not lead to MCR abandonment.

Table 4-2.6-3 Supporting Requirements (SR) for HLR-FSS-B

The fire PRA shall include an analysis of potential fire scenarios leading to the MCR abandonment (HLR-FSS-B).

Index No. FSS-B	Capability Category I	Capability Category II	Capability Category III
FSS-B1 [Note (1)]	SPECIFY and JUSTIFY the conditions that are assumed to cause MCR abandonment and/or reliance on ex-control room operator actions including remote and/or alternate shutdown actions.		
FSS-B2 [Notes (2) and (3)]	SELECT one or more fire scenarios, either in the MCR or elsewhere, leading to MCR abandonment and/or a reliance on ex-control room operator actions including remote and/or alternate shutdown actions, consisting of a combination of an ignition source (or group of ignition sources), such that the selected scenarios provide reasonable assurance that the MCR abandonment fire risk contribution can be bounded.	SELECT one or more fire scenarios, either in the MCR or elsewhere, leading to MCR abandonment and/or a reliance on ex-control room operator actions including remote and/or alternate shutdown actions, consisting of a combination of an ignition source (or group of ignition sources), such that the selected scenarios provide reasonable assurance that the MCR abandonment fire risk contribution can be realistically characterized.	SELECT one or more fire scenarios, either in the MCR or elsewhere, leading to MCR abandonment and/or a reliance on ex-control room operator actions including remote and/or alternate shutdown actions, consisting of a combination of an ignition source (or group of ignition sources), such that the selected scenarios provide reasonable assurance that the fire risk contribution of the MCR abandonment can be realistically characterized and such that the risk contributions can be correlated to specific ignition sources and locations within the MCR.

NOTES:
(1) In justifying the selected abandonment conditions, consideration should reflect the assumptions that control room abandonment may be required should the control room itself become untenable for human habitation (e.g., heat buildup sufficient to cause pain to human skin or smoke buildup sufficient to substantially impede operator performance), or as a result of a loss of a sufficient set of plant controls or indications such that operator performance would be substantially impeded, or as required by plant procedures.
(2) MCR abandonment and ex-control room operator actions are not relevant to all physical analysis units. The intent of Requirement FSS-B2 is to require the development of these scenarios wherever plant procedures include a reliance on either alternate or remote shutdown operator actions.
(3) Requirement FSS-B2 deals with the selection of MCR abandonment scenarios only. It is intended that the fire scenarios selected based on Requirements FSS-A1 through FSS-A6 will include, as appropriate, fire scenarios that may affect the MCR but do not lead to MCR abandonment.

Table 4-2.6-4 Supporting Requirements (SR) for HLR-FSS-C

The fire PRA shall characterize the factors that will influence the timing and extent of fire damage for each combination of an ignition source and damage target sets selected per Requirement HLR-FSS-A (HLR-FSS-C).

Index No. FSS-C	Capability Category I	Capability Category II	Capability Category III
FSS-C1 [Notes (1) and (2)]	For fire scenarios selected in accordance with Requirements HLR-FSS-A and HLR-FSS-B, ASSIGN characteristics to the ignition sources that bound potentially risk-contributing fire events in the context of both fire intensity and duration given the nature of the fire ignition sources present.	For risk-significant ignition sources where currently accepted methods support a probabilistic representation of fire characteristics, REPRESENT the ignition source using fire characteristics that reflect a range of fire intensities and durations that includes the contribution of low-likelihood but potentially more challenging fires.	
FSS-C2 [Note (3)]	CHARACTERIZE ignition-source intensity such that the fire is initiated at full-peak intensity (i.e., heat release rate).	For those scenarios that represent significant contributors to fire risk, CHARACTERIZE ignition-source intensity using a realistic time-dependent fire growth profile (i.e., a time-dependent heat release rate) appropriate to the ignition source.	
FSS-C3 [Note (4)]	If fire burnout is included in the analysis, JUSTIFY the burnout time and conditions.	JUSTIFY the heat release rate profile stages included in the analysis (i.e., fire growth, steady burning, or decay stages).	
FSS-C4 [Note (5)]	If a severity factor is credited in the analysis, ENSURE that *(a)* the severity factor remains independent of other quantification factors *(b)* the severity factor reflects the fire event set used to estimate fire frequency *(c)* the severity factor bounds the conditions and assumptions of the specific fire scenarios under analysis *(d)* a technical basis supporting the severity factor's determination is provided	APPLY severity factors for risk-significant fire scenarios. ENSURE that *(a)* the severity factor remains independent of other quantification factors *(b)* the severity factor reflects the fire event set used to estimate fire frequency *(c)* the severity factor reflects the conditions and assumptions of the specific fire scenarios under analysis *(d)* a technical basis supporting the severity factor's determination is provided	APPLY severity factors for risk-significant fire scenarios. ESTABLISH a direct relationship between the severity factor and the fire characteristics assumed in the analysis. ENSURE that *(a)* the severity factor remains independent of other quantification factors *(b)* the severity factor reflects the fire event set used to estimate fire frequency *(c)* the severity factor reflects the conditions and assumptions of the specific fire scenarios under analysis *(d)* a technical basis supporting the severity factor's determination is provided

Table 4-2.6-4 Supporting Requirements (SR) for HLR-FSS-C (Cont'd)

The fire PRA shall characterize the factors that will influence the timing and extent of fire damage for each combination of an ignition source and damage target sets selected per Requirement HLR-FSS-A (HLR-FSS-C).

Index No. FSS-C	Capability Category I	Capability Category II	Capability Category III
FSS-C5	JUSTIFY that the damage criteria used in the fire PRA are representative of the damage targets associated with each fire scenario.		JUSTIFY that the damage criteria used in the fire PRA are representative of the damage targets associated with each fire scenario and reflect the damage criteria of plant-specific damage targets, where available.
FSS-C6	ASSUME target damage occurs when the exposure environment exceeds the damage threshold.		ESTIMATE target damage times based on the thermal response of the damage target.
FSS-C7	If multiple suppression paths are credited, MODEL dependencies among the credited paths, including dependencies associated with recovery of a failed fire suppression system, if such recovery is credited.		
FSS-C8 [Note (6)]	If raceway fire wraps, other passive fire barrier elements, or active fire barrier elements within a single physical analysis unit are credited in the analysis of fire scenarios *(a)* SPECIFY a technical basis for their fire-resistance rating *(b)* CONFIRM that the fire wrap or other passive fire-protection features will not be subjected to either mechanical damage or damage from direct flame impingement from a high-hazard ignition source unless the wrap has been subject to qualification or other proof of performance testing under these conditions *(c)* INCLUDE treatment of fire scenarios involving the failure of the credited barrier element		

NOTES:

(1) In the context of this Standard, an ignition source is characterized based on parameters such as its intensity (e.g., heat release rate), type (e.g., oil pool fire, electrical fire, high-energy arcing fault, etc.), location (e.g., close to walls or ceilings that could affect the behavior of the ignition source), duration, and transient profile.

(2) Meeting Requirement FSS-C1, Capability Categories II and III will require judgment, and the measures required may vary depending on the nature of both the fire-ignition source and the threatened fire-damage targets. In simple cases, it may be sufficient to apply a two-point fire-intensity model. Under this approach, the analyst first determines the minimum fire intensity capable of causing fire spread to intervening combustibles and/or fire-induced damage to at least one member of the target set. Risk is then quantified based on that fraction of fires that exceed this minimum threshold (the severity factor approach). The two-point fire-intensity model uses two discrete fire-intensity values to represent the spectrum of fires larger than the minimum damaging fire. In other cases, a higher level of resolution may be required to more accurately reflect risk contributions or to suit a specific risk application (e.g., a fire-intensity model characterized by three or more discrete values). For ignition sources that present the potential for two or more fire types with fundamentally different fire characteristics (e.g., pumps that might involve both electrical motor fires and oil fires), a two-point modeling approach might be applied to each fire type. The SR as written does allow an exception for those fire sources or fire types where currently accepted methods do not yet support a multipoint fire characterization model. For example, current accepted practice generally applies a single-point fire characterization approach for high-energy arc faults, bus duct faults, fires associated with catastrophic failures of the turbine-generator set, hydrogen fires, and catastrophic failures in oil-filled transformers.

Table 4-2.6-4 Supporting Requirements (SR) for HLR-FSS-C (Cont'd)

NOTES: (Cont'd)

(3) In Capability Category I, the intent is to consider the full range of ignition sources present based on the application of conservative assumptions regarding fire burning behavior. In Capability Categories II and III, this practice is acceptable for those ignition sources that are not significant contributors to fire risk. However, those ignition sources that are significant contributors to fire risk should receive a more detailed treatment that uses more realistic fire characterization assumptions where available and as appropriate to the ignition source.

(4) The intent for Capability Category I is to allow consideration of burnout due, for example, to depletion of the available fuels (including the potential for fire spread to any secondary combustibles that might be present). For Capability Categories II and III, a more realistic treatment of the fire is expected including consideration of the fire growth behavior and the fire decay (burnout) behavior, again including the consideration of potential fire spread to secondary combustibles.

(5) The phrase "conditions and assumptions of the specific fire scenarios under analysis" refers to those characteristics of the fire scenario that could influence whether or not a fire will damage targets. Examples would include the distance between fire source and target, position of the targets relative to the fire source, the damage threshold of the targets, and the mode of fire exposure (e.g., buoyant plume exposure versus radiant heating). The intent of Requirement FSS-C4 for Capability Categories II and III is, in part, that such factors would be an explicit consideration in quantifying the severity factor. The intent for Capability Category I is to allow for the application of generic severity factors that reflect, more generally, those fire events that contributed to the fire ignition frequency but without explicit consideration of such case-specific factors so long as the severity factor applied is consistent with, and independent of, other quantification factors.

(6) Requirement HLR-FSS-G and its SRs provide for the treatment of fire scenarios impacting adjacent physical analysis units (the multicompartment fire analysis). Requirement FSS-C8 is intended, in part, to ensure that a similar treatment is provided for cases where barriers exist within a single physical analysis unit (i.e., the barriers exist but were not credited during plant partitioning). If the analysis of fire scenarios within a single physical analysis unit credits these barriers (e.g., with limiting fire damage, or delaying the spread of fire or the onset of fire damage), then Requirement FSS-C8 requires an analysis of fire scenarios involving the failure of the credited barrier that is analogous to the multicompartment fire analysis. Such barriers may include passive barriers (e.g., nonrated partition walls, cable wraps, or radiant energy shields) or active barriers (e.g., normally open fire doors or water curtains).

Table 4-2.6-5 Supporting Requirements (SR) for HLR-FSS-D

The fire PRA shall quantify the likelihood of risk-relevant consequences for each combination of an ignition source and damage target sets selected per Requirement HLR-FSS-A (HLR-FSS-D).

Index No. FSS-D	Capability Category I	Capability Category II	Capability Category III
FSS-D1 [Note (1)]	SELECT appropriate fire modeling tools for estimating fire growth and damage behavior considering the physical behaviors relevant to the selected fire scenarios.		
FSS-D2	USE fire models that have sufficient capability to model the conditions of interest and only within known limits of applicability.		
FSS-D3 [Note (2)]	USE conservative assumptions regarding the likelihood and/or extent of fire damage in the analysis of each fire scenario such that the fire-risk contribution of each unscreened physical analysis unit is bounded.	For any physical analysis unit that represents a significant contributor to fire risk, SELECT and APPLY fire modeling tools such that the scenario analysis provides reasonable assurance that the fire-risk contribution can be accurately characterized.	For any physical analysis unit that represents a significant contributor to fire risk, SELECT and APPLY fire modeling tools such that the scenario analysis provides reasonable assurance that the fire-risk contribution can be accurately characterized and such that the risk contributions can be correlated to specific ignition sources and locations within the physical analysis unit.
FSS-D4	SPECIFY a technical basis for fire modeling tool input values used in the analysis given the context of the fire scenarios being analyzed.		
FSS-D5 [Note (3)]	SPECIFY a technical basis for any applied statistical models in the context of the fire scenarios being analyzed.		SPECIFY a technical basis for any applied statistical models in the context of the fire scenarios being analyzed. INCLUDE plant-specific updates to generic statistical models when *(a)* appropriate data are available to support the update *(b)* updating of the statistical model might impact the quantification of one or more significant contributors to fire risk
FSS-D6 [Note (4)]	SPECIFY a technical basis for any applied empirical models in the context of the fire scenarios being analyzed by *(a)* providing a reference basis, or *(b)* developing a basis if *(1)* basis is not provided in referenced documentation (e.g., technical reports describing the empirical models), or *(2)* the empirical models are used outside the recommended scenario conditions		

Table 4-2.6-5 Supporting Requirements (SR) for HLR-FSS-D (Cont'd)

The fire PRA shall quantify the likelihood of risk-relevant consequences for each combination of an ignition source and damage target sets selected per Requirement HLR-FSS-A (HLR-FSS-D).

Index No. FSS-D	Capability Category I	Capability Category II	Capability Category III
FSS-D7 [Notes (5)–(7)]	In crediting fire detection and suppression systems, USE generic estimates of total system unavailability provided that *(a)* the credited system is installed and maintained in accordance with applicable codes and standards (b) the credited system is in a fully operable state during plant operation	In crediting fire detection and suppression systems, USE generic estimates of total system unavailability provided that *(a)* the credited system is installed and maintained in accordance with applicable codes and standards *(b)* the credited system is in a fully operable state during plant operation *(c)* the system has not experienced outlier behavior relative to system unavailability	In crediting fire detection and suppression systems, USE plant-specific information and CALCULATE realistic parameter estimates for total system unavailability consistent with Requirement DA-D1, Capability Category II (see Part 2).
FSS-D8 [Note (8)]	INCLUDE an assessment of fire detection and suppression systems effectiveness in the context of each fire scenario analyzed.		
FSS-D9 [Note (9)]	No requirement to evaluate the potential for smoke damage	EVALUATE the potential for smoke damage to fire PRA equipment on a qualitative basis, and INCLUDE the results of this assessment in the definition of fire scenario target sets.	
FSS-D10 [Note (10)]	CONDUCT walkdowns to confirm that the combinations of fire sources and target sets that were selected per Requirement FSS-A5 appropriately represent as-built plant conditions for those physical analysis units that represent significant contributors to fire risk.	CONDUCT walkdowns to confirm that the combinations of fire sources and target sets that were selected per Requirement FSS-A5 appropriately represent the as-built plant conditions.	
FSS-D11 [Note (11)]	CONDUCT walkdowns to verify that other aspects of the selected fire scenarios not covered by Requirement FSS-D10 have been characterized appropriately for each analyzed fire scenario.		

NOTES:

(1) The selection of appropriate fire modeling tools may be driven by a number of factors. For example, the relative risk significance of a fire scenario may influence the choice of fire modeling tools. Low-risk scenarios may be analyzed using simple fire modeling tools, whereas higher risk scenarios might be analyzed using more sophisticated tools such as a compartment fire model. As a second example, a fire PRA that is simply seeking conservative screening level results may use conservative damage state assumptions in lieu of detailed fire growth and damage analyses. As a third example, the fire phenomena of interest would also be a factor. If damage targets are all located directly above the fire source, then a plume modeling correlation may be appropriate, but if targets are in other locations, then radiant heating, ceiling jet, and/or hot gas layer predictions may be needed.

(2) In Capability Categories II and III, the intent is to allow for a preliminary assessment of a physical analysis unit's risk significance based on the application of conservative assumptions (e.g., consistent with the Capability Category I requirement), but to require that physical analysis units or scenarios that are significant contributors to fire risk will be analyzed in greater detail through the application of appropriate fire modeling tools.

Table 4-2.6-5 Supporting Requirements (SR) for HLR-FSS-D (Cont'd)

NOTES: (Cont'd)

(3) It is anticipated that some aspects of fire growth and damage analysis (including suppression) may be treated using various types of statistical models; that is, a model in which a parameter or behavior is treated as a random variable with specified statistical characteristics. For example, fire spread behavior within electrical panels or the main control board has been modeled statistically. A second example might be the modeling of fire intensity using a probability distribution.

(4) It is anticipated that some aspects of fire modeling may be treated using various types of empirical models; that is, models based on experience or observation alone. For example, fire suppression by the manual fire brigade is often based on an empirical relationship derived from a statistical analysis of fire suppression times reported in past operating experience. A second example is characterizing high-energy arcing faults in electrical switching equipment based on characteristics observed in past events. A third example is the wide range of closed-form empirical correlations documented in sources such as textbooks or engineering handbooks.

(5) Typical fire PRA practice involves the application of a nonsuppression probability; that is, the probability that suppression efforts fail to suppress the fire before the onset of the postulated equipment/cable damage. Hence, the nonsuppression probability estimate includes an assessment of effectiveness (including the relative timing of fire damage versus detection/suppression and fire brigade performance), discussed in Requirement FSS-D8, as well as an overall assessment of system unavailability. The intent of Requirement FSS-D7 is to require increasing levels of plant specificity in assessing system unavailability with increasing capability category.

(6) The applicable codes and standards will generally be the relevant NFPA code(s) of record.

(7) The intent for Capability Category II is to additionally require a review of plant records to determine if the generic unavailability credit is consistent with actual system unavailability. Outlier experience would be any experience indicating that actual system is unavailable more frequently than would be indicated by the generic values.

(8) Fire detection or suppression system effectiveness depends on, at a minimum, the following:

(a) system design compliance with applicable codes and standards, and current fire protection engineering practice

(b) the time available to suppress the fire prior to target damage

(c) specific features of physical analysis unit and fire scenario under analysis (e.g., pocketing effects, blockages that might impact plume behaviors or the "visibility" of the fire to detection and suppression systems, and suppression system coverage)

(d) suitability of the installed system given the nature of the fire source being analyzed

(9) Fire scenarios that assume widespread damage (e.g., damage across an entire physical analysis unit) will generally capture potential smoke damage within the limits of the assumed fire damage (e.g., assuming the loss of all equipment in a physical analysis unit given a fire, as might be employed during early stages of a screening analysis).

(10) One aspect of confirmation by walkdown is the verification of information obtained from engineering drawings or other plant documentation. However, the objectives of walkdowns also include the confirmation of configuration-specific factors that influence fire growth and damage behaviors to ensure that these factors have been properly accounted for in the fire growth and damage analyses (i.e., in the fire modeling efforts).

(11) It is anticipated that the scope of the confirmatory walkdowns will be commensurate with both the risk importance of the physical analysis units under analysis and with the overall level of detail and sophistication associated with fire scenario analysis. For example, a screening level fire scenario analysis that assumes widespread fire damage within a physical analysis unit would only require verification of ignition sources present in the physical analysis unit. In the case of a detailed analysis of a fire scenario that is a significant contributor to fire risk, confirmation of additional factors would be appropriate such as the location of damage targets relative to ignition sources, proximity and configuration of secondary combustibles, placement and effectiveness of fire detection and suppression equipment, etc.

Table 4-2.6-6 Supporting Requirements (SR) for HLR-FSS-E

The parameter estimates used in fire modeling shall be based on relevant generic industry and plant-specific information. Where feasible, generic and plant-specific evidence shall be integrated using acceptable methods to obtain plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty (HLR-FSS-E).

Index No. FSS-E	Capability Category I	Capability Category II	Capability Category III
FSS-E1	For any fire modeling parameters not covered by Requirement HLR-FSS-C or HLR-FSS-D, USE plant-specific parameter estimates for fire modeling if available, or use generic information modified as discussed in Requirement FSS-E2; USE generic information for the remaining parameter estimates.		
FSS-E2	If neither plant-specific data nor generic parameter estimates are available for a fire modeling parameter, USE data or estimates for the most similar situation, adjusting if necessary to account for differences. Alternatively, USE expert judgment and document the rationale behind the choice of parameter values.		
FSS-E3 [Note (1)]	For each combination of a fire-ignition source and a target set (e.g., see Requirement FSS-A5) whose analysis has taken credit for fire suppression prior to fire damage, CALCULATE a point estimate of the nonsuppression probability. CHARACTERIZE the uncertainty in the estimated nonsuppression probability. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	For each combination of a fire-ignition source and a target set (e.g., see Requirement FSS-A5) whose analysis has taken credit for fire suppression prior to fire damage, the following actions apply: *(a)* For risk-significant fire scenarios, CALCULATE a mean value of the nonsuppression probability and PROVIDE the probabilistic representation of the uncertainty in the estimated nonsuppression probability. *(b)* For the non–risk-significant fire scenarios, CALCULATE a point estimate value of the nonsuppression probability and CHARACTERIZE the uncertainty in the estimated nonsuppression probability. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	For each combination of a fire-ignition source and a target set (e.g., see Requirement FSS-A5) whose analysis has taken credit for fire suppression prior to fire damage, CALCULATE a point estimate of the nonsuppression probability. PROVIDE the probabilistic representation of the uncertainty in the estimated nonsuppression probability. Acceptable methods for developing a probabilistic representation include Bayesian updating or the use of expert judgment.
FSS-E4 [Note (2)]	CHARACTERIZE the uncertainties associated with cases where cable routing has been assumed based on Requirement CS-A10 or CS-A11, or both.		

Table 4-2.6-6 Supporting Requirements (SR) for HLR-FSS-E (Cont'd)

NOTE:

(1) The nonsuppression probability is the common name applied to the conditional probability that given fire ignition the postulated target set suffers fire-induced damage.

(2) Uncertainties associated with cases where cable routing was assumed may be associated with the exact location of the cables with respect to the ignition sources, and fire-resistance characteristics and fire protection (e.g., fire-resistant covers) of the cables.

Table 4-2.6-7 Supporting Requirements (SR) for HLR-FSS-F

The fire PRA shall search for and analyze risk-relevant scenarios with the potential for causing fire-induced failure of exposed structural steel (HLR-FSS-F).

Index No. FSS-F	Capability Category I	Capability Category II	Capability Category III
FSS-F1 [Note (1)]	INDENTIFY any locations within the fire PRA global analysis boundary that meet both of the following conditions: *(a)* exposed structural steel is present *(b)* a high-hazard fire source is present in that location If such locations are identified, SELECT those fire scenarios that could potentially damage, including collapse, the exposed structural steel for each identified location.		IDENTIFY and CONFIRM, by walkdown, the fireproofing of structural steel, and IDENTIFY any locations within the fire PRA global analysis boundary that meet both of the following conditions: *(a)* exposed structural steel is present *(b)* a high-hazard fire source is present in that location If such locations are identified, SELECT those fire scenarios that could potentially damage, including collapse, the exposed structural steel for each identified location.
FSS-F2 [Note (2)]	No requirement to establish or justify criteria for collapse of structural steel	If, per Requirement FSS-F1, one or more scenarios are selected, ESTABLISH and JUSTIFY criteria for structural collapse due to fire exposure.	
FSS-F3 [Note (3)]	If, per Requirement FSS-F1, one or more scenarios are selected, COMPLETE a qualitative assessment of the risk of the selected fire scenarios, including collapse of the exposed structural steel.	If, per Requirement FSS-F1, one or more scenarios are selected, COMPLETE a quantitative assessment of the risk of the selected fire scenarios in a manner consistent with Requirements HLR-FQ-A through HLR-FQ-F, including collapse of the exposed structural steel.	

NOTES:

(1) The prototypical fire scenario leading to failure of structural steel would be catastrophic failure of the turbine itself (e.g., a blade ejection event) and an ensuing lube-oil fire. For the lube-oil fire, the possibility of effects of pooling, the flaming oil traversing multiple levels, and spraying from continued lube-oil pump operation should be considered. However, the analysis should also consider scenarios involving other high-hazard fire sources as present in the relevant physical analysis units (e.g., oil storage tanks, hydrogen storage tanks and piping, mineral oil-filled transformers).

(2) Various resources exist in the public literature dealing with the failure of exposed structural steel in a fire including Chapter 4-9 of *The SFPE* [Society of Fire Protection Engineers] *Handbook of Fire Protection Engineering* (*SFPE Handbook*) [4-3] and Section 12-4 of the current National Fire Protection Association (NFPA) *Fire Protection Handbook* (*NFPA Handbook*) [4-4] (see Section 7-4 of earlier editions of the *NFPA Handbook*).

(3) The intent of Requirement FSS-F3 is to highlight that, for Capability Categories II/III, selected fire scenarios are flagged for quantification per the FQ technical element. For Capability Category I, scenarios are assessed qualitatively and therefore not quantified per the FQ technical element.

Table 4-2.6-8 Supporting Requirements (SR) for HLR-FSS-G

The fire PRA shall evaluate the risk contribution of multicompartment fire scenarios (HLR-FSS-G).

Index No. FSS-G	Capability Category I	Capability Category II	Capability Category III
FSS-G1 [Note (1)]	APPLY all the supporting requirements listed in Requirements FSS-C1 through FSS-C8 for fire modeling of single physical analysis units to the modeling of multicompartment fire scenarios.		
FSS-G2	SPECIFY screening criteria for multicompartment fire scenarios that provide reasonable assurance that the contribution of the physical analysis unit combinations that are screened out are of low risk significance.		
FSS-G3	APPLY the screening criteria defined per Requirement FSS-G2 to all physical analysis unit combinations within the global analysis boundary (as defined in plant partitioning) using a systematic methodology. For each physical analysis unit combination that is not screened out, SELECT one or more multicompartment fire scenario(s) to represent the potential consequences of fires impacting the physical analysis unit combination.		
FSS-G4 [Note (2)]	If passive fire barriers with a fire-resistance rating are credited in the fire PRA, ENSURE that the credit is consistent with the fire-resistance rating as demonstrated by conformance to applicable test standards.	If passive fire barriers with a fire-resistance rating are credited in the fire PRA *(a)* ENSURE that the credit is consistent with the fire-resistance rating as demonstrated by conformance to applicable test standards *(b)* ASSESS the effectiveness, reliability, and availability of any credited passive fire barrier feature *(c)* EVALUATE the potential for fire-induced or random failure of credited passive fire barrier features	If passive fire barriers with a fire-resistance rating are credited in the fire PRA *(a)* ENSURE that the credit is consistent with the fire-resistance rating as demonstrated by conformance to applicable test standards *(b)* CALCULATE the reliability and availability of any credited passive fire barrier feature that accounts for its effectiveness *(c)* EVALUATE the potential for fire-induced or random failure of credited passive fire barrier features
FSS-G5 [Note (3)]	For any scenario selected per Requirement FSS-G3, if the adjoining physical analysis units are separated by active fire barrier elements, ASSESS qualitatively the effectiveness, reliability, and availability of the active fire barrier element.	For any scenario selected per Requirement FSS-G3, if the adjoining physical analysis units are separated by active fire barrier elements *(a)* QUANTIFY the reliability and availability of the active fire barrier element *(b)* CONFIRM that the active fire barrier element will be effective given the nature of the fire threat being postulated	
FSS-G6 [Note (4)]	PROVIDE a qualitative assessment of the potential risk importance of any selected multicompartment fire scenarios.	CALCULATE the risk contribution of any selected multicompartment fire scenarios in a manner consistent with Requirements HLR-FQ-A through HLR-FQ-F.	

NOTES:

(1) In applying Requirements FSS-C1 through FSS-C7, additional phenomena associated with multicompartment fire scenarios, beyond those associated with scenarios of single physical analysis units, may need to be addressed. For example, the modeling of hot gas flow through openings and ducts from the physical analysis unit of fire origin may be necessary.

Table 4-2.6-8 Supporting Requirements (SR) for HLR-FSS-G (Cont'd)

NOTES: (Cont'd)

(2) Passive fire barrier features that may have been credited in plant partitioning or scenario analysis include items such as walls, normally closed fire doors, penetration seals, and other similar features that require no action (manual or automatic) to perform their intended function. This requirement would apply to all passive fire barrier elements credited in the fire PRA, including the plant partitioning, as well as in the fire-scenario selection and analysis. The fire-resistance rating of passive fire barrier features is typically established in accordance with the ASTM E 119-07a [4-5] test standard and/or other similar, related, or subsidiary standards. The intent of Requirement FSS-G4 is to allow an analysis to credit passive fire barrier features that do have an established fire-resistance rating consistent with that fire-resistance rating.

(3) Active fire barrier elements include items such as normally open fire doors, dampers, water curtains, and other similar items that require that some action (manual or automatic) occur for the element to perform its intended function. The intent of Requirement FSS-G5 is to ensure that the potential failure of active fire barrier elements (both random and fire induced) is included in the assessment of the risk importance of selected multicompartment fire scenarios.

(4) The intent of Requirement FSS-G6 is to highlight that for Capability Categories II/III, selected fire scenarios are flagged for quantification per the FQ technical element. For Capability Category I, scenarios are assessed qualitatively and therefore not quantified per the FQ technical element.

Table 4-2.6-9 Supporting Requirements (SR) for HLR-FSS-H

The fire PRA shall document the results of the fire scenario and fire modeling analyses including supporting information for scenario selection, underlying assumptions, scenario descriptions, and the conclusions of the quantitative analysis, in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-FSS-H).

Index No. FSS-H	Capability Category I	Capability Category II	Capability Category III
FSS-H1	For each fire scenario analyzed, DOCUMENT *(a)* the nature and characteristics of the ignition source *(b)* the nature and characteristics of the damage target set *(c)* any applied severity factors *(d)* the calculated nonsuppression probability all in a manner that facilitates fire PRA applications, upgrades, and peer review.		
FSS-H2	DOCUMENT a basis for target damage mechanisms and thresholds used in the analysis.	DOCUMENT a basis for target damage mechanisms and thresholds used in the analysis, including references for any plant-specific or target-specific performance criteria applied in the analysis.	
FSS-H3	DOCUMENT a basis for the selection of the applied fire modeling tools.		
FSS-H4	DOCUMENT the fire modeling tool input values used in the analysis of each fire scenario.		

Table 4-2.6-9 Supporting Requirements (SR) for HLR-FSS-H (Cont'd)

The fire PRA shall document the results of the fire scenario and fire modeling analyses including supporting information for scenario selection, underlying assumptions, scenario descriptions, and the conclusions of the quantitative analysis, in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-FSS-H).

Index No. FSS-H	Capability Category I	Capability Category II	Capability Category III
FSS-H5	DOCUMENT fire modeling output results for each analyzed fire scenario in a manner that facilitates fire PRA applications, upgrades, and peer review.	DOCUMENT fire modeling output results for each analyzed fire scenario, including the results of parameter uncertainty evaluations (as performed) in a manner that facilitates fire PRA applications, upgrades, and peer review.	DOCUMENT fire modeling output results for each analyzed fire scenario, including the results of parameter uncertainty evaluations (as performed) in a manner that facilitates fire PRA applications, upgrades, and peer review. DOCUMENT insights related to the impact of uncertainties for significant input parameters in the context of the resulting fire-risk estimates.
FSS-H6	DOCUMENT *(a)* a technical basis for any statistical models applied in the analysis, including applicability *(b)* a technical basis for any plant-specific updates applied to generic statistical models *(c)* the plant-specific data applied in any plant-specific updates		
FSS-H7	DOCUMENT the assumptions made related to credited firefighting activities including fire detection, fire suppression systems, and any credit given to manual suppression efforts.		
FSS-H8	DOCUMENT the methodology used to select potentially risk-significant multicompartment fire scenarios, the results of the multicompartment fire scenario analysis including the applied screening criteria; results of the screening analysis; the identification of any multicompartment fire scenarios identified as potentially risk significant; and the quantitative results for any scenarios analyzed quantitatively in a manner that facilitates fire PRA applications, upgrades, and peer review.		
FSS-H9	DOCUMENT the sources of model uncertainty and related assumptions associated with the FSS technical element.		
FSS-H10 [Note (1)]	DOCUMENT the walkdown process and results.		

NOTE:
(1) Typical walkdown results may include the purpose of each walkdown conducted, dates and participants, supporting calculations (if any), and information gained.

4-2.7 IGNITION FREQUENCY

4-2.7.1 Objectives

The objectives of the ignition frequency (IGN) element are to
(a) establish the plant-wide frequency of fires of various types on a generic basis for a nuclear power plant
(b) tailor the generic fire frequency values to reflect a particular plant
(c) apportion fire frequencies to specific physical analysis units, and/or fire scenarios

4-2.7.2 Fire Ignition Frequency

The fire ignition frequency is a key factor contributing to fire risk quantification. It is multiplied with various conditional probabilities (conditional on occurrence of the postulated fire) to generate CDF and LERF risk estimates. Conditional probabilities may address fire severity (referred to as severity factor), probability of nonsuppression, and conditional probability of core damage. Typically, two types of ignition frequencies are employed in a fire risk analysis:

(a) the frequency of a fire in a physical analysis unit or plant area

(b) the frequency of fire ignition involving a specific ignition source (e.g., an electrical panel)

A large number of fire events have occurred in the nuclear power industry. These events have served as the basis for establishing fire ignition frequencies and associated uncertainties that have been reported in several public and proprietary sources. It may also be acceptable to use applicable data from nonnuclear power industry sources when there is no similar experience in the nuclear power industry, with appropriate justification.

An analyst is expected to include generic nuclear power plant experience when developing plant-specific fire frequencies. The analyst may apply plant-specific experience in an updating of the generic fire frequencies but may not develop fire frequencies based exclusively on plant-specific experience. The only exception would be where plant-specific experience involves a unique fire ignition source not otherwise found in the generic data. It is important to note that this Standard prohibits assigning zero ignition frequency to a plant area. For example, a transient combustible fire may occur at any location of the plant, thereby rendering an assumption of zero ignition frequency inappropriate. Administrative controls may be a consideration in assigning the relative frequency of transient fires to various physical analysis units, but because they might be violated, they cannot fully preclude transient fires from any given physical analysis unit.

Table 4-2.7-1 High Level Requirement for Ignition Frequency (IGN)

Designator	Requirement
HLR-IGN-A	The fire PRA shall develop fire ignition frequencies for every physical analysis unit that has not been qualitatively screened.
HLR-IGN-B	The fire PRA shall document the fire frequency estimation in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.7-2 Supporting Requirements (SR) for HLR-IGN-A

The fire PRA shall develop fire ignition frequencies for every physical analysis unit that has not been qualitatively screened (HLR-IGN-A).

Index No. IGN-A	Capability Category I	Capability Category II	Capability Category III
IGN-A1	Except as allowed by Requirements IGN-A2 and IGN-A3, USE current nuclear power industry event history that includes power plants of similar type, characteristics, and vintage to establish ignition frequencies on a per reactor-year basis. JUSTIFY excluded data that are not considered to be applicable (e.g., due to changes in industry practices).		
IGN-A2 [Note (1)]	Except as allowed by Requirement IGN-A3, USE applicable data from nonnuclear power industry sources only when there is no similar experience in the nuclear power industry. JUSTIFY all nonnuclear power industry sources used for establishing fire ignition frequencies by demonstrating the applicability of information provided in those sources to the specific ignition source being studied. In justifying the use of nonnuclear power industry data, INCLUDE verification that applicable nuclear industry data do not exist; a description of the data being applied, including its source; discussion of the data analysis approach and methods used to estimate per reactor-year fire frequencies; and verification of the applicability of the applied data to nuclear power plant conditions and the fire scenario(s) being analyzed.		

Table 4-2.7-2 Supporting Requirements (SR) for HLR-IGN-A (Cont'd)

The fire PRA shall develop fire ignition frequencies for every physical analysis unit that has not been qualitatively screened (HLR-IGN-A).

Index No. IGN-A	Capability Category I	Capability Category II	Capability Category III
IGN-A3 [Note (2)]	In cases where nuclear power industry and nonnuclear industry data are not available, USE engineering judgment.		
IGN-A4 [Note (3)]	No requirement to review or incorporate plant-specific fire event experience in fire frequencies	REVIEW plant-specific experience for fire event outlier experience, and UPDATE fire frequencies if outliers are found.	UPDATE fire frequencies to reflect plant-specific experience.
IGN-A5 [Note (4)]	ESTIMATE generic fire ignition frequencies or plant-specific fire frequency updates on a reactor-year basis (generic fire frequencies are typically reported on this same basis). INCLUDE in the fire frequency estimation the plant availability, such that the frequencies are weighted by the fraction of time the plant is at-power.		
IGN-A6	When combining evidence from generic and plant-specific data, USE a Bayesian update process or equivalent statistical process. JUSTIFY the selection of any informative prior distribution used on the basis of industry experience.		
IGN-A7 [Note (5)]	USE a plant-wide consistent methodology based on parameters that are expected to influence the likelihood of ignition to apportion high-level ignition frequencies (e.g., plant-wide values) to estimate physical analysis unit or ignition source level frequencies.		
IGN-A8	ASSIGN an ignition frequency greater than zero to every plant physical analysis unit.		ASSIGN an ignition frequency, greater than zero to every plant physical analysis unit, and fire risk-relevant ignition source.
IGN-A9	POSTULATE the possibility of transient combustible fires for all physical analysis units regardless of the administrative restrictions.		
IGN-A10	CHARACTERIZE (e.g., discuss qualitatively) the uncertainty intervals for significant fire ignition frequencies.	ESTIMATE a mean value of and a statistical representation of the uncertainty intervals for the parameters for the significant fire ignition frequencies.	ESTIMATE a mean value of and a statistical representation of the uncertainty intervals for the parameters for all fire ignition frequencies.

NOTES:

(1) It is recognized that nonnuclear power industry sources may be of sufficient quality to be used for developing ignition frequencies as a supplement to nuclear plant sources provided that an appropriate level of applicability, robustness, and fidelity can be demonstrated. At a minimum, any analysis of nonnuclear power industry fire event data would need to demonstrate the following:

(a) The underlying data set is applicable to the specific ignition source being studied.

(b) The underlying data set is applicable to nuclear power plant conditions and the fire scenario(s) being analyzed.

(c) The scope and completeness of the underlying data set is adequate to support robust statistical treatment.

(d) The total population base and equivalent years of operating experience represented by the underlying data set can be quantified.

Table 4-2.7-2 Supporting Requirements (SR) for HLR-IGN-A (Cont'd)

NOTES: (Cont'd)

(e) The fire frequencies calculated are consistent with, and maintain statistical independence from, other aspects of the fire PRA, including, in particular, any applied fire severity (e.g., fire severity factor) treatments and/or any mitigation credit applied for fire detection and suppression prior to target damage including the analysis of both timing and effectiveness.

The underlying data set and all analyses performed would also need to be available for review by both peer reviewers and, if applicable, the authority responsible for approval or acceptance of the specific fire PRA application. If nonnuclear power industry sources are identified in the future that can meet the above requirements, it is expected that this Standard would be revised to allow the use of nonnuclear sources.

(2) Refer to 1-4.3 on probabilistic risk assessment for nuclear power plant applications for discussions relevant to the application of engineering judgment.

(3) Outlier experience includes cases where the plant has experienced more fires of any given type than would be expected given the generic industry experience, or where the plant has experienced a type of fire that is potentially risk relevant but is not reflected in the generic event database.

(4) That is, the analysis accounts for the fraction of the year that the plant is in at-power operational state.

(5) The term "plant-wide consistent methodology" indicates that the selected approach for apportioning generic frequencies to physical analysis units must be consistent throughout the plant. For example, if equipment count is chosen as the approach for determining physical analysis unit apportioning factors, counting rules should be established and applied consistently throughout all the physical analysis units in the plant. In addition, the plant-wide fire frequency must be conserved.

Table 4-2.7-3 Supporting Requirements (SR) for HLR-IGN-B

The fire PRA shall document the fire frequency estimation in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-IGN-B).

Index No. IGN-B	Capability Category I	Capability Category II	Capability Category III
IGN-B1	DOCUMENT all frequencies and event data used in the analysis in a manner that facilitates fire PRA applications, upgrades, and peer review.		
IGN-B2	DOCUMENT references for fire events and fire ignition frequency sources used.		
IGN-B3	DOCUMENT the apportioning methodology and bases of selected values in a manner that facilitates fire PRA applications, upgrades, and peer review.		
IGN-B4	DOCUMENT the plant-specific frequency updating process. INCLUDE in the documentation *(a)* the selected plant-specific events *(b)* the basis for the selection and or exclusion of events *(c)* the analysis supporting the plant-specific reactor-years *(d)* the Bayesian process for updating generic frequencies		
IGN-B5	DOCUMENT the sources of model uncertainty and related assumptions associated with the ignition frequency analysis.		

4-2.8 QUANTITATIVE SCREENING

The objective of the quantitative screening (QNS) element is to screen out physical analysis units from further (e.g., more detailed quantitative) consideration based on preliminary estimates of fire risk contribution and by using established quantitative screening criteria.

The HLRs below begin with the phrase "If quantitative screening is performed . . . ." The QNS element is optional because a fire PRA can include detailed quantitative analysis of all fire areas. A fire PRA with no quantitative screening and detailed fire PRA of all areas is deemed to satisfy Capability Category III for the QNS element.

The potential risk contribution of all fire compartments is reexamined in the multicompartment fire scenario analysis regardless of the fire compartment's disposition during qualitative screening (see Requirement FSS-G3 in 4-2.6).

Most of the SRs for the SR QNS element are nominally the same across the three capability categories, except for Requirement QNS-B3. This requirement distinguishes among the three categories ensuring that the higher the fire PRA category, the more physical analysis units will be analyzed with detailed quantitative analysis in subsequent parts. As with the QLS element, an implied distinction exists due to the intimate relationship between the QNS element and the prior tasks above. These prior tasks define the predominant factors assessed in the quantitative screening criteria, namely, the physical analysis units being examined, the list of relevant equipment, the list of relevant cables, the mapping of equipment and cables to fire compartments, the fire frequencies, and development of the fire PRA plant response model. Hence, the scope defined by these prior tasks will largely define the scope and level of rigor associated with quantitative screening. The intent is to ensure that the quantitative screening task is performed to a scope and level of rigor in a manner consistent with these prior and related tasks.

Table 4-2.8-1 High Level Requirement for Quantitative Screening (QNS)

Designator	Requirement
HLR-QNS-A	If quantitative screening is performed, the fire PRA shall establish quantitative screening criteria to ensure that the estimated cumulative impact of screened-out physical analysis units on CDF and LERF is small.
HLR-QNS-B	If quantitative screening is performed, the fire PRA shall identify those physical analysis units that are screened out as individual risk contributors.
HLR-QNS-C	VERIFY that the cumulative impact of screened-out physical analysis units on CDF and LERF is small.
HLR-QNS-D	The fire PRA shall document the results of quantitative screening in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.8-2 Supporting Requirements (SR) for HLR-QNS-A

If quantitative screening is performed, the fire PRA shall establish quantitative screening criteria to ensure that the estimated cumulative impact of screened-out physical analysis units on LERF and CDF is small (HLR-QNS-A).

Index No. QNS-A	Capability Category I	Capability Category II	Capability Category III
QNS-A1 [Note (1)]	SPECIFY quantitative screening criteria that ensure that the cumulative impact of screened-out physical analysis units on CDF and LERF is small.		

NOTE:

(1) The criteria established in Requirement QNS-A1 should support Requirement QNS-C1, which may require iteration in revising the criteria. Since the CDF is different for each plant, a single criterion is not possible. A plant with a lower overall CDF will require a lower quantitative screening criterion than a plant with a higher overall CDF to ensure that significant contributors are not screened out. Requirement QNS-C1 provides the verification of this process.

Table 4-2.8-3 Supporting Requirements (SR) for HLR-QNS-B

If quantitative screening is performed, the fire PRA shall identify those physical analysis units that are screened out as individual risk contributors (HLR-QNS-B).

Index No. QNS-B	Capability Category I	Capability Category II	Capability Category III
QNS-B1	APPLY the quantitative screening criteria to each physical analysis unit defined by the plant partitioning analysis not previously screened out qualitatively.		
QNS-B2 [Note (1)]	RETAIN for risk quantification or scenario development each physical analysis unit that does not meet the defined quantitative screening criteria.		

NOTE:

(1) It is acceptable for the quantitative screening analysis to retain any physical analysis units for risk quantification analysis without a rigorous application of the defined quantitative screening criteria.

Table 4-2.8-4 Supporting Requirements (SR) for HLR-QNS-C

Verify that the cumulative impact of screened-out physical analysis units on CDF and LERF is small (HLR-QNS-C).

Index No. QNS-C	Capability Category I	Capability Category II	Capability Category III
QNS-C1 [Notes (1) and (2)]	ENSURE that the quantitative screening process does not screen out the highest-risk physical analysis units.	ENSURE that *(a)* the quantitative screening process does not screen out the highest-risk physical analysis units *(b)* the sum of the CDF contributions for all screened-out fire compartments is < 10% of the estimated total CDF for fire events *(c)* the sum of the LERF contributions for all screened-out fire compartments is < 10% of the estimated total LERF for fire events	ENSURE that *(a)* the quantitative screening process does not screen out the highest-risk physical analysis units *(b)* the sum of the CDF contributions for all screened-out fire compartments is < 1% of the estimated total CDF for fire events *(c)* the sum of the LERF contributions for all screened-out fire compartments is < 1% of the estimated total LERF for fire events

NOTES:

(1) For Capability Category I, the highest risk fire areas are any areas that have a fire risk within an order of magnitude of the highest risk fire area. For example, if the highest risk area has a CDF of 1E-5/yr, any area with a CDF of 1E-5/yr to 1E-6/yr is considered a "highest risk fire area."

(2) For subpara. (a), some fire compartments within a fire area may be screened out, but as long as the highest risk fire compartments within the area are retained, the fire area is not considered screened out. However, the estimate of fire area fire risk includes the estimated risk associated with screened-out compartments (i.e., the truncation error).

Table 4-2.8-5 Supporting Requirements (SR) for HLR-QNS-D

The fire PRA shall document the results of quantitative screening in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-QNS-D).

Index No. QNS-D	Capability Category I	Capability Category II	Capability Category III
QNS-D1	DOCUMENT the disposition per Requirement HLR-QNS-B of each physical analysis unit defined by the plant partitioning analysis as either screened out or retained for quantitative analysis, and the cumulative impact of the quantitative screening per Requirement HLR-QNS-C in a manner that facilitates fire PRA applications, upgrades, and peer review.		
QNS-D2	DOCUMENT the CDF and LERF values used for quantitative screening and the cumulative impact of quantitative screening, for each physical analysis unit defined in the plant partitioning analysis that has been screened out in a manner that facilitates fire PRA applications, upgrades, and peer review.		

4-2.9 CIRCUIT FAILURES

The objectives of the circuit failure (CF) element are to

(a) refine the understanding and analysis of fire-induced circuit failures on an individual fire scenario basis

(b) ensure that the consequences of each fire scenario on the damaged cables and circuits have been addressed

The overall scope of circuits examined in the fire PRA is addressed in 4-2.2 and 4-2.3. However, the CS element addressed in 4-2.3 contains some simplifications and was performed without consideration of certain limiting cable failure combinations and circuit failure modes. Accordingly, certain cable failure combinations or failure modes might not actually jeopardize the credited equipment function on an individual fire scenario basis. In addition, the specific circuit failure mode of concern might have a conditional probability of occurrence given circuit failure that is not unity. A circuit analysis is performed given these circuit failures to determine the scope and extent of equipment functional impacts and the conditional probability of the specific circuit failure mode needed to cause those impacts.

The scope of the CF requirements is limited to only those elements of fire-induced consequences that are attributable to cable and circuit failures.

Table 4-2.9-1 High Level Requirement for Circuit Failures (CF)

Designator	Requirement
HLR-CF-A	The fire PRA shall determine the applicable conditional probability of the cable and circuit failure mode(s) that would cause equipment functional failure and/or undesired spurious operation based on the credited function of the equipment in the fire PRA.
HLR-CF-B	The fire PRA shall document the development of the circuit failure analysis in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.9-2 Supporting Requirements (SR) for HLR-CF-A

The fire PRA shall determine the applicable conditional probability of the cable and circuit failure mode(s) that would cause equipment functional failure and/or undesired spurious operation based on the credited function of the equipment in the fire PRA (HLR-CF-A).

Index No. CF-A	Capability Category I	Capability Category II	Capability Category III
CF-A1 [Note (1)]	REVIEW the conditional failure probabilities for fire-induced circuit failures. ASSIGN the appropriate industry-wide generic values.	REVIEW the conditional failure probabilities for fire-induced circuit failures. ASSIGN the appropriate industry-wide generic values for risk-significant contributors based on the specific circuit configuration under consideration.	
CF-A2	CHARACTERIZE the uncertainty in the conditional failure probability estimates assigned per Requirement CF-A1. This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding.	PROVIDE a statistical representation of the uncertainty intervals for the conditional failure probability estimates assigned per Requirement CF-A1.	

NOTES:

(1) Requirement CF-A1 is not intended to preclude the use of new and/or plant-specific cable failure modes and effects testing insights. Requirement CF-A1 is also not intended to preclude the use of screening values or conservative treatment in Category I, or screening values or conservative treatment for non–risk-significant contributors for Category II/III. Duration of spurious operation may be considered as part of the conditional failure probability.

Table 4-2.9-3 Supporting Requirements (SR) for HLR-CF-B

The fire PRA shall document the development of the circuit failure analysis in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-CF-B).

Index No. CF-B	Capability Category I	Capability Category II	Capability Category III
CF-B1	DOCUMENT the results of the circuit failure analyses in sufficient detail to describe the treatment of fire-induced circuit failures and the application of associated conditional failure probability values and in a manner that facilitates fire PRA applications, upgrades, and peer review.		

4-2.10 HUMAN RELIABILITY ANALYSIS (HRA)

4-2.10.1 Objectives

The objectives of the human reliability analysis (HRA) element are to

(a) identify the human actions and resulting HFEs to be included in the fire PRA

(b) quantify the human error probabilities (HEPs) for these HFEs

4-2.10.2 HFEs

In this task, any prior postinitiator HFEs adopted for use in (or imported directly into) the fire PRA (e.g., from the internal-events PRA that has been assessed against Part 2) need to be modified to incorporate fire location and fire scenario-specific changes in assumptions, modeling structure, and performance shaping factors. Additionally, HFEs need to be included in the fire PRA to address the use of procedures that

(a) are not modeled in other analyses

(b) direct special actions that the operators take to maintain acceptable plant configurations and achieve safe shutdown given a fire

Preinitiator HFEs can impact fire risk through errors that affect operability/functionality of

(a) systems and equipment used for safe shutdown, such as an auxiliary feedwater valve, or

(b) fire protection systems (active or passive) and program elements (e.g., transient combustible control or fire brigade training program)

While it is expected that preinitiator HFEs under subpara. (a) above continue to be addressed in the fire PRA just as in an internal-events PRA that is assessed against Part 2, preinitiator HFEs under subpara. (b) above are addressed differently. Such errors affecting operability/functionality of fire protection systems, features, and program elements are already addressed under other parts/elements of this Standard that are assumed to rely on a combination of historical and experimental data with regard to operability/functionality of fire protection systems (active and passive) including fire suppression and fire barriers that include preinitiator human errors. Hence, no specific requirements are provided here with regard to treatment of preinitiator HFEs unique to fire-related issues. This lack of requirements does not prevent a user from performing preinitiator HRA of these possible errors if it is decided to do so. Under those circumstances, the identification and quantification of such errors should follow Part 2 requirements for preinitiator HFEs used for internal-events PRAs.

Table 4-2.10-1 High Level Requirement for Human Reliability Analysis (HRA)

Designator	Requirement
HLR-HRA-A	The fire PRA shall identify human actions relevant to the sequences in the fire PRA plant response model.
HLR-HRA-B	The fire PRA shall include events where appropriate in the fire PRA that represent the impacts of incorrect human responses associated with the identified human actions.
HLR-HRA-C	The fire PRA shall quantify HEPs associated with the incorrect responses accounting for the plant-specific and scenario-specific influences on human performance, particularly including the effects of fires.
HLR-HRA-D	The fire PRA shall include recovery actions only if it has been demonstrated that the action is plausible and feasible for those scenarios to which it applies, particularly accounting for the effects of fires.
HLR-HRA-E	The fire PRA shall document the HRA, including the unique fire-related influences of the analysis, in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.10-2 Supporting Requirements (SR) for HLR-HRA-A

The fire PRA shall identify human actions relevant to the sequences in the fire PRA plant response model (HLR-HRA-A).

Index No. HRA-A	Capability Category I	Capability Category II	Capability Category III
HRA-A1	For each fire scenario, for each safe shutdown action carried over from the internal-events PRA, DETERMINE whether or not each action remains relevant and valid in the context of the fire PRA consistent with the scope of selected equipment per the ES technical element (see 4-2.2) and plant response model per the PRM technical element (see 4-2.5), and in accordance with Requirement HLR-HR-E and its SRs in Part 2 with the following clarifications: *(a)* Where Requirement HR-E1 mentions "in the context of the accident scenarios," specific attention is to be given to the fact that these are fire scenarios. *(b)* Requirement HR-E3 has been modified in Requirement HRA-A4 to address fire scenarios. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under HLR-HR-E in Part 2.		
HRA-A2 [Notes (1) and (2)]	For each fire scenario, IDENTIFY any new fire-specific safe shutdown actions called out in the plant fire response procedures (e.g., de-energizing equipment per a fire procedure for a specific fire location) in a manner consistent with the scope of selected equipment from the ES and PRM technical elements (see 4-2.2 and 4-2.5), and in accordance with Requirement HLR-HR-E and its SRs in Part 2 with the following clarifications: *(a)* Where Requirement HR-E1 discusses procedures, it is to be extended to procedures for responding to fires. *(b)* Where Requirement HR-E1 mentions "in the context of the accident scenarios," specific attention is to be given to the fact that these are fire scenarios. *(c)* Another source for Requirement HR-E1 is likely to be the current Fire Safe Shutdown/ Appendix R analysis. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-HR-E in Part 2.		
HRA-A3	No requirement to identify new undesired operator actions arising from fire-induced spurious indications	For each fire scenario, IDENTIFY any new, undesired operator action that could result from fire-induced spurious indications resulting from failure of a single instrument, per Requirement ES-C2 (e.g., due to verbatim compliance with the instruction in an alarm response procedure, when separate confirmation is not available or required).	For each fire scenario, IDENTIFY any new, undesired operator action that could result from fire-induced spurious indications resulting from failure of up to and including two instruments at a time, per Requirement ES-C2 (e.g., due to verbatim compliance with the instruction in an alarm response procedure, when separate confirmation is not available or required).

Table 4-2.10-2 Supporting Requirements (SR) for HLR-HRA-A (Cont'd)

The fire PRA shall identify human actions relevant to the sequences in the fire PRA plant response model (HLR-HRA-A).

Index No. HRA-A	Capability Category I	Capability Category II	Capability Category III
HRA-A4	REVIEW the interpretation of the procedures associated with actions identified in Requirements HRA-A1 and HRA-A2 with plant operations or training personnel to confirm that the interpretation is consistent with plant operational and training practices.	TALK THROUGH (i.e., review in detail) with plant operations and training personnel the procedures and sequence of events to confirm that interpretation of the procedures relevant to actions identified in Requirements HRA-A1, HRA-A2, and HRA-A3 is consistent with plant operational and training practices.	

NOTES:

(1) Requirements HRA-A1 and HRA-A2 are complementary requirements. Requirement HRA-A1 requires the reassessment of human actions that were carried over into the fire PRA from the internal-events PRA. Requirement HRA-A2 deals with the treatment of those human actions that were not included in the internal-events PRA but will be included in the fire PRA because they are specific to the fire response procedures.

(2) The graded application of both of the above SRs is based on the gradations in Part 2 for the SRs under Requirement HLR-HR-E. Note also that the gradation associated with 4-2.2 and 4-2.5 will affect what operator actions are addressed (e.g., if a system is not going to be addressed per Reqirement HLR-ES-B in 4-2.2 and so not subsequently modeled following 4-2.5, operator actions associated with that system are not addressed).

Table 4-2.10-3 Supporting Requirements (SR) for HLR-HRA-B

The fire PRA shall include events where appropriate in the fire PRA that represent the impacts of incorrect human responses associated with the identified human actions (HLR-HRA-B).

Index No. HRA-B	Capability Category I	Capability Category II	Capability Category III
HRA-B1 [Note (1)]	INCLUDE and MODIFY, if necessary, HFEs corresponding to the actions identified per Requirement HRA-A1 in the fire PRA plant response model in a manner consistent with 4-2.2 and 4-2.5, such that the HFEs represent the impact of the human failures at the function, system, train, or component level as appropriate. Failures to correctly perform several responses may be grouped into one HFE if the impact of the failures is similar or can be conservatively bounded.		INCLUDE and MODIFY, if necessary, HFEs corresponding to the actions identified per Requirement HRA-A1 in the fire PRA plant response model consistent with 4-2.2 and 4-2.5, such that the HFEs represent the impact of the human failures at the function, system, train, or component level as appropriate.
HRA-B2	INCLUDE new fire-related safe shutdown HFEs corresponding to the actions identified per Requirement HRA-A2 in the fire PRA plant response model in a manner consistent with 4-2.2 and Section 4-2 and in accordance with Requirement HLR-HR-F and its SRs in Part 2. SPECIFY a defined basis to support the claim of nonapplicability of any of the SRs under Requirement HLR-HR-F in Part 2.		

Table 4-2.10-3 Supporting Requirements (SR) for HLR-HRA-B (Cont'd)

The fire PRA shall include events where appropriate in the fire PRA that represent the impacts of incorrect human responses associated with the identified human actions (HLR-HRA-B).

Index No. HRA-B	Capability Category I	Capability Category II	Capability Category III
HRA-B3	COMPLETE the definition of the HFEs identified in Requirements HRA-B1 and HRA-B2 by specifying the following, taking into account the context presented by the fire scenarios in the fire PRA: *(a)* accident sequence–specific timing of cues, and time window for successful completion *(b)* accident sequence–specific procedural guidance (e.g., AOPs, EOPs) *(c)* the availability of cues or other indications for detection and evaluation errors *(d)* the complexity of the response (Task analysis is not required.)	COMPLETE the definition of the HFEs identified in Requirements HRA-B1 and HRA-B2 by specifying the following, taking into account the context presented by the fire scenarios in the fire PRA: *(a)* accident sequence–specific timing of cues, and time window for successful completion *(b)* accident sequence–specific procedural guidance (e.g., AOPs, EOPs) *(c)* the availability of cues or other indications for detection and evaluation errors *(d)* the specific high-level tasks (e.g., train-level) required to achieve the goal of the response	COMPLETE the definition of the HFEs identified in Requirements HRA-B1 and HRA-B2 by specifying the following, taking into account the context presented by the fire scenarios in the fire PRA: *(a)* accident sequence–specific timing of cues, and time window for successful completion *(b)* accident sequence–specific procedural guidance (e.g., AOPs, EOPs) *(c)* the availability of cues or other indications for detection and evaluation errors *(d)* the specific detailed tasks (e.g., at the level of individual components, such as pumps and valves) required to achieve the goal of the response
HRA-B4 [Notes (2) and (3)]	No requirement to include HFEs where fire-induced failures might lead to an undesired operator action	INCLUDE HFEs for cases where fire-induced instrumentation failure of any single instrument could cause an undesired operator action, consistent with Requirement HLR-ES-C of this Part and in accordance with Requirement HLR-HR-F and its SRs in Part 2. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-HR-F in Part 2.	INCLUDE HFEs for cases where fire-induced instrumentation failure of up to and including two instruments at a time could cause an undesired operator action, consistent with Requirement HLR-ES-C of this Part and in accordance with Requirement HLR-HR-F and its SRs in Part 2. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-HR-F in Part 2.

NOTES:

(1) HFEs related to actions previously modeled in an analysis such as the internal-events PRA may have to be modified because the fire may change the scenario characteristics such as timing, cues, or specific actions that would have to be taken (e.g., due to fire-induced circuit failures that affect the manner in which certain components may be operated). These changes would therefore require alteration of a previously defined HFE to fit the applicable fire situation in the fire PRA.

Table 4-2.10-3 Supporting Requirements (SR) for HLR-HRA-B (Cont'd)

NOTES (Cont'd):
(2) The intent of this requirement is to recognize that in cases where instrumentation required for an operator action could be affected by a fire, the implication is that there is a potentially significant likelihood that the operator will either fail to perform an action or take an inappropriate action (e.g., shut down a pump because of a spurious pump high temperature alarm) due to the failed instrumentation. This requirement is to ensure that these types of HFEs are not overlooked in recognition that the corresponding HEPs could be high.
(3) One of the modes of failure to be considered is spurious operation of the instrument.

Table 4-2.10-4 Supporting Requirements (SR) for HLR-HRA-C

The fire PRA shall quantify HEPs associated with the incorrect responses accounting for the plant-specific and scenario-specific influences on human performance, particularly including the effects of fires (HLR-HRA-C).

Index No. HRA-C	Capability Category I	Capability Category II	Capability Category III
HRA-C1 [Notes (1) and (2)]	For each selected fire scenario, CALCULATE the HEPs for all HFEs in accident sequences that survive initial quantification and INCLUDE relevant fire-related effects using conservative estimates (e.g., screening values), in accordance with the SRs for Requirement HLR-HR-G in Part 2 set forth under Capability Category I, with the following clarifications: *(a)* Attention is to be given to how the fire situation alters any previous assessments in nonfire analyses as to the influencing factors and the timing considerations covered in Requirements HR-G3, HR-G4, and HR-G5 in Part 2. *(b)* SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-HR-G in Part 2.	For each selected fire scenario, CALCULATE the HEPs for all HFEs and INCLUDE relevant fire-related effects using detailed analyses for significant HFEs and conservative estimates (e.g., screening values) for nonsignificant HFEs, in accordance with the SRs for Requirement HLR-HR-G in Part 2 set forth under at least Capability Category II, with the following clarifications: *(a)* Attention is to be given to how the fire situation alters any previous assessments in nonfire analyses as to the influencing factors and the timing considerations covered in Requirements HR-G3, HR-G4, and HR-G5 in Part 2. *(b)* SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-HR-G in Part 2.	For each selected fire scenario, CALCULATE the HEPs for all HFEs and INCLUDE relevant fire-related effects using detailed analyses, in accordance with the SRs for Requirement HLR-HR-G in Part 2 set forth under at least Capability Category III, with the following clarification: *(a)* Attention is to be given to how the fire situation alters any previous assessments in nonfire analyses as to the influencing factors and the timing considerations covered in Requirements HR-G3, HR-G4, and HR-G5 in Part 2. *(b)* SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-HR-G in Part 2.

NOTE:
(1) The fire PRA context introduces new aspects to those performance shaping factors (PSFs) already identified in the Part 2 requirements (e.g., the effects of the environmental conditions would need to consider relevant fire environments), or might introduce new PSFs (e.g., the fact that one operator is generally assigned as member of the fire brigade or the added burden associated with postfire operator actions). The intent of Requirement HRA-C1 is to ensure inclusion of such factors.
(2) The term "detailed analyses" is intended to refer to the use of HRA methods that take into account the context presented by the fire scenario, including the impacts of relevant PSFs. Methods characterized as "screening" or "scoping" in nature or that involve applying generic multipliers to values obtained from prior detailed analyses (e.g., from the internal-events PRA) do not qualify as detailed analyses. These more simplified approaches may be used to satisfy Capability Category I and to satisfy Capability Category II for any HFEs that are not risk significant.

Table 4-2.10-5 Supporting Requirements (SR) for HLR-HRA-D

The fire PRA shall include recovery actions only if it has been demonstrated that the action is plausible and feasible for those scenarios to which it applies, particularly accounting for the effects of fires (HLR-HRA-D).

Index No. HRA-D	Capability Category I	Capability Category II	Capability Category III
HRA-D1	INCLUDE operator recovery actions that can restore the functions, systems, or components on an as-needed basis to provide a more realistic evaluation of CDF and LERF.	INCLUDE operator recovery actions that can restore the functions, systems, or components on an as-needed basis to provide a more realistic evaluation of significant accident sequences.	INCLUDE operator recovery actions that can restore the functions, systems, or components on an as-needed basis to provide a more realistic evaluation of modeled accident sequences.
HRA-D2 [Note (1)]	For any operator recovery actions identified in Requirement HRA-D1 *(a)* INCLUDE relevant fire-related effects, including any effects that may preclude a recovery action or alter the manner in which it is accomplished, in accordance with Requirements HR-H2 and HR-H3 in Part 2 *(b)* SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirements HR-H2 and HR-H3 in Part 2		

NOTE:
(1) An example of a fire-related effect that must be considered carefully in identifying and evaluating recovery actions is the potential for a circuit failure that could both defeat automatic operation of a valve and prevent remote manual operation.

Table 4-2.10-6 Supporting Requirements (SR) for HLR-HRA-E

The fire PRA shall document the HRA, including the unique fire-related influences of the analysis, in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-HRA-E).

Index No. HRA-E	Capability Category I	Capability Category II	Capability Category III
HRA-E1	DOCUMENT the fire PRA HRA, including *(a)* those fire-related influences that affect the methods, processes, or assumptions used as well as the identification and quantification of the HFEs/HEPs in accordance with Requirement HLR-HR-I and its SRs in Part 2, and SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-HR-I in Part 2 *(b)* any defined bases to support the claim of nonapplicability of any of the referenced requirements in Part 2 beyond that already covered by the clarifications in this Part		

4-2.11 SEISMIC FIRE

The objective of the seismic fire (SF) element is to qualitatively assess the potential risk implications of seismic/fire interaction issues.

Part 5 specifically addresses seismic PRA for nuclear power plants. However, it does not explicitly address seismically induced fire events and/or hazards caused by an integrity failure or spurious operation of a suppression system. Also, it does not address degradations in fire suppression systems and capabilities as a result of an earthquake. Therefore, the effects of an earthquake on fire-related issues are addressed in 4-2.11.

The Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues (SAND88-0177, NUREG/CR-5088, 1988) [4-6] identifies the following four seismic/fire interaction issues:

(a) seismically induced fires
(b) degradation of fire suppression systems and features
(c) spurious operation of suppression and/or detection systems
(d) degradation of manual firefighting effectiveness

Accepted methods for quantifying the risk contribution of these issues are not currently available. Hence, the final results of a fire PRA (i.e., CDF and LERF) would likely not include quantitative results for fire scenarios initiated by an earthquake, and this Standard provides no requirements for quantification of seismic/fire interactions. However, during the individual plant examination of external events (IPEEE) process, qualitative methods for identifying and assessing plant configurations and practices with respect to each of these four issues were established and were applied successfully by licensees in their IPEEE fire studies. Hence, the SF requirements follow the precedent set by the IPEEE process with the expectation that a fire PRA will address the above listed issues qualitatively.

Table 4-2.11-1 High Level Requirement for Seismic Fire (SF)

Designator	Requirement
HLR-SF-A	The fire PRA shall include a qualitative assessment of potential seismic/fire interaction issues in the fire PRA.
HLR-SF-B	The fire PRA shall document the results of the seismic/fire interaction assessment in a manner that facilitates fire PRA applications, upgrades, and peer review.

Table 4-2.11-2 Supporting Requirements (SR) for HLR-SF-A

The fire PRA shall include a qualitative assessment of potential seismic/fire interaction issues in the fire PRA (HLR-SF-A).

Index No. SF-A	Capability Category I	Capability Category II	Capability Category III
SF-A1	For those physical analysis units within the fire PRA global analysis boundary *(a)* IDENTIFY for fire ignition source scenarios that might arise as the result of an earthquake that would be unique from those postulated during the general analysis of each physical analysis unit *(b)* PROVIDE a qualitative assessment of the potential risk significance of any unique fire ignition source scenarios identified		
SF-A2	For those physical analysis units within the fire PRA global plant analysis boundary *(a)* REVIEW installed fire detection and suppression systems and provide a qualitative assessment of the potential for either failure (e.g., rupture or unavailability) or spurious operation during an earthquake *(b)* EVALUATE the potential impact of system rupture or spurious operation on postearthquake plant response including the potential for flooding relative to water-based fire suppression systems, loss of habitability for gaseous suppression systems, and the potential for diversion of suppressants from areas where they might be needed for those fire suppression systems associated with a common suppressant supply		
SF-A3	EVALUATE the potential for common-cause failure of multiple fire suppression systems due to the seismically induced failure of supporting systems such as fire pumps, fire water storage tanks, yard mains, gaseous suppression storage tanks, or building standpipes.		
SF-A4	REVIEW plant seismic response procedures. EVALUATE qualitatively the potential that a seismically induced fire, or the spurious operation of fire suppression systems, might compromise postearthquake plant response.		
SF-A5	REVIEW *(a)* plant fire brigade training procedures and ASSESS the extent to which training has prepared firefighting personnel to respond to potential fire alarms and fires in the wake of an earthquake *(b)* the storage and placement of firefighting support equipment and fire brigade access routes ASSESS the potential that an earthquake might compromise one or more of these features.		

Table 4-2.11-3 Supporting Requirements (SR) for HLR-SF-B

The fire PRA shall document the results of the seismic/fire interaction assessment in a manner that facilitates fire PRA applications, upgrades, and peer review (HLR-SF-B).

Index No. SF-B	Capability Category I	Capability Category II	Capability Category III
SF-B1	DOCUMENT the results of the seismic/fire interaction analysis, including the results and insights gained from any unique fire scenarios that were identified, in a manner that facilitates fire PRA applications, upgrades, and peer review.		

4-2.12 FIRE RISK QUANTIFICATION

The objectives of the fire risk quantification (FQ) element are to

(a) quantify the fire-induced CDF and LERF contributions to plant risk

(b) understand what are the significant contributors to the fire-induced CDF and LERF

The final fire risk is determined on the basis of quantifying the fire PRA plant response model developed per the requirements in 4-2.5 having integrated the results of all the other technical elements of the fire PRA.

The approach to quantification and the quantified risk measures are virtually the same as is specified for internal-events PRA results per Part 2 but are modified to also include results as to the significant fires (and fire scenarios) and fire locations (e.g., compartments). This modified approach ensures that the quantified results are performed in a way to provide fire-unique related insights (e.g., important fires).

Table 4-2.12-1 High Level Requirement for Fire Risk Quantification (FQ)

Designator	Requirement
HLR-FQ-A	The fire-induced CDF shall be quantified.
HLR-FQ-B	The fire-induced CDF quantification shall use appropriate models and codes and shall account for method-specific limitations and features.
HLR-FQ-C	Model quantification shall determine that all identified dependencies are addressed appropriately.
HLR-FQ-D	The frequency of different containment failure modes leading to a fire-induced large early release shall be quantified and aggregated, thus determining the fire-induced LERF.
HLR-FQ-E	The fire-induced CDF and LERF quantification results shall be reviewed, and significant contributors to CDF and LERF, such as fires and their corresponding plant initiating events, fire locations, accident sequences, basic events (equipment unavailabilities and human failure events), plant damage states, containment challenges, and failure modes, shall be identified. The results shall be traceable to the inputs and assumptions made in the fire PRA.
HLR-FQ-F	The documentation of CDF and LERF analyses shall be consistent with the applicable SRs.

Table 4-2.12-2 Supporting Requirements (SR) for HLR-FQ-A

The fire-induced CDF shall be quantified (HLR-FQ-A).

Index No. FQ-A	Capability Category I	Capability Category II	Capability Category III
FQ-A1	For each fire scenario selected per the FSS technical element (see 4-2.6) that will be quantified as a contributor to fire-induced plant CDF and/or LERF, REPRESENT the equipment and cable failures, including specification of the failure modes, defined per the FSS technical element as basic events in the fire PRA plant response model, including consideration of insights from the circuit failure analysis (see 4-2.9).		
FQ-A2 [Notes (1) and (2)]	For each fire scenario selected per the FSS technical element (see 4-2.6) that will be quantified as a contributor to fire-induced plant CDF and/or LERF, IDENTIFY the appropriate initiating event or events (e.g., general transient, LOOP) that will be used to quantify CDF and LERF such that quantification properly accounts for the damage caused by the fire and scenario timing.		
FQ-A3	For each fire scenario selected per the FSS requirements that will be quantified as a contributor to fire-induced plant CDF and/or LERF, QUANTIFY the fire PRA plant response model reflecting the scenario-specific quantification factors (i.e., circuit failure likelihoods per Requirements HLR-CF-A and HLR-CF-B, HEP values for HFEs quantified per the HRA requirements, and the fire-induced equipment and cable failures per Requirement FQ-A1).		
FQ-A4 [Note (3)]	CALCULATE the fire-induced CDF in accordance with Requirement HLR-QU-A and its SRs in Part 2 with the following clarifications: *(a)* Quantification is to include the fire ignition frequency per the IGN technical element (see 4-2.7) and fire-specific conditional damage probability factors per Requirements HLR-FSS-A through HLR-FSS-H. *(b)* Requirement QU-A4 in Part 2 is to be met based on meeting Requirement HLR-HRA-D in 4-2.10. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-QU-A in Part 2.		

NOTES:

(1) In some cases, a given fire scenario could lead to more than one initiating event. For example, in the case of a pump control cable failure, spurious operation of the pump might imply one initiating event, whereas a loss of function failure might imply a different initiating event. For screening purposes, the selection of the most conservative (i.e., the most challenging from the CDF and LERF perspectives) initiating event might be assumed with a conditional probability of 1.0 for the corresponding pump failure mode. Quantification might also consider both initiators with a split fraction applied to reflect each pump failure mode. The intent of Requirement FQ-A2 is to ensure that the selected initiating event, or events, encompasses the risk contribution from all applicable initiating events.

(2) When quantifying fire scenarios based on an internal-events initiating-event sequence, there may be a difference in success criteria, timing of human actions, and other elements of the PRA model for a fire-induced system failure that causes a demand for a reactor trip and the same failure if it occurs after a reactor trip. If, for example, the fire PRA model employs a general transient as the initiating event, with all of the fire impacts incorporated as failures subsequent to that trip, then to meet the intent of Requirement FQ-A2, it would be appropriate to ensure that any differences with respect to selecting a more specific initiating event are negligible.

(3) It is understood that quantification is performed using the fire PRA plant response model that meets 4-2.5.

Table 4-2.12-3 Supporting Requirements (SR) for HLR-FQ-B

The fire-induced CDF quantification shall use appropriate models and codes and shall account for method specific limitations and features (HLR-FQ-B).

Index No. FQ-B	Capability Category I	Capability Category II	Capability Category III
FQ-B1	PERFORM the quantification in accordance with HLR-QU-B and its SRs in Part 2. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-QU-B in Part 2.		

Table 4-2.12-4 Supporting Requirements (SR) for HLR-FQ-C

Model quantification shall determine that all identified dependencies are addressed appropriately (HLR-FQ-C).

Index No. FQ-C	Capability Category I	Capability Category II	Capability Category III
FQ-C1	INCLUDE dependencies during the fire PRA plant response model quantification in accordance with Requirement HLR-QU-C and its SRs in Part 2. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-QU-C in Part 2.		

Table 4-2.12-5 Supporting Requirements (SR) for HLR-FQ-D

The frequency of different containment failure modes leading to a large early release shall be quantified and aggregated thus determining the fire-induced LERF (HLR-FQ-D).

Index No. FQ-D	Capability Category I	Capability Category II	Capability Category III
FQ-D1	CALCULATE LERF from use of the fire PRA plant response model developed per 4-2.5 in accordance with Requirement HLR-LE-E and its SRs in Part 2 with the following clarifications: *(a)* Requirement LE-E1 of Part 2 is to be met in a manner consistent with 4-2.10. *(b)* Requirement LE-E1 of Part 2 is to be met in a manner consistent with 4-2.5 to the extent 4-2.5 modifies the requirements of 2-2.6 of Part 2. *(c)* Requirement LE-E4, including the "Discussion" for that SR of Part 2, is to be met following Requirements FQ-A1, FQ-B1, and FQ-C1. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under Requirement HLR-LE-E in Part 2.		

Table 4-2.12-6 Supporting Requirements (SR) for HLR-FQ-E

The fire-induced CDF and LERF quantification results shall be reviewed, and significant contributors to CDF and LERF, such as fires and their corresponding plant initiating events, fire locations, accident sequences, basic events (equipment unavailabilities and human failure events), plant damage states, containment challenges, and failure modes, shall be identified. The results shall be traceable to the inputs and assumptions made in the fire PRA (HLR-FQ-E).

Index No. FQ-E	Capability Category I	Capability Category II	Capability Category III
FQ-E1 [Note (1)]	IDENTIFY significant contributors in accordance with Requirements HLR-QU-D and HLR-LE-F and their SRs in Part 2 with the following clarifications: *(a)* Requirements QU-D6 and QU-D7 of Part 2 are to be met including identification of which fire scenarios and which physical analysis units (consistent with the level of resolution of the fire PRA such as fire area or fire compartment) are significant contributors. *(b)* Requirement QU-D7 of Part 2 is to be met recognizing that "component" in Part 2 is generally equivalent to "equipment" in Part 4. *(c)* Requirement QU-D4 for comparison to similar plants is not applicable. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under these sections in Part 2.		

NOTE:
(1) There is no requirement for a comparison of fire PRA results for similar plants under this SR, due to lack of fire PRA results using the updated industry fire PRA methods [4-A-6]. Additionally, small differences in geometry, plant layout, and the Fire Safe Shutdown Procedures may result in significant differences in risk that may be difficult to understand without detailed fire PRA results from plants being compared

Table 4-2.12-7 Supporting Requirements (SR) for HLR-FQ-F

Documentation of the CDF and LERF analyses shall be consistent with the applicable SRs (HLR-FQ-F).

Index No. FQ-F	Capability Category I	Capability Category II	Capability Category III
FQ-F1	DOCUMENT the CDF and LERF analyses in accordance with Requirements HLR-QU-F and HLR-LE-G and their SRs in Part 2 with the following clarifications: *(a)* Requirements QU-F2 and QU-F3 of Part 2 are to be met, including identification of which fire scenarios and which physical analysis units (consistent with the level of resolution of the fire PRA such as fire area or fire compartment) are significant contributors. *(b)* Requirement QU-F4 of Part 2 is to be met consistently with 4-2.13. *(c)* Requirements LE-G2 (uncertainty discussion) and LE-G4 of Part 2 are to be met consistently with 4-2.13. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under these sections in Part 2.		
FQ-F2	DOCUMENT any defined bases to support the claim of nonapplicability of any of the referenced requirements in Part 2 beyond that already covered by the clarifications in this Part.		

4-2.13 UNCERTAINTY AND SENSITIVITY ANALYSIS

The objectives of the uncertainty and sensitivity analysis (UNC) element are to
(a) identify sources of analysis uncertainty
(b) characterize these uncertainties
(c) assess their potential impact on the CDF and LERF estimates

This Part provides the requirements aimed at ensuring that uncertainties (i.e., those sources of uncertainty that can affect the use of a fire PRA's results in a risk-informed decision-making process) are appropriately identified and characterized with their potential impacts on the fire PRA understood.

For this technical element, an HLR for documentation is not included. Documentation of uncertainty is encompassed by Requirement HLR FQ-F.

Table 4-2.13-1 High Level Requirement for Uncertainty and Sensitivity Analysis (UNC)

Designator	Requirement
HLR-UNC-A	The fire PRA shall identify sources of CDF and LERF uncertainties and related assumptions and modeling approximations. These uncertainties shall be characterized such that their potential impacts on the results are understood.

Table 4-2.13-2 Supporting Requirements (SR) for HLR-UNC-A

The fire PRA shall identify sources of CDF and LERF uncertainties and related assumptions and modeling approximations. These uncertainties shall be characterized such that their impacts on the results are understood (HLR-UNC-A).

Index No. UNC-A	Capability Category I	Capability Category II	Capability Category III
UNC-A1 [Note (1)]	PERFORM the uncertainty analysis in accordance with Requirement HLR-QU-E and its SRs in Part 2 as well as Requirements LE-F2 and LE-F3 in Part 2. SPECIFY a defined basis to support the claim of nonapplicability of any of the requirements under these sections in Part 2.		
UNC-A2	INCLUDE the treatment of uncertainties, including their documentation, as called out in Requirements PRM-A4, FQ-F1, IGN-A10, IGN-B5, FSS-E3, FSS-E4, FSS-H5, FSS-H9, and CF-A2 and that required by performing Part 2 referenced requirements throughout this Standard.		

NOTE:
(1) It is intended that the uncertainty analysis include that which has been included in the quantification as affecting the quantified fire-induced CDF and LERF for the fire scenarios quantified per Requirement FQ-A3. Hence, it is not intended that uncertainty analysis include that which has been screened out by virtue of meeting the technical elements of this Standard (i.e., that screened out per 4-2.4 or 4-2.8 or any other justified screening performed).

Section 4-3
Peer Review for the Internal Fire Analysis

4-3.1 PURPOSE

This Section provides requirements for peer review of a Level 1/LERF fire PRA at-power.

4-3.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATIONS

In addition to the knowledge base specified in 1-6.2.4, the peer review team shall have knowledge and collective experience of systems engineering, fire PRA (including fire HRA), 10CFR50 Appendix R (or equivalent) Fire Safe Shutdown Analysis, circuit failure analyses, fire modeling, and fire protection programs and their elements, as applicable to the scope of the review.

4-3.3 REVIEW OF FIRE PRA ELEMENTS TO CONFIRM THE METHODOLOGY

The peer review team shall use the requirements of this Part for the fire PRA elements being reviewed to determine if the methodology and the implementation of the methodology for each fire PRA element meet the requirements of this Standard. Where an SR in this Part refers to SRs in other parts of this Standard, the peer review team shall evaluate the referenced SRs during the peer review against the fire PRA. The judgment of the reviewer shall be used to determine the specific depth of the review in each fire PRA element. The results of the overall fire PRA and the results of each fire 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 of Section 4-2 shall be used by the peer review team in assessing the completeness of a fire PRA element.

Prior to performing the initial fire PRA peer review, the peer review team shall verify that the internal-events PRA has been reviewed against Part 2. The results of the internal-events PRA peer review shall be reviewed as a part of the fire PRA peer review. This review shall be used in support of the determination for the capability category for SRs above referencing Part 2 requirements.

If the peer review is being performed as a result of PRA maintenance or upgrade, a previously completed internal-events PRA peer review of the revised analysis is not required. Instead, the internal-events PRA and fire PRA peer reviews may be done simultaneously or near the same time. However, the results of the internal-events PRA peer review shall be reviewed for its effect on the fire PRA once the internal-events PRA peer review report is complete.

4-3.3.1 Plant Partitioning (PP)

A review shall be performed on the plant partitioning analysis. The plant partitioning analysis verification typically includes the following:

(a) The global analysis boundary is appropriate to the overall fire PRA scope and the intended fire PRA applications.

(b) The criteria used to partition the plant into physical analysis units are defined and appropriate.

(c) All fire areas within the global analysis boundary have been clearly identified.

(d) In those cases where a fire barrier that lacks a fire resistance rating or spatial separation has been credited as a partitioning feature, use a selective review to show that an appropriate multi-compartment fire scenario analysis has been conducted.

(e) A selective review, by walkdown, is recommended to confirm the plant partitioning analysis.

The review of the plant partitioning analysis shall be closely coordinated with the review of the corresponding multi-compartment analysis.

4-3.3.2 Equipment Selection (ES)

A review shall be performed on the equipment selection process. The equipment selection process verification typically includes the following:

(a) The equipment selection process has captured the potentially risk significant equipment and their failure modes (including spurious operation) sufficient to meet the needs of the fire PRA application.

(b) The credited functions needed to support human actions in the fire PRA have been identified in a manner consistent with the fire PRA Capability Category being addressed (or otherwise that the fire PRA has assumed the worst failure mode for any non-credited equipment).

4-3.3.3 Cable Selection and Location (CS)

A review shall be performed on the cable selection and location process. The cable selection and location process verification typically includes the following:

(a) The cable selection process is consistent with the equipment selection and associated failure modes and

captures other support equipment (including locations) needed to provide the credited functions.

(b) That power supply and distribution systems have been treated in the cable selection process including fuse/breaker coordination.

(c) The cable location information (including cable endpoint location) is of sufficient depth and scope so as to support the intended fire PRA applications and is consistent with the physical analysis units as defined by Plant Partitioning.

(d) The fire PRA has appropriately treated those instances where specific cable location information is lacking.

4-3.3.4 Qualitative Screening (QLS)

If a qualitative screening analysis has been performed, the peer review shall be performed on it. The qualitative screening analysis verification typically includes the following:

(a) Appropriate qualitative screening criteria have been established.

(b) The criteria have been uniformly applied and a justification is provided for any physical analysis units screened out of the analysis with assurance that the screening process does not cause a significant risk contributor to be missed.

(c) A disposition has been documented for all physical analysis units within the global analysis boundary.

4-3.3.5 Fire PRA Plant Response Model (PRM)

A review shall be performed on the fire PRA plant response model. The fire PRA plant response model verification typically includes the following:

(a) The fire-induced initiating events are properly identified.

(b) The equipment (e.g., structures, systems, components, instrumentation, barriers) are properly modeled with the appropriate fire relevant failure modes, including spurious operation and accounting for the appropriate fire scenarios.

(c) The modeled equipment and HFEs reflect the as-built plant considering the reactor type, design vintage, and specific design.

(d) The human failure events are properly modeled including both non–fire-specific and fire-relevant actions.

4-3.3.6 Fire Scenario Selection and Analysis (FSS)

A review shall be performed on the fire scenario selection and analysis process. The fire PRA fire scenario selection and analysis process verification typically includes the following:

(a) The fire scenario selection and analysis element has identified and analyzed a representative set of fire scenarios that adequately cover potential risk-significant scenarios involving fire for both single- and multi-compartment scenarios as appropriate.

(b) The selected target sets are reasonable and appropriately reflect potential post-fire cable and equipment failures, including specification of failure modes, such as spurious operations, given the nature of the fire sources present and target locations.

(c) Fire detection and suppression considerations have been treated appropriately.

(d) Appropriate fire modeling tools have been selected, and that fire modeling tools have been applied within their capabilities and limitations by personnel knowledgeable of their use.

4-3.3.7 Ignition Frequency (IGN)

A review shall be performed on the ignition frequency analysis. The ignition frequency analysis verification typically includes the following:

(a) The ignition frequencies have included generic industry data and experience.

(b) As appropriate to the Capability Category, the ignition frequency analysis has considered plant outlier experience (Capability Category II) and/or has included plant-specific frequency updates (Capability Category III).

(c) The apportionment process applied to estimate fire area, fire compartment, and/or fire scenario frequencies has appropriately preserved the original plant-wide fire frequencies for all ignition sources.

4-3.3.8 Quantitative Screening (QNS)

If a quantitative screening analysis has been performed, the peer review shall be performed on it. The quantitative screening analysis verification typically includes the following:

(a) The quantitative screening criteria have been established, and that the applied criteria are consistent with the quantitative goals established for this technical element and for the required Capability Category.

(b) The criteria have been uniformly applied.

(c) A disposition has been documented for all physical analysis units within the global analysis boundary that survived qualitative screening.

4-3.3.9 Circuit Failures (CF)

A review shall be performed on the circuit failure analysis. The circuit failure analysis verification typically includes the following:

(a) For a selected set of representative cases, the circuit failure analysis has appropriately identified the relevant fire-induced circuit failure modes.

(b) For a selected set of representative cases, the circuit failure mode probability evaluations have appropriately quantified the likelihood of fire-related failure modes that could cause equipment functional failure and/or spurious operation.

4-3.3.10 Human Reliability Analysis (HRA)

A review shall be performed on the HRA. The HRA verification typically includes the following:

(a) The HRA adequately accounts for the additional influences caused by fire.

(b) HFEs adopted from an internal-events PRA have been modified as appropriate to reflect fire effects.

(c) New HFEs are included to account for specific fire-related actions that are consistent with plant procedures that were not covered by the internal-events PRA.

4-3.3.11 Seismic Fire (SF)

A review shall be performed on the seismic fire interactions review. The seismic fire interactions review verification typically includes a documented qualitative seismic fire interaction analysis whose findings are reasonable.

4-3.3.12 Fire Risk Quantification (FQ)

A review shall be performed on the fire risk quantification. The fire risk quantification verification typically includes the following:

(a) The CDF and LERF for each quantified fire scenario is properly quantified.

(b) The fire PRA provides the results and insights needed for risk-informed decisions.

(c) The CDF and LERF estimates and uncertainties have been reported.

(d) The significant risk contributors have been identified and discussed.

4-3.3.13 Uncertainty and Sensitivity (UNC)

A review shall be performed on the uncertainty and sensitivity analysis. The portion of the uncertainty and sensitivity analysis verification typically includes the following:

(a) Sources of uncertainty that can significantly affect the fire PRA conclusions have been identified.

(b) The effects of identified uncertainties have been properly estimated or that these uncertainties have been propagated during quantification and that the impacts on the results have been discussed and evaluated.

(c) Sufficient sensitivity analyses have been performed so as to provide an understanding of

(1) the level of robustness of the results

(2) how sensitive the acceptability of any risk-informed decisions may be to realistic changes in the value of uncertain parameters

Section 4-4
References

[4-1] EPRI TR-1011989–NUREG/CR-6850: EPRI/NRCRES Fire PRA Methodology for Nuclear Power Facilities, 8 NRC has endorsed the original 2001 version of NFPA-805 but not the 2006 revision. A joint EPRI/NRC publication, EPRI TR-1011989, Palo Alto, CA, and NUREG/CR-6850,U.S. NRC, Washington, DC, September 2005 (a report in two volumes); Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852

[4-2] NFPA Standard 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001; Publisher: National Fire Protection Association (NFPA), 1 Batterymarch Park, Quincy, MA 02169

[4-3] SFPE Handbook: The SFPE Handbook of Fire Protection Engineering, a joint publication of the Society of Fire Protection Engineers and the National Fire Protection Association, 3rd edition, 2002; Publisher: National Fire Protection Association (NFPA), 1 Batterymarch Park, Quincy, MA 02169

[4-4] NFPA Handbook: *Fire Protection Handbook, 19th Edition, 2003;* Publisher: National Fire Protection Association (NFPA), 1 Batterymarch Park, Quincy, MA 02169

[4-5] ASTM Standard E119 10b, Standard Test Methods for Fire Tests of Building Construction and Materials, October 2010; Publisher: American Society for Testing and Materials (ASTM International), 100 Barr Harbor Drive, P.O. Box C700, West Conshohocken, PA 19428

NONMANDATORY APPENDIX 4-A FIRE PRA METHODOLOGY

4-A.1 FIRE RISK ASSESSMENT METHODS

4-A.1.1 Overview

This Appendix contains discussion of various published methods for assessment of risk associated with fires internal to the plant in nuclear power plants (NPPs).[1] The intent is to provide the users of this Standard with insights into how their existing fire analyses may compare against the expectations for a fire risk analysis carried out in concert with this Standard.

Before the Individual Plant Examination for External Events (IPEEE) program, starting with WASH-1400 [4-A-1] and ending with NUREG/CR-4840 [4-A-2], a number of documents offered methods for estimating fire-induced risk. These documents made a significant contribution to the state of the art in fire probabilistic risk assessment (fire PRA); however, it is not the intent of this Appendix to provide a complete historical bibliography listing, relative to these methods. The focus of this Appendix is on

(a) the methods used by plants during the IPEEE program, because the entire fleet of U.S. nuclear power plants possesses an IPEEE fire risk analysis

(b) the most recently documented post-IPEEE methods

4-A.1.2 FIVE Methodology

The vast majority of the IPEEE fire studies followed one of two methodologies, namely, the "Fire-Induced Vulnerability Evaluation (FIVE)" methodology [4-A-3] and/or the "Fire PRA Implementation Guide" [4-A-4]. For comparison purposes, the "Fire Protection Significance Determination Process" (FPSDP) [4-A-5] and the Fire Risk Requantification Study Method as provided in EPRI 1011989-NUREG/CR-6850 [4-A-6], each of which has been developed since 2000, are also considered.

In 1988, the U.S. Nuclear Regulatory Commission (NRC) initiated the Individual Plant Examination (IPE) program. A supplement in 1989 outlined the need for examination of vulnerabilities resulting from external events including internal fires (IPEEE). In response to this need, the Electric Power Research Institute (EPRI) developed FIVE [4-A-3] and the "Fire PRA Implementation Guide" [4-A-4]. Nearly every plant in the U.S. used a combination of these two methods in response to the IPEEE program. It is important to note that the IPEEE process was a vulnerability search; hence, a full-scope fire PRA was not required to meet the IPEEE objectives.

FIVE [4-A-3] was developed as a screening methodology to search for vulnerabilities in NPPs. The methodology relied heavily on the existing plant fire protection analyses and documentation. This method was first piloted at two plants leading to a draft for NRC review. This review resulted in a final publication of FIVE in 1992 that was approved by the NRC, with qualifications, to meet the objectives of the IPEEE program.

4-A.1.3 Fire PRA Implementation Guide

In the early 1990s, EPRI initiated development of a second method, documented as the "Fire PRA Implementation Guide" [4-A-4]. This method was intended to offer improvements in key technical areas where FIVE did not offer a specific approach to reduce conservatism. The Fire PRA Implementation Guide [4-A-4] offered additional guidance and technical bases in a number of technical areas including

(a) development and evaluation of a fire PRA plant response model, including human actions

(b) fire characterization, determination of heat release rate, and fire severity

(c) assessment of fire growth and damage, detection, and suppression

(d) control room fires, including control room abandonment fire scenarios

(e) quantitative methods for the screening and assessment of fire involving multiple fire areas

The Fire PRA Implementation Guide [4-A-4] was reviewed by the NRC in 1997 and issued as EPRI/NRC 97-501, Review of the EPRI Fire PRA Implementation Guide [4-A-7].

This review [4-A-7] raised a number of technical issues with the method that culminated with 16 generic requests for additional information (RAIs) specific to the objectives of the IPEEE program [4-A-8]. Working with the Nuclear Energy Institute (NEI) and the NRC,

[1] The discussion here is focused on methods that have been developed and published with the intent of industry-wide application rather than approaches used by individual studies. In particular, before the IPEEE process, each individual fire PRA tended to build upon predecessor analyses, and each employed somewhat unique approaches and assumptions. The authors have made no attempt to characterize or describe these earliest fire PRAs. Also, it is important to recognize the specific approaches and important assumptions used in a fire risk assessment, as a study is likely to use a combination and/or variation of these published methods.

EPRI developed EPRI SU-105928, Guidance for Development of Response to Generic Request for Additional Information on Fire Individual Plant Examination for External Events (IPEEE), A Supplement to EPRI Fire PRA Implementation Guide (TR-105928) [4-A-9] in March of 2000 that addressed the 16 generic RAIs. The NRC, in a letter [4-A-10], approved the use of the Fire PRA Implementation Guide [4] with its supplement [4-A-9] in support of the IPEEE program.[2]

4-A.1.4 EPRI and RES Joint Project: Fire PRA Methodology for Nuclear Power Facilities

In late 2000, EPRI and the NRC Office of Regulatory Research (RES) initiated discussion of a joint project for developing improvements needed for the fire risk analysis methods to support risk-informed fire protection related decisions. In September 2005, EPRI and RES published EPRI 1011989-NUREG/CR-6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]. This method is a consolidation of the state of the art in fire PRA that reflects the consensus of EPRI and RES. This document received review from utilities, the Office of Nuclear Reactor Regulation (NRR), the Advisory Committee for Reactor Safeguards (ACRS), and the general public. This methodology offers significant improvements in key technical areas of fire PRA that are discussed in detail in Vol. 1 of EPRI 1011989-NUREG/CR-6850 [4-A-6]. Some of these improvements are as follows:

(a) New Tasks

(1) circuit selection and analysis, including consideration of multiple spurious equipment including instrument operations and probabilistic analysis of circuit failure modes

(2) approach for estimating damage from high-energy arcing faults

(b) Significant Changes: Change/Addition of Method

(1) ignition frequency model and use of the data

(2) postfire HRA, especially screening human error probabilities

(3) fire modeling, including fire characterization, severity factor definition, and modeling of fire detection and suppression processes

From 2003 to 2004, the NRC developed a significant revision of the approach used to assess the safety significance of fire protection inspection findings known as the FPSDP [4-A-5]. This method uses many of the same technical bases and databases that are used in EPRI 1011989-NUREG/CR-6850 [4-A-6] with simplifications that are intended to allow quicker examinations.

A summary comparison of EPRI FIVE [4-A-3], the Fire PRA Implementation Guide [4-A-4], the EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6], and the FPSDP [4-A-5] is offered in Table 4-A-1. This table is intended to highlight the key differences (with insights as to the strengths and weaknesses) of these methods. Some knowledge of these methods is needed for understanding the contents of this table as the text is summarized in the interest of maintaining a reasonable size. Readers are strongly advised to refer to the reference for details of each method.

[2] The stated objective of the IPEEE program was identifying potential vulnerabilities. NRC's approval was provided within this context.

Table 4-A-1 Overview of the Selected Methods for Fire Analysis

Technical Discipline/ Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Plant analysis boundary and partitioning	Use of the plant fire hazards analysis (FHA) and fire areas.	Similar to FIVE.	Clarification of guidance in FIVE and "Fire PRA Implementation Guide." Stronger focus on fire compartments rather than fire areas.	Relies on plant FHA and fire areas but allows use of fire PRA for further compartmentalization.
Screening				
Qualitative	Based on equipment credited (in this case safe shutdown components) and whether there can be a plant trip initiator.	Based on equipment credited in internal-events PRA.	Based on the fire PRA equipment that is based on internal-events PRA accident sequence model, an analysis of the internal events initiating events and circuit analysis.	Qualitative screening in FPSDP is applied to the findings as opposed to fire compartments or scenarios.
Quantitative	CDF < 1E-6/ reactor-year.	Similar to FIVE.	Added LERF and allows analyst flexibility to suit application needs. Accounts for the cumulative fire risk and fire versus internal risk. [Note (2)]	Screening applied to the findings as opposed to fire compartments or scenarios. Findings are screened against RG 1.174 [4-A-11] criteria.

Table 4-A-1 Overview of the Selected Methods for Fire Analysis (Cont'd)

Technical Discipline/ Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Fire initiation, propagation, mitigation, and damage				
Fire ignition frequency	A location/ component-based model. The model was built directly based on the operating history of fires at U.S. nuclear power plants.	Similar to FIVE.	Component-based ignition frequency model that begins from plant-wide fire frequency for a given group of components. Based on 2000 version of EPRI fire event database. [Note (3)] Review and exclusion of nonchallenging fires. Use of two-stage Bayesian method.	A simplified version of the EPRI/NRC-RES method is used. Some ignition source bins are combined for a net of fewer unique bins. Otherwise, data analysis is identical.
	Note that all of these methods begin from the plant-wide fire frequency derived from operating experience. With the exception of FPSDP, all methods preserve this plant-wide fire frequency in the partitioning process. FPSDP uses a component-based partitioning approach that uses generic estimates of plant component populations. As a result, the plant-wide frequency may not be fully preserved.			
Initial fire characterization, including heat release rate (HRR) and severity factor	Use of single-point peak HRR. Electrical fires: no suggested duration. Oil fires: confined and unconfined spills. Others: no specific guidance.	Use of single-point peak HRR and fixed severity factor. Electrical fires: A 30-min duration suggested based on Sandia National Laboratory (SNL) cabinet fire tests. Oil fires: Similar to FIVE.	Electrical fires: Multiple-point HRR or single-point HRR and scenario-adjusted severity factor. Suggested fire duration based on SNL cabinet fire tests. Oil fires: Similar to FIVE transient fires; confined and unconfined spills similar to FIVE. Multiple-point HRR or single-point HRR and scenario-adjusted severity factor.	Electrical fires: Two-point HRR model (expected and high confidence) and scenario-adjusted severity factor (0.9 and 0.1, respectively). Oil fires: Similar to FIVE transient fires; confined and unconfined spills similar to FIVE. Two-point HRR and scenario-adjusted severity factor for other transients.

Table 4-A-1 Overview of the Selected Methods for Fire Analysis (Cont'd)

Technical Discipline/ Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Fire growth and propagation	Hand calculations (Refs. [4-A-12] and [4-A-13]).	Hand calculations (Refs. [4-A-12] and [4-A-13]) augmented by optional zone models. Special model for *(a)* electrical cabinet to adjacent cabinet; *(b)* hydrogen fires.	Left to the user to select and use appropriate model. Special models for *(a)* cable fires; *(b)* electrical cabinet to adjacent cabinet (same as EPRI SU-105928) [4-A-9] *(1)* high-energy arcing faults, *(2)* hydrogen fires, *(3)* main control board fires, *(4)* turbine/ generator fires.	Zone of influence derived from hand calculations and simple spreadsheet formulations (reference [4-A-14]). Special models for high-energy arcing faults and cable fires adopted from EPRI/NRC-RES method.
Fire detection and suppression	Automatic suppression: Generic unreliability used with effectiveness based on scenario geometry and timing. Manual suppression: Fire brigade unreliability derived from plant-specific drill results. Effectiveness based on scenario geometry and timing.	Automatic suppression: Generic unreliability similar to FIVE. Effectiveness tied loosely to design, installation, and maintenance in addition to scenario geometry and timing. Manual suppression: Prompt suppression by plant personnel based on historical evidence. Fire brigade unreliability derived from generic data. Effectiveness tied loosely to design, installation, and maintenance in addition to scenario geometry and timing.	Improved and clearer guidance on analysis of detection and suppression in an event tree format. Automatic suppression: Generic unreliability similar to FIVE. Effectiveness tied loosely to design, installation, and maintenance in addition to scenario geometry and timing. Manual suppression: Prompt suppression by plant personnel based on historical evidence. High-energy arcing faults: Unique suppression curve derived from HEAF events.	Similar to EPRI/ NRC-RES, except that the use of data is limited to post-1988 and fire brigade response time is embedded in suppression time data (total fire duration is used in lieu of suppression time). Root data are taken from EPRI/ NRC-RES analysis.

Table 4-A-1 Overview of the Selected Methods for Fire Analysis (Cont'd)

Technical Discipline/ Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Main control room fires	No specific guidance for treatment of control room fires.	Methods for derivation of fire frequency in the individual control room electrical panels and control room evacuation scenarios, including postevacuation operator manual action reliability. Time to evacuation based on generic main control room (MCR) suppression reliability.	Data on severity factors for control board fires. Method for fire frequency in the individual control room electrical panels similar to Fire PRA Implementation Guide. Time to evacuation derived from analysis of plant-specific MCR fire scenarios.	Not treated explicitly in current Phase II process. FPSDP does provide fire frequency estimates and a fire duration curve derived from EPRI/NRC-RES method.
Fire barriers and multi-compartment fires	A qualitative approach to derivation of fires crossing fire barriers with a fire-resistance rating.	A quantitative method that considers failure of fire barriers with a fire-resistance rating based on data in NUREG/CR-4840 [4-A-2].	Similar to Fire PRA Implementation Guide.	Only considered when finding is related to a degraded fire barrier. Treatment is similar to Fire PRA Implementation Guide, but barriers are credited based on degradation of nominal fire-resistance rating. No treatment of random failures.
Electrical raceway fire barrier systems (ERFBS) and other passive fire protection systems	Credited as 100% effective (follows Fire Safe Shutdown/ Appendix R).	Guidance based on limited available fire test for specific ERFBS, solid-bottom trays and some coatings.	Similar to Fire PRA Implementation Guide.	Credited at nominal fire-resistance rating or at a degraded rating if there is a finding against the barrier.
Cable damage/ ignition temperature	. . .	. . .	Utilizes available data from fire testing and from certain equipment qualification tests.	Derived from EPRI/NRC-RES method documentation.

Table 4-A-1 Overview of the Selected Methods for Fire Analysis (Cont'd)

Technical Discipline/ Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Fire-induced risk				
Definition of risk	The definition of risk is, for the most part, the core damage frequency (CDF). The methodology requires qualitative investigation of containment systems availability in the event of fire, somewhat similar to Fire Safe Shutdown Analysis.	Similar to FIVE.	The definition of risk covers CDF and LERF.	Focus is on risk change given a finding of degradation against some element of the fire protection program. Primary measure is CDF although an LERF SDP is also available.
Fire PRA components	Components credited in the safe shutdown analysis (SSA).	Combination of the components credited in the internal-events PRA and SSA.	In addition to internal-events PRA, there are significant additions: components leading to internal events initiating events. Consideration of multiple spurious operations. Identification of diagnostic instrumentation faults that may impact operator actions.	Utilizes the plant's SDP model with supplements as deemed appropriate by the SRA.

Table 4-A-1 Overview of the Selected Methods for Fire Analysis (Cont'd)

Technical Discipline/ Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Fire PRA sequences	Uses unavailability of the SSA credited components to derive conditional core damage probability (CCDP). This implies use of SSA sequence/ strategy. Loss of off-site power assumed for most cases.	Derived based on the internal-events PRA model.	Similar to Fire PRA Implementation Guide with the following addition: Specific guidance to review plant procedures to search for fire-specific sequences either as the result of fire-specific procedures or accident initiators/sequences resulting from single/multiple fire-induced spurious operations.	Maps sequences to internal events using plant SDP models.
Circuits and failure modes	Relies entirely on the plant's SSA.	Similar to FIVE.	Specific guidance for selection and analysis of circuit failure modes and their likelihood within the context of risk.	Uses insights of the EPRI/NRC-RES method
Postfire operator manual actions	Not addressed.	Guidance for assigning postfire HEPs to actions inside and outside the MCR.	New method for screening postfire HRA with quantification. Partial development of detailed fire HRA with focus on performance shaping factors to be addressed.	Screening level estimates of manual action reliability are used. (Anything beyond screening level is Phase III.)

Table 4-A-1 Overview of the Selected Methods for Fire Analysis (Cont'd)

Technical Discipline/ Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Seismic/fire interactions	Qualitative approach based on review of plant analyses and walkdowns.	Similar to FIVE.	Similar to FIVE.	Not addressed.
Uncertainty	Not addressed.	Not addressed.	Identification of the sources of uncertainty and suggestions as to how they may be treated.	Not addressed.

GENERAL NOTES:
Acronyms:
CDF = core damage frequency
EPRI = Electric Power Research Institute
ERFBS = electrical raceway fire barrier system
FHA = fire hazards analysis
FIVE = EPRI Fire-Induced Vulnerability Evaluation method
FPRAIG = EPRI Fire PRA Implementation Guide
LERF = large early release frequency
HEAF = high-energy arcing fault
HEP = human error probability
HRA = human reliability analysis
MCR = main control room
NRC/RES = U.S. Nuclear Regulatory Commission Office of Nuclear Regulatory Research
SDP = significance determination process
SSA = safe shutdown analysis

NOTES:
(1) FPSDP also includes a Phase 3 process where the full spectrum of fire PRA tools, methods, and data can be applied. This table is limited to consideration of the FPSDP Phase I and II procedures.
(2) The objective of vulnerability evaluation is identifying combinations of hazards and plant response that lead to high risk. In the process, an acceptable and often exercised practice is to screen low-risk contributors. On the other hand a risk assessment needs to ensure that a reasonable profile of risk is obtained regardless of how small the individual contributors may be. For example, consider a plant with 100 fire compartments where the fire-induced CDF for one compartment is 1E-6/reactor-year and the remaining 99 compartments have a fire-induced CDF of 1E-8/reactor-year each. In a vulnerability evaluation, the high-risk compartment is clearly the vulnerability, and the remaining compartments are of less concern and therefore may not be reported in the results. On the other hand, in a risk assessment, the 99 low-risk compartments contribute to the total fire risk at the plant and should be included in the fire risk profile reported.
(3) The fire ignition frequency model in reference [4-1] is one step closer than references [4-A-3] and [4-A-9] toward a component-based ignition frequency model in that it does not use the location of a fire source as a contributor to its fire frequency. Both methods still conserve the total plant-wide fire frequency for each component type.

4-A.2 EXAMINATION OF THE FIRE RISK METHODS AGAINST THE CAPABILITY CATEGORIES OF THIS STANDARD

This Part is an examination of selected fire risk methods against the capability category requirements in the main body of this Standard. This examination assigns capability categories at rather high levels as compared with the main body of this Standard. For example, the fire ignition frequency is represented with a single entry in this table while the same technical element is defined with several HLRs and SRs in the main body of this Standard. It is not the intent of this Appendix to go beyond this level of detail at this time.

The following considerations are critical to the use of the information in Table 4-A-2:

(a) The process of establishing quality requires careful consideration of the details that are embedded in the HLRs and SRs discussed in the main body of this Standard. This table should only be used as supplemental information when trying to establish a relationship between these methods and the requirements contained in the main body of this Standard.

(b) The level of the detail of this table also leads to the need for assigning multiple capability categories for the same technical discipline/task [e.g., while parts of the Fire Scenarios Selection and Analysis in EPRI "Fire PRA Implementation Guide" and its supplement ([4-A-4], [4-A-9]) may be CAT I, other parts of it may be classified as CAT II].

(c) For this examination, a method is defined as it is documented and not as it may be implemented. Many fire risk assessments tend to take various pieces of different methodologies and therefore should be evaluated in that context.

(d) The examination presented here should not be taken as a universal assessment applicable to every implementation of a given method. Even within a given method, there is generally wide latitude for the use of analyst judgment. Analyst choices, even if they fall within the overall guidance of a particular method, could shift the capability category (either up or down). Each application should be judged on its own merits.

Table 4-A-2 Examination of the Fire Risk Methods Against the Capability Categories of This Standard

Technical Discipline/ Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Plant partitioning (4-2.1 in the main body of this Standard)	**CAT I/II:** Follows the FHA and does not require examination of all locations "within the licensee-controlled area where a fire could adversely affect any function or equipment to be credited in the Fire PRA model."	**CAT I/II**: Same as FIVE	**CAT III**: . . .	**CAT I:** Plant partitions are generally based on plant fire areas per compliance documents, but inspectors are given latitude to consider fire scenarios in fire compartments consistent with the EPRI/NRC-RES method.

Table 4-A-2 Examination of the Fire Risk Methods Against the Capability Categories of This Standard (Cont'd)

Technical Discipline/Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Equipment selection and location (4-2.12 in the main body of this Standard)	**Less than CAT I:** Will likely not include multiple spurious operation considerations and probably not instrumentation faults.	**CAT I/II:** Although more complete than EPRI FIVE (includes PRA), still will likely not include multiple spurious operation considerations and probably not instrumentation faults.	**CAT III:** If carried out to its complete potential, will meet the Cat III requirements of this Standard.	**Less than CAT I:** [Note (1)] Relies on equipment credited in plant SDP system notebooks with selective updating by SRA if required to support analysis.
Cable selection and location (4-2.3 in the main body of this Standard)	**CAT I:** . . .	**CAT I:** . . .	**CAT III:** . . .	**Less than CAT I:** FPSDP requires no supplemental cable tracing beyond information available at the plant site. Supplemental information can be used if available.
Qualitative screening (4-2.4 in the main body of this Standard)	**CAT III:** . . .	**CAT III:** . . .	**CAT III:** . . .	**Less than CAT I:** Qualitative screening criteria are defined in Phase I of the process but are applied to each *identified* finding rather than to fire *compartments.*

Table 4-A-2 Examination of the Fire Risk Methods Against the Capability Categories of This Standard (Cont'd)

Technical Discipline/Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Fire PRA plant response model (4-2.5 in the main body of this Standard)	**Less than CAT I:** The "Unavailability of the protected train" concept of FIVE is not likely to meet criteria set for CAT 1 for the PRM element and, in particular, those SRs related to spurious actuations.	**CAT I/II:** Based on the scope of the internal-events PRA, the major portions of this model should be between that called out under CAT I and CAT II of this standard. The analysis will not likely have addressed multiple spurious operations. Not having addressed instrumentation faults results in less than CAT I relative to this consideration.	**CAT III:** Method generally meets CAT III requirements. In the area of spurious operation analysis, SR ES-A4 is not explicitly covered by the method. The method does not set an upper bound on spurious actuation considerations and can readily accommodate the expanded analysis scope implied by these requirements.	**Less than CAT I:** SDP plant system notebooks are applied with selective updating as deemed necessary by the supporting SRA. Supplemental models may be applied if available.
Fire scenario selection and analysis (4-2.6 in the main body of this Standard)	**CAT I:** Lack of guidance for analysis of some scenarios.	**CAT I/II:** Lack of guidance for analysis of some scenarios.	**CAT I/II/III:** Different levels of fire modeling are applied depending on the nature of the scenario and its risk importance.	**CAT I/II:** FPSDP focuses on identification and quantification of credible fire scenarios. Fire modeling tools applied are based largely on NUREG-1805 [4-A-14].
Ignition frequency (4-2.7 in the main body of this Standard)	**CAT I:** Application of generic fire frequencies based on industry-wide experience without plant-specific updates.	**CAT I:** Method uses generic fire frequencies based on industry-wide experience. Plant-specific updates are possible, but guidance does not cover "outlier experience."	**CAT III:** State-of-the-art statistics including plant-specific updates (recommended) and consideration of uncertainty.	**CAT I:** Approach derived directly from EPRI/NRC-RES method but is simplified such that plant-specific equipment counts are not required (uses generic industry-wide statistics).

Table 4-A-2 Examination of the Fire Risk Methods Against the Capability Categories of This Standard (Cont'd)

Technical Discipline/Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Quantitative screening (4-2.8 in the main body of this Standard)	**CAT I:** Method does not consider cumulative contribution of screened areas for CDF or LERF as a screening criteria per QNS-C1.	**CAT I:** Method does not consider cumulative contribution of screened areas for CDF or LERF as a screening criteria per QNS-C1.	**CAT I/II/III:** Quantitative screening criteria are recommended, but implementation is left to analyst discretion. Screening criteria are also left to the analyst to define. Hence, analyst choices would govern the capability category achieved.	**Less than CAT I:** FPSDP uses a continuous quantitative screening approach, but findings are screened, not *physical analysis units.*
Circuit failures (4-2.9 in the main body of this Standard)	**CAT I:** This method relies on plant SSA for selection and analysis of fire-induced circuit failures.	**CAT I:** This method relies on plant SSA for selection and analysis of fire-induced circuit failures.	**CAT II/III:** Specific guidance that allows search for multiple spurious operations (even though it does not ensure all to be found). Actual depth of the analysis is left to analyst discretion as required to suit intended application.	**Less than CAT I:** The treatment of circuit failures is dependent on the nature of the finding and is implemented at the discretion of the supporting SRA. Generally pursued only for circuit-related findings.

Table 4-A-2 Examination of the Fire Risk Methods Against the Capability Categories of This Standard (Cont'd)

Technical Discipline/Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Human reliability analysis (4-2.10 in the main body of this Standard)	**Less than CAT I:** This method does not offer instructions as to how to model these events.	**CAT I:** The instructions recognize and offer simple approach to account for the fire impact on the actions and dependency. However, relative to considering human errors from spurious instrumentation signals, this method does not meet even CAT I.	**CAT II/III:** Methodology is limited to screening methods though allows for accounting of appropriate performance shaping factors including fire effects. However, choice of a detailed HRA method and specific implementation guidance is left to analyst discretion.	**CAT I:** The FPSDP provides only a limited high level screening approach for human reliability.
Seismic fire (4-2.11 in the main body of this Standard)	**CAT I:** Qualitative assessment through review verification supplemented by walkdown.	**CAT I:** Same as FIVE.	**CAT II/III:** Still qualitative assessment, but expanded review and verification guidance.	**Less than CAT I:** Not considered in FPSDP Phase I/II.
Fire risk quantification (4-2.12 in the main body of this Standard)	**Less than CAT I:** The Vulnerability evaluation method did not require quantification of final fire risk results. No risk was calculated if all compartments dropped below screening criteria. May be a CAT I if high risk dictated "the significant contributors to the final fire risk results."	**CAT I:** This method includes CDF quantification for the most significant contributors. The method provides some consideration of containment bypass scenarios and the impact of fire on containment functions (Step 9.1), but did not explicitly quantify fire-induced LERF.	**CAT III:** This method should fall into the same category for quantification in ASME-RA-2002/ RA-Sb-2005 since it follows similar fundamentals and depth of analysis.	**Less than CAT I:** Risk quantification is based on *findings* not on *physical analysis units.*

Table 4-A-2 Examination of the Fire Risk Methods Against the Capability Categories of This Standard (Cont'd)

Technical Discipline/Task	EPRI FIVE [4-A-3]	EPRI Fire PRA Implementation Guide and Its Supplement [4-A-4], [4-A-9]	EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities [4-A-6]	FPSDP-Phase I/II [Note (1)] [4-A-5]
Uncertainty and sensitivity (4-2.13 in the main body of this Standard)	**Less than CAT I:** This method does not offer instructions as to how to account for uncertainties.	**Less than CAT I:** This method does not offer instructions as to how to account for uncertainties.	**CAT I/II/III:** Method provides discussion of uncertainty sources and methods, but extent of implementation is left to the analyst.	**Less than CAT I:** Not considered in FPSDP Phase I/II.

NOTE:
(1) In this table "Less than CAT I" is used when the treatment of the technical Discipline/Task in the method does not satisfy the high-level requirements described in the main body of this Standard. Note that in some cases the method may not have been intended to produce the results associated with technical discipline/task.

4-A.3 REFERENCES

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[4-A-2] NUREG/CR-4840, "Recommended Procedures for the Simplified External Event Risk Analyses for NUREG 1150," September 1989; Publisher: U.S. Nuclear Regulatory Commission (NRC), 11555 Rockville Pike, Rockville, MD 20852

[4-A-3] EPRI TR-100370, Fire-Induced Vulnerability Evaluation (FIVE), May 1992; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

[4-A-4] EPRI TR-105928, Fire PRA Implementation Guide, December 1995; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

[4-A-5] "Fire Protection Inspection Significance Determination Process," Inspection Manual Chapter 0609, Appendix F, U.S. NRC, Washington, DC, February 2005; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

[4-A-6] EPRI TR-1011989, NUREG/CR-6850, EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, a joint publication: Electric Power Research Institute/U.S. Nuclear Regulatory Commission, September 2005, in two volumes; Publisher: U.S. Nuclear Regulatory Commission (NRC), 11555 Rockville Pike, Rockville, MD 20852

[4-A-7] EPRI/NRC 97-501, "Review of the EPRI Fire PRA Implementation Guide," Letter Report to the U.S. Nuclear Regulatory Commission, August 1997; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

[4-A-8] "Request for Additional Information on the EPRI Fire Probabilistic Risk Analysis Implementation Guide," Letter from M. W. Hodges (NRC/RES) to D. Modeen (NEI), December 3, 1997

[4-A-9] EPRI SU-105928, Guidance for Development of Response to Generic Request for Additional Information on Fire Individual Plant Examination for External Events (IPEEE), A Supplement to EPRI Fire PRA Implementation Guide (TR-105928), March 2000; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

[4-A-10] EPRI Guidance for Development of Response to the NRC's Generic Request for Additional Information on the EPRI Fire PRA Implementation Guide, Letter from T. L. King (NRC) to D. Modeen (NEI), June 15, 1999

[4-A-11] RG 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, available on the Internet at http://www.nrc.gov/reading-rm/adams.html under ML003740133, Regulatory Guide 1.174, July 1998; Publisher: U.S. Nuclear Regulatory Commission (NRC), 11555 Rockville Pike, Rockville, MD 20852

[4-A-12] EPRI TR-100443, Methods for Quantitative Fire Hazards Analysis, May 1992; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

[4-A-13] EPRI 1002981, Fire Modeling Guide for Nuclear Power Plant Applications, August 2002; Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

[4-A-14] NUREG-1805, Fire Dynamics Tools (FDTs): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program, December 2004; Publisher: U.S. Nuclear Regulatory Commission (NRC), 11555 Rockville Pike, Rockville, MD 20852

(a) PART 5
(b) REQUIREMENTS FOR SEISMIC EVENTS AT-POWER PRA

Section 5-1
Overview of Seismic-PRA Requirements At-Power

5-1.1 PRA SCOPE

This Part establishes technical requirements for a Level 1 and large early release frequency (LERF) analysis of seismic events while at power.

5-1.2 COORDINATION WITH OTHER PARTS OF THIS STANDARD

This Part is intended to be used with Parts 1 and 2 of this Standard.

Standards ANSI/ANS-2.27-2008, Criteria for Investigations of Nuclear Facility Sites for Seismic Hazard Analysis [5-1], and ANSI/ANS-2.29-2008, Probabilistic Seismic Hazards Analysis [5-2], are available as background information.

5-1.3 SEISMIC EVENTS SCOPE

The requirements herein cover

(a) a Level 1 analysis of the core damage frequency (CDF) and

(b) a limited Level 2 analysis sufficient to evaluate the large early release frequency (LERF).

The approach to any external hazard PRA typically uses as its starting point the internal-events PRA model, to which must be added a number of structures, or systems, or components, or a combination thereof (SSCs) not included in that model but that could fail due to the external hazard. Some "trimming" of that model is also common, to eliminate parts of it not relevant to the external hazard analysis (see Requirement SPR-A4 for more discussion of these issues). Both the part of the internal-events model dealing with CDF and the part dealing with LERF are used as starting points.

The analysis of the LERF endpoint proceeds in the same way as the analysis of the CDF endpoint, with one major exception, as follows: There are some accident sequences, leading to core damage but not to large early releases in the internal-events PRA model, that need to be designated as potential LERF sequences when caused by an external hazard. One set of sequences is those where the effects of the external hazard might compromise containment integrity and thereby possibly contribute to LERF. The other set is sequences in which off-site protective action (specifically, the evacuation of nearby populations) is impeded due to the external hazard. The same sequence that might not be a LERF sequence due to any internal hazard may perhaps affect nearby populations that cannot evacuate as effectively.

These sequences would fall into the LERF category because the word "early" in the definition of LERF does not refer to a specific point in time but rather to the issue of whether a large release might occur before effective protective actions (e.g., evacuation and sheltering) can be implemented to protect surrounding populations.

For example, suppose that an earthquake that triggers an accident sequence at the nuclear plant were to damage the only road available to evacuate close-in populations. Without effective evacuation, these populations may be exposed to radioactive releases that they would not be exposed to were the same accident sequence to arise from an internal hazard.

Therefore, in analyzing external hazards that have the potential to impede effective emergency evacuation, the analysis must examine whether any accident sequences that are not in the LERF category in the internal-events PRA model need to be included in that category for the particular event being evaluated. The LERF part of the PRA analysis would require expansion accordingly.

5-1.4 THE PHRASE "ACCEPTABLE METHOD"

In many places, the commentary contains words such as, "Reference X provides an acceptable method for performing this aspect of the analysis." The plain meaning of this wording should be clear, namely, that using the methodology or data or approach in Reference X is one way to meet this Standard. The intent of any requirement that uses this language is to be *permissive*, meaning that the analysis team can use another method without prejudice.

However, it is important to understand that the intent of this Standard goes beyond the plain meaning, as follows: Whenever the phrasing "acceptable method" is used herein, the *intent* is that if the analysis uses another method, the other method must accomplish the stated objective with an appropriate level of detail and scope. It is not acceptable to use another method that does not accomplish the intent of the requirement. Whenever an alternative to the acceptable method is selected, it is understood that the peer-review team will pay particular attention to this choice.

5-1.5 FIDELITY: PLANT VERSUS SEISMIC PRA

It is important that the PRA or SMA reasonably reflect the actual as-built, as-operated nuclear power plant being analyzed. Several mechanisms are used to achieve this fidelity between plant and analysis. One key mechanism is called "plant familiarization." During this phase, plant information is collected and examined. This involves

(a) information sources, including design information, operational information, maintenance information, and engineering information

(b) or plant walkdowns, both inside and outside the plant

Later, if the plant or the PRA is modified, it remains important to ensure that fidelity is preserved, and hence, further plant-familiarization work is necessary.

Throughout this Standard, requirements can be found whose objective is to ensure fidelity between plant and analysis. Because seismic PRAs depend critically on plant walkdowns, both inside and outside the plant, to ascertain the physical configurations of important SSCs and the environments to which they are exposed, this Section places special emphasis on *walkdowns*, through requirements in the relevant sections dealing with SSC fragilities due to earthquakes (5-2.2) and with peer review (5-2.3).

5-1.6 USE OF GENERIC FRAGILITY INFORMATION

Supporting Requirement SFR-F2 includes the use of generic fragilities. Seismic-PRA analysts sometimes find the need to use generic rather than plant-specific seismic-fragility information. There are many reasons for this, and using generic seismic fragilities is an efficient approach if done with appropriate care. However, there have been a few seismic PRAs in the past where the use of generic fragilities has been inappropriate, either because the generic seismic fragilities that were used did not apply to the specific plant, or because the generic fragilities relied on were themselves more generally erroneous, having been compiled for a different purpose. For example, generic fragilities developed as achievable for SSCs of advanced reactor designs may not be appropriate for current plant SSCs. Analysts should apply caution in the use of generic fragilities and provide justification that the generic fragilities are applicable to the plant-specific SSCs and plant conditions, including an understanding of the purpose and scope of the source of the generic fragilities. Peer reviews should focus on the use of generic fragilities to ensure that their use is appropriate and justified.

Section 5-2
Technical Requirements for Seismic PRA At-Power

The technical requirements for seismic PRA have been developed based on a wealth of experience over the past 20 yr, including a very large number of full-scope seismic PRAs for nuclear power plants and a large number of methodology guidance documents and methodology reviews. Nonmandatory Appendix 5-A contains a short introduction and review of the seismic-PRA methodology. Other useful references include references [5-6], [5-9], [5-10], [5-16], [5-17], and [5-18]. The earliest important guidance on seismic-PRA methods is described in references [5-16], [5-19], and [5-20]. The proceedings of an international conference sponsored by the Organization for Economic Cooperation and Development (OECD) Nuclear Energy Agency (NEA) in Tokyo [5-21] contain a number of methodological advances. The principal guidance on seismic hazard analysis is in references [5-22] and [5-23]. The major PRA technical elements of a seismic PRA are

(a) probabilistic seismic hazard analysis
(b) seismic-fragility evaluation
(c) seismic plant respone analysis

The technical requirements for each of these are given in the following subsections.

Seismic PRA is an integrated activity requiring close interactions among specialists from different fields (for example, seismic hazard analysis, systems analysis, and fragility evaluation). Although the methodology for seismic PRA and the supporting data have evolved and advanced over the past 30 yr, the analysis still requires judgment and extrapolation beyond observed data. Therefore, the analyst is strongly urged to review published seismic-PRA reports and to compare his/her plant-specific seismic PRA to the published studies of similar reactor types and system designs. This will promote consistency among similar PRAs and risk-informed applications and will also promote reasonableness in the numerical results and risk insights. The peer review is also directed in part toward this same objective.

5-2.1 PROBABILISTIC SEISMIC HAZARD ANALYSIS

Requirements for the probabilistic seismic hazard analysis PSHA address two situations. The first situation deals with cases where no prior study exists, and the site-specific PSHA must be generated anew. In the second situation, the PSHA analyst may have the option to use an existing study to form the basis for a site-specific assessment. For example, the Lawrence Livermore National Laboratory (LLNL) and Electric Power Research Institute (EPRI) regional hazard studies [5-24, 5-25] for east of the Rocky Mountains can be used to develop a site-specific PSHA for most of the central and eastern U.S. (CEUS) sites after certain checks or revisions are made (see Requirement HLR-SHA-H in Table 5-2.1-9).

Since the publication of the LLNL and EPRI studies in the late 1980s and early 1990s, significant updates have occurred in connection with early site permit (ESP) and combined license (COL) applications. For example, reference [5-41] contains updates to seismic sources. EPRI also revised its ground-motion models in the early 2000s [5-42], and these revised models have been used in the ESP and COL applications. A joint research program among NRC, DOE, and EPRI to update the CEUS seismic sources has also concluded [5-43]. A joint program to update the CEUS ground motion, known as Next Generation Attenuation (NGA) East, is expected to be completed in 2015. Whether an update of a given seismic PRA is needed will depend on the application.

As discussed in Requirement HLR-SHA-H, these studies and many hazard studies conducted for plant-specific PRAs are considered to meet the overall requirements of this Part, subject to any revisions as necessary. The intent of this requirement is not to repeat the entire hazard exercise or calculations, unless new information and interpretations that affect the site have been established and affect the usefulness of the seismic PRA for the intended application. The peer review should concentrate on this aspect and report the findings as to the suitability of using existing analysis.

The primary objective of the PSHA for most sites is to estimate the probability or frequency of exceeding different levels of vibratory ground motion, and the requirements described in this Part address this objective in detail. If site conditions make it necessary to include other seismic hazards, such as fault displacement, landsliding, soil liquefaction, soil settlement, and earthquake-induced external flooding, the objective is similar — to estimate the probability or frequency either of hazard occurrence as a function of its size or intensity, or of hazard consequences.

The "level" (complexity and efforts related to use of expert judgment, expert elicitation, integration, etc.) of hazard analysis depends on two primary considerations:

(a) intended use of the seismic PRA (linked with the Capability Category needed for that application)

(b) the complexity of the seismic environment

When dealing with a particular issue that will affect the results of the PSHA, the NRC/EPRI/DOE Senior Seismic Hazard Analysis Committee's (SSHAC's) so-called "SSHAC" report [5-22] lists the following factors that affect the choice of level for the hazard analysis:

(1) the significance of the issue to the final results of the PSHA

(2) the issue's technical complexity and level of uncertainty

(3) the amount of technical contention about the issue in the technical community

(4) important nontechnical considerations such as budgetary, regulatory, scheduling, or other concerns

Based on considerations of the above, with respect to the issues identified and other factors, the SSHAC report has identified and provided guidance for four "levels" of hazard analysis. When viewed in the context of this Standard's Capability Categories, the SSHAC Levels 1 and 2 will generally correspond to Capability Category I; Levels 2 and 3 will generally correspond to Capability Category II; and Levels 3 and 4 will correspond to Capability Category III. Level 1 or 2 analysis, based primarily on the use of available information, by its very nature will contain more uncertainties and will need to be demonstrably adequate or conservative for the intended application. On the other hand, accurate characterization and reduction of uncertainties are deemed essential features of Capability Category III applications, requiring development of detailed site-specific information possibly including field investigations (Levels 3 and 4).

The LLNL [5-24] and EPRI [5-25] seismic hazard studies are considered SSHAC Level 3 studies and, therefore, meet the requirements of this Part as stated earlier.

To illustrate further, using generic or regional hazard analyses or mean hazard estimates, as was often done in various IPEEE applications, would be examples of Capability Category I. Using a site-specific hazard analysis performed for a particular site (e.g., Seabrook) or using the LLNL and EPRI hazard analyses are examples of Capability Category II. The Diablo Canyon study [5-26] and the Yucca Mountain study [5-27] represent Capability Category III seismic hazard studies.

The detailed description of these four levels is contained in the SSHAC report [5-22]. While basic constituent elements of a PSHA are the same in all applications, the SSHAC levels are roughly in order of increasing resources and sophistication. It is important, ultimately, to show that the PSHA characterization is robust for the intended application and accounts for the uncertainties.

There are 10 high-level requirements for PSHA, as follows:

Table 5-2.1-1 High Level Requirements for Seismic Probabilistic Risk Assessment: Technical Requirements for Probabilistic Seismic Hazard Analysis (SHA)

Designator	Requirement
HLR-SHA-A	The frequency of seismic ground motion at the site shall be based on a site-specific probabilistic seismic hazard analysis (existing or new) that reflects the composite distribution of the informed technical community. The level of analysis shall be determined based on the intended application and on site-specific complexity.
HLR-SHA-B	To provide inputs to the probabilistic seismic hazard analysis, comprehensive up-to-date data shall be compiled that include geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties. A catalog of historical, instrumental, and paleoseismicity information shall also be compiled.
HLR-SHA-C	To account for the frequency of occurrence of seismic ground motions in the site region, the probabilistic seismic hazard analysis shall examine all credible sources of potentially damaging earthquakes.
HLR-SHA-D	The probabilistic seismic hazard analysis shall examine mechanisms influencing vibratory ground motion that can occur at a site given the occurrence of an earthquake of a certain type (e.g., strike slip, normal, reverse) and magnitude, and at a certain location. Uncertainties shall be addressed characterizing the ground motion propagation.
HLR-SHA-E	The probabilistic seismic hazard analysis shall account for the effects of local site response.
HLR-SHA-F	Uncertainties in each step of the hazard analysis shall be propagated and displayed in the final quantification of hazard estimates for the site.
HLR-SHA-G	For further use in the seismic PRA, the spectral shape shall be based on a site-specific evaluation taking into account results of the probabilistic seismic hazard analysis.
HLR-SHA-H	When use is made of an existing study for probabilistic seismic hazard analysis purposes, it shall be confirmed that the basic data and interpretations are still valid in light of current information.
HLR-SHA-I	A screening analysis shall be performed to assess whether in addition to the vibratory ground motion, other seismic hazards, such as fault displacement, landslide, soil liquefaction, or soil settlement, need to be included in the seismic PRA.
HLR-SHA-J	Documentation of the probabilistic seismic hazard analysis shall be consistent with the applicable supporting requirements.

Table 5-2.1-2 Supporting Requirements for HLR-SHA-A

The frequency of seismic ground motion at the site shall be based on a site-specific probabilistic seismic hazard analysis (existing or new) that reflects the composite distribution of the informed technical community. The level of analysis shall be determined based on the intended application and on site-specific complexity (HLR-SHA-A).

<table>
<tr><th>Index No. SHA-A</th><th>Capability Category I</th><th>Capability Category II</th><th>Capability Category III</th></tr>
<tr><td>SHA-A1 [Note (1)]</td><td>In performing the probabilistic seismic hazard analysis (PSHA), BASE it on, and MAKE it consist of, the collection and evaluation of available information and data, consideration of the uncertainties in each element of the PSHA, and a defined process and documentation to make the PSHA traceable.</td><td colspan="2">In performing the probabilistic seismic hazard analysis (PSHA), BASE it on, and MAKE it consist of, the collection and evaluation of available information and data, evaluation of the uncertainties in each element of the PSHA, and a defined process and documentation to make the PSHA traceable.</td></tr>
<tr><td>SHA-A2</td><td colspan="3">As the parameter to characterize both hazard and fragilities, USE the spectral accelerations, or the average spectral acceleration over a selected band of frequencies, or peak ground acceleration.</td></tr>
<tr><td>SHA-A3</td><td colspan="3">If spectral acceleration or average spectral acceleration over a band of frequencies is used, INCLUDE the response frequencies of SSCs that are significant in the PRA results and insights.</td></tr>
<tr><td>SHA-A4 [Note (2)]</td><td colspan="3">In developing the probabilistic seismic hazard analysis results for use in accident sequence quantification, whether they are characterized by spectral accelerations, peak ground accelerations, or both, EXTEND them to large-enough values (consistent with the physical data and interpretations) so that the truncation does not produce unstable final numerical results, such as core damage frequency, and the delineation and ranking of seismic-initiated sequences are not affected.</td></tr>
<tr><td>SHA-A5 [Note (3)]</td><td colspan="3">SPECIFY a lower-bound magnitude (or probabilistically defined characterization of magnitudes based on a damage parameter) for use in the hazard analysis, such that earthquakes of magnitude less than this value are not expected to cause significant damage to the engineered structures or equipment.</td></tr>
</table>

GENERAL NOTE: The need for determining the composite distribution is discussed in reference [5-22].

NOTES:

(1) The guidance and process given in reference [5-22] address the above requirement and may be used as an acceptable methodology. In general, Levels 1 and 2 of this reference correspond to Capability Category I; Levels 2 and 3 to Capability Category II; and Levels 3 and 4 to Capability Category III. The distinction between the consideration of uncertainties (for Capability Category I) and the evaluation of them (Capability Categories II and III) is important. The latter means a numerical evaluation.

The Regulatory Guide 1.208 [5-28] published in March 2007 provides guidance for conducting the probabilistic seismic hazard analysis for the Early Site Permit and Combined License applicants. This reference also provides guidance on the use of the existing information as acceptable starting points for the evaluations and performing updates as necessary. Although the scope of this Standard covers only PRAs for operating plants, the guidance for new plants is available in RG 1.208.

Table 5-2.1-2 Supporting Requirements for HLR-SHA-A (Cont'd)

NOTES: (Cont'd)

(2) It is necessary to make sure that the hazard estimation used in the accident sequence quantification is carried out to large-enough values (consistent with the physical data and interpretations) so that when combined with the plant or component level fragility, the resulting failure frequencies are robust estimates and do not change if the acceleration range is extended. A sensitivity study can be conducted to define the upper-bound value. NUREG-1407 [5-7] provides the additional guidance. Another factor that may dictate how far the hazard should extend is the required resolution in the final results. For example, if the interest is in computing the likelihood of component or plant failure down to 1E-7/yr, this can be achieved with the hazard characterization down to 1E-8/yr exceedance. Peer reviews need to be attentive to this aspect.

(3) The value of the lower-bound magnitude used in analyzing the site-specific hazard is based on engineering considerations [5-25]. Based on the evaluation of earthquake experience data, earthquakes with moment magnitudes less than 5.0 are not expected to cause damage to safety-related SSCs. A lower-bound magnitude value of 5.0 was used for both the Lawrence Livermore National Laboratory and Electric Power Research Institute studies. The latest research in this area recommends using a probabilistically defined characterization of what magnitudes are expected to cause damage based on the Cumulative Absolute Velocity (CAV) parameter. Reference [5-40] provides additional guidance on the use of the CAV parameter. Note that this consideration of lower-bound cutoffs applies only to the magnitude range considered in the final hazard quantification, not to the characterization and determination of seismicity parameters for the sources, for which lower-magnitude earthquakes have to be included.

Table 5-2.1-3 Supporting Requirements for HLR-SHA-B

To provide inputs to the probabilistic seismic hazard analysis, comprehensive up-to-date data shall be compiled that include geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties. A catalog of historical, instrumental, and paleoseismicity information shall also be compiled (HLR-SHA-B).

Index No. SHA-B	Capability Category I	Capability Category II	Capability Category III
SHA-B1 [Note (1)]	In performing the probabilistic seismic hazard analysis (PSHA), USE available or developed geological, seismological, geophysical, and geotechnical data that reflect the current state of the knowledge and that are used by experts/analysts to develop interpretations and inputs to the PSHA.		In performing the probabilistic seismic hazard analysis (PSHA), USE available and developed comprehensive geological, seismological, geophysical, and geotechnical data that reflect the current state of the knowledge and that are used by experts/analysts to develop interpretations and inputs to the PSHA. INCLUDE site-specific laboratory data for site soils, including their potential uncertainty, to characterize local site response effects.
SHA-B2 [Note (2)]	ENSURE that the database and information used are adequate to characterize all credible seismic sources that may contribute significantly to the frequency of occurrence of vibratory ground motion at the site, considering regional attenuation of ground motions and local site effects. If the existing probabilistic seismic hazard analysis (PSHA) studies are to be used in the seismic PRA, ENSURE that any new data or interpretations that could affect the PSHA are adequately incorporated in the existing data and analysis.		ENSURE that the size of the region to be investigated and the scope of investigations is adequate to characterize all credible seismic sources that may contribute significantly to the frequency of occurrence of vibratory ground motion at a site, considering regional attenuation of ground motions and local site effects. If the existing probabilistic seismic hazard analysis studies are to be used in the seismic PRA, ENSURE that the investigations are of sufficient scope to determine whether there are new data or interpretations that are not adequately incorporated in the existing data and analysis.

Table 5-2.1-3 Supporting Requirements for HLR-SHA-B (Cont'd)

To provide inputs to the probabilistic seismic hazard analysis, comprehensive up-to-date data shall be compiled that include geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties. A catalog of historical, instrumental, and paleoseismicity information shall also be compiled (HLR-SHA-B).

Index No. SHA-A	Capability Category I	Capability Category II	Capability Category III
SHA-B3 [Note (3)]	INCLUDE an appropriate existing catalog of historically reported earthquakes, instrumentally recorded earthquakes, and earthquakes reported through geological investigations. USE reference [5-30] requirements or equivalent.		As a part of data collection, COMPILE a catalog (or AUGMENT an existing catalog) of historically reported earthquakes, instrumentally recorded earthquakes, and earthquakes reported through geological investigations. USE reference [5-30] requirements or equivalent.

NOTES:

(1) It is important that a comprehensive compilation of data be shared and used by all experts in developing the interpretations. The availability of the database also facilitates the review process. References [5-28] and [5-29] give acceptable guidance on the scope and types of data required for use in the seismic source characterization, ground motion modeling, and local site response evaluations to meet this requirement.

(2) Reference [5-28] defines four levels of investigations, with the degree of their detail based on distance from the site, the nature of the Quaternary tectonic regime, the geological complexity of the site and region, the existence of potential seismic sources, the nature of sources, the potential for surface deformation, etc. This guidance can be used to determine scope and size of region for investigations. The guidance in reference [5-30] may be used to meet this requirement.

One definition of significant contribution used in the past has been that all modeled sources represent at least 99% of the hazard (annual frequency) at the amplitude of interest for 10 Hz and 1 Hz over the range of the ground motion parameter that contributes importantly to the hazard.

(3) In general, the catalog typically includes events of size modified Mercalli intensity (MMI) or equivalent greater than or equal to IV and magnitude greater than or equal to 3.0 that have occurred within a radius of 320 km of a site [5-30]. For the earthquakes listed, the catalog typically contains information such as event date and time, epicentral location, earthquake magnitudes (measured and calculated, possibly several earthquake magnitude metrics, e.g., M_b, M_L, and M_w), magnitude uncertainty, uncertainty in the event location, epicentral intensity, intensity uncertainty, hypocentral depth, references, and data sources.

Table 5-2.1-4 Supporting Requirements for HLR-SHA-C

To account for the frequency of occurrence of seismic ground motions in the site region, the probabilistic seismic hazard analysis shall examine all credible sources of potentially damaging earthquakes (HLR-SHA-C).

Index No. SHA-C	Capability Category I	Capability Category II	Capability Category III
SHA-C1 [Note (1)]	In the probabilistic seismic hazard analysis, EVALUATE sources of earthquakes that have the potential to contribute significantly to the probabilistic hazard at the site. IDENTIFY and CHARACTERIZE seismic sources taking into account previous compilations of seismic sources, based on regional and site geological and geophysical data, historical and instrumental seismicity data, and geological evidence of prehistoric earthquakes.		
SHA-C2 [Note (2)]	ENSURE that any expert elicitation process used to characterize the seismic sources is compatible with the level of analysis discussed in Requirement HLR-SHA-A, and USE a structured approach.		
SHA-C3 [Note (3)]	The seismic sources are characterized by alternative source representation and source geometry, maximum earthquake magnitude, and earthquake recurrence. ENSURE that the total uncertainties in these characterizations are accounted for.	The seismic sources are characterized by alternative source representation and source geometry, maximum earthquake magnitude, and earthquake recurrence. INCLUDE the aleatory and epistemic uncertainties explicitly in these characterizations, where significant.	
SHA-C4 [Note (4)]	If an existing seismic source model is used, DEMONSTRATE that any new seismic sources that have been identified or were uncharacterized when the existing models were developed are not significant, or INCLUDE them in the update of the hazard estimates.		

NOTES:
(1) A useful reference is Regulatory Guide 1.208 [5-28].
(2) Guidance given in reference [5-22], which provides a structured approach, is one acceptable way to meet this requirement. See 1-4.3 for further discussion of the use of experts.
(3) Although the explicit display of the uncertainties or the distinction between aleatory or epistemic uncertainties (see Part 2, "Definitions," and Nonmandatory Appendix B of this Part for brief explanations of these terms) in the final results may not always be necessary, it is essential in the probabilistic seismic hazard analysis to characterize the uncertainties properly so as to make the process transparent and results interpretable. Uncertainties in the hazard estimates typically contribute most to the overall uncertainties in the final seismic-PRA results, and it is therefore crucial to understand the sources and nature of these uncertainties in making decisions. Reference [5-22] gives detailed discussion and acceptable guidance on a process to be used for determination and quantification of uncertainties to meet this requirement. For Capability Category I, it is not necessary to decompose the total or composite uncertainties into different components. However, the total uncertainty needs to be accounted for (e.g., see NUREG-1407 [5-7]).
(4) Reference [5-28] is one acceptable method. It gives detailed guidance on how to assess the significance of new information, including new interpretations.

Table 5-2.1-5 Supporting Requirements for HLR-SHA-D

The probabilistic seismic hazard analysis shall examine mechanisms influencing vibratory ground motion that can occur at a site given the occurrence of an earthquake of a certain type (e.g., strike slip, normal, reverse) and magnitude, and at a certain location. Uncertainties shall be addressed in characterizing the ground motion propagation (HLR-SHA-D).

Index No. SHA-D	Capability Category I	Capability Category II	Capability Category III
SHA-D1 [Note (1)]	In the vibratory ground motion analysis, INCLUDE *(a)* credible mechanisms governing estimates of vibratory ground motion that can occur at a site *(b)* available historical and instrumental seismicity data (including strong motion data) *(c)* current attenuation models for the ground motion estimates		
SHA-D2 [Note (2)]	ENSURE that any expert elicitation process used to characterize the ground motion or any other elements of the ground motion analysis is compatible with the level of analysis discussed in Requirement HLR-SHA-A, and USE a structured approach.		
SHA-D3 [Note (3)]	ENSURE that the uncertainties that contribute most in the ground motion characterization are accounted for.	INCLUDE both the aleatory and epistemic uncertainties separately in the ground motion characterization in accordance with the level of analysis identified for Requirement HLR-SHA-A.	
SHA-D4 [Note (4)]	If existing ground motion models are used, DEMONSTRATE that new information not previously used or which was unknown when the existing models were developed would not significantly affect the PSHA results, or INCLUDE it in the update of the hazard estimates.		

NOTES:
(1) It is important to note that in the guideline documents [5-2, 5-22, 5-28], the probabilistic seismic hazard estimates are first performed for the real or assumed rock conditions in the free field. For the nonrock sites, the site-specific estimates are performed, taking into account the local site conditions and properties including aleatory and epistemic uncertainties as discussed under Requirement HLR-SHA-E. Further discussion on this issue can be found in reference [5-31].
(2) The structured approach given in reference [5-22] is one acceptable way to meet this requirement. See 1-4.3 for further discussion of the use of experts.
(3) The characterization of ground motion includes the equation (attenuation relationship) that predicts the median level of ground motion parameter of engineering interest (spectral acceleration, displacements, peak ground acceleration, etc.) as a function of magnitude and distance; an estimate of the aleatory variability in ground motion, which quantifies the unexplained scatter in ground motion and the event-to-event variability of earthquakes of the same magnitude; and an estimate of the epistemic uncertainty taking into account the possible existence of several different applicable ground motion models. As discussed in Requirement SHA-D3, it is necessary to characterize properly the uncertainties in the hazard estimates. Reference [5-22] gives guidance on an acceptable process to be used for determination and quantification of uncertainties, including the distinction between aleatory and epistemic uncertainties.
(4) Reference [5-28] gives detailed guidance on how to assess the significance of the new information, including new interpretations.

Table 5-2.1-6 Supporting Requirements for HLR-SHA-E

The probabilistic seismic hazard analysis shall account for the effects of local site response (HLR-SHA-E).

Index No. SHA-E	Capability Category I	Capability Category II	Capability Category III
SHA-E1 [Note (1)]	DEMONSTRATE that the probabilistic seismic hazard analysis accounts for the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site.	In the probabilistic seismic hazard analysis, INCLUDE the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site.	
SHA-E2 [Note (2)]	ENSURE that the uncertainties that contribute most to the ground motion characterization are accounted for.	INCLUDE both the aleatory and epistemic uncertainties in the local site response analysis.	

NOTES:

(1) The purpose of a local site response analysis is to quantify the influence of surficial geologic conditions on site ground motions. Two approaches are generally used to account for surficial conditions at a site as part of the estimation of ground motion. The first is to utilize ground motion attenuation relationships appropriate for the site conditions (i.e., relationships that have been developed for the type of subsurface conditions that exist at a site). The second is to develop site-specific transfer functions that can be used to modify the rock ground motions for the site characteristic [5-31]. The existing probabilistic seismic hazard analysis studies should be shown to account for the local site effects or should be revised. Probabilistic estimates of site properties should be used in determining the site-specific functions.

(2) Consistent with the source characterization and ground motion estimates, it is essential that the uncertainties are properly characterized and propagated in this step. Reference [5-22] gives guidance on an acceptable process to be used for determination and quantification of uncertainties, including the distinction between aleatory and epistemic uncertainties.

Table 5-2.1-7 Supporting Requirements for HLR-SHA-F

Uncertainties in each step of the hazard analysis shall be propagated and displayed in the final quantification of hazard estimates for the site (HLR-SHA-F).

Index No. SHA-F	Capability Category I	Capability Category II	Capability Category III
SHA-F1 [Note (1)]	In the final quantification of the seismic hazard, USE the mean estimate.	In the final quantification of the seismic hazard, INCLUDE uncertainties through a family of hazard curves.	
SHA-F2 [Note (2)]	In the probabilistic seismic hazard analysis, INCLUDE appropriate sensitivity studies and intermediate results to identify factors that are important to the site hazard and that make the analysis traceable.		
SHA-F3 [Note (3)]	CALCULATE the following results as a part of the hazard quantification process, compatible with needs for the level of analysis determined in Requirement HLR-SHA-A: *(a)* mean hazard curves for peak ground acceleration and spectral accelerations *(b)* mean uniform hazard response spectrum	CALCULATE the following results as a part of the hazard quantification process, compatible with needs for the level of analysis determined in Requirement HLR-SHA-A: *(a)* fractile and mean hazard curves for each ground motion parameter considered in the probabilistic seismic hazard analysis *(b)* uniform hazard response spectra	CALCULATE the following results as a part of the hazard quantification process, compatible with needs for the level of analysis determined in Requirement HLR-SHA-A: *(a)* fractile and mean hazard curves for each ground motion parameter considered in the probabilistic seismic hazard analysis *(b)* uniform hazard response spectra *(c)* magnitude-distance deaggregation for the median and mean hazard *(d)* seismic source deaggregation *(e)* mean magnitude and distance

NOTES:

(1) The seismic hazard quantification involves the combination of seismic source and ground motion inputs to compute the frequency of exceedance of ground motions at a site (i.e., the seismic hazard curve). Thus, the principal result of the probabilistic seismic hazard analysis is a set of seismic hazard curves, each curve quantifying the aleatory uncertainty and the set of curves representing the epistemic uncertainties in the site hazard. This is typically presented in terms of a set of fractile seismic hazard curves and a mean hazard curve. Two acceptable approaches have been used to propagate epistemic uncertainties: logic tree enumeration and Monte Carlo simulation [5-25, 5-32]. For Capability Category I, use of a single mean hazard curve may be appropriate.

(2) Sensitivity studies and intermediate results provide important information to reviewers about how some of the key assumptions affect the final results of this complex seismic hazard process. Examples of useful sensitivity studies include an evaluation of alternate schemes used to assign weights to the individual expert models and an evaluation of the way different experts make different assignments of the regional seismicity to different zonation maps.

Table 5-2.1-7 Supporting Requirements for HLR-SHA-F (Cont'd)

NOTES: (Cont'd)
(3) The magnitude-distance deaggregation and seismic source deaggregation [5-33] are useful when the application of the seismic PRA depends on the quantitative results and full understanding of sources of uncertainties is essential. These aspects become important when relative comparisons are to be made among risks resulting from different earthquake magnitudes. The magnitude-distance deaggregation helps in identifying the earthquake events (magnitude and distance) that contribute most to the hazard, particularly for some regions of the eastern United States where it is difficult to identify seismic faults associated with the observed seismicity. This in turn allows the analyst to characterize the nature of ground motion properly for use in the response and fragility analyses.

Fractile curves are generally plotted for the 5, 16, 50, 84, and 95 percentiles.

The uniform hazard response spectrum provides hazard information for spectral accelerations at several discrete frequencies for one or more probabilities of exceedance.

Table 5-2.1-8 Supporting Requirements for HLR-SHA-G

For further use in the seismic PRA, the spectral shape shall be based on a site-specific evaluation taking into account results of the probabilistic seismic hazard analysis (HLR-SHA-G).

Index No. SHA-G	Capability Category I	Capability Category II	Capability Category III
SHA-G1 [Note (1)]	ENSURE that the spectral shape used in the seismic PRA uses or bounds the site-specific considerations.	ENSURE that the spectral shape used in the seismic PRA uses site-specific evaluations performed for the PSHA.	ENSURE that the response spectral shape (horizontal and vertical) used on the seismic PRA uses site-specific evaluations performed for the PSHA, and uses or bounds the characteristic spectral shapes associated with the mean magnitude and distance pairs determined in the PSHA for the important ground motion levels.

NOTE:
(1) The issue of which spectral shape should be used in the screening of SSCs and in quantification of seismic-PRA results requires careful consideration. For screening purposes, the spectral shape used should have amplification factors such that the demand resulting from the use of this shape is higher than that based on the design spectra. This will preclude premature screening of components and will avoid anomalies such as the screened components (e.g., surrogate elements) being the dominant risk-contributing components. Additional discussion on this issue can be found in reference [5-12].

In the quantification of fragilities and of final risk results, it is important to use as realistic a shape as possible. For rock sites, NUREG/CR-6728 [5-31] is one source. For soil sites, site amplification studies will define site spectral shapes over a wide range of frequencies. The uniform hazard response spectral (UHS) shape is acceptable for screening unless evidence comes to light (e.g., within the technical literature) that this shape does not reflect the spectral shape of the site-specific events.

Table 5-2.1-9 Supporting Requirements for HLR-SHA-H

When use is made of an existing study for probabilistic seismic hazard analysis purposes, it shall be confirmed that the basic data and interpretations are still valid in light of established current information (HLR-SHA-H).

Index No. SHA-H	Capability Category I	Capability Category II	Capability Category III
SHA-H1 [Note (1)]	CONFIRM that the basic data and interpretations for any existing studies used remain valid in light of established current information, consistent with the Requirements HLR-SHA-A through HLR-SHA-G, and DESCRIBE the bases and methodology used.		

NOTE:
(1) When using the existing studies, the intent of this requirement is not to repeat the entire hazard exercise or calculations, unless new information and interpretations prepared to a comparable technical level have been established that may affect the usefulness of the seismic PRA for the intended application. Depending upon the application, sensitivity studies, modest extensions of the existing analysis, or approximate estimates of the differences between using an existing hazard study and applying the newer one may be sufficient. Additionally, an assessment may be sufficient to demonstrate that the impact on the application of information or data that is less extensive than a new hazard study is not significant.

Table 5-2.1-10 Supporting Requirements for HLR-SHA-I

A screening analysis shall be performed to assess whether, in addition to the vibratory ground motion, other seismic hazards, such as fault displacement, landslide, soil liquefaction, or soil settlement, need to be included in the seismic PRA (HLR-SHA-I).

Index No. SHA-I	Capability Category I	Capability Category II	Capability Category III
SHA-I1 [Note (1)]	DOCUMENT the bases and methodology used for any screening out of the seismic hazards other than vibratory ground motion.		
SHA-I2 [Note (2)]	For those hazards not screened out, INCLUDE their effect through assessment of the frequency of hazard occurrence and the magnitude of hazard consequences.		

NOTES:
(1) It is expected that only a few sites will require consideration of other seismic hazards considered in this requirement. The best guidance is available in a few case studies that needed to address some of the above hazards and original investigations conducted in support of site selection.
(2) An example of probabilistic fault displacement hazard analysis methodology is contained in reference [5-44].

Table 5-2.1-11 Supporting Requirements for HLR-SHA-J

Documentation of the probabilistic seismic hazard analysis shall be consistent with the applicable supporting requirements (HLR-SHA-J).

Index No. SHA-J	Capability Category I	Capability Category II	Capability Category III
SHA-J1	DOCUMENT the probabilistic seismic hazard analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
SHA-J2	DOCUMENT the process used in the probabilistic seismic hazard analysis. For example, this documentation is typically consistent with reference [5-28] and includes a description of *(a)* the specific methods used for source characterization and ground motion characterization *(b)* the scientific interpretations that are the basis for the inputs and results, and *(c)* if an existing PSHA is used, documentation to ensure that it is adequate to meet the spirit of the requirements herein		
SHA-J3	DOCUMENT the sources of model uncertainty and related assumptions associated with the probabilistic seismic hazard analysis.		

5-2.2 SEISMIC-FRAGILITY ANALYSIS

The seismic fragility of an SSC is defined as the conditional probability of its failure at a given value of seismic motion parameter (e.g., PGA, peak spectral acceleration at different frequencies, or floor spectral acceleration at the equipment frequency). The methodology for evaluating seismic fragilities of SSCs is documented in the PRA Procedures Guide [5-6] and is more specifically described for application to nuclear power plants in references [5-10] and [5-37]. Nonmandatory Appendix 5-A provides a brief description of how seismic-fragility curves are developed for any SSC. Seismic fragilities used in a seismic PRA should be realistic and plant-specific based on actual conditions of the SSCs in the plant, as confirmed through a detailed walkdown of the plant. Seismic-fragility evaluation has been conducted for more than 40 nuclear power plants in the U.S. and other countries. Based on the experience and insights gained in these studies, certain methodological improvements and simplifications have been proposed in reference [5-12].

The primary objective of the seismic-fragility analysis is to estimate the seismic fragilities of SSCs whose failure may contribute to core damage or large early release, or both. The fragility characterization can be based on generic or plant-specific information as required to meet the capability category targeted. If it can be demonstrated that an SSC capacity is high enough (i.e., very low fragility), it may be screened out from further consideration for the assessment of seismic risk to the plant.

Note that in performing a seismic PRA, the seismic-fragility evaluation is performed before the integration and quantification that are the subjects of Requirement HLR-SPR-E. Thus, the order of the requirements herein is different from the order in which the analysis work must be performed.

There are seven high-level requirements under "Seismic-Fragility Analysis," as described in Table 5-2.2-1.

Table 5-2.2-1 High Level Requirements for Seismic Probabilistic Risk Assessment: Technical Requirements for Seismic-Fragility Analysis (SFR)

Designator	Requirement
HLR-SFR-A	The seismic-fragility evaluation shall be performed to estimate seismic fragilities of SSCs whose failure may contribute to core damage or large early release, or both.
HLR-SFR-B	If screening of high-seismic-capacity components is performed, the basis for the screening shall be fully described.
HLR-SFR-C	The seismic-fragility evaluation shall be based on a seismic response that the SSCs experience at their failure levels.
HLR-SFR-D	The seismic-fragility evaluation shall be performed for critical failure modes of SSCs such as structural failure modes and functional failure modes identified through the review of plant design documents, supplemented as needed by earthquake experience data, fragility test data, generic qualification test data, and a walkdown.
HLR-SFR-E	The seismic-fragility evaluation shall incorporate the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support, and potential systems interactions.
HLR-SFR-F	The calculation of seismic-fragility parameters such as median capacity and variabilities shall be based on plant-specific data or, if necessary, on earthquake experience data, fragility test data, and generic qualification test data. Use of such generic data shall be justified.
HLR-SFR-G	Documentation of the seismic-fragility evaluation shall be consistent with the applicable supporting requirements.

Table 5-2.2-2 Supporting Requirements for HLR-SFR-A

The seismic-fragility evaluation shall be performed to estimate seismic fragilities of SSCs whose failure may contribute to core damage or large early release, or both (HLR-SFR-A).

Index No. SFR-A	Capability Category I	Capability Category II	Capability Category III
SFR-A1 [Note (1)]	CALCULATE seismic fragilities for SSCs identified by the systems analysis (see Requirement SPR-D1).		
SFR-A2 [Note (2)]	Generic data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data) may be used to develop seismic fragilities. However, DEMONSTRATE that any use of such generic data is applicable.	CALCULATE the seismic fragilities based on plant-specific data, and ENSURE that they are realistic (median with uncertainties). Generic data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data) may be used for screening of certain SSCs and for calculating their seismic fragilities by applying the Requirement HLR-SFR-F, which permits use of such generic data under specified conditions. However, DEMONSTRATE that any use of such generic data is applicable.	CALCULATE the seismic fragilities based on plant-specific data, and ENSURE that they are realistic (median with uncertainties).

NOTES:
(1) Seismic fragilities are needed for SSCs identified by the systems analysis that are modeled in the event trees and fault trees. Failure of one or more of these may contribute to core damage or large early release, or both. Requirements for developing this list of SSCs are given under the Systems Analysis section (see Requirement SPR-D1). See also the Requirement HLR-SFR-B on screening.
(2) The objective of a seismic PRA is to obtain a realistic seismic risk profile for the plant using plant-specific and site-specific data. It has been demonstrated in several seismic PRAs that the risk estimates and insights on seismic vulnerabilities are very plant specific, even varying between supposedly identical units at a multiunit plant. To minimize the effort on nonsignificant items and to focus the resources on the more critical aspects of the seismic PRA, certain high-seismic-capacity components are screened out using generic data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data). It is important to be conservative in the use of such generic data.

Table 5-2.2-3 Supporting Requirements for HLR-SFR-B

If screening of high-seismic-capacity components is performed, the basis for the screening shall be fully described (HLR-SFR-B).

Index No. SFR-B	Capability Category I	Capability Category II	Capability Category III
SFR-B1 [Note (1)]	If screening of high-seismic-capacity components is performed, DESCRIBE the basis for screening and the supporting documents and SELECT the screening level high enough that the contribution to core damage frequency and large early release frequency from the screened-out components is not significant.		SCREEN OUT high-seismic-capacity components only if the components' failures can be considered as fully independent of the remaining components.

NOTES:
(1) When screening of high-seismic-capacity components is performed, the basis for screening and supporting documents is to be fully described. Guidance given in EPRI NP-6041-SL, Rev. 1 [5-3] and NUREG/CR-4334 [5-4] may be used to screen out high-seismic-capacity components after satisfying the caveats. Note that the screening guidance in these documents has been developed generally for U.S.-vendored equipment and based on U.S. seismic design practice. Care should be used in applying the screening criteria for other situations. The use of generic fragility information is acceptable for screening if the SSCs can be shown to fall within the envelope of the generic fragility caveats.
The screening level chosen should be based on the seismic hazard at the site and on the plant seismic design basis and should be high enough that the contribution to core damage frequency and large early release frequency from the screened-out components is not significant. (See Requirement SHA-G1.) For a discussion of possible approaches to the selection of the screening level, the reader is referred to reference [5-10].

Table 5-2.2-4 Supporting Requirements for HLR-SFR-C

The seismic-fragility evaluation shall be based on seismic response that the SSCs experience at their failure levels (HLR-SFR-C).

Index No. SFR-C	Capability Category I	Capability Category II	Capability Category III
SFR-C1 [Note (1)]	ESTIMATE the seismic responses that the components experience at their failure levels using input earthquake response spectra in three orthogonal directions, anchored to a ground motion parameter such as peak ground acceleration or average spectral acceleration over a given frequency band, and ENSURE that the spectral shape used bounds the site-specific conditions.		ESTIMATE the seismic responses that the components experience at their failure levels on a realistic basis using site-specific earthquake response spectra in three orthogonal directions, anchored to a ground motion parameter such as peak ground acceleration or average spectral acceleration over a given frequency band.

Table 5-2.2-4 Supporting Requirements for HLR-SFR-C (Cont'd)

The seismic-fragility evaluation shall be based on seismic response that the SSCs experience at their failure levels (HLR-SFR-C).

Index No. SFR-C	Capability Category I	Capability Category II	Capability Category III
SFR-C2 [Note (2)]	If probabilistic response analysis is performed to obtain structural loads and floor response spectra, ENSURE that the number of simulations done (e.g., Monte Carlo simulation and Latin Hypercube Sampling) is large enough to obtain stable median and 85% nonexceedance responses. INCLUDE the entire spectrum of input ground motion levels displayed in the seismic hazard curves.		PERFORM probabilistic seismic response analysis taking into account the uncertainties in the input ground motion and site soil properties, and structural parameters, and ESTIMATE joint probability distributions of the responses of different components in the building.
SFR-C3 [Note (3)]	If scaling of existing response analysis is used, JUSTIFY it based on the adequacy of structural models, foundation characteristics, and similarity of input ground motion.		Addressed in Requirement SFR-C2
SFR-C4	When the existing response analysis models are judged not to be realistic and state of the art, or when the existing input ground motion is significantly different from the site-specific input motion, PERFORM new analysis to obtain realistic structural loads and floor response spectra for use in the seismic PRA.		Addressed in Requirement SFR-C2
SFR-C5 [Note (4)]	If median-centered response analysis is performed, ESTIMATE the median response (i.e., structural loads and floor response spectra) and variability in the response using established methods.		Addressed in Requirement SFR-C2
SFR-C6 [Note (5)]	When soil-structure interaction (SSI) analysis is conducted, ENSURE that it is median centered using median properties, at soil strain levels corresponding to the input ground motions that contribute most to the seismically induced core damage frequency. INCLUDE the uncertainties in the SSI analysis.		Addressed in Requirement SFR-C2

NOTES:

(1) NUREG-1407 [5-7] recommends the use of 10,000-yr return period UHS median spectral shapes provided in reference [5-32] along with variability estimates that reflect the site-specific shapes as discussed in Note (1) of Table 5-2.1-8. Any UHS should be used cautiously to ensure that the spectral shape reflects the contributions from dominating events as discussed under Requirement SHA-G1. See Note (1) of Table 5-2.1-8 for further discussion on this topic.

Table 5-2.2-4 Supporting Requirements for HLR-SFR-C (Cont'd)

NOTES: (Cont'd)
(2) For a description of the probabilistic seismic response analysis, the reader is referred to references [5-38] and [5-31].
(3) The scaling procedures given in reference [5-3] may be used. Scaling of responses from existing analysis is not permitted for Capability Category III.
(4) Reference [5-10] gives an acceptable method.
(5) Further details about the basis of this requirement can be found in reference [5-15].

Table 5-2.2-5 Supporting Requirements for HLR-SFR-D

The seismic-fragility evaluation shall be performed for critical failure modes of SSCs such as structural failure modes and functional failure modes identified through the review of plant design documents, supplemented as needed by earthquake experience data, fragility test data, generic qualification test data, and a walkdown (HLR-SFR-D).

Index No. SFR-D	Capability Category I	Capability Category II	Capability Category III
SFR-D1 [Note (1)]	IDENTIFY realistic failure modes of structures (e.g., sliding, overturning, yielding, and excessive drift), equipment (e.g., anchorage failure, impact with adjacent equipment or structures, bracing failure, and functional failure), and soil (e.g., liquefaction, slope instability, and excessive differential settlement) that interfere with the operability of equipment during or after the earthquake, through a review of the plant design documents and the walkdown.		
SFR-D2 [Note (2)]	EVALUATE all relevant failure modes identified in Requirement SFR-D1, and EVALUATE fragilities for critical failure modes.		

NOTES:
(1) Note that sometimes failure modes such as drift and yielding may be more relevant for the functionality of attached equipment than gross structural failures (e.g., partial collapse or complete collapse).
(2) Published references and past seismic PRAs could be used as guidance. Examples include references [5-3], [5-10], and [5-26].

Table 5-2.2-6 Supporting Requirements for HLR-SFR-E

The seismic-fragility evaluation shall incorporate the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support, and potential systems interactions (HLR-SFR-E).

Index No. SFR-E	Capability Category I	Capability Category II	Capability Category III
SFR-E1 [Note (1)]	CONDUCT a detailed walkdown of the plant, focusing on equipment anchorage, lateral seismic support, spatial interactions, and potential systems interactions (both structural and functional interactions).		
SFR-E2	DOCUMENT the walkdown procedures, walkdown team composition and its members' qualifications, walkdown observations, and conclusions.		
SFR-E3	If components are screened out during or following the walkdown, DOCUMENT the basis, including any anchorage calculations that justify such a screening.		
SFR-E4 [Note (2)]	During the walkdown, EVALUATE the potential for seismically induced fire and flooding by focusing on the issues described in NUREG-1407 [5-7].		
SFR-E5 [Note (3)]	During the walkdown, EVALUATE potential sources of interaction (e.g., II/I issues, impact between cabinets, masonry walls, flammable and combustion sources, flooding, and spray) and consequences of such interactions on equipment contained in the systems model.		

NOTES:

(1) The seismic walkdown is an important activity in the seismic PRA. The purposes of such a walkdown are to find as-designed, as-built, and as-operated seismic weaknesses in the plant and to ensure that the seismic fragilities are realistic and plant specific. It should be done in sufficient detail and documented in a sufficiently complete fashion so that the subsequent screening or fragility evaluation is traceable. For guidance on walkdowns, the analyst is referred to references [5-3] and [5-4]. (See Requirement SPR-B9.)

(2) Seismically induced fires and floods are to be addressed as described in NUREG-1407 [5-7]. The effects of seismically induced fires and impact of inadvertent actuation of fire protection systems on safety systems should be assessed. The effects of seismically induced external flooding and internal flooding on plant safety should be included. The scope of the evaluation of seismically induced flood, in addition to that of the external sources of water (e.g., tanks and upstream dams), should include the evaluation of some internal flooding that is consistent with the discussion in Appendix F of EPRI NP-6041-SL, Rev. 1 [5-3].

(3) A "II/I issue" refers to situations where a nonseismically qualified object could fall on and damage a seismically qualified item of safety equipment, and also situations where a low seismic capacity object falls on and damages an SSC item with higher seismic capacity. In such cases, the fragility of the higher capacity SSC may be controlled by the low capacity object.

Table 5-2.2-7 Supporting Requirements for HLR-SFR-F

The calculation of seismic-fragility parameters such as median capacity and variabilities shall be based on plant-specific data or, if necessary, on earthquake experience data, fragility test data, and generic qualification test data. Use of such generic data shall be justified (HLR-SFR-F).

Index No. SFR-F	Capability Category I	Capability Category II	Capability Category III
SFR-F1 [Note (1)]	CALCULATE component seismic-fragility parameters such as median capacity and variabilities (logarithmic standard deviations reflecting randomness and uncertainty) based on plant-specific data or, if necessary, on earthquake experience data, fragility test data, and generic qualification test data. *Exception*: JUSTIFY the use of generic fragility for any SSC as being appropriate for the plant.		CALCULATE component seismic-fragility parameters such as median capacity and variabilities (logarithmic standard deviations reflecting randomness and uncertainty) based on plant-specific data or, if necessary, on earthquake experience data, fragility test data, and generic qualification test data. CALCULATE component fragility as a function of the local response parameter. DERIVE the joint probability distribution of the seismic capacities of different components.
SFR-F2 [Note (2)]	For all SSCs that appear in the significant accident sequences, ENSURE that they have site-specific fragility parameters that are derived based on plant-specific information, such as anchoring and installation of the component or structure and plant-specific material test data. *Exception:* JUSTIFY the use of generic fragility for any SSC as being appropriate for the plant.		For all SSCs that appear in the significant accident sequences, ENSURE that they have site-specific fragility parameters that are derived based on plant-specific information, such as anchoring and installation of the component or structure and plant-specific material test data.
SFR-F3 [Note (3)]	PERFORM screening to identify low-ruggedness relays. CALCULATE seismic fragilities of essential low-ruggedness relays.	CALCULATE seismic fragilities for relays identified to be essential and that are included in the systems-analysis model.	
SFR-F4 [Note (4]	CALCULATE seismic fragilities for SSCs that are identified in the systems model as playing a role in the large early release frequency part of the seismic PRA. (See Requirements SPR-A1 and SPR-A3.)		

Table 5-2.2-7 Supporting Requirements for HLR-SFR-F (Cont'd)

NOTES: (Cont'd)

(1) Typically, the seismic fragility of a component is characterized by a double lognormal model whose parameters are median capacity, β_R and β_U. β_R is the logarithmic standard deviation of the capacity and represents the variability (known as "aleatory variability") due to the randomness of the earthquake characteristics for the same acceleration and to the structural response parameters that relate to these characteristics. β_U is the logarithmic standard deviation of the median capacity and represents the uncertainties (known as "epistemic uncertainties") in models and model parameters. For some applications, it may be sufficient to develop a mean fragility curve characterized by a lognormal probability distribution with parameters of A_m and β_c, where $\beta_c = (\beta_R^2 + \beta_U^2)^{1/2}$ is the logarithmic standard deviation of composite variability. An approach suggested in reference [5-12] is to first calculate the high confidence of low probability of failure (HCLPF) capacity based on the conservative deterministic failure margin (CDFM) method. This HCLPF capacity is taken as the 1% conditional-probability-of-failure value, and a generic β_C is estimated for typical SSCs. Using these, the median capacity and hence the mean fragility curve are approximated. For further discussion on the uses and limitations of these approximations, refer to references [5-10] and [5-12]. The use of generic fragilities for some SSCs is common and acceptable, but must be justified as appropriately reflecting the plant-specific SSCs and plant conditions.

(2) The objective of the fragility analysis is to derive fragility parameters that are as realistic as possible. They should reflect the as-built conditions of the equipment and should use plant-specific information. Use of conservative fragilities would distort the contribution of the seismic events to core damage frequency and large early release frequency. Note that the use of conservative fragilities may underestimate the frequencies of some accident sequences involving "success" terms. Therefore, generic fragilities, if used, should not be overly conservative and should be appropriate for the seismic risk profile of the plant. For further discussion, refer to 5-1.6. Peer reviews need to be attentive to this aspect.

(3) Guidance on evaluation of relay chatter effects is given in references [5-3], [5-7], and [5-14] (see Requirement SPR-B4). Essential relays are defined in reference [5-14].

(4) Generally, the concern is the seismically induced early failure of containment functions. NUREG-1407 [5-7] describes these functions as containment integrity, containment isolation, prevention of bypass functions, and some specific systems depending on the containment design (e.g., igniters, suppression pools, or ice baskets).

Table 5-2.2-8 Supporting Requirements for HLR-SFR-G

Documentation of the seismic-fragility evaluation shall be consistent with the applicable supporting requirements (HLR-SFR-G).

Index No. SFR-G	Capability Category I	Capability Category II	Capability Category III
SFR-G1	DOCUMENT the seismic-fragility analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
SFR-G2 [Note (1)]	DOCUMENT the process used in the seismic-fragility analysis. For example, this typically includes a description of *(a)* the methodologies used to quantify the seismic fragilities of SSCs, together with key assumptions *(b)* the SSC fragility values that includes the method of seismic qualification, the dominant failure mode(s), the source of information, and the location of the component *(c)* the fragility parameter values (i.e., median acceleration capacity, β_R and β_U) and the technical bases for them for each analyzed SSC, and *(d)* the different elements of seismic-fragility analysis, such as *(1)* the seismic response analysis *(2)* the screening steps *(3)* the walkdown *(4)* the review of design documents *(5)* the identification of critical failure modes for each SSC, and *(6)* the calculation of fragility parameter values for each SSC modeled		

NOTE:
(1) The documentation requirements given in NUREG-1407 [5-7] and followed in the Diablo Canyon Long Term Seismic Program [5-26] and Bohn and Lambright [5-17] studies may be used as guidance.

5-2.3 SEISMIC PLANT-RESPONSE ANALYSIS

The primary objectives of the plant-response analysis are to develop a plant systems model that includes seismically induced initiating events and other failures and the plant's response to them; to develop accident sequences based on the plant configuration and the initiating events and failures; and to integrate the seismic-hazard analysis and the seismic-fragilities analysis with the systems model to quantify the model — that is, to estimate the probability or frequency of reaching the undesired end states of core damage or a large early release for each of the important accident sequences.

It is assumed in the systems-analysis requirements contained herein that the seismic-PRA analysis team possesses a full-scope internal-events, at-power Level 1 and Level 2 LERF PRA, developed either prior to or concurrently with the seismic PRA. It is further assumed that this internal-events PRA is then used as the basis for the seismic-PRA systems analysis. If these assumptions are not valid, then such a PRA generally would be needed before the seismic-PRA systems-analysis work can proceed. It is also assumed that the internal-events, at-power PRA is in general conformance with Part 2.

Systems analysis for seismic PRA generally consists of both adding some earthquake-related basic events to the internal-events systems model and also "trimming" some aspects of that model that do not apply or can be screened out on a sound basis. Examples of trimming include eliminating the part of the model covering recovery from LOSP, which is usually not feasible after a large earthquake; eliminating event trees that start with very unlikely events unrelated to earthquakes; and screening out of low-probability nonseismic failures and human-error events. Thus, the seismic-PRA systems model is generally substantially simpler than the corresponding model for internal events, even though it also contains some added complexity related to earthquake-caused failures.

In special circumstances, it is acceptable to develop an ad hoc systems model tailored especially to the seismic-PRA situation being modeled, instead of starting with the internal-events model and adapting it. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects. Whichever approach is used, either adapting the internal-events systems model or building an ad hoc systems model, it is important that the systems model include all important failures, including both failures caused by the earthquake and nonseismic failures and human errors.

There are six high-level requirements for systems analysis, as follows.

Table 5-2.3-1 High Level Requirements for Seismic Probabilistic Risk Assessment: Technical Requirements for Systems Analysis (SPR)

Designator	Requirement
HLR-SPR-A	The seismic-PRA systems models shall include seismic-caused initiating events and other failures including seismically induced SSC failures, nonseismically induced unavailabilities, and human errors that give rise to significant accident sequences and/or significant accident progression sequences.
HLR-SPR-B	The seismic-PRA systems model shall be adapted to incorporate seismic-analysis aspects that are different from corresponding aspects found in the at-power, internal-events PRA systems model.
HLR-SPR-C	The seismic-PRA systems model shall reflect the as-built and as-operated plant being analyzed.
HLR-SPR-D	The list of SSCs selected for seismic-fragility analysis shall include the SSCs that participate in accident sequences included in the seismic-PRA systems model.
HLR-SPR-E	The analysis to quantify core damage and large early release frequencies shall appropriately integrate the seismic hazard, the seismic fragilities, and the systems-analysis aspects.
HLR-SPR-F	The seismic-PRA analysis shall be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review.

Table 5-2.3-2 Supporting Requirements for HLR-SPR-A

The seismic-PRA systems model shall include seismic-caused initiating events and other failures including seismic-induced SSC failures, non–seismic-induced unavailabilities, and human errors, that give rise to significant accident sequences and/or significant accident progression sequences (HLR-SPR-A).

Index No. SPR-A	Capability Category I	Capability Category II	Capability Category III
SPR-A1 [Note (1)]	ENSURE that earthquake-caused initiating events that give rise to significant accident sequences and/or significant accident progression sequences are included in the seismic-PRA system model using a systematic process.		
SPR-A2 [Note (2)]	In the initiating-event selection process, DEVELOP a hierarchy to ensure that every earthquake greater than a certain defined size produces a plant shutdown within the systems model.		
SPR-A3 [Note (3)]	USE the accident sequences and the systems logic model from the at-power, internal-event PRA model as the basis for the seismic-PRA model.		
SPR-A4 [Note (4)]	Under special circumstances based on the judgment of the analyst, DEVELOP an ad hoc systems model tailored especially to the seismic-PRA configurations or issues being modeled, instead of starting with the internal-events model and adapting it, as in Requirement SPR-A3. If this approach is used, ENSURE that the resulting model is consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures.		
SPR-A5 [Note (3)]	ENSURE that the PRA systems models reflect earthquake-caused failures and nonseismically induced unavailabilities and human errors that give rise to significant accident sequences or significant accident progression sequences.		

NOTES:

(1) It is very important that site-specific failure events, usually earthquake-caused structural, mechanical, and electrical failures, be thoroughly investigated. One approach that has been used successfully is to perform an FMEA of the seismic failures identified by the fragility analysis. The usual list of seismically induced initiating events considered in seismic PRAs includes, for example

(a) failure of the reactor pressure vessel or of another very large component such as a steam generator,a recirculation pump, or the pressurizer

(b) loss-of-coolant accidents of various sizes and in all relevant locations

(c) transients, of which loss of off-site power (LOSP) is usually the most important

There are two general types of transients that should be considered: those in which the power conversion system (PCS) or heat-transport system has failed as a direct consequence of the earthquake (for example, following LOSP) and those in which the PCS is initially available.

Other types of transient initiating events include, for example, losses of key support systems such as service water or direct-current power.

Also, multiple-unit impacts and dependencies should be considered, as appropriate, including recovery resources that could be affected by a large earthquake.

Attention to both the core damage frequency (CDF) endpoint and the large early release frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement. This means that initiating events and SSC failures that could lead to LERF-type consequences need to be included in the systems model even if the CDF frequency is quite low. [Concerning LERF, see Requirement SFR-F4 and its corresponding note, Note (4).]

A qualitative rationale may be used to exclude low-level seismic events that do not lead to significant plant challenges.

(2) It is generally a requirement at all nuclear reactor stations that any earthquake larger than a certain size — usually defined as the operating-basis earthquake (OBE) — will require the plant to shut down (terminate the chain reaction and move toward a safe, stable shutdown state) to reduce energies that may cause loss-of-coolant accidents (LOCAs) and to enable inspection for possible earthquake-caused damage. (Some plants are designed to be shut down when certain earthquakes smaller than the OBE occur.) The purpose of the initiating event (IE) hierarchy is to ensure that given an earthquake that exceeds this threshold, the sum total of all of the IE conditional probabilities adds to unity (100%). Another purpose is to ensure that the most important accident sequences are treated first in the model, that is, in the proper order vis-à-vis

Table 5-2.3-2 Supporting Requirements for HLR-SPR-A (Cont'd)

NOTES: (Cont'd)

their potential consequences. If this means that a manual-shutdown sequence must be added to account for those circumstances when no automatic post-earthquake shutdown will occur, then such manual actions must be added to the systems model. Usually, this involves adding these manual-shutdown sequences to the group of transients in which the power conversion system is initially available.

The order of the hierarchy is usually defined so that if one earthquake-caused IE occurs, the occurrence of other IEs down the hierarchy is of no significance in terms of the systems model. Thus, for example, if the earthquake causes a large LOCA, there is no concern in the systems model for the simultaneous occurrence of a small LOCA. Implicit in the IE hierarchy is the notion that basic failure events that define an IE cannot occur in the accident sequences corresponding to IEs lower in the hierarchy, so as to avoid duplication within the sequence modeling. For example, a failure of the reactivity-control function (control rod failure) usually is modeled so that it can occur as a basic event in sequences in which a large LOCA is modeled as the IE, but not vice-versa — when seismic-caused control rod failure is modeled as the IE, large LOCAs are not included there. If the seismically caused IE hierarchy is constructed logically, the various types of sequences will automatically conform to this hierarchy. For additional details, see Bohn and Lambright (reference [5-17]).

(3) Note that part of the discussion below touches on issues related also to Requirement HLR-SPR-B. The analysis may group earthquake-caused failures if the leading failure in the group is modeled. The event trees and fault trees from the internal-events, at-power PRA model are generally used as the basis for the seismic event trees. This is done both to capture the thinking that has gone into their development and to assist in allowing comparisons between the internal-events PRA and the seismic PRA to be made on a common basis. (As mentioned in the text in 5-1.3, considerable screening out and "trimming" of the internal-events PRA systems model is also common where appropriate. The lumping of certain groups of individual components into so-called "supercomponents" in the systems model is also a valid approximation in many situations.) However, it is cautioned that supercomponents should be used in a manner that they will not become significant contributors to the seismic CDF.

Earthquakes can cause failures that are not explicitly represented in the internal-events models, primarily (but not exclusively) due to damage to structures and other passive items such as distribution systems (electrical raceways, piping runs, ductwork, instrument tubing, etc.), vessels, large tanks, and all supports and anchorage and spatial interactions that can then affect safety functions. The principal challenge in meeting this requirement is ensuring that these passive-failure events are included. Other categories of seismically induced failures that are typically not modeled in the internal-events PRA are seismically induced relay-chatter and related events (see Requirement SPR-B4 in Table 5-2.3-2), and seismic-caused damage that can block personnel access to safety equipment or controls, thereby inhibiting manual operability actions, in either the control room or another location, that might otherwise be credited (see Requirement SPR-B6). Also, some failures that are modeled as one basic event in the internal-events-PRA model (for example, failure of a diesel generator) may be modeled differently, as several different basic events, in the seismic PRA model. (For example, in seismic PRAs the diesel generator itself is sometimes modeled separately from its day tank or its control circuitry.)

The principal way in which the seismic-PRA trees differ from those used in internal-events PRA analysis, besides adding in the passive SSCs, is the need to consider the physical locations and proximity of SSCs. This need exists both because secondary failures such as spatial interactions must be considered — this aspect is usually taken into account in the seismic walkdowns — and because response correlations can be important and are related to colocation of similar items. After the seismic-capacity-engineering work has been accomplished, the systems analysis needs to introduce response correlations into the models where appropriate.

Introducing these aspects into the systems analysis can be done in any of several different ways: basic events can be added directly to the fault trees and the "gates" appropriately modified, or an event (such as liquefaction or building failure) that globally affects an entire safety function or accident sequence can be added directly to the Boolean expression, or linked event trees can be used along with a "seismic pretree" with associated conditional split fractions in the plant-response part of the model, or the fragility definition of a (stronger) SSC can be redefined in terms of the fragility of another (weaker) SSC whose failure can cause the undesired failure of the stronger SSC.

Table 5-2.3-2 Supporting Requirements for HLR-SPR-A (Cont'd)

NOTES: (Cont'd)

Sometimes, the knowledge that a given SSC is very rugged to resist earthquakes can save the systems-analysis team the work of developing a model that includes that SSC's failure. This may be true, for example, of certain structures, pressure-retaining components, or piping and duct runs. Thus, a round of iteration with the seismic-capacity-engineering aspect of the seismic PRA can be useful when the systems-analysis work is underway. The SSCs to be considered in this aspect include both SSCs that can act as (or contribute to) seismically induced initiating events (IEs), and SSCs that appear as nodes in event trees or as basic events in fault trees.

Attention to both the core damage frequency (CDF) endpoint and the large early release frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement. This means that IEs and SSC failures that could lead to LERF-type consequences need to be included in the systems model even if the CDF frequency is quite low. [See Requirement SFR-F4 and its corresponding note, Note (4).]

(4) If this approach is used, it is especially important that the special circumstances requiring it or making it useful be documented. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects.

Table 5-2.3-3 Supporting Requirements for HLR-SPR-B

The seismic-PRA systems model shall be adapted to incorporate seismic-analysis aspects that are different from corresponding aspects found in the at-power, internal-events PRA systems model (HLR-SPR-B).

Index No. SPR-B	Capability Category I	Capability Category II	Capability Category III
SPR-B1 [Note (1)]	In each of the following aspects of the seismic-PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. SPECIFY a basis to support the claimed nonapplicability of any exceptions. The aspects governed by this requirement are *(a)* initiating-event analysis *(b)* accident-sequence analysis *(c)* success-criteria analysis *(d)* systems analysis *(e)* data analysis *(f)* human-reliability analysis *(g)* use of expert judgment		
SPR-B2	INCLUDE the following seismic impacts on performance-shaping factors (PSFs) for the control room and ex-control room post-initiator actions as appropriate to the human reliability analysis (HRA) methodology used: *(a)* additional post-earthquake workload and stress that can increase the likelihood of human errors or inattention *(b)* seismic failures that impact access *(c)* cue availability		INCLUDE the following seismic impacts on performance-shaping factors (PSFs) for the control room and ex-control room post-initiator actions as appropriate to the human reliability analysis (HRA) methodology used: *(a)* additional post-earthquake workload and stress that can increase the likelihood of human errors or inattention *(b)* seismic failures that impact access *(c)* cue availability When calculating the human error probabilities (HEPs) for seismic PRA, USE detailed HRA analysis in accordance with the applicable HRA requirements in Part 2.
SPR-B3 [Note (2)]	PERFORM an analysis of seismic-caused correlations in a way so that any screening of SSCs appropriately accounts for those correlations. USE bounding or generic correlation values and PROVIDE the basis for such use.		PERFORM an analysis of seismic-caused correlations in a way so that any screening of SSCs appropriately accounts for those correlations. USE plant-specific correlation values throughout.

Table 5-2.3-3 Supporting Requirements for HLR-SPR-B (Cont'd)

The seismic-PRA systems model shall be adapted to incorporate seismic-analysis aspects that are different from corresponding aspects found in the at-power, internal-events PRA systems model (HLR-SPR-B).

Index No. SPR-B	Capability Category I	Capability Category II	Capability Category III
SPR-B4 [Note (3)]	INCLUDE the effects of the chatter of relays and similar devices in the systems model.		
SPR-B4a [Note (4)]	If screening out on the basis of seismic capacity is performed in the systems model, SPECIFY the screening criterion.		
SPR-B4b [Note (4)]	If post-earthquake recovery actions are included in the systems model, INCLUDE them on a documented basis.		
SPR-B5 [Note (5)]	In the systems-analysis models, for each basic event that represents a significant seismically caused failure, INCLUDE the complementary "success" state where applicable to a particular SSC, and SPECIFY the criteria used for the term "significant" in this activity.		In the systems-analysis models, for each basic event that represents a seismically caused failure, INCLUDE the complementary "success" state where applicable to a particular SSC.
SPR-B6 [Note (6)]	EVALUATE the possibility that a large earthquake can cause damage that blocks personnel access to safety equipment or controls, thereby inhibiting operator actions that might otherwise be credited.		
SPR-B7 [Note (7)]	DO NOT INCLUDE recoveries that are specially tailored to the fact that the initiator is an earthquake. It is acceptable to include recoveries embedded in the internal-events systems model unless they would be precluded by conditions introduced by the seismic event.	EVALUATE the likelihood that system recoveries modeled in the internal-events PRA may be more complex or even not possible after a large earthquake, and ADJUST the recovery models accordingly. It is acceptable to use generic or conservative recovery values.	EVALUATE the likelihood that system recoveries modeled in the internal-events PRA may be more complex or even not possible after a large earthquake, and ADJUST the recovery models accordingly. USE plant-specific recovery values where available.
SPR-B8 [Note (8)]	ASSUME the existence of an earthquake-caused "very small loss-of-coolant accident" in the seismic-PRA accident sequences and system modeling, unless it is demonstrated that such a LOCA can be excluded, based on a walkdown or on another examination of the possible sources of such a LOCA.		INCLUDE the fragility (i.e., probability of failure) of an earthquake-caused "very small loss-of-coolant accident," and incorporate the effects into the seismic-PRA accident sequences and system modeling, unless it is demonstrated that such a LOCA can be excluded, based on a walkdown or on another examination of the possible sources of such a LOCA.
SPR-B9 [Note (9)]	If the seismic-PRA walkdown (see Requirement SFR-E4) identifies the potential for seismically induced fires and flooding, INCLUDE potential significant contributions to accident sequences in the systems model.		

Table 5-2.3-3 Supporting Requirements for HLR-SPR-B (Cont'd)

GENERAL NOTE: While the most common procedure for developing the seismic-PRA systems model is to start with the internal-events systems model and adapt it by adding and trimming, in some circumstances it is acceptable instead to develop an ad hoc seismic-PRA systems model tailored especially to the situation being modeled. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects. See 5-2.3 and also Note (3) of Table 5-2.3-2 for further commentary.

NOTES:

(1) These Sections of Part 2 are effectively incorporated here by reference. A few aspects, however, do not apply in detail. Whenever an exception is taken, the PRA analyst team needs to be cognizant of the underlying rationale for the specific Part 2 requirement so as to ensure that this rationale is considered when the exception is taken.

(2) It is vital that the analysis capture the important correlations among seismic-caused failures. Of course, this is generally true in all PRAs, but because the earthquake will affect all SSCs at the same time with the same incoming motion, special care must be taken on this subject when performing a seismic PRA. (See Requirement SPR-E4 where the requirement to deal with correlations in the integration/quantification is covered.) Some papers at the Organization for Economic Cooperation and Development/Nuclear Energy Agency Workshop in Tokyo [5-21] provide useful discussion and guidance on this issue.

One reasonable approach to take, which is usually bounding, is to assume 100% response correlation as a starting point. If the issue of correlation then seems to make a difference to the overall results or insights, one can do a sensitivity analysis by assuming zero response correlation to ascertain how important the correlation might be. If there is a major difference, the analyst must then attempt to determine just what the best assumption really is for treating the correlation.

The screening-out step must be done conservatively because once an SSC is screened out, it is "lost" from the rest of the analysis. Before SSCs are screened out on what is an otherwise well-defined basis, it is important to check that possible correlations do not invalidate the screening-out step. This requirement is intended to capture this practice. An acceptable method for this screening is found in reference [5-17], which provides more detail for an approach similar to that described above.

Requirement SPR-E1 has additional requirements and commentary about correlations.

A concern with seismic PRA today is that the overall state of knowledge about the amount of correlation among earthquake-induced SSC failures is limited. Specifically, when similar items are co-located (for example, adjacent), the analyst typically will assume full-response correlation, whereas if SSCs are quite different or found in very different locations, then the typical assumption is to assign small or zero correlation. Because of the broad range of variables in the types of SSCs, and the available test or experience data, there may not be high confidence in estimating correlation. Thus, it is standard practice among seismic-PRA analysts to perform sensitivity analyses to test how much difference emerges in the final PRA "results" when different amounts of correlation are assigned.

(3) The analysis of relay and contactor chatter has become a standardized part of seismic PRA, and several reports and guidance documents exist [5-14, 5-34, 5-35, 5-36]. After the list of relays and contactors involved in key safety functions has been developed, it is usually more efficient to screen out those with very high seismic capacities, or whose chatter will not affect the proper execution of a safety function, before including the others in the systems model. Typically, only a small subset of the relays and contactors survive these screening-out steps. Reference [5-14] provides an acceptable methodology for performing this aspect of the analysis. Requirement SFR-F3 has the requirements for analyzing the seismic fragility of relays and similar devices.

(4) To make the systems-analysis models more manageable, it is common practice to screen out some of the nonseismic failures and human errors from the model if their contribution to the results is demonstrably very small. One acceptable approach to accomplish this screening is given in NUREG/CR-5679 [5-13]. In meeting Requirement SPR-B4b, the analyst should refer to Requirement HLR-HR-H in Part 2 concerning recovery.

Table 5-2.3-3 Supporting Requirements for HLR-SPR-B (Cont'd)

NOTES: (Cont'd)

(5) At intermediate earthquake levels, many SSCs whose seismically induced failure is important to safety at higher levels will not fail or will fail with only modest probability. The modeling of the nonfailure (that is, the "success") of such SSCs is an important aspect of the systems model, and excluding these "success" states can lead to erroneous PRA results. For Categories I and II, only significant failures need be modeled in this way, and the criterion used to distinguish the significant failures from the others needs to be defined.

(6) This information is most effectively gathered during the walkdown, which should be structured to search for access issues. Coordination with the human-reliability-analysis aspect of the PRA is important. If access problems are identified, the systems model needs to be modified so as to assign the (weaker) seismic fragility of the failure causing the access problem to each (presumably stronger) SSC to which access is thereby impaired. In making these evaluations, it may be assumed that portable lighting is available and that breathing devices are available for confined spaces, if in fact the plant configuration includes them.

(7) The restoration of safety functions after an earthquake can be inhibited by any of several types of causes; these include damage or failure, access problems, confusion, loss of supporting personnel to other post-earthquake-recovery functions, potential for aftershocks, and so on. Careful consideration of these must be given before recoveries are credited in the initial period after a large earthquake. This is especially true for earthquake-caused loss of off-site power (LOSP), given that the damage could be to switchyard components or to the off-site grid towers, which are generally difficult to fix quickly. While this Part does not require the analyst to assume an unrecoverable LOSP after a large earthquake, the general practice in seismic PRAs has been to make such an assumption.

(8) It may not be feasible in a seismic-PRA walkdown to evaluate every small impulse line connected to the primary circuit, whose failure in an earthquake could cause a so-called "very small loss-of-coolant accident" (LOCA) (a leak with an area from one to a few square-centimeters) in the primary circuit. If the walkdown evaluation is not done, or if a seismically caused break in a given small impulse line cannot otherwise be excluded on the basis of acceptable evidence, then it needs to be included as the Requirement states. Typically, breaks in one or a very few such lines cannot always be precluded, given the large number of such lines and their unusual configurations in many cases. Therefore, it is a common (although not a universal) practice in seismic PRAs to include such a very small LOCA as an additional assumed fault in most or even every accident sequence, in addition to whatever other failures are modeled. (See Requirement SM-B4 in Part 10.) This has the effect of making "success" (that is, reaching a safe stable state) in those sequences dependent on the availability of at least enough makeup water to the primary system to replace the inventory loss at high pressure from such a break. This requirement is intended to ensure that adding such a very small LOCA basic event to each relevant accident sequence is *considered* and is done unless a justification for omitting such can be supported.

(9) Extensive experience with seismic PRAs at U.S. nuclear plants indicates that only rarely is the PRA analysis team faced with the task of quantifying a core damage frequency or large early release frequency for these types of scenarios using a full seismic-fire-PRA analysis, but if so, then this analysis must quantify the hazard, the fragilities, and the systems-analysis aspect as in any other aspect of the seismic PRA. The walkdown that supports this aspect should be linked with the walkdown that examines seismic spatial interactions. (See Requirement HLR-SFR-E.) NUREG-1407 [5-7] contains guidance on how to do this evaluation.

Table 5-2.3-4 Supporting Requirements for HLR-SPR-C

The seismic-PRA systems model shall reflect the as-built and as-operated plant being analyzed (HLR-SPR-C).

Index No. SPR-C	Capability Category I	Capability Category II	Capability Category III
SPR-C1	To ensure that the systems-analysis model reflects the as-built, as-operated plant, JUSTIFY any conservatisms or other distortions that do not adequately reflect the as-built, as-operated plant.		

Table 5-2.3-5 Supporting Requirements for HLR-SPR-D

The list of SSCs selected for seismic-fragility analysis shall include the SSCs that participate in accident sequences included in the seismic-PRA systems model (HLR-SPR-D).

Index No. SPR-D	Capability Category I	Capability Category II	Capability Category III
SPR-D1	USE the PRA systems model as the basis for developing the seismic equipment list to support the fragility analysis of 5-2.2. INCLUDE structures and passive components that may not be present in the internal-events model but that require consideration in the seismic PRA. SUPPLEMENT the list based on the review of industry seismic-PRA seismic equipment lists (SELs), if available.		

Table 5-2.3-6 Supporting Requirements for HLR-SPR-E

The analysis to quantify core damage frequency and large early release frequency shall appropriately integrate the seismic hazard, the seismic fragilities, and the systems-analysis aspects (HLR-SPR-E).

Index No. SPR-E	Capability Category I	Capability Category II	Capability Category III
SPR-E1 [Note (1)]	In the quantification of core damage frequency and large early release frequency, PERFORM the integration using the seismic hazard, fragility, and systems analyses.		
SPR-E2 [Note (2)]	PERFORM seismic-sequence quantification in accordance with the applicable requirements described in 2-2.7.		
SPR-E3 [Note (3)]	USE the quantification process to confirm and support the screening of SSCs (refer to Requirement SFR-B1).		
SPR-E4 [Note (4)]	In the integration/quantification analysis, INCLUDE the significant correlations that affect the results. It is acceptable to use generic correlation values. If used, SPECIFY the basis for such use.		In the integration/quantification analysis, INCLUDE all significant correlations that affect the results. USE plant-specific correlation values throughout.
SPR-E5 [Note (5)]	USE the mean hazard, composite fragilities, and the systems analysis to generate point estimates for core damage frequency (CDF) and large early release frequency (LERF). ESTIMATE the uncertainties in overall CDF and LERF.	In the integration/quantification analysis, INCLUDE in the uncertainties in core damage frequency and large early release frequency results that arise from each of the several inputs (the seismic hazard, the seismic fragilities, and the systems-analysis aspects).	In the integration/quantification analysis, QUANTIFY the uncertainties in core damage frequency and large early release frequency results that arise from each of the several inputs (the seismic hazard, the seismic fragilities, and the systems-analysis aspects).
SPR-E6 [Note (6)]	In the analysis of LERF, SATISFY the LERF requirements in 2-2.8, where applicable.		

NOTES:

(1) The integration step is where the various earlier and supporting parts of the seismic PRA are brought together and integrated to produce and quantify the final results in terms of core damage frequency (CDF) and large early release frequency (LERF) and in terms of identifying the "important contributors."

Seismic-PRA practitioners possess different tools to accomplish this integration and quantification. Analysts usually use an iterative process in which an interim and approximate quantification is done, after which certain parts of the overall systems model are screened out on the basis that they do not contribute importantly to the results. The quantification is then finalized. Seismic screening of an SSC (refer also to Requirements SFR-B1 and SPR-B4a) can be done on the basis that its seismic capacity is very strong, so that it does not contribute importantly to any seismically induced accident sequences, above some defined cutoff level. Screening of a nonseismic failure or of a human-error basic event in the model can be done on the basis that its contribution to any seismically induced accident sequences is below a defined cutoff (refer also to Requirement SPR-B4a). Whatever the basis for the screening (see the supporting requirements below on this subject), that basis must be defined, and the selection of a cutoff should be done very carefully.

Table 5-2.3-6 Supporting Requirements for HLR-SPR-E (Cont'd)

NOTES: (Cont'd)

While details vary, one typical systems-analysis approach is to add seismic-related basic events (or sometimes entire new "branches") to the internal-events fault tree models that are adapted from the internal-events-PRA Level 1 and Level 2 LERF analysis. Considerable screening out or "trimming" of the systems model is also a common practice. The quantification would then typically consist of a series of hazard-specific quantifications: the model is quantified several times for a range of different hazard intervals, and these quantifications are then summed. In this approach, for each hazard interval and for each SSC/basic event, the hazard, response, and fragility analyses are integrated to produce a "probability of seismically induced failure" — actually a distribution of the analyst's state of knowledge of that probability, taking into account the uncertainties in hazard, response, and fragility. This probability is then inserted into the relevant fault tree, which is solved. Typically, each fault tree would be solved separately, and then these would be integrated into the relevant event tree(s) to produce a set of accident-sequence-specific values for CDF conditional on the hazard interval being evaluated. (Other methods are also in use in which the integration over the hazard is not done on a fault-tree-specific basis but rather at the event-tree level; logically, the outcome should be the same.)

The one issue that requires great care is the treatment of seismic-related correlations among the seismic failures: in particular

(a) the linking of the various basic events to capture their correlated failures

(b) the screening out of SSCs and other nonseismic basic events in light of these correlations (see Requirements SPR-B4a, SPR-E4, and SPR-E6 on these subjects)

The relevant seismic correlations arise, of course, because in a given earthquake event, every SSC in the plant is exposed to the exact same earthquake input motion (although modified —amplified, damped, frequency shifted, etc. — as the earthquake energy propagates from the earth below the site to the location of the SSC at issue). There are a number of different approaches in use to treat these correlations, and this Standard does not single out any one of them. Acceptable methods can be found in references [5-17] and [5-26].

(2) The intent of this requirement is to ensure that key information about each accident sequence (or cutset) is retained rather than simply "lost" in the production of overall integrated values for core damage frequency and large early release frequency. Of course, it is common to group cutsets of accident sequences when they are so similar that phenomenologically they cannot be distinguished very well; such grouping is entirely acceptable if its basis is defined.

(3) SSC screening — the elimination from the model of SSCs — is done throughout the process of performing any PRA. A defined set of criteria must be developed and used to ensure that this screening does not eliminate elements of the model that should have been retained. (See Requirement SPR-B4a.) The intent of this requirement is to ensure that the quantification process is used to check that the screening has not erroneously eliminated important SSCs. It is recognized that this type of work is an iterative process, in which approximate interim quantifications are done during which the screening decisions are checked, and only then is a final quantification done. However, the check need be of only limited scope, depending on the circumstances. There are many different approaches in current use among seismic-PRA analysts to accomplish this step. Reference [5-17] contains a useful discussion on this aspect.

(4) As discussed earlier, treating earthquake-specific correlations properly is vital to achieving a successful seismic PRA. This requirement is intended to ensure that this issue is covered. A discussion of this type of correlation analysis is found in reference [5-17]. See Requirement SPR-B3, where the requirement to deal with correlations in initial screening is covered.

(5) All seismic-PRA analyses are characterized by large numerical uncertainties not only in the seismic hazard aspect but also in the seismic-fragility and systems-analysis aspects as well. Examples of other analysis areas where uncertainties arise in seismic PRA that are different from those encountered in internal-events PRA are the human-reliability-analysis aspect, the issue of earthquake-caused correlations, relay chatter, and the recovery analysis.

Table 5-2.3-6 Supporting Requirements for HLR-SPR-E (Cont'd)

NOTES: (Cont'd)

It is essential that estimates of the uncertainties in the analysis team's state of knowledge about each aspect be developed in the base PRA model and that these be carried through to be incorporated quantitatively into the integration/quantification step. Experience has shown that to do otherwise can produce "results" that may not be relied on in terms of both overall insights and the details. For specific applications, a graded approach to uncertainty analysis is appropriate, depending on the application. Also note that the requirement to "include" the various uncertainties recognizes that not all of them must necessarily be quantified explicitly, especially if they are small. [See also the comment of Table 5-2.3-7, Note (3).]

There are numerous methods in current use to accomplish this requirement, ranging from numerical-integration schemes to schemes that approximate the various empirical distributions by well-defined analytical forms (such as lognormal forms) that are more amenable to numerical integration. "INCLUDE" means the analyst need not quantify some of the uncertainties if they are small in one category compared to those in another category. "QUANTIFY" means quantify everywhere.

(6) Those aspects of LERF analysis that are common to internal-events PRA and seismic PRA are referred to here. Also, the discussion of LERF analysis in the last four paragraphs of 5-1.3 is broadly applicable and should be referred to as background information.

Table 5-2.3-7 Supporting Requirements for HLR-SPR-F

The seismic-PRA analysis shall be documented in a manner that facilitates applying the PRA and updating it, and that enables peer review (HLR-SPR-F).

Index No. SPR-F	Capability Category I	Capability Category II	Capability Category III
SPR-F1 [Note (1)]	DOCUMENT the seismic plant response analysis and quantification in a manner that facilitates PRA applications, upgrades, and peer review.		
SPR-F2 [Note (2)]	DOCUMENT the process used in the seismic plant response analysis and quantification.		
SPR-F3 [Note (2)]	DOCUMENT the sources of model uncertainty and related assumptions associated with the seismic plant response model development.		

NOTES:

(1) The major outputs of a seismic PRA, such as mean CDF, mean LERF, uncertainty distributions on CDF and LERF, results of sensitivity studies, and significant risk contributors, are examples of the PRA results that are generally documented.

(2) This documentation typically includes a description of
(a) the specific adaptations made in the internal-events PRA model to produce the seismic-PRA model, and their motivation
(b) the major outputs of a seismic PRA, such as mean CDF, mean LERF, uncertainty distributions on CDF and LERF, results of sensitivity studies, and significant risk contributors

(3) While many of these uncertainties must necessarily be expressed in terms of numerical distributions of the analysis team's state of knowledge about a numerical result, not all of them must be expressed in such numerical terms. Also, see Note (4) in Table 5-2.3-6. As in Requirement SPR-E4, which uses the word "INCLUDE," the word "DOCUMENT" here implies a recognition that not all of the various uncertainties must necessarily be quantified explicitly, especially if they are small. However, this requirement does require a description of each of the important uncertainties.

Section 5-3
Peer Review for Seismic Events PRA At-Power

5-3.1 PURPOSE

This Section provides requirements for peer review of a seismic-events PRA at-power.

5-3.2 PEER-REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATIONS

In addition to the general requirements of Section 1-6, the peer-review team shall have knowledge and collective experience in the areas of systems engineering, seismic hazard, seismic capability engineering, and seismic PRAs or seismic margin methodologies. The reviewer(s) focusing on the seismic-fragility work shall have demonstrated experience in seismic walkdowns of nuclear power plants.

5-3.3 REVIEW OF SEISMIC-PRA ELEMENTS TO CONFIRM THE METHODOLOGY

5-3.3.1 Seismic Hazard

The peer-review team shall evaluate whether the seismic hazard study used in the PRA is appropriately specific to the site and has met the relevant requirements of this Standard.

5-3.3.2 Seismic-Fragility Analysis

5-3.3.2.1 Seismic Response Analysis. The peer-review team shall evaluate whether the seismic response analysis used in the development of seismic fragilities meets the relevant requirements of this Standard. Specifically, the review should focus on the input ground motion (i.e., spectrum or time history), structural modeling including soil-structure interaction effects, parameters of structural response (e.g., structural damping and soil damping), and the reasonableness of the calculated seismic response.

5-3.3.2.2 Seismic Walkdown. The peer-review team shall review the seismic walkdown of the plant to ensure the reasonableness of the findings of the seismic review team on screening, seismic spatial interactions, and the identification of critical failure modes.

5-3.3.2.3 SSC Fragility Analysis. The peer-review team shall evaluate whether the methods and data used in the fragility analysis of SSCs are adequate for the purpose.

5-3.3.3 Seismic Plant-Response Analysis

5-3.3.3.1 Seismic-Induced Initiating Events. The peer-review team shall evaluate whether the seismically induced initiating events are properly identified and analyzed.

5-3.3.3.2 Seismic-Accident-Sequence Analysis. The peer-review team shall evaluate whether, in the systems analysis, the SSCs are properly modeled and the accident sequences are properly analyzed and quantified. The review team shall ensure that the seismic equipment list is reasonable for the plant considering the reactor type, design vintage, and specific design.

5-3.3.3.3 Seismic Quantification. The peer-review team shall evaluate whether the seismic quantification method used in the seismic PRA is appropriate and provides all the results and insights needed for risk-informed decisions. The review shall focus on the core damage frequency and large early release frequency estimates and uncertainty bounds and on the significant risk contributors.

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NONMANDATORY APPENDIX 5-A
SEISMIC PROBABILISTIC RISK ASSESSMENT METHODOLOGY: PRIMER

5-A.1 BACKGROUND

Seismic probabilistic risk assessments (PRAs) have been conducted for more than 50 nuclear power plants worldwide in the last 20 yr. The methodology has been well established, and the necessary data on the parameters of the PRA model have been generally collected. Detailed description of the procedures used in seismic PRA is given in several published reports and technical papers: PRA Procedures Guide [5-A-1], PSA Procedures Guide [5-A-2] and references [5-A-3], [5-A-4], [5-A-5], [5-A-6], and [5-A-7].

5-A.1.1 Differences Between Seismic and Internal-Event PRAs

Seismic PRA is different from an internal-event PRA in several important ways:

(a) Earthquakes could cause initiating events different from those considered in the internal-event PRA.

(b) All possible levels of earthquakes along with their frequencies of occurrence and consequential damage to plant systems and components should be considered.

(c) Earthquakes could simultaneously damage multiple redundant components. This major common-cause effect should be properly accounted for in the risk quantification.

5-A.1.2 Seismic PRA Objectives

The objectives of a seismic PRA include the following:

(a) Develop an appreciation of accident behavior (i.e., consequences and role of operator).

(b) Understand the most likely accident sequences induced by earthquakes (useful for accident management).

(c) Gain an understanding of the overall likelihood of core damage induced by earthquakes.

(d) Identify the dominant seismic risk contributors.

(e) Identify the range of peak ground acceleration (PGA) that contributes significantly to the plant risk (this is helpful in making judgments on seismic margins).

(f) Compare seismic risk with risks from other events and establish priorities for plant upgrading.

5-A.2 KEY ELEMENTS OF SEISMIC PROBABILISTIC RISK ASSESSMENT

The key elements of a seismic PRA can be identified as

(a) Seismic hazard analysis: to develop frequencies of occurrence of different levels of ground motion (e.g., PGA) at the site.

(b) Seismic-fragility evaluation: to estimate the conditional probability of failure of important structures and equipment whose failure may lead to unacceptable damage to the plant (e.g., core damage); plant walkdown is an important activity in conducting this task.

(c) Systems/accident sequence analysis: modeling of the various combinations of structural and equipment failures that could initiate and propagate a seismic core damage sequence.

(d) Risk quantification: assembly of the results of the seismic hazard, fragility, and systems analyses to estimate the frequencies of core damage and plant damage states. Assessment of the impact of seismic events on the containment and consequence analyses, and integration of these results with the core damage analysis to obtain estimates of seismic risk in terms of effects on public health (e.g., early deaths and latent cancer fatalities).

The process is shown schematically in Fig. 5-A.1 and is described in detail in reference [5-A-1]. Following is a brief description of the four steps utilized in the seismic PRA process.

5-A.2.1 Seismic Hazard Analysis

Seismic hazard is usually expressed in terms of the frequency distribution of the peak value of a ground motion parameter (e.g., PGA) during a specified time interval. The different steps of this analysis are as follows:

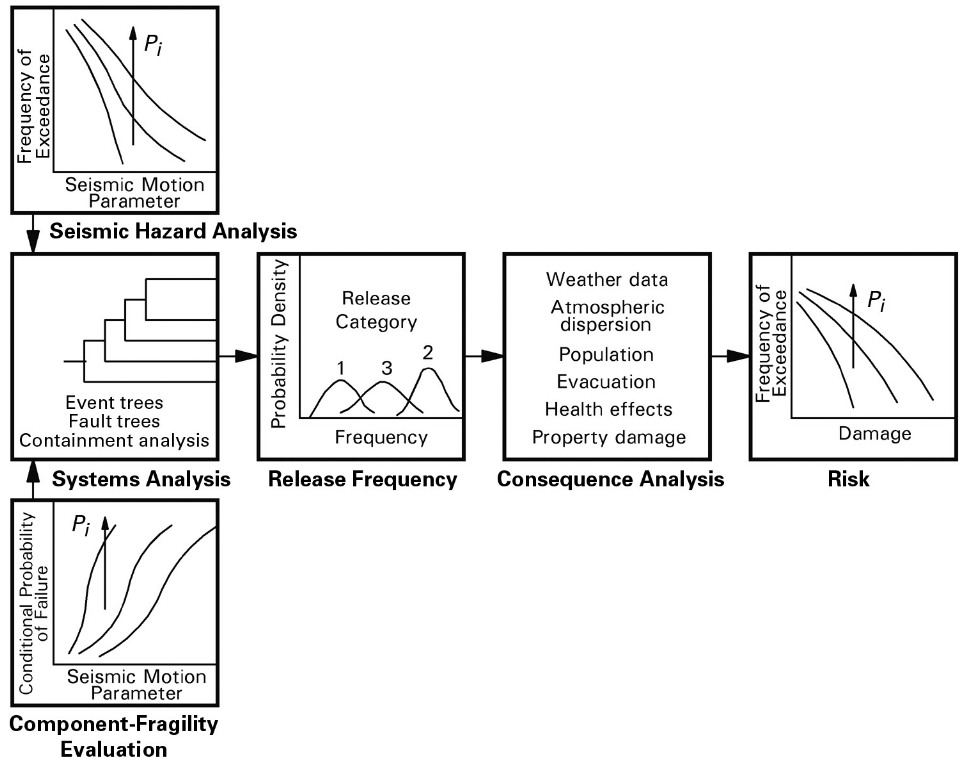
(a) identification of the sources of earthquakes, such as faults and seismotectonic provinces

(b) evaluation of the earthquake history of the region to assess the frequencies of occurrence of earthquakes of different magnitudes or epicentral intensities

(c) development of attenuation relationships to estimate the intensity of earthquake-induced ground motion (e.g., PGA) at the site

(d) integration of the above information to estimate the frequency of exceedance for selected ground motion parameters

Fig. 5-A.1 Schematic Overview of a Seismic PRA



GENERAL NOTE: P_i = subjective probability weight assigned to each curve, $_i$

The hazard estimate depends on uncertain estimates of attenuation, upper-bound magnitudes, and the geometry of the postulated seismic sources. Such uncertainties are included in the hazard analysis by assigning probabilities to alternative hypotheses about these parameters. A probability distribution for the frequency of occurrence is thereby developed. The annual frequencies for exceeding specified values of the ground motion parameter are displayed as a family of curves with different probabilities or with different fractiles (Fig. 5-A.2).

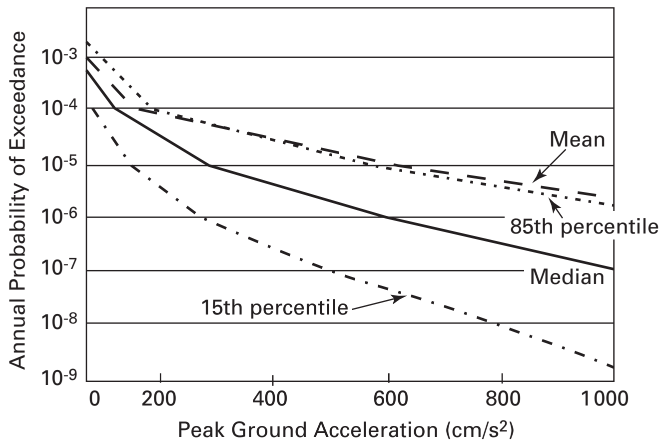
A mean estimate of the frequency of exceedance at any PGA is obtained as the weighted sum of the frequencies of exceedance at this acceleration given by the different hazard curves; the weighting factor is the probability assigned to each hazard curve. Thus, the probabilistic seismic hazard analysis (PSHA) embeds uncertainties in the core of the methodology, and results are expressed in terms of likelihood — estimated probabilities in a given time period or estimated frequencies — that earthquakes producing various sizes of ground motion will occur at a given site. These results reflect two different classes of uncertainties. Lack-of-knowledge uncertainties or *epistemic* uncertainties arise from imperfect scientific understanding that can, in principle, be further reduced through additional research and acquisition of data. The *aleatory* or random uncertainties are those uncertainties that, for all practical purposes, cannot be known in detail or cannot be reduced. Although in some applications it may not be necessary to display, this distinction in the nature of uncertainties (e.g., NUREG-1407 [5-A-3]) allowed the use of the mean hazard curve that includes combined uncertainties instead of the full family of hazard curves for identification of vulnerabilities and ranking dominants sequences and contributors), it is crucial that in the development of a PSHA, this distinction is maintained to understand and communicate the sources of uncertainties.

For further details on seismic hazard analysis methods, the reader is referred to references [5-A-7] and [5-A-8]. Typical results of a PSHA include families of seismic hazard curves in terms of PGA or spectral acceleration values at different frequencies, and site-specific ground motion response spectra.

5-A.2.2 Seismic-Fragility Evaluation

The methodology for evaluating seismic fragilities of structures and equipment is documented in references [5-A-4] and [5-A-9]. Seismic fragility of a structure or equipment item is defined as the conditional probability

Fig. 5-A.2 Typical Seismic Hazard Curves for a Nuclear Power Plant Site



of its failure at a given value of the seismic input or response parameter (e.g., PGA, stress, moment, or spectral acceleration). Seismic fragilities are needed in a PRA to estimate the conditional probabilities of occurrence of initiating events [i.e., loss of emergency alternating-current (AC) power, loss of forced circulation cooling systems], and the conditional failure probabilities of different mitigating systems (e.g., auxiliary feedwater system).

The objective of fragility evaluation is to estimate the ground motion capacity of a given component and its uncertainty. This capacity is defined either in terms of average spectral acceleration value or PGA value for which the seismic response of a given component located at a specified point in the structure exceeds the component's resistance capacity, resulting in its failure. Although the average spectral acceleration is preferable, PGA has been used in many seismic PRAs and is acceptable provided that the uncertainties in the spectral shape are not too large. The ground acceleration capacity of the component is estimated using information on plant design bases, responses calculated at the design analysis stage, as-built dimensions, and material properties. Because there are many variables in the estimation of this ground acceleration capacity, component fragility is described by a family of fragility curves; a probability value is assigned to each curve to reflect the uncertainty in the fragility estimation. This family of fragility curves may be described by three parameters: the median acceleration capacity, A_m, and logarithmic standard deviations, β_R and β_U, for randomness and uncertainty.

In seismic margin assessments, the high confidence of low probability of failure (HCLPF) capacity is used as a measure of seismic margin. HCLPF capacity is a ground motion value at which there is 95% confidence that the probability of failure is < 5%. If the fragility curve is described by the median, A_m, the randomness, β_R, and uncertainty, β_U, where the β's are logarithmic standard deviations, the HCLPF may be computed from

$$\text{HCLPF} = A_m \exp[-1.65(\beta_R + \beta_U)] \qquad \text{(5-A.1)}$$

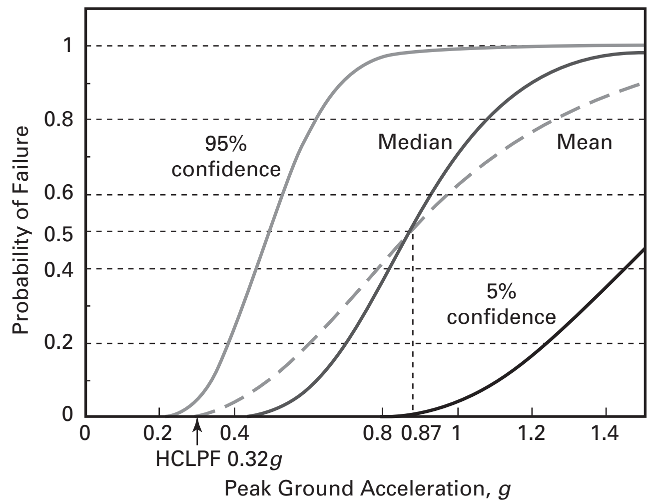
An example family of seismic-fragility curves is shown in Fig. 5-A.3. The component is designed for a safe shutdown earthquake of 0.17g. Its median capacity for overturning (resulting in failure of attached piping) is calculated as 0.87g; the logarithmic standard deviations β_R and β_U are estimated as 0.25 and 0.35, respectively. The HCLPF capacity of the component is calculated from eq. (5-A.1) as 0.32g. Figure 5-A.3 shows the median, 5% confidence and 95% confidence fragility curves. The mean fragility curve is also shown, which is obtained from the lognormal probability distribution with A_m and $\beta_c = (\beta_R^2 + \beta_U^2)^{1/2}$. For some applications, exclusive use of mean fragility curves is judged to be sufficient.

Seismic fragilities of structures and equipment are calculated using many sources: plant-specific seismic design and qualification data, fragility test data, generic seismic qualification test data, and earthquake experience data. In a typical seismic PRA, more than 500 components are identified as requiring evaluations. A plant walkdown is performed to screen out a large number of these components based on their generically high seismic capacities and on lack of obvious seismic deficiencies (such as poor anchorage and inadequate lateral support) and spatial interactions (e.g., a nonseismically qualified component failing and falling on a component modeled in the seismic PRA). For the remaining components, seismic fragilities are calculated using one or more of the data sources.

5-A.2.3 Analysis of Plant Systems and Accident Sequences

Frequencies of severe core damage and radioactive release to the environment are calculated by combining plant logic with component fragilities and seismic hazard estimates. Event and fault trees are constructed to

Fig. 5-A.3 Typical Family of Fragility Curves for a Component



identify the accident sequences that may lead to severe core damage and radioactive release.

The plant systems and sequence analyses used in seismic PRAs are based on the PRA Procedures Guide [5-A-1] and can generally be summarized as follows:

(a) The analyst constructs fault trees reflecting

(1) failures of key system components or structures that could initiate an accident sequence

(2) failures of key system components or structures that would be called on to stop the accident sequence

(b) The fragility of each such component (initiators and mitigators) is estimated.

(c) Fault trees are used to develop Boolean expressions for severe core damage that lead to each distinct plant damage state sequences.

(d) Considering possible severe core damage sequences and containment mitigation systems (e.g., fan coolers, containment sprays, and containment), Boolean expressions are developed for each release category.

As an example, the Boolean expression for severe core damage in the Limerick seismic PRA is

$$CM = 3 + 4 + 1 * \{(6 + 7 + 8 + 9 + 10 + 11 + 12 + 13 + 14) + [(2 + 15) * (3 + 16)]\} \quad (5\text{-A.2})$$

The numbers represent components for which seismic fragilities have been developed or which represent non-seismic failures. The symbols "+" and "*" indicate Boolean "OR" and "AND" operations, respectively. Plant level fragility curves are obtained by combining the fragilities of individual components according to eq. (5-A.2), using either Monte Carlo simulation or numerical integration. The plant level fragility is defined as the conditional probability of severe core damage as a function of the PGA at the site. The uncertainty in plant level fragility is displayed by developing a family of fragility curves; the weight (probability) assigned to each curve is derived from the fragility curves of components appearing in the specific plant damage state accident sequence.

5-A.2.4 Evaluation of Core Damage Frequency

Plant level fragilities are convolved with the seismic hazard curves to obtain a set of doublets for the plant damage state frequency:

$$\{\langle P_{ij}, f_{ij} \rangle\} \quad (5\text{-A.3})$$

where

f_{ij} = the seismically induced plant damage state frequency

P_{ij} = the discrete probability of this frequency

$$P_{ij} = q_i P_j \quad (5\text{-A.4})$$

$$f_{ij} = \int_0^\infty f_i(a) \frac{dH_i}{da} da \quad (5\text{-A.5})$$

where

f_i = ith plant damage fragility curve

H_j = jth hazard curve

P_j = probability associated with the jth hazard curve

q_i = probability associated with the ith fragility curve

Equations (5-A.3), (5-A.4), and (5-A.5) state that the convolution between the seismic hazard and plant level fragility is carried out by selecting hazard curve j and fragility curve i; the probability assigned to the plant damage frequency resulting from the convolution is the product of the probabilities P_j and q_i assigned to these two curves.

The convolution operation given by eq. (5-A.5) consists of multiplying the occurrence frequency of an earthquake PGA between a and $a + da$ (obtained as the derivative of H_j with respect to a) with the conditional probability of the plant damage state, and integrating such products over the entire range of PGAs from 0 to ∞. In this manner, a probabilistic distribution on the frequency of a plant damage state can be obtained.

Severe core damage occurs if any one of the plant damage states occurs. By probabilistically combining the plant damage states, the plant level fragility curves for severe core damage are obtained. Integration of the family of fragility curves over the family of seismic hazard curves yields the probability distribution function of the occurrence frequency of severe core damage. By extending this procedure, probability distribution functions of the occurrence of different release categories are obtained.

5-A.3 OUTPUTS OF SEISMIC PROBABILISTIC RISK ASSESSMENT

The outputs of a seismic PRA are

(a) seismic fragilities of components and seismic margins

(b) seismic fragilities of accident sequences and seismic margins

(c) seismic accident sequence frequencies and uncertainty distributions

(d) impact of nonseismic unavailabilities on seismic risk

(e) identification of dominant risk contributors: components, systems, sequences, and procedures

(f) distribution on range of accelerations contributing to seismic risk

(g) risk reduction as a function of seismic upgrading to aid in backfit decisions

5-A.4 REFERENCES

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PART 6 REQUIREMENTS FOR SCREENING AND CONSERVATIVE ANALYSIS OF OTHER HAZARDS AT-POWER (a) (b)

Section 6-1 Approach for Screening and Conservative Analysis

6-1.1 GENERAL

Generally, the evaluation covered by the requirements in this Part is one of the critical tasks undertaken in a full-scope other hazards PRA. Through the work required herein, the analysis team ascertains which of the hazards can be screened out so that no further PRA analysis is needed. This screening out allows the team to focus on those hazards that remain within the analysis. Experience has shown that earthquakes can never be screened out by using the methods required herein; that sometimes high winds and external flooding can be screened out but sometimes they require further analysis, either a bounding analysis, a semiquantitative analysis, or perhaps even a full PRA; and that occasionally one or more other hazards also require a full PRA. Subsequent Parts of this Standard cover requirements for a full PRA of the hazards that may not be screened out.

6-1.2 OTHER HAZARDS SCOPE

The term "other hazard" refers to hazards other than internal events, internal flood, internal fire, and earthquakes, which must be addressed per the requirements in Parts 2, 3, 4, and 5 of this Standard, respectively. Nonmandatory Appendix 6-A provides a list of "other hazards" that may be applicable to a specific site or application. This list has been adapted from NUREG/CR-2300 [6-1].

6-1.3 COORDINATION WITH OTHER PARTS OF THIS STANDARD

This Part deals with screening out hazards from further consideration. For those hazards that cannot be screened out, the requirements in Part 7, 8, or 9 of this Standard, which are used in conjunction with requirements in Parts 1 and 2 of this Standard, also apply. This Part is not applicable to any hazard covered by Part 2, 3, 4, or 5 of this Standard.

6-1.4 SCOPE OF "OTHER HAZARDS" AND APPLICABILITY

For this Part, which deals with analysis of an entire category of hazard, the term "hazard" in the singular is used for a single and entire category of similar events, or hazard group, and the category "hazard group" is intended to include all "sizes" of such events within the category. For example, the hazard group for "extreme temperature" includes all extreme-temperature conditions, no matter how extreme or how infrequent; the hazard group "transportation accidents" includes all such accidents arising from nearby transport modes. Within that hazard group, the hazard "aircraft impact" includes crashes of all aircraft, of all sizes; and so on.

Section 6-2
Technical Requirements for Screening and Conservative Analysis

6-2.1 GENERAL

The requirements in this Part are concerned with screening out. The term "screening out" is used here for the process whereby a hazard is excluded from further consideration in a PRA. Even though, as written, it contemplates the screening out of an entire hazard, it is not intended to restrict the analyst from screening out specific hazard events resulting from the hazard if the screening can be done on a defined basis and if differentiation from the remaining hazard events is clear. For example, suppose that for a given site all transportation accidents except aircraft impact can be screened out based on bounding CDF, and within the aircraft impact the only important risk potential arises from military jet overflights. Suppose that large commercial jets can be screened out on the basis of a very low annual frequency and that small crop-duster planes can be screened out on the basis of not being able to cause enough damage. It is completely acceptable to subdivide transportation accidents into individual hazards to screen all except aircraft impact, then subdivide the hazard "aircraft impact" into specific aircraft impact events to screen the large jets and crop dusters on a defined basis, and then to subject only the military jet subcategory to detailed PRA analysis using the requirements in Part 9.

Note that the above discussion does not mention screening an entire hazard group. Although hazards can be grouped by common approach, methods, and data, each hazard must be screened individually.

6-2.2 RATIONALE

There is a three-part underlying rationale for the requirements in this Part.

(a) All potential hazards (both internal and external) that may affect the facility must be considered, and each of them must be either screened out on a defined basis (following the requirements in this Part) or subjected to analysis using a PRA, either a limited PRA or perhaps a detailed PRA (following the requirements in Part 2, 3, 4, 5, 7, 8, or 9).

(b) A set of screening criteria is provided, which offers a defensible basis for screening out a hazard.

(c) If a hazard cannot be screened out using these screening criteria, then a demonstrably conservative or bounding analysis, when used together with quantitative screening criteria, can also provide a defensible basis for screening out the hazard, without the need for detailed analysis.

The burden of demonstrating that a given bounding analysis is "demonstrably conservative" falls on the analyst; different circumstances will require different approaches. The general notion is that the conservatism is demonstrated in part by accounting for all uncertainties, approximations, or simplifications that might invalidate the demonstration if not accounted for appropriately.

6-2.3 SCREENING CRITERIA

There are three fundamental screening criteria embedded in the requirements here, as follows. A hazard can be screened out if

(a) it meets the criteria in the U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) [6-2] or a later revision; or

(b) it can be shown, by using a demonstrably conservative analysis, that the mean value of the frequency of the design-basis hazard event used in the plant design is less than $\sim 10^{-5}$/yr and that the conditional core damage probability is $<10^{-1}$, given the occurrence of the design-basis-hazard event; or

(c) it can be shown, by using a demonstrably conservative analysis, that the CDF is $<10^{-6}$/yr

It is important to recognize that a demonstratively conservative estimate of a mean value is not a point estimate. When uncertainties are large, the mean frequency can fall above the 95th percentile of the distribution. Therefore, it is incumbent on the analyst to document the evidence that justifies estimates of uncertainties, approximations, or simplifications leading to the estimate of the mean event frequency or CDF. The discussion of the high-level requirements below further explains an acceptable approach for ensuring demonstratively conservative screening.

Concerning LERF, note that there is an implicit assumption that if a hazard is screened out using one or another of the screening criteria herein, then neither the CDF nor the LERF arising due to that event is of concern. This assumption is made even though only limited consideration is given in the screening to LERF issues (for example, during the walkdown, a review of spatial interactions is required). This assumption may not be conservative.

A hazard that cannot be screened out using any of these criteria must be subjected to the requirements in Part 7, 8, or 9. However, a full-scope realistic PRA analysis is not always necessary to satisfy this requirement; a limited PRA, a conservative/bounding PRA, or some other intermediate approach may be sufficient for the purpose at hand.

Table 6-2-1 High Level Requirements for Other Hazards: Requirements for Screening and Conservative Analysis (EXT)

Designator	Requirement
HLR-EXT-A	All potential other hazards (i.e., all other internal and external hazards) that may affect the site shall be identified.
HLR-EXT-B	Preliminary screening, if used, shall be performed by using a defined set of screening criteria.
HLR-EXT-C	A bounding or demonstrably conservative analysis, if used for screening, shall be performed by using defined quantitative screening criteria.
HLR-EXT-D	The basis for the screening out of a hazard shall be confirmed through a walkdown of the plant and its surroundings.
HLR-EXT-E	Documentation of the screening out of a hazard shall be consistent with the applicable supporting requirements.

GENERAL NOTES:

(a) It should be understood that Requirements HLR-EXT-B, HLR-EXT-C, HLR-EXT-D, and HLR-EXT-E are applicable when a hazard is selected for screening rather than for detailed analysis. At any time during the screening process, a decision can be made to bypass that process and go directly to the detailed-analysis requirements in Part 7, 8, or 9. Nonmandatory Appendix 6-A contains a list of hazards to be considered, and using this list is one acceptable approach to meeting this requirement. (See Requirement EXT-A1.)

(b) Requirements for detailed analyses of a hazard that cannot be screened out by using either the qualitative criteria under Requirement HLR-EXT-B or the quantitative criteria under Requirement HLR-EXT-C are provided in Parts 7, 8, and 9.

Table 6-2-2 Supporting Requirements for HLR-EXT-A

All potential other hazards (i.e., all other internal or external hazards) that may affect the site shall be identified (HLR-EXT-A).

Index No. EXT-A	Requirement
EXT-A1	In the list of hazards, INCLUDE as a minimum those that are enumerated in the PRA Procedures Guide, NUREG/CR-2300 [6-1] and in NUREG-1407 [6-3] and examined in past studies such as NUREG-1150 [6-4]. Nonmandatory Appendix 6-A contains the list adapted from NUREG/CR-2300, and this list provides one acceptable way to meet this requirement.
Commentary: None	
EXT-A2	INCLUDE any site-specific and plant-unique hazards with the list considered in Requirement EXT-A1.
Commentary: The purpose of this requirement is to ensure that an unusual type of hazard is not inadvertently omitted simply because it does not fit into any of the list of hazards commonly considered and listed in the standard references in Requirement EXT-A1. Examples are possible detritus or zebra mussels growth in the river affecting the intake (although they may be considered to have been included in the category "biological events"), or possible shoreline-slump effects (although they may be considered to have been included under "landslide or seiche").	

Table 6-2-3 Supporting Requirements for HLR-EXT-B

Preliminary screening, if used, shall be performed using a defined set of screening criteria (HLR-EXT-B).

Index No. EXT-B	Requirement
EXT-B1	Initial Preliminary Screening: For screening out a hazard other than internal events, internal flood, internal fire, and seismic events, USE any of the following five screening criteria, each of which provides an acceptable basis: *Criterion 1:* The hazard is of equal or lesser damage potential than the hazards for which the plant has been designed. This screening out requires an evaluation of plant design bases in order to estimate the resistance of plant structures and systems to a particular hazard. *Criterion 2:* The hazard has a significantly lower mean frequency of occurrence than another hazard, taking into account the uncertainties in the estimates of both frequencies, and the hazard could not result in worse consequences than the consequences from the other hazard. *Criterion 3:* The hazard cannot occur close enough to the plant to affect it. This criterion must be applied taking into account the range of magnitudes of the hazard for the recurrence frequencies of interest. *Criterion 4:* The hazard is included in the definition of another hazard. *Criterion 5:* The hazard is slow in developing, and it can be demonstrated that there is sufficient time to eliminate the source of the threat or to provide an adequate response.
Commentary: *[This commentary has been deleted.]*	
EXT-B2	Second Preliminary Screening: For screening out an external hazard other than seismic events, USE the following screening criterion, if applicable. The criterion is that the design basis for the hazard meets the criteria in the NRC Standard Review Plan [6-2]. JUSTIFY any screening out of an external hazard based *solely* on conformance to SRP.
Commentary: In some past PRAs, certain external hazards were screened out on the grounds that the design basis for the hazard event met the NRC Standard Review Plan [6-2]. For certain external hazards, the SRP requires the selection of the design-basis event at annual frequencies of occurrence between 10^{-7} and 10^{-6} [e.g., design-basis tornado following Regulatory Guide (RG) 1.76, design-basis explosions on transportation routes near the plant following RG 1.91, and turbine missile protection per RG 1.112]. For other events, conservative maximum sizes or intensities are specified (e.g., design-basis flooding per RG 1.59). Based on current information, the design-basis flooding evaluation per RG 1.59 needs to be reassessed for any screening out of the external flooding in the PRA. It is expected that the analyst performs a review of any changes in the site environs (see Requirement EXT-B4) to confirm that the data and models used in the selection of design-basis event per SRP are still valid. Therefore, the PRA analyst should use caution when screening out an external hazard based *solely* on conformance to the SRP.	
EXT-B3	BASE the application of the screening criteria for a given hazard on a review of information on the plant's design hazard and licensing basis relevant to that hazard.
Commentary: In the siting and plant design stage, most site-specific natural and man-made hazards will have been addressed and included in the design basis, unless they were screened out using the licensing criteria described in the NRC Standard Review Plan and Regulatory Guides.	

Table 6-2-3 Supporting Requirements for HLR-EXT-B (Cont'd)

Preliminary screening, if used, shall be performed using a defined set of screening criteria (HLR-EXT-B).

Index No. EXT-B	Requirement
EXT-B4	REVIEW any significant changes since the NRC operating license was issued. In particular, review all of the following: *(a)* military and industrial facilities within 8 km of the site *(b)* on-site storage or other activities involving hazardous materials *(c)* nearby transportation *(d)* any other developments that could affect the original design conditions
Commentary: This short list [(a), (b), and (c)] is specifically identified because it represents the most common areas where a significant change might have occurred since the issuance of the operating license. The 8-km distance is defined in the NRC Standard Review Plan [6-2].	

Table 6-2-4 Supporting Requirements for HLR-EXT-C

A bounding or demonstrably conservative analysis, if used for screening, shall be performed using defined quantitative screening criteria (HLR-EXT-C).

Index No. EXT-C	Requirement
EXT-C1	For screening out a hazard other than internal events, internal flood, internal fire, and seismic events, USE either of the following two screening criteria, each of which provides an acceptable basis for bounding analysis or demonstrably conservative analysis: *Criterion A:* The current design-basis hazard event has a mean frequency $<10^{-5}$/yr, and the mean value of the conditional core damage probability (CCDP) is assessed to be $<10^{-1}$. *Criterion B:* The core damage frequency, calculated using a bounding or demonstrably conservative analysis, has a mean frequency $<10^{-6}$/yr.
Commentary: The bounding or demonstrably conservative analysis is intended to provide a conservative calculation showing, if true, either that the hazard would not result in core damage or that the core damage frequency (CDF) is acceptably low. Some or all of the key elements of the external-hazard risk analysis could be used to reach and support this conclusion: hazard analysis, fragility analysis, or systems analysis (plant-systems analysis, human-reliability analysis, accident-sequence analysis, etc.). In some cases, Criterion A can allow an efficient way to verify that the original design-basis hazard (frequency) is low and that the CDF is also acceptably low. Using Criterion A requires a refined modeling of the hazard and an approximate evaluation of conditional core damage probability (CCDP). The numerical screening values in Criteria A and B are set low enough so that if either of them is met, the external hazard can be screened out.	
EXT-C2	ESTIMATE the mean frequency and the other parameters of the design-basis hazard, or the bound on them, by using hazard modeling and recent data (e.g., annual maximum wind speeds at the site, aircraft activity in the vicinity, or precipitation data).
Commentary: The spirit of a bounding or demonstrably conservative analysis is such that it is acceptable to use demonstrably conservative modeling and data for the hazard evaluation. Evaluation of the uncertainties in both modeling and data is part of the needed analysis. Although the bounding or demonstrably conservative analysis is the minimum requirement here, if the mean-frequency approach is used, it should be so stated by the analyst in the documentation for clarity and to allow review.	
EXT-C3	In estimating the mean conditional core damage probability (CCDP), USE a bounding analysis or a demonstrably conservative analysis that employs a systems model of the plant that meets the systems-analysis requirements in Part 2 insofar as they apply.
Commentary: None	

Table 6-2-4 Supporting Requirements for HLR-EXT-C (Cont'd)

A bounding or demonstrably conservative analysis, if used for screening, shall be performed using defined quantitative screening criteria (HLR-EXT-C).

Index No. EXT-C	Requirement
EXT-C4	IDENTIFY those SSCs required to maintain the plant in operation or that are required to respond to an initiating event to prevent core damage, that are vulnerable to the hazard. DETERMINE the failure modes for those SSCs.
Commentary: None	
EXT-C5	CALCULATE the CCDP taking into account the initiating events caused by the hazard, and the systems or functions rendered unavailable. Modifying the internal-events PRA model as appropriate, using conservative assessments of the impact of the hazard (fragility analysis), is an acceptable approach.
Commentary: None	
EXT-C6	ESTIMATE the mean core damage frequency by using models and data that are either realistic or demonstrably conservative. This includes not only the hazard analysis but also any fragility analysis that is applicable.
Commentary: Calculation of this CDF may be done using different demonstrably conservative assumptions, as explained by the following example. Typically, nuclear power plants are sited such that the accidental impact of plant structures by aircraft is highly unlikely. As part of the hazard PRA, the risk from aircraft accidents may be assessed at different levels. The mean annual frequency of aircraft impact during takeoff, landing, or in flight may be determined. If this hazard frequency is very low (e.g., 10^{-7}/yr), then the aircraft impact as a hazard may be eliminated from further study. This approach assumes that the aircraft impact results in damage of the structure leading to core damage or large early release (this assumption is likely to be highly conservative). If the frequency of aircraft impacting the plant structures is estimated to be larger, the fragility of the structures may be evaluated to make a refined estimate of the frequency of core damage. Further refinements could include *(a)* eliminating certain structural failures as not resulting in core damage (e.g., damage of diesel generator building may not result in core damage if off-site electrical power is available) *(b)* performing a plant-systems and accident-sequence analysis to calculate the CDF This example shows that for some hazards, it may be sufficient to perform only the hazard analysis; for some others, the hazard analysis and a simple fragility analysis may be needed; in rare cases, a plant-systems and accident-sequence analysis may be necessary. Other examples of bounding (demonstrably conservative) analysis can be found in references [6-4], [6-6], [6-7], and [6-8]. As indicated in the commentary for Requirement EXT-C1, the numerical screening criteria are set low enough so that if any of them is met using either realistic or conservative analysis, the hazard can be screened out.	
EXT-C7	If none of the screening criteria in this entire Part 6 can be met for a given hazard, then PERFORM additional analysis. (See Parts 7, 8, and 9.)
Commentary: None	

Table 6-2-5 Supporting Requirements for HLR-EXT-D

The basis for the screening out of a hazard shall be confirmed through a walkdown of the plant and its surroundings (HLR-EXT-D).

Index No. EXT-D	Requirement
EXT-D1	CONFIRM the basis for the screening out of a hazard through a walkdown of the plant and its surroundings.
Commentary: The general hazards-screening walkdown should concentrate, although not exclusively, on outdoor facilities that could be affected by high winds and flooding, on-site storage of hazardous materials, and off-site developments such as increased usage of new airports/airways, highways, and gas pipelines.	
EXT-D2	If the screening out of any specific hazard depends on the specific plant layout, then CONFIRM that layout with a walkdown. For most hazards, this confirmation typically necessitates a walkdown that evaluates the site layout outside the plant buildings as well as inside.
Commentary: None	

Table 6-2-6 Supporting Requirements for HLR-EXT-E

Documentation of the screening out of a hazard shall be consistent with the applicable supporting requirements (HLR-EXT-E).

Index No. EXT-E	Requirement
EXT-E1	DOCUMENT the hazard screening and conservative analyses in a manner that facilitates PRA applications, upgrades, and peer review.
EXT-E2	DOCUMENT the process used in the hazard screening and conservative analyses. For example, this documentation typically includes a description of *(a)* the approach used for the screening (preliminary screening or demonstrably conservative analysis) and the screening criteria used for each hazard that is screened out *(b)* any engineering or other analysis performed to support the screening out of a hazard or in the conservative assessment of a hazard

Section 6-3
Peer Review for Screening and Conservative Analysis

6-3.1 PURPOSE

This Section provides requirements for peer review of screening and conservative analyses of hazards.

6-3.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATIONS

The peer review team shall have knowledge and collective experience in the areas of systems engineering, evaluation of the hazards being considered for screening or conservative analysis, and evaluation of how the hazards being considered for screening or conservative analysis could damage the nuclear plant's SSCs, as applicable to the scope of the review. Section 1-6 provides general requirements for peer review. Subsection 1-6.2 specifies requirements for peer review team knowledge and collective experience. Paragraph 1-6.1.1 specifies requirements regarding peer review scope.

6-3.3 REVIEW TECHNICAL ELEMENTS TO CONFIRM THE METHODOLOGY

The peer review shall focus on the potential for the hazard to cause core damage and/or large early release.

The peer review team shall evaluate whether the hazard information is appropriately specific to the site and has met the relevant requirements of this Standard.

The peer review team shall evaluate whether the basis for applying any deterministic and/or quantitative screening criteria is appropriately specific to the site and has met the relevant requirements of this Standard.

The peer review team shall evaluate whether the plant initiating events postulated to be caused by the hazard are properly identified, the SSCs are properly modeled, and any accident sequences considered are properly quantified.

The peer review team shall review the walkdown of the plant in order to ensure the validity of the findings of the analysis in terms of screening, any spatial interactions, and the identification of critical failure modes.

The peer review team shall evaluate whether the quantification method used in the screening analysis is appropriate and provides all of the results and insights needed for risk-informed decisions. The peer review team shall review the validity of the screening assumptions.

Section 6-4
References

[6-1] J. Hickman et al., "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Report NUREG/CR-2300, American Nuclear Society, Institute of Electrical and Electronic Engineers, and U.S. Nuclear Regulatory Commission (1983)

[6-2] "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," Report NUREG-0800, formerly issued as NUREG-75/087, U.S. Nuclear Regulatory Commission (1975)

[6-3] "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Report NUREG-1407, U.S. Nuclear Regulatory Commission (1991)

[6-4] J. A. Lambright, M. P. Bohn, S. L. Daniel, J. J. Johnson, M. K. Ravindra, P. S. Hashimoto, M. J. Mraz, W. H. Tong, and D. A. Brosseau, "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events," Report NUREG/CR-4550, Vol. 4, Rev. 1, Part 3, Sandia National Laboratories and U.S. Nuclear Regulatory Commission (1990)

[6-5] M. K. Ravindra and A. M. Nafday, "State-of-the-Art and Current Research Activities in Extreme Winds Relating to Design and Evaluation of Nuclear Power Plants," Report NUREG/CR-5497, UCID-21933-Rev-1, Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission (1990)

[6-6] M. K. Ravindra and H. Bannon, "Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development," Report NUREG/CR-4839, Sandia National Laboratories and U.S. Nuclear Regulatory Commission (1992)

[6-7] M. K. Ravindra and H. Bannon, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP) Volume 7: External Event Scoping Quantification," Report NUREG/CR-4832/7 of 10, Sandia National Laboratories and U.S. Nuclear Regulatory Commission (1992)

[6-8] C. Y. Kimura and R. J. Budnitz, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Report NUREG/CR-5042, Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission (1987)

NONMANDATORY APPENDIX 6-A
LIST OF HAZARDS REQUIRING CONSIDERATION

Adapted from NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants" [6-A-1]

Hazard Group [Note (1)]	Hazard	Applicable Screening Criteria: Requirement EXT-B1 Describes These Five Criteria [Note (2)]	Remarks [Notes (3), (4)]
Biological events	Biological events	1, 5	Includes events such as detritus and zebra mussels.
External fire	Forest fire	1, 3	Fire cannot propagate to the site because the site is cleared; plant design and fire-protection provisions are adequate to mitigate the effects.
	Grass fire	1, 3	Fire cannot propagate to or on the site because site is cleared; plant design and fire-protection provisions are adequate to mitigate the effects.
	Nonsafety building fire	1, 3	Fire cannot propagate to safety areas of plant; separation, plant design, and fire-protection provisions are adequate to mitigate the effects.
Extraterrestrial events	Meteorite or satellite strikes	2	Can be excluded for all sites.
Extreme temperature	Frost	1	Snow and ice govern.
	High summer temperature	1	Can often be excluded where the ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms. Evaluation is needed of possible loss of air cooling due to high temperatures.
	Ice cover	1, 4	Ice blockage of river included in flood; loss of cooling-water flow is considered in plant design.

Hazard Group [Note (1)]	Hazard	Applicable Screening Criteria: Requirement EXT-B1 Describes These Five Criteria [Note (2)]	Remarks [Notes (3), (4)]
Extreme temperature (Cont'd)	Low winter temperature	1, 5	Thermal stresses and embrittlement are usually insignificant or covered by design codes and standards for plant design; generally, there is adequate warning of icing on the ultimate heat sink so that remedial action can be taken.
Ground shifts	Avalanche	3	Can be excluded for most sites in the United States.
	Coastal erosion	4, 5	Included in the effects of external flooding.
	Landslide	3	Can be excluded for most nuclear plant sites in the United States; confirm through walkdown.
	Sinkholes	1, 5	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard.
	Soil shrink–swell	1, 5	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard.
Heat-sink effects	Drought	1, 5	Can often be excluded where there are multiple sources of ultimate heat sink or where the ultimate heat sink is not affected by drought (e.g., cooling tower with adequately sized basin).
	Low lake or river water level	1, 5	Can often be excluded where the ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms.
	River diversion	1, 4	Considered in the evaluation of the ultimate heat sink; should diversion become a hazard, adequate storage is usually provided. Requires detailed site or plant study.
Heavy-load drop	Heavy-load drop	. . .	Site specific; requires detailed study.

Hazard Group [Note (1)]	Hazard	Applicable Screening Criteria: Requirement EXT-B1 Describes These Five Criteria [Note (2)]	Remarks [Notes (3), (4)]
High winds	Extreme winds and tornadoes	. . .	Site specific; requires detailed study.
	Hail	1	Other missiles govern.
	Hurricane	4	Included under external flooding; wind forces are covered under extreme winds and tornadoes.
	Sandstorm	1, 4	Included under extreme winds and tornadoes; potential blockage of air intakes with particulate matter is generally considered in plant design.
Industrial accidents	Industrial or military facility accident	. . .	Site specific; requires detailed study.
	Pipeline accident	. . .	Site specific; requires detailed study.
	Release of chemicals from on-site storage	. . .	Plant specific; requires detailed study.
	Toxic gas	. . .	Site specific; requires detailed study.
Lightning	Lightning	1	Considered in plant design.
Site flooding	External flooding	. . .	Site specific; requires detailed study.
	High tide	4	Included under external flooding.
	Precipitation, intense	4	Included under external and internal flooding. Roof loading and its effect on building integrity must be checked.
	Seiche	4	Included under external flooding.
	Storm surge	4	Included under external flooding.
	Tsunami	4	Included under external flooding and seismic events.
	Waves	4	Included under external flooding.
Snow	Snow	1, 4	Plant designed for higher loading; snowmelt causing river flooding is included under external flooding.

Hazard Group [Note (1)]	Hazard	Applicable Screening Criteria: Requirement EXT-B1 Describes These Five Criteria [Note (2)]	Remarks [Notes (3), (4)]
Transportation accidents	Aircraft impacts	. . .	Site specific; requires detailed study.
	Fog	1	Could increase the frequency of man-made hazard involving surface vehicles or aircraft; accident data include the effects of fog.
	Ship impact	. . .	Site specific; requires detailed study.
	Vehicle impact	. . .	Plant specific; requires detailed study.
	Vehicle or ship explosion	. . .	Plant specific; requires detailed study.
Turbine-generated missiles	Turbine-generated missiles	1, 2	Plant specific; requires detailed study.
Volcanic activity	Volcanic activity	3	Can be excluded for most sites in the United States.

NOTE:
(1) In accordance with the limitation noted in 1-1.2, the occurrence of any listed hazard that results from sabotage or terrorism is excluded from consideration.
(2) The criteria indicated are those that have commonly been applied successfully in screening the listed hazards. It should not be assumed that the hazard can always be screened out with the indicated criteria or that other criteria would not apply.
(3) The screening guidance provided here only addresses screening out of hazards using the criteria in Requirement EXT-B1 (and Requirement EXT-B2, if applicable). The remark "Site specific; requires detailed study" should not be taken to imply that PRA following the requirements in Parts 7 through 9 of this Standard is required. Rather, detailed study could be limited to showing that the hazard can be screened out using the criteria in Requirement EXT-C1.
(4) The idea behind the screening remark that something is screened because it is "included under" or "covered by" another hazard is that it is not evaluated separately but is inherently included in another data set. Taking hurricanes as an example, the data set used for external flooding would be expected to include historic site levels for all causes upon which a site level hazard would be developed. This hazard would be "cause independent." Similarly for winds, the wind data set would include winds from all causes (based on daily or hourly meteorological station readings, again independent of cause).

Section 6-A-1
References

[6-A-1] J. Hickman et al., "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Report NUREG/CR-2300, American Nuclear Society, Institute of Electrical and Electronic Engineers, and U.S. Nuclear Regulatory Commission (1983)

PART 7 (a)
REQUIREMENTS FOR HIGH-WIND EVENTS AT-POWER PRA (b)

Section 7-1
Overview of High-Wind At-Power PRA Requirements

7-1.1 PRA SCOPE

This Part establishes technical requirements for a Level 1 and large early release frequency (LERF) analysis of the high-wind hazard group while at-power.

7-1.2 COORDINATION WITH OTHER PARTS OF THE STANDARD

This Part is intended to be used together with Parts 1 and 2 of this Standard. A PRA developed in accordance with Part 2 is the starting point for assessing the conditional core damage probability and conditional large early release probability.

7-1.3 HIGH-WIND EVENTS SCOPE

There are several types of high-wind events that need to be considered, depending on the site. These include

(a) tornado winds and other tornado effects

(b) tropical cyclone winds (cyclones, hurricanes, and typhoons)

(c) extratropical straight winds (thunderstorms, squall lines, weather fronts, etc.)

It is assumed that the analyst team has employed screening methods (see Part 6) to eliminate from consideration those high-wind events that are not important at the site under study, so that the requirements in this Part will be used to analyze only those high-wind phenomena that have not been screened out.

If it has been decided that the only effect on the plant, from a particular wind hazard, is to induce a loss of offsite power, and that has been incorporated into the model for internal events, then that wind hazard need not be addressed using this Part.

Section 7-2
Technical Requirements for High-Wind Events At-Power PRA

It should be noted that PRA of high winds has been carried out for several U.S. nuclear power plants, and in a few cases it involved detailed analysis. Also, the hazard and plant analysis carried out during the design stage provide a basis for the screening analyses and demonstrably conservative analyses using the approaches in Part 6. These approaches have usually shown that the contribution of high winds (other than those resulting in losses of offsite power) to CDF is insignificant. Therefore, the collective experience with high-winds PRA is limited. Because of this limited experience, the analyst team may need to improvise its approach to high-winds PRA analysis following the overall methodology requirements in this Part.

The technical requirements for high-winds PRA are similar, with adaptations, to those for seismic PRA. The major elements are wind hazard analysis, wind fragility analysis, and plant response analysis including quantification. The analyst should refer to Nonmandatory Appendix 5-A ("Seismic Probabilistic Risk Assessment Methodology: Primer").

It is further assumed here that the high-winds-PRA team possesses an internal-events, at-power Level 1 and Level 2 LERF PRA, developed either prior to or concurrently with the high-winds PRA; that this internal-events PRA is used as the basis for the high-winds-PRA systems model; and that the technical basis for the internal-events, at-power PRA is Part 2.

References that are useful in developing a high-winds PRA include [7-1], [7-2], and [7-3] through [7-6]. The relevant references for wind-hazard analysis are provided in the commentary below adjacent to the relevant wind hazard technical requirements (7-2.1).

The high-winds-PRA technical requirements consist of four high-level requirements, under which are organized the several supporting technical requirements.

Although the usual format for these supporting technical requirements is to organize them into three "Capability Categories," there are no entries here for Capability Category I. Before applying the requirements in this Part, the analyst has presumably subjected the "high-winds" hazard to a screening analysis following the requirements in Part 6, but it was not possible to screen it out. Therefore, it is necessary to perform a more detailed analysis using the requirements in this Part. In this version of the Standard, it is assumed for many SRs that if a more detailed analysis of this hazard group is needed, then the analyst will desire that the PRA have capability at least corresponding to Capability Category II and perhaps even to Capability Category III for some issues. In these cases, the Capability Category I requirements are not defined. Some SRs call for the use or adaptation of the internal-events PRA. In these cases, it is recognized that portions of the PRA adopted for the plant response model used will correspond to the associated Capability Category from Part 2.

Rationale and Structure of the Requirements: There are three technical elements in the PRA of a high-wind hazard. They are described briefly below:

(a) High-Wind Hazard Analysis (WHA). This element involves the evaluation of the frequency of occurrence of different intensities of high winds based on a site-specific probabilistic evaluation reflecting recent available data and site-specific information.

(b) High-Wind Fragility Evaluation (WFR). This element evaluates the fragilities of the SSCs as a function of the intensity of the high wind using plant-specific, SSC-specific information and an accepted engineering method for evaluating the postulated failure.

(c) High-Wind Plant Response Model (WPR). This element develops a plant response model that addresses the initiating events and other failures resulting from the effects of high wind that can lead to core damage or large early release. The model is based on the internal-events, at-power PRA model to incorporate those aspects that are different, due to the effects of high wind, from the corresponding aspects of the internal-events, at-power model. The conditional CDF and LERF obtained from this model is combined with the frequency of the plant damage states obtained by convoluting the wind hazard and wind fragility curves to estimate the unconditional CDF and LERF.

7-2.1 HIGH-WIND HAZARD ANALYSIS (WHA)

The objective of the hazard analysis is to assess the frequency of occurrence of high wind as a function of intensity on a site-specific basis.

Table 7-2.1-1 High Level Requirements for Wind Hazard Analysis (WHA)

Designator	Requirement
HLR-WHA-A	The frequency of high winds at the site shall be based on site-specific probabilistic wind hazard analysis (existing or new) that reflects recent available regional and site-specific information. Uncertainties in the models and parameter values shall be properly accounted for and fully propagated in order to obtain a family of hazard curves from which a mean hazard curve can be derived.
HLR-WHA-B	Documentation of the wind hazard analysis shall be consistent with the applicable supporting requirements.

Table 7-2.1-2 Supporting Requirements for HLR-WHA-A

The frequency of high winds at the site shall be based on a site-specific probabilistic wind hazard analysis (existing or new) that reflects recent available regional and site-specific information. Uncertainties in the models and parameter values shall be properly accounted for and fully propagated to obtain a family of hazard curves from which a mean hazard curve can be derived (HLR-WHA-A).

Index No. WHA-A	Capability Category I	Capability Category II	Capability Category III
WHA-A1 [Note (1)]	Not Defined	In the tornado wind hazard analysis, USE the state-of-the-art methodology and up-to-date databases on tornado occurrences, intensities, etc. PROPAGATE uncertainties in the models and parameter values to obtain a family of hazard curves from which a mean hazard curve can be derived.	
WHA-A2 [Note (2)]	Not Defined	In evaluating the hazard from hurricanes, USE the state-of-the-art hurricane hazard analysis methodology and up-to-date databases on hurricane occurrences, intensities, etc. PROPAGATE uncertainties in the models and parameter values in order to obtain a family of hazard curves from which a mean hazard curve can be derived.	
WHA-A3 [Note (3)]	Not Defined	In evaluating the hazard from extratropical windstorms and other high straight wind phenomena, USE recorded wind-speed data appropriate to the site.	
WHA-A4 [Note (4)]	Not Defined	EVALUATE the hazard from wind-generated missiles by using a high-wind missile hazard analysis methodology. In this evaluation, EXAMINE specific features of exterior barriers (i.e., walls and roofs) of safety-related structures, any weather-exposed SSCs, and the consequences of this damage from wind-borne missile impact that may result in core damage or large early release.	
WHA-A5	Not Defined	SURVEY the plant building and surroundings to assess the number, types, and locations of potential missiles.	SURVEY the plant building and surroundings to assess and catalog the number, types, and locations of potential missiles.

Table 7-2.1-2 Supporting Requirements for HLR-WHA-A (Cont'd)

GENERAL NOTE: The models used for frequency and intensity calculations can sometimes be unduly influenced by recent, short-term trends in the frequencies of high-wind events. Such influence could result in erroneous results. It is the analyst's responsibility to demonstrate that the model is not biased by this issue. One acceptable approach is to incorporate at least the worst weather conditions experienced historically at the site or in the region, when applicable.

NOTES:

(1) State-of-the-art methodologies are given by references [7-7] and [7-8]. Examples of tornado hazard analysis for nuclear facilities using these methodologies can be found in references [7-9], [7-10], and [7-11].

Tornado wind hazard analysis typically includes the following elements:

(a) variation of tornado intensity with occurrence frequency (the frequency of tornado occurrence decreases rapidly with increased intensity)

(b) correlation of tornado width and length of damage area; longer tornadoes are usually wider

(c) correlation of tornado area and intensity; stronger tornadoes are usually larger than weaker tornadoes

(d) variation in tornado intensity along the damage path length; tornado intensity varies throughout its life cycle

(e) variation of tornado intensity across the tornado path width

(f) variation of tornado differential pressure across the tornado path width

(2) In the United States, hurricanes predominantly affect the Gulf of Mexico and the Atlantic coastline. Hurricanes rapidly decay during their movement over land because of friction from terrain. Hence, it is sufficient to consider their impact only up to a few hundred kilometers or so from the coastline, and a hurricane risk analysis is not required farther inland. However, wind hazard frequencies for a site can usually be generated from direct wind measurements at the site except for the largest recurrence intervals [7-12]. Because of the absence of direct wind measurements at many sites of interest for significant time periods, numerical simulation techniques are commonly used to generate hurricane wind hazard frequencies for a site. A stochastic model of hurricane occurrences is used, and the hazard analysis considers the occurrence rate of hurricanes for each coastal segment, distribution of central pressure, radius of maximum winds, storm decay over land, wind field characteristics, and coast crossing location. Available probabilistic models are discussed in reference [7-13].

Numerical simulations based on these models simulate the hurricane wind field using random variables that model the size, intensity, translation speed, direction, and location of the site with respect to the coastal line. The probability density functions of these variables are developed using hurricane data compiled by Batts et al. [7-14] and Jarvinen et al. [7-15].

Such a simulation procedure was used in developing the hurricane wind hazard curves for the Indian Point site [7-7].

(3) For inland sites in the United States, the hazard (i.e., annual probability of exceedance) at lower wind speeds is typically at higher annual frequencies from extratropical straight windstorms than from tornadoes or hurricanes. Therefore, the evaluation of risks from extratropical straight windstorms is needed, especially if the plant structures have not been designed to withstand tornadoes. Typically, the annual maximum wind-speed data recorded at a weather station appropriate to the site are fitted by a Type I extreme value probability distribution. Since the site-specific wind-speed data may be available over only a short period (e.g., < 50 yr), there is considerable uncertainty in the hazard, especially at higher wind speeds [7-12]. It is customary to assume that the uncertainty in the hazard comes mainly from the sampling error due to the small number and duration of records. (See reference [7-16].) This standard deviation is taken into account to obtain a family of hazard curves with assigned subjective probabilities (e.g., reference [7-6]). Other uncertainties that arise from lack of weather station data near the site, terrain differences, and so on should be accounted for properly in developing the wind hazard curves.

Table 7-2.1-2 Supporting Requirements for HLR-WHA-A (Cont'd)

NOTES: (Cont'd)

(4) An acceptable method for evaluating wind-borne missile risk is given in references [7-13] and [7-17]. It models the tornado wind field, trajectory of missiles (injection and transportation), and impact effects of missiles onto safety-related buildings and exposed equipment. A survey of the plant buildings and their surroundings is made to assess the number and types of objects that could be picked up by a tornado and could become potential missiles. Using the results of the detailed tornado missile risk analysis, Reed and Ferrell [7-6] have developed missile strike probabilities per unit area of buildings. Note that tornado missile risk is judged to be acceptably small if the plant design meets the 1975 NRC Standard Review Plan Criteria [7-18]. Note also that wind-generated missiles from other high-wind phenomena (hurricanes, etc.) can be analyzed using an adaptation of the tornado-missile method.

Table 7-2.1-3 Supporting Requirements for HLR-WHA-B

Documentation of the wind hazard analysis shall be consistent with the applicable supporting requirements.

Index No. WHA-B	Capability Category I	Capability Category II	Capability Category III
WHA-B1	DOCUMENT the wind hazard analysis manner that facilitates PRA applications, upgrades, and peer review.		
WHA-B2	DOCUMENT the process used to identify wind hazards. For example, this documentation typically includes a description of *(a)* the specific methods used for determining the high-wind hazard curves *(b)* the associated wind pressure, pressure distributions, missile and differential pressure effects *(c)* the scientific interpretations that are the basis for the inputs and results		
WHA-B3	DOCUMENT the sources of model uncertainty and related assumptions associated with the wind hazard analysis.		

7-2.2 HIGH-WIND FRAGILITY ANALYSIS (WFR)

The objective of the fragility analysis is to identify those SSCs that are susceptible to the effects of high winds and to determine their plant-specific failure probabilities as a function of the intensity of the wind.

Table 7-2.2-1 High Level Requirements for Wind Fragility Analysis (WFR)

Designator	Requirement
HLR-WFR-A	A wind fragility evaluation shall be performed to estimate plant-specific, realistic wind fragilities for those SSCs whose failure contributes to core damage or large early release, or both.
HLR-WFR-B	Documentation of the wind fragility analysis shall be consistent with the applicable supporting requirements.

Table 7-2.2-2 Supporting Requirements for HLR-WFR-A

A wind fragility evaluation shall be performed to estimate plant-specific, realistic wind fragilities for those SSCs whose failure contributes to core damage or large early release, or both (HLR-WFR-A).

Index No. WFR-A	Capability Category I	Capability Category II	Capability Category III
WFR-A1 [Note (1)]	Not Defined	In evaluating wind fragilities of structures and components (e.g., tanks, transformers, diesel-generator exhaust stack, piping, and intake pumps), USE plant-specific data. In the assessment, INCLUDE nonsafety structures that could fall into/onto safety-related structures, thereby causing damage. In this evaluation, INCLUDE the findings of a plant walkdown.	
WFR-A2	IDENTIFY plant SSCs that are vulnerable to the wind hazards. INCLUDE both wind effect and wind-borne missiles effect.		

NOTE:

(1) Wind fragility is evaluated using the same general methodology as for seismic fragilities. (See the requirements in 5-2.2 for seismic-fragility evaluation and the seismic-fragility discussion in Nonmandatory Appendix 5-A). Typically, the entire family of fragility curves for an SSC corresponding to a particular failure mode is expressed in terms of the median wind-speed capacity, V_m, and the logarithmic standard deviations, and β_R and β_U, representing randomness in capacity and uncertainty in median capacity, respectively. Such fragility parameters are estimated for the credible failure modes of the SSC. Failure of structures could be overall, such as failure of a shearwall or moment resisting frame, or local, such as out-of-plane wall failure or pull-off of metal siding.

Wind pressure loading is based on the methodology contained in wind design standards [7-19]. The effect of wind-borne missiles on SSCs can be found in references [7-20] and [7-21].

The development of fragility curves for structures is done in terms of the factor-of-safety, defined as the resistance capacity divided by the response associated with the design-basis loads from extreme winds. The variability of the factor-of-safety depends on the variability of strength capacity and the response to specified loads. Wind capacity is modeled as a product of random variables and is expressed in terms of wind speed. Besides the strength characteristics, the capacity of a structure for the effects of wind pressure also depends on a number of factors affecting wind pressure/force relationship.

For example, shielding effects of various structures at the site results in an increase of wind speed through a constricted space or a decrease where it may be slowed down due to obstructions. Such funneling characteristics describing the channeling of winds around structures have a very important influence on the wind forces. The actual forces are also determined by the structural shapes because wind pressure and forces are related to the wind velocity by a shape factor. Another factor important in this regard is the vertical distribution of wind velocity, which is a function of terrain roughness. Examples of the development of wind fragilities for structures can be found in references [7-3], [7-4], and [7-6].

Most nuclear power plant structures have excellent wind resistance. Major vulnerabilities have sometimes been identified for nonseismic Capability Category II structures due to their potential for collapsing on safety-related structures or equipment. These structures include exhaust stacks, unprotected walls, outside wiring and cabling, etc. Similarly, many older plants have safety-related equipment such as tanks and equipment located outdoors that are vulnerable to wind-borne missiles. They should be identified during the walkdown.

In analyzing the failure of indoor equipment (within the structures), it is conservatively assumed that the failure of a structure causes the failure of all equipment dependent on or within the structure. It is possible that the structure may not collapse, but the indoor equipment may still be damaged from pressure drop due to passage of a tornado. This damage occurs because of inadequate venting in the structure. There is a rapid pressure drop due to passage of a tornado, and this results in escape of air from the building; if the exit is not rapid enough, it causes internal pressure. This pressure might lead to failure of block walls, which could collapse onto safety-related structures. Indoor equipment is also susceptible to damage from missiles entering through louvres, vents, etc. Damage to internal SSCs may also be caused by wind-induced pressurization through openings in the structure.

Table 7-2.2-3 Supporting Requirements for HLR-WFR-B

Documentation of the wind fragility analysis shall be consistent with the applicable supporting requirements.

Index No. WFR-B	Capability Category I	Capability Category II	Capability Category III
WFR-B1	DOCUMENT the wind fragility analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
WFR-B2	DOCUMENT the process used in the wind fragility analysis. For example, this documentation typically includes a description of *(a)* the methodologies used to quantify the high-wind fragilities of SSCs, together with key assumptions *(b)* a detailed list of SSC fragility values that includes the method of analysis, the dominant failure mode(s), the sources of information, and the location of each SSC *(c)* the basis for the screening out of any generic high-capacity SSCs		
WFR-B3	DOCUMENT the sources of model uncertainty and related assumptions associated with the wind fragility analysis.		

7-2.3 HIGH-WIND PLANT RESPONSE MODEL (WPR)

The objectives of this element are to

(a) develop a wind plant response model by modifying the internal-events, at-power PRA model to include the effects of the wind in terms of initiating events and failures caused

(b) quantify this model to provide the conditional core damage probability (CCDP) and conditional large early release probability (CLERP) for each defined wind plant damage state

(c) evaluate the unconditional CDF and LERF by integrating the CCDP/CLERP with the frequencies of the plant damage states obtained by combining the wind hazard and wind fragility analyses

Table 7-2.3-1 High Level Requirements for High-Wind Plant Response Model and Quantification (WPR)

Designator	Requirement
HLR-WPR-A	The high-wind PRA systems model shall include wind-caused significant initiating events, and other failures that are significant contributors, that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate wind-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model.
HLR-WPR-B	The analysis to quantify core damage and large early release frequencies shall appropriately integrate the wind hazard, the wind fragilities, and the systems-analysis aspects.
HLR-WPR-C	Documentation of the high-wind plant response model development and quantification shall be consistent with the applicable supporting requirements.

Table 7-2.3-2 Supporting Requirements for HLR-WPR-A

The high-wind PRA systems model shall include wind-caused significant initiating events, and other failures that are significant contributors, that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate wind-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model.

Index No. WPR-A	Capability Category I	Capability Category II	Capability Category III
WPR-A1 [Note (1)]	ENSURE that wind-caused initiating events that give rise to significant accident sequences and/or significant accident progression sequences are included in the wind PRA system model using a systematic process.		
WPR-A2	USE the event trees and fault trees from the internal-events, at-power PRA model as the basis for the high-wind accident sequence analysis.		
WPR-A3 [Note (2)]	ENSURE that the PRA systems models reflect wind-caused failures as well as other unavailabilities and human errors that give rise to significant accident sequences or significant accident progression sequences.		
WPR-A4 [Note (3)]	In each of the following aspects of the high-wind PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. SPECIFY a defined basis to support the claimed non-applicability of any exceptions. The aspects governed by this requirement are *(a)* initiating-event analysis *(b)* accident-sequence analysis *(c)* success-criteria analysis *(d)* systems analysis *(e)* data analysis *(f)* human-reliability analysis *(g)* use of expert judgment When the Part 2 requirements are used, USE the Capability Category designations in Part 2 and, for consistency, USE the same Capability Category in this analysis.		
WPR-A5 [Note (4)]	In the human reliability analysis (HRA) aspect, EVALUATE additional stresses that can increase the likelihood of human errors or inattention, compared to the likelihood assigned in the internal events HRA when the same activities are undertaken in non–high-wind-event accident sequences. Whether or not increases in error probabilities are used, JUSTIFY the basis for this decision about what error rates to use.		
WPR-A6	If any screening is performed, PERFORM it using defined criteria that are documented in the PRA.		
WPR-A7 [Note (5)]	PERFORM an analysis of wind-hazard-caused dependencies and correlations in a way so that any screening out of SSCs appropriately includes those dependencies.		

Table 7-2.3-2 Supporting Requirements for HLR-WPR-A (Cont'd)

The high-wind PRA systems model shall include wind-caused significant initiating events, and other failures that are significant contributors, that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate wind-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model.

Index No. WPR-A	Capability Category I	Capability Category II	Capability Category III
WPR-A8 [Note (6)]	ENSURE that any screening out of human-error basic events and non–wind-caused-failure basic events does not eliminate any significant accident sequences or significant accident progression sequences.		
WPR-A9 [Note (7)]	In the systems-analysis models, for each basic event that represents a wind-caused failure, INCLUDE the complementary "success" state where applicable to a particular SSC in cases where the wind-caused failure probability is high.		
WPR-A10 [Note (8)]	EVALUATE the possibility that the high wind can cause damage or plant conditions that preclude personnel access to safety equipment or controls, thereby inhibiting operator actions that might otherwise be credited.		
WPR-A11 [Note (9)]	Not Defined	EVALUATE the likelihood that system recoveries modeled in the internal-events PRA may be more complex or even not possible after a high-wind event, and ADJUST the recovery models accordingly.	

GENERAL NOTE: While the most common procedure for developing the hazard-specific PRA systems model is to start with the internal-events systems model and adapt it by adding and trimming, in some circumstances it is acceptable instead to develop an ad hoc wind-hazard-specific PRA plant response tailored especially to the situation being modeled. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects.

NOTES:

(1) It is very important that site-specific failure events, usually wind-caused structural, mechanical, and electrical failures, be thoroughly investigated. Also, multiple-unit impacts and dependencies should be considered, as appropriate, including recovery resources that could be affected by high wind.

Attention to both the core damage frequency (CDF) endpoint and the large early release frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement. This means that initiating events and SSC failures that could lead to LERF-type consequences need to be included in the systems model even if the CDF frequency is quite low.

(2) The analysis may group wind-caused failures if the leading failure in the group is modeled. The event trees and fault trees from the internal-events, at-power PRA model are generally used as the basis for the wind-initiated accident sequences/event trees. This captures the thinking that has gone into their development and assists in allowing comparisons between the internal-events PRA and the wind event PRA to be made on a common basis.

In some circumstances, it is acceptable to develop an ad hoc systems model tailored especially to the wind-hazard-specific PRA situation being modeled, instead of starting with the internal-events model and adapting it. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures.

Attention to both the core damage frequency (CDF) endpoint and the large early release frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement. This means that initiating events and SSC failures that could lead to LERF-type consequences need to be included in the systems model even if the CDF frequency is quite low.

Table 7-2.3-2 Supporting Requirements for HLR-WPR-A (Cont'd)

NOTES: (Cont'd)

(3) These Sections of Part 2 are effectively incorporated here by reference. A few aspects, however, do not apply in detail. Whenever an exception is taken, the PRA analyst team needs to be cognizant of the underlying rationale for the specific Part 2 requirement so as to ensure that this rationale is considered when the exception is taken.

(4) The human-error probabilities may be increased for some high-wind event actions, compared to the probabilities assigned in analogous internal-events-initiated sequences. The corresponding technical requirements in Part 2 should be consulted in performing the HRA aspect of a high-wind PRA.

(5) It is vital that the analysis capture the important dependencies among high-wind-caused failures, e.g. spatial or environmental dependencies. Common-cause failure analysis is important in all PRAs, but particularly for high-wind hazards.

(6) To make the systems-analysis models more manageable, some of the non-wind caused failures and human errors may be screened out of the model if their contribution to the results is demonstrably very small.

(7) For some hazards, some SSCs whose hazard-induced failure is important to safety at higher levels will not fail or will fail with only modest probability. The modeling of the non-failure (that is, the "success") of such SSCs is an important aspect of the systems model, and excluding these "success" states can lead to erroneous PRA results.

(8) This information is most effectively gathered during the walkdown, which must be structured to search for access issues. Coordination with the human-reliability–analysis aspect of the PRA is important. In making these evaluations, it may be assumed that portable lighting is available and that breathing devices are available, if in fact the plant configuration includes them.

(9) The restoration of safety functions can be inhibited by several types of causes; these include damage or failure, access problems, confusion, loss of supporting personnel to other post–wind-event recovery functions, and so on. Careful consideration of these causes must be given before recoveries are credited in the initial period after the wind hazard event. This is especially true for externally caused loss of off-site power (LOOP), given that the damage could be to switchyard components or to the off-site grid towers, which are generally difficult to fix quickly.

Table 7-2.3-3 Supporting Requirements for HLR-WPR-B

The analysis to quantify core damage and large early release frequencies shall appropriately integrate the wind hazard, the wind fragilities, and the plant response aspects (HLR-WPR-B).

Index No. WPR-B	Capability Category I	Capability Category II	Capability Category III
WPR-B1 [Note (1)]	Not Defined	EVALUATE accident sequences initiated by high winds to estimate core damage frequency and large early release frequency contribution. In the analysis, USE the site-specific wind hazard curves and the fragilities of structures and equipment.	
WPR-B2 [Note (2)]	Not Defined	In the integration-quantification, INCLUDE the uncertainties in each of the inputs and for all important dependencies and correlations.	

GENERAL NOTE: In special circumstances, it is acceptable to develop an ad hoc systems model tailored especially to the high-wind phenomenon being analyzed, instead of starting with the internal-events systems model and adapting it. If this approach is used, it is especially important that a peer review be undertaken that concentrates on this aspect. Judgment is necessary to determine the scope of this requirement. The intent is to evaluate only important initiating events.

NOTES:

(1) The wind-PRA systems-analysis model is almost always based on the internal-events, at-power PRA systems model, to which are added basic failure events derived from the information developed in the wind fragility analysis. Considerable screening out of parts of the internal-events systems model is also common, where appropriate. The analysis consists of developing event trees and fault trees in which the initiating event can be either the extreme wind effect itself or a transient or loss-of-coolant accident induced by the extreme winds. Various accident sequences that lead to core damage or large early release are identified, and their conditional probabilities of occurrence are calculated. The frequency of core damage or large early release is obtained by a convolution over the relevant range of hazard intensities.

The procedure for determining the accident sequences is similar to that used in seismic-PRA systems analysis, and following the requirements therein represents one acceptable approach, after they are adapted to apply to the wind-PRA situation. Other factors to be considered include non–wind-related unavailabilities or failures of equipment, operator errors, any warning time available to take mitigating steps (e.g., in the case of hurricanes), the possibility of recovery actions by operators and replacement by substitute equipment to accomplish the needed function, and the likelihood of common-cause failures.

Examples of systems analysis for high winds can be found in the Indian Point Individual Plant Examination of External Events (IPEEE) report [7-3] and the several so-called "TAP A-45" reports that Sandia National Laboratories performed for the U.S. Nuclear Regulatory Commission [7-5].

(2) The usefulness of the "final results" of the PRA for high winds is dependent on performing enough assessment to understand the uncertainties, dependencies, and correlations and to account for them quantitatively if they are important.

Table 7-2.3-4 Supporting Requirements for HLR-WPR-C

Documentation of the high-wind plant response model development and quantification shall be consistent with the applicable supporting requirements.

Index No. WPR-C	Capability Category I	Capability Category II	Capability Category III
WPR-C1	DOCUMENT the wind plant response analysis and quantification in a manner that facilitates PRA applications, upgrades, and peer review.		
WPR-C2	DOCUMENT the process used in the wind plant response analysis and quantification. For example, this documentation typically includes a description of *(a)* the specific adaptations made to the internal-events PRA model to produce the high-wind-PRA model, and their motivation *(b)* the final results of the PRA analysis in terms of core damage frequency and large early release frequency, as well as selected intermediate results		
WPR-C3	DOCUMENT the sources of model uncertainty and related assumptions associated with the high-wind plant response model development.		

Section 7-3
Peer Review for High-Wind At-Power PRA

7-3.1 PURPOSE

This Section provides requirements for peer review of a Level 1 and LERF at-power PRA of high-wind events.

7-3.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATIONS

The peer review team shall have knowledge and collective experience in the subjects of systems engineering, evaluation of the high-wind hazard, and evaluation of how the high-wind hazard could damage the nuclear plant's SSCs, as applicable to the scope of the review. Section 1-6 provides general requirements for peer review. Subsection 1-6.2 specifies requirements for peer review team knowledge and collective experience. Paragraph 1-6.1.1 specifies requirements regarding peer review scope.

7-3.3 REVIEW OF HIGH-WIND PRA ELEMENTS TO CONFIRM THE METHODOLOGY

7-3.3.1 High-Wind Hazard Selection

The peer review team shall evaluate whether the high-wind hazard used in the PRA is appropriately specific to the site and has met the relevant requirements of this Standard.

7-3.3.2 Wind-Induced Initiating Events

The peer review team shall evaluate whether the initiating events postulated to be caused by the high-wind events are properly identified, the SSCs are properly modeled, and the accident sequences are properly quantified.

7-3.3.3 "Fragility" Analysis Methods

The peer review team shall evaluate whether the methods and data used in the "fragility" analysis of SSCs are adequate for the purpose and meet the relevant requirements of this Standard. The review team should perform independent fragility calculations of a selected sample of SSCs covering different categories and contributions to core damage frequency and large early release frequency.

7-3.3.4 Plant Walkdown

The peer review team shall review the walkdown of the plant to ensure the validity of the findings of the analysis in terms of screening, any spatial interactions, and the identification of critical failure modes.

7-3.3.5 Quantification Method

The peer review team shall evaluate whether the quantification method used in the PRA is appropriate and provides all of the results and insights needed for risk-informed decisions. If the analysis contains screening assumptions, or assumptions that the analysis team claims to be demonstrably conservative, the peer review team shall review the validity of these assumptions. The review shall focus on the core damage frequency and large early release frequency estimates and uncertainty bounds and on the significant contributors.

Section 7-4
References

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PART 8 REQUIREMENTS FOR EXTERNAL FLOOD EVENTS AT-POWER PRA

(a)
(b)

Section 8-1 Overview of External Flood At-Power PRA Requirements

8-1.1 PRA SCOPE

This Part establishes technical requirements for a Level 1 and large early release frequency (LERF) analysis of the external flood hazard group while at-power.

8-1.2 COORDINATION WITH OTHER PARTS OF THIS STANDARD

This Part is intended to be used together with Parts 1 and 2 of this Standard. A PRA developed in accordance with Part 2 is the starting point for assessing the conditional core damage probability and conditional large early release probability.

8-1.3 EXTERNAL FLOOD EVENTS SCOPE

There are several types of external-flooding phenomena that need to be considered, depending on the site. These include both natural phenomena (high river or lake water, ocean flooding such as from high tides or wind driven storm surges, extreme precipitation, tsunamis, seiches, flooding from landslides, etc.), and man-made events (principally failures of dams, levees, and dikes). It is also important to consider rational probabilistic-based combinations of the above phenomena. The consequences of heavy rain and other flooding, such as water collected on rooftops and in low-lying plant areas, are also within the scope of this Part.

Section 8-2
Technical Requirements for External Flood Events At-Power PRA

PRA of external flooding has been carried out for several U.S. nuclear power plants, and in a few cases, it involved detailed analysis. Also, the hazard and plant analyses carried out during the design stage provide a basis for the screening analyses and demonstrably conservative analyses using the approaches in Part 6. These approaches, based on a combination of using of the recurrence intervals for the design-basis floods and analyzing the effectiveness of mitigation measures to prevent core damage, have usually shown that the contribution to CDF is insignificant.

The collective experience with PRA external-flooding analysis is limited. Because of this limited experience, and the unavailability of any detailed methodology guidance documents, the analysis team may need to improvise its approach to external-flooding analysis following the overall methodology requirements in this Part. Given the above, an extensive peer review is very important if an analysis under this Part is undertaken.

The technical requirements for external-flooding PRA including local precipitation are similar, with adaptations, to those for internal-flooding PRA and seismic PRA. The major elements of the PRA methodology are flooding hazard analysis, flooding fragility analysis (involving analysis of flooding pathways and water levels), and systems analysis including quantification. The analyst should refer to the requirements on internal flooding and seismic PRA and commentary in Parts 3 and 5, respectively, and Nonmandatory Appendix 5-A ("Seismic Probabilistic Risk Assessment Methodology: Primer"). Specifically, some aspects of external-flooding PRA, especially concerning how flooding causes the failure of SSCs, are similar to internal-flooding PRA.

Usually, it is assumed that the analyst team has employed screening methods (see Part 6) to eliminate from consideration those external-flooding phenomena that are not important at the site under study and therefore that the requirements in this Part will be used to analyze only those flooding phenomena that have not been screened out.

It is further assumed that the external-flooding-PRA analysis team possesses an internal-events, at-power Level 1 and LERF PRA, developed either prior to or concurrently with the external-flooding PRA; that this internal-events PRA is used as the basis for the external-flooding-PRA systems model; and that the technical basis for the internal-events, at-power PRA is Part 2. Fragility analysis for both capacity and demand may be based on the standard methodology used for seismic events, with appropriate modifications unique to the flooding event being studied.

As mentioned above, external-flooding risks are generally not found to be important contributors to overall risk at nuclear power plants. One major reason is that the siting requirements are intended to assure this outcome, and by and large, they have been successful in that regard (references [8-1] through [8-7]). Another key reason is that most large external floods occur only after significant warning time or over a long enough duration to allow the plant operating personnel to take appropriate steps to secure the plant and its safety-related SSCs. The PRA team is therefore urged to take as much credit for warning time and compensatory actions as the plant's planning and procedures allow (see Requirement XFPR-B1).

References [8-8], [8-9], [8-10], and [8-11] are useful in developing an external-flooding PRA.

The external-flooding PRA technical requirements consist of four high-level requirements, under which are organized several supporting technical requirements.

Although the usual format for these supporting technical requirements is to organize them into three "Capability Categories," there are no entries here for Capability Category I. Before applying the requirements of this Part, the analyst has presumably subjected the "external-flooding" hazard to a screening analysis following the requirements in Part 6, but it was not possible to screen it out. Therefore, it is necessary to perform a more detailed analysis using the requirements in this Part. It is assumed for many SRs that if a more detailed analysis is needed, then the analyst will desire that the PRA have capability at least corresponding to Capability Category II and perhaps even to Category III for some issues. In these cases, is recognized that portions of the PRA adopted for the plant response model used will correspond to the associated Capability Category from Part 2.

Rationale and Structure of the Requirements: There are three technical elements in the PRA of an external flood hazard. They are described briefly below:

(a) External Flood Hazard Analysis (XFHA). This element involves the evaluation of the frequency of

occurrence of different external flood severities based on a site-specific probabilistic evaluation reflecting recent available data and site-specific information.

(b) External Flood Fragility Evaluation (XFFR). This element evaluates the fragility of plant SSCs as a function of the severity of the external flood using plant-specific, SSC-specific information and an accepted engineering method for evaluating the postulated failure.

(c) External Flood Plant Response Model and Quantification (XFPR). This element develops a plant response model that addresses the initiating events and other failures resulting from the effects of external flooding that can lead to core damage or large early release. The model is based on the internal-events, at-power PRA model to incorporate those aspects that are different, due to the effects of an external flood, from the corresponding aspects of the internal-events, at-power model. The conditional CDF and LERF obtained from this model is combined with the frequency of the plant damage states obtained by convoluting the external flooding hazard and external flooding effects (i.e., fragility) to estimate the unconditional CDF and LERF.

8-2.1 EXTERNAL FLOODING HAZARD ANALYSIS (XFHA)

The objective of the hazard analysis is to assess the frequency of occurrence of external floods as a function of severity on a site-specific basis.

Table 8-2.1-1 High Level Requirements for External Flooding Hazard Analysis (XFHA)

Designator	Requirement
HLR-XFHA-A	The frequency of external flooding at the site shall be based on site-specific probabilistic hazard analysis (existing or new) that reflects recent available regional and site-specific information. The external-flooding hazard analysis shall use up-to-date databases. Uncertainties in the models and parameter values shall be properly accounted for and fully propagated to obtain a family of hazard curves from which a mean hazard curve can be derived.
HLR-XFHA-B	Documentation of the external flood hazard analysis shall be consistent with the applicable supporting requirements.

Table 8-2.1-2 Supporting Requirements for HLR-XFHA-A

The frequency of external flooding at the site shall be based on a site-specific probabilistic hazard analysis (existing or new) that reflects recent available site-specific information. The external flooding hazard analysis shall use up-to-date databases. Uncertainties in the models and parameter values shall be properly accounted for and fully propagated in order to obtain a family of hazard curves from which a mean hazard curve can be derived (HLR-XFHA-A)

Index No. XFHA-A	Capability Category I	Capability Category II	Capability Category III
XFHA-A1 [Note (1)]	Not Defined	In the hazard analysis for extreme local precipitation, USE up-to-date data for the relevant phenomena. It is acceptable to use both site-specific and regional data.	
XFHA-A2 [Note (2)]	Not Defined	In the hazard analysis for extreme river flooding, including floods due to single or cascading dam failures, USE up-to-date data for the relevant phenomena. It is acceptable to use both site-specific and regional data.	
XFHA-A3 [Note (3)]	Not Defined	In the hazard analysis for extreme ocean (coastal and estuary) flooding, USE up-to-date data for the relevant phenomena. It is acceptable to use both site-specific and regional data.	
XFHA-A4 [Note (4)]	Not Defined	In the hazard analysis for extreme lake flooding, USE up-to-date data for the relevant phenomena. INCLUDE high water levels, surges, and wind–wave effects.	
XFHA-A5 [Note (5)]	Not Defined	In the hazard analysis for extreme tsunami flooding, USE up-to-date data for the relevant phenomena. It is acceptable to use both site-specific and regional or oceanwide data.	
XFHA-A6 [Note (6)]	Not Defined	In the hazard analysis for flooding caused by the failure of a dam, levee, or dike, USE up-to-date data for the failure probabilities and effects.	

NOTES:

(1) The usual methodologies for analyzing extreme local precipitation depend on modeling of intense local rain over very short time periods (a few minutes up to, say, an hour), coupled with computer-based stochastic studies, such as Monte Carlo–type analysis, to generate the likelihood of several severe rains or snows in a longer period such as an 8-hr period. The limitations on these methods are principally that not enough is known about the correlations among extreme short-duration storms. Attempts have been made to develop correlations, either spatial over short distances or temporal over a few hours, based on the proposition that one can develop an understanding of how a severe storm might move (or not) in time, but these attempts have not generally been successful.

Site-specific historical records of precipitation may be used to predict extreme precipitation effects in much the same manner that such statistical data are used to define wind design criteria [8-12, 8-13].

There is a general consensus that some limited extrapolation beyond the site-specific historical record, using data from other sites, can be justified. However, for the most extreme rainfalls, say, those with frequencies below 0.001/yr, the problem is that these rare events seem to involve more than one extreme phenomenon in time correlation and that the correlations are neither understood from empirical information nor modeled satisfactorily. The technical basis for such a correlation model is not understood for most sites. See reference [8-14] for more discussion on these methods. The U.S. Nuclear Regulatory Commission's guidance in this area is in Regulatory Guide 1.59 [8-5].

Table 8-2.1-2 Supporting Requirements for HLR-XFHA-A (Cont'd)

NOTES: (Cont'd)

(2) The river-flooding design basis for most nuclear power plants is based on the Army Corps of Engineers "Probable Maximum Flood" (PMF). Although the method for selecting the PMF is not directly linked to its annual frequency or return period, the PMF annual frequencies are typically in the range of from 0.01/yr to 0.001/yr [8-9].

It is difficult to develop hazard curves for much larger river floods, with annual frequencies much below 0.001/yr. One prestigious study by a government advisory committee [8-14] was very pessimistic about the technical basis for such hazard curves, but another study [8-13] was more optimistic, believing that methods do exist for making estimates down to the range of 0.001/yr or even lower, if appropriate watershed data can be obtained. The fundamental problem is that when extrapolations beyond the historical record must be made, there is a need to understand the correlations between weather phenomena, which correlations are neither understood theoretically nor reliably known from actual data at most sites. See reference [8-9] for a discussion of these issues. The U.S. Nuclear Regulatory Commission's guidance in this area is in Regulatory Guide 1-59 [8-5]. Because this hazard aspect is difficult to analyze, the peer review team should concentrate on it.

(3) For most U.S. coastal sites, the historical record, going back perhaps a century or sometimes two or more, provides a reasonable basis for a limited extrapolation beyond the actual record. For example, data for a longer section of coastline can be used to strengthen the database, provided that care is taken to account for the specific site topography, both beneath the adjacent sea surface and on the land. The largest coastal floods sometimes involve the coincident arrival of a large storm surge when the tides are also very high, and it is necessary to use a joint probability distribution to account for this phenomena. Unfortunately, the correlations are not well understood for the largest storms. This presents a major difficulty for analyses that attempt to extrapolate the hazard frequency well beyond the historical record (say, beyond about one order of magnitude). Various extreme-value distributions have been used. (See references [8-9] and [8-15].) Because this hazard aspect is difficult to analyze, the peer review team should concentrate on it.

(4) In the United States, the issue of extreme lake flooding arises mostly for the several nuclear power plants located on the Great Lakes, where the problem is principally due to the possible (but rare) combination of several effects such as storm-driven wave run-up, wind-generated waves, and an unusually high lake level. For the Great Lakes, only slightly more than 100 yr of reliable data exist. (For other lakes, the record may be somewhat longer.) Effects of extreme winds, including both wind-driven waves and wind setup along the shore, are often much larger than the variations in the lake levels themselves. (See reference [8-9].) Theoretical analysis of wind-wave effects is reasonably well grounded and can support modest extrapolations beyond the historical record when local subsurface topographical features are accounted for.

(5) The historical database for tsunamis extends for several hundred years in both the Pacific and Atlantic Ocean basins, with less reliable historical data going back somewhat further. Given a distant tsunami arriving at a specific location, it is feasible to determine how large the tsunami-induced flood will be, taking into account the local offshore subsurface topography. Usually, an engineering analysis is sufficient to screen out tsunamis. If a site-specific probabilistic (numerical) analysis of the hazard frequency is required, the uncertainties are often large and therefore must be accounted for properly.

(6) See also Requirement XFHA-A2. Several generic databases exist on U.S. dam failures, categorized by the different dam types (earthfill dams, concrete dams, etc.). See references [8-16] and [8-17]. These databases must be used with care, depending on how closely the specific dam fits into the database. The mean failure rate for all U.S. dams is in the range between about 10^{-4}/yr and 10^{-5}/yr [8-9]. However, for some modern dams with extensive engineering, values below 10^{-5}/yr have been quoted [8-18], while for older, poorly constructed dams, values near 10^{-3}/yr could be appropriate. An accurate and useful probabilistic analysis of any specific dam would require detailed engineering evaluations.

Table 8-2.1-3 Supporting Requirements for HLR-XFHA-B

Documentation of the external flood hazard analysis shall be consistent with the applicable supporting requirements.

Index No. XFHA-B	Capability Category I	Capability Category II	Capability Category III
XFHA-B1	DOCUMENT the external flood hazard analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
XFHA-B2	DOCUMENT the process used to identify external flood hazards. For example, this documentation typically includes a description of the specific methods used for determining the external flooding hazard curves, including the scientific interpretations that are the basis for the inputs and results.		
XFHA-B3	DOCUMENT the sources of model uncertainty and related assumptions associated with the external flood hazard analysis.		

8-2.2 EXTERNAL FLOOD FRAGILITY ANALYSIS (XFFR)

The objective of the external flood fragility analysis is to identify those SSCs that are susceptible to the effects of external floods and to determine their plant-specific failure probabilities as a function of the severity of the external flood.

Table 8-2.2-1 High Level Requirements for External Flood Fragility Analysis (XFFR)

Designator	Requirement
HLR-XFFR-A	A flooding fragility evaluation shall be performed to estimate plant-specific, realistic flooding fragilities for those SSCs whose failure contributes to core damage or large early release.
HLR-XFFR-B	Documentation of the external flood fragility analysis shall be consistent with the applicable supporting requirements.

Table 8.2.2-2 Supporting Requirements for HLR-XFFR-A

A flooding fragility evaluation shall be performed to estimate plant-specific, realistic flooding fragilities for those SSCs whose failure contributes to core damage or large early release (HLR-XFFR-A).

Index No. XFFR-A	Capability Category I	Capability Category II	Capability Category III
XFFR-A1 [Note (1)]	Not Defined	In the evaluation of flood fragilities of structures and exposed equipment (equipment located at low elevations, intake and ultimate-heat-sink equipment, etc.), USE plant-specific data. In this evaluation, INCLUDE the findings of a plant walkdown. It is acceptable in the fragility analysis for both capacity and demand to apply the standard methodology used for seismic events, with appropriate modifications unique to the flooding event being studied.	
XFFR-A2	IDENTIFY plant SSCs that are vulnerable to the flood hazards.		

NOTE:

(1) Flood-caused failure of equipment is typically due to immersion, although in some instances, particularly applicable to structures, the failure may be due to flow-induced phenomena. The analyst needs to account for the ability to survive and to function for each equipment item susceptible to flooding.

Usually, it is assumed that equipment submerged by the flood waters and not specially protected will "fail," meaning that it will fail to perform its safety function. The analysis should include length of warning time, since plant personnel may be able to secure equipment in a safe configuration. Further, the analysis must include whether the "failure" of an item of equipment would leave it in a fail-safe position. Also, flood waters may only partially submerge an item of equipment, so the analysis must determine how much partial submersion would be sufficient to cause the "failure."

Failure of structures could be overall, such as due to a foundation failure, or local, such as failure of a wall or barrier leading to leakage or major flooding through the wall or barrier. Most nuclear power plant structures have excellent resistance to flooding, by design. Major vulnerabilities have sometimes been identified for certain structures, but usually, the equipment housed therein is not crucial to overall plant safety. The walkdown should play a major role in identifying potential problems, supplemented by an evaluation of structural drawings. As the requirement states, fragility analysis for both capacity and demand may be based on the standard methodology used for seismic events, with appropriate modifications unique to the flooding event being studied. The modifications need to be subject to a peer review.

Table 8-2.2-3 Supporting Requirements for HLR-XFFR-B

Documentation of the external flood hazard analysis shall be consistent with the applicable supporting requirements.

Index No. XFFR-B	Capability Category I	Capability Category II	Capability Category III
XFFR-B1	DOCUMENT the external flood fragility analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
XFFR-B2	DOCUMENT the process used in the external flood fragility analysis. For example, this documentation typically includes a description of *(a)* methodologies used to quantify the flooding-caused fragilities of SSCs, together with key assumptions *(b)* the basis for the screening out of any SSCs for which the screening basis is other than the SSC being located where flooding does not occur *(c)* SSC fragility values that includes the method of analysis, the dominant failure mode(s), the sources of information, and the location of each SSC		
XFFR-B3	DOCUMENT the sources of model uncertainty and related assumptions associated with the external flood fragility analysis.		

8-2.3 EXTERNAL FLOOD PLANT RESPONSE MODEL (XFPR)

The objectives of this element are to

(a) develop a external flood plant response model by modifying the internal-events, at-power PRA model to include the effects of the external flood in terms of initiating events and failures caused

(b) quantify this model to provide the conditional core damage probability (CCDP) and conditional large early release probability (CLERP) for each defined external flood plant damage state

(c) evaluate the unconditional CDF and LERF by integrating the CCDP/CLERP with the frequencies of the plant damage states obtained by combining the external flood hazard analysis and external flood fragility analysis

Table 8-2.3-1 High Level Requirements for External Flood Plant Response Model and Quantification (XFPR)

Designator	Requirement
HLR-XFPR-A	The external flooding PRA systems model shall include flood-caused significant initiating events, and other failures that are significant contributors, that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate flood-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model.
HLR-XFPR-B	The analysis to quantify core damage and large early release frequencies shall appropriately integrate the external flood hazard, the external flood fragilities, and the systems-analysis aspects.
HLR-XFPR-C	Documentation of the external flood plant response model development and quantification shall be consistent with the applicable supporting requirements.

Table 8-2.3-2 Supporting Requirements for HLR-XFPR-A

The external flooding PRA systems model shall include flood-caused significant initiating events, and other failures that are significant contributors, that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate flood-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model.

Index No. XFPR-A	Capability Category I	Capability Category II	Capability Category III
XFPR-A1 [Note (1)]	ENSURE that external flood-caused initiating events that give rise to significant accident sequences and/or significant accident progression sequences are included in the external flood PRA system model using a systematic process.		
XFPR-A2	USE the event trees and fault trees from the internal-events, at-power PRA model as the basis for the external flood accident sequence analysis.		
XFPR-A3 [Note (2)]	ENSURE that the PRA systems models reflect external flood-caused failures as well as other unavailabilities and human errors that give rise to significant accident sequences or significant accident progression sequences.		
XFPR-A4 [Note (3)]	In each of the following aspects of the external flood PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where 8-2.3 includes additional requirements. SPECIFY a defined basis to support the claimed non-applicability of any exceptions. The aspects governed by this requirement are *(a)* initiating-event analysis *(b)* accident-sequence analysis *(c)* success-criteria analysis *(d)* systems analysis *(e)* data analysis *(f)* human-reliability analysis *(g)* use of expert judgment When the Part 2 requirements are used, USE the Capability Category designations in Part 2, and for consistency, USE the same Capability Category in this analysis.		
XFPR-A5 [Note (4)]	In the human reliability analysis (HRA), EVALUATE additional stresses that can increase the likelihood of human errors or inattention, compared to the likelihood assigned in the internal events HRA when the same activities are undertaken in non-external flood event accident sequences. Whether or not increases in error probabilities are used, JUSTIFY the basis for this decision about what error rates to use.		
XFPR-A6	If any screening out is performed, PERFORM it using defined criteria that are documented in the PRA.		
XFPR-A7 [Note (5)]	PERFORM an analysis of external flood-caused dependencies and correlations in a way so that any screening out of SSCs appropriately includes those dependencies.		

Table 8-2.3-2 Supporting Requirements for HLR-XFPR-A (Cont'd)

The external flooding PRA systems model shall include flood-caused significant initiating events, and other failures that are significant contributors, that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate flood-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model.

Index No. XFPR-A	Capability Category I	Capability Category II	Capability Category III
XFPR-A8 [Note (6)]	ENSURE that any screening of human-error basic events and non–external flood-caused-failure basic events does not eliminate any significant accident sequences or significant accident progression sequences.		
XFPR-A9 [Note (7)]	In the systems-analysis models, for each basic event that represents a external flood-caused failure, INCLUDE the complementary "success" state where applicable to a particular SSC in cases where the external flood-caused failure probability is high.		
XFPR-A10 [Note (8)]	EVALUATE the possibility that the external flood can cause damage or plant conditions that preclude personnel access to safety equipment or controls, thereby inhibiting operator actions that might otherwise be credited.		
XFPR-A11 [Note (9)]	Not Defined	EVALUATE the likelihood that system recoveries modeled in the internal-events PRA may be more complex or even not possible after an external flood event, and ADJUST the recovery models accordingly.	

GENERAL NOTE: While the most common procedure for developing the hazard-specific PRA systems model is to start with the internal-events systems model and adapt it by adding and trimming, in some circumstances it is acceptable instead to develop an ad hoc external flood PRA plant response tailored especially to the situation being modeled. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects.

NOTES:

(1) It is very important that site-specific failure events, usually external flood-caused structural, mechanical, and electrical failures, be thoroughly investigated.

Also, multiple-unit impacts and dependencies should be evaluated, as appropriate, including recovery resources that could be affected by external floods.

Attention to both the core damage frequency (CDF) endpoint and the large early release frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement. This means that initiating events and SSC failures that could lead to LERF-type consequences need to be included in the systems model even if the CDF frequency is quite low.

Table 8-2.3-2 Supporting Requirements for HLR-XFPR-A (Cont'd)

NOTES: (Cont'd)

(2) The analysis may group external flood-caused failures if the leading failure in the group is modeled. The event trees and fault trees from the internal-events, at-power PRA model are generally used as the basis for the external flood-initiated accident sequences/event trees. This captures the thinking that has gone into their development and assists in allowing comparisons between the internal-events PRA and the external flood PRA to be made on a common basis.

In some circumstances, it is acceptable to develop an ad hoc systems model tailored especially to the external flood PRA situation being modeled, instead of starting with the internal-events model and adapting it. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause-effect relationships of the failures.

Attention to both the core damage frequency (CDF) endpoint and the large early release frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement. This means that the intiating event and SSC failures that could lead to LERF-type consequences need to be included in the systems model even if the CDF frequency is quite low.

(3) These Sections of Part 2 are effectively incorporated here by reference. A few aspects, however, do not apply in detail. Whenever an exception is taken, the PRA analyst team needs to be cognizant of the underlying rationale for the specific Part 2 requirement so as to ensure that this rationale is considered when the exception is taken.

(4) The human-error probabilities may be increased for some external flood actions, compared to the probabilities assigned in analogous internal-events-initiated sequences. The corresponding technical requirements in Part 2 should be consulted in performing the HRA aspect of a external flood PRA.

(5) It is vital that the analysis capture the important dependencies among external flood-caused failures (e.g., spatial or environmental dependencies). Common-cause failure analysis is important in all PRAs, but particularly for external-flood hazards.

(6) To make the systems-analysis models more manageable, some of the non-external flood caused failures and human errors may be screened out of the model if their contribution to the results is demonstrably very small.

(7) For some hazards, some SSCs whose hazard-induced failure is important to safety at higher levels will not fail or will fail with only modest probability. The modeling of the non-failure (that is, the "success") of such SSCs is an important aspect of the systems model, and excluding these "success" states can lead to erroneous PRA results.

(8) This information is most effectively gathered during the walkdown, which must be structured to search for access issues. Coordination with the human-reliability-analysis aspect of the PRA is important. In making these evaluations, it may be assumed that portable lighting is available and that breathing devices are available, if in fact the plant configuration includes them.

(9) The restoration of safety functions can be inhibited by any of several types of causes, including damage or failure, access problems, confusion, and loss of supporting personnel to other post–external flood recovery functions. Careful consideration of these causes must be given before recoveries are credited in the initial period after the external flood event. This is especially true for externally caused loss of off-site power (LOOP), given that the damage could be to switchyard components or to the off-site grid towers, which are generally difficult to fix quickly.

Table 8-2.3-3 Supporting Requirements for HLR-XFPR-B

The analysis to quantify core damage and large early release frequencies shall appropriately integrate the external flood hazard, the external flood fragilities, and the systems-analysis aspects. (HLR-XFPR-B).

Index No. XFPR-B	Capability Category I	Capability Category II	Capability Category III
XFPR-B1 [Note (1)]	Not Defined	To estimate core damage frequency and large early release frequency contributions, EVALUATE accident sequences initiated by external flooding. USE, where applicable, the appropriate flooding hazard curves and the fragilities of structures and equipment.	
XFPR-B2 [Note (2)]	Not Defined	In the integration-quantification, INCLUDE the uncertainties in each of the inputs, dependencies, and correlations.	

GENERAL NOTE: In special circumstances, it is acceptable to develop an ad hoc systems model tailored especially to the particular flooding phenomenon being analyzed, instead of adapting the internal-events systems model. If this approach is used, it is especially important that a peer review be undertaken that concentrates on this aspect.

NOTES:

(1) The external-flooding-PRA systems-analysis model is almost always based on the internal-events, at-power PRA systems model, to which are added basic failure events derived from the information developed in the flooding fragility analysis. Considerable screening out of parts of the internal-events systems model is also common, where appropriate. The analysis consists of developing event trees and fault trees in which the initiating event can be either the extreme flood itself or a transient or loss-of-coolant accident induced by the extreme flood. Various accident sequences that lead to core damage or large early release are identified, and their conditional probabilities of occurrence are calculated. The frequency of core damage or large early release is obtained by a convolution over the relevant range of hazard intensities.

The procedure for determining the accident sequences is similar to that used in seismic-PRA systems analysis, and following the requirements therein represents one acceptable approach, after they are adapted to apply to the external-flooding-PRA situation. (See the requirements and commentary in 5-2.3 and the discussion about seismic PRA methods in Nonmandatory Appendix 5-B). Other factors to be considered include non–flooding-related unavailabilities or failures of equipment, operator errors, any warning time available to take mitigating steps, the possibility of recovery actions by operators and replacement by substitutes to accomplish the needed function — and the likelihood of common-cause failures. The clogging of intake structures and other flow paths by debris related to the flooding must also be considered, and a walkdown is important to ensure that this issue has been evaluated properly.

One key consideration is that most large external floods occur only after significant warning time, which allows the plant operating personnel to take appropriate steps to secure the plant and its key equipment. This warning time and the typical situation in which the plant grade is well above any credible flooding phenomena are the principal reasons why external-flooding risks are not often found to be important contributors to overall risks. The analysis team is therefore urged to take as much credit for warning time and compensatory actions as the plant's planning and procedures allow.

(2) The usefulness of the "final results" of the PRA for external flooding are dependent on performing enough assessment to understand the uncertainties, dependencies, and correlations and to account for them quantitatively if they are important.

Table 8-2.3-4 Supporting Requirements for HLR-XFPR-C

Documentation of the external flood plant response model development and quantification shall be consistent with the applicable supporting requirements.

Index No. XFPR-C	Capability Category I	Capability Category II	Capability Category III
XFPR-C1	DOCUMENT the external flood plant response analysis and quantification in a manner that facilitates PRA applications, upgrades, and peer review.		
XFPR-C2	DOCUMENT the process used in the external flood plant response analysis and quantification. For example, this documentation typically includes a description of *(a)* the specific adaptations made to the internal-events PRA model to produce the external flooding-PRA model, and their motivation *(b)* the final results of the PRA analysis in terms of core damage frequency and large early release frequency, as well as selected intermediate results		
XFPR-C3	DOCUMENT the sources of model uncertainty and related assumptions associated with the external flood plant response model development.		

Section 8-3
Peer Review for External Flood At-Power PRA

8-3.1 PURPOSE

This Section provides requirements for peer review of a Level 1 and LERF at-power PRA of external flood events.

8-3.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATIONS

The peer review team shall have knowledge and collective experience in the subjects of systems engineering, evaluation of the external flood hazard, and evaluation of how the external flood hazard could damage the nuclear plant's SSCs, as applicable to the scope of the review. Section 1-6 provides general requirements for peer review. Subsection 1-6.2 specifies requirements for peer review team knowledge and collective experience. Paragraph 1-6.1.1 specifies requirements regarding peer review scope.

8-3.3 REVIEW OF EXTERNAL FLOOD PRA ELEMENTS TO CONFIRM THE METHODOLOGY

8-3.3.1 External Flood Hazard Selection

The peer review team shall evaluate whether the external flood hazard used in the PRA is appropriately specific to the site and has met the relevant requirements of this Standard.

8-3.3.2 External Flood-Induced Initiating Events

The peer review team shall evaluate whether the initiating events postulated to be caused by the external flood are properly identified, the SSCs are properly modeled, and the accident sequences are properly quantified.

8-3.3.3 "Fragility" Analysis Methods and Data

The peer review team shall evaluate whether the methods and data used in the "fragility" analysis of SSCs are adequate for the purpose and meet the relevant requirements of this Standard. The review team should perform independent fragility calculations of a selected sample of SSCs covering different categories and contributions to core damage frequency and large early release frequency.

8-3.3.4 Plant Walkdown

The peer review team shall review the walkdown of the plant to ensure the validity of the findings of the analysis in terms of screening out, any spatial interactions, and the identification of critical failure modes.

8-3.3.5 Quantification Method

The peer review team shall evaluate whether the quantification method used in the PRA is appropriate and provides all of the results and insights needed for risk-informed decisions. If the analysis contains screening assumptions, or assumptions that the analysis team claims to be demonstrably conservative, the peer review team shall review the validity of these assumptions. The review shall focus on the core damage frequency and large early release frequency estimates and uncertainty bounds and on the significant contributors.

Section 8-4
References

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(a)
(b)

PART 9 REQUIREMENTS FOR OTHER HAZARDS AT-POWER PRA

Section 9-1 Overview of Requirements for Other Hazards At-Power PRA

9-1.1 PRA SCOPE

This Part establishes technical requirements for a Level 1 and large early release frequency (LERF) analysis of other hazards while at-power. Note that each hazard for which a unique approach is developed will constitute its own hazard group.

9-1.2 COORDINATION WITH OTHER PARTS OF THIS STANDARD

This Part is intended to be used with Parts 1 and 2 of this Standard. A PRA developed in accordance with Part 2 is the starting point for assessing the conditional core damage probability and conditional large early release probability.

9-1.3 OTHER HAZARDS SCOPE AND APPLICABILITY

(a) Scope. The term "other hazard" refers to internal or external hazards other than those for which requirements are provided in Parts 2, 3, 5, 7, and 8 of this Standard (e.g., internal events, internal floods, internal fires, earthquakes, high winds, external floods). Nonmandatory Appendix 6-A includes a list of hazards that may apply as specific sites.

For high winds and external flooding, either this Part or Parts 7 and 8 can be used.

(b) Applicability. This Part applies to other hazards that cannot be screened out (that is, cannot be excluded from further consideration in the PRA analysis) using the processes and criteria in Part 6, "Requirements for Screening and Conservative Analysis of Other Hazards At-Power" or in instances where a baseline PRA of a hazard is needed for a specific application. The requirements in Part 6 can be used for the analysis of any other hazard. Alternatively, the requirements in Part 7 ("Requirements for High-Wind Events At-Power PRA") or Part 8 ("Requirements for External Flood Events At-Power PRA") can be used for those hazards. If either Part 7 or 8 is used, then all of the requirements therein apply.

(c) Terminology: "Hazard" in the Singular. For this Part, which deals with analysis of an entire category of hazard, the term "hazard" in the singular is used for a single and entire category of similar events, or hazard group, and the hazard group is intended to include all "sizes" of such events within the category. For example, the hazard group for "extreme temperature" includes all extreme-temperature conditions, no matter how extreme or how infrequent; the hazard group "transportation accidents" includes all such accidents arising from nearby transport modes. Within that hazard group, the hazard group "aircraft impact accidents" includes crashes of all aircraft, of all sizes; and so on.

This set of requirements is concerned with detailed PRA analysis of a hazard group. Even though as written it contemplates the analysis of an entire hazard group, it is not intended to restrict the analyst from analyzing only a subgroup or particular hazard events if the differentiation of the subgroup or hazard event from the remainder of the larger hazard group makes sense, presumably because only the subgroup is important and the remainder can be screened out. For example, suppose that for a given site the real risk potential from transportation accidents is from accidental aircraft crashes, and within aircraft crashes, the impact arises from military jet overflights. As part of this example, suppose that, by using Part 6 and a demonstrably conservative analysis, all transportation accidents except aircraft can be screened out on the basis of a low core damage frequency. Continuing this example, further suppose that, by using Part 6, large commercial jets can be screened out on the basis of a very low annual frequency, and that small crop-duster planes can be screened out on the basis of not being able to cause enough damage. It is completely acceptable to subdivide the hazard for "aircraft impact" into specific aircraft crash events by using judgment and approximate analysis and then to subject only the military jet subgroup to detailed PRA analysis by using the requirements herein.

(d) Large Early Release Frequency. In applying the analyses covered in this Part, it is necessary to be attentive to both CDF and LERF. In this regard, the definition of LERF is applicable and should be taken into account. Also, the analyst is urged to be especially attentive to effects of the hazard that might compromise, challenge, or degrade containment integrity and thereby possibly contribute to LERF-type accident sequences.

(e) General Guidance. The PRA Procedures Guide [9-1] and the Probabilistic Safety Assessment Procedures Guide [9-2] contain detailed discussions that provide general guidance on how to approach the PRA of a hazard. Some of the commentary herein is adapted from these guides.

Section 9-2
Technical Requirements for Other Hazards At-Power PRA

(a) Screening, Realistic Analysis, and Conservative Analysis. Presumably, if a hazard cannot be screened out based on the criteria in Part 6, it is because the hazard fails to meet those criteria — or at least, the hazard cannot be shown to meet those criteria by using the screening-out methods or demonstrably conservative analysis methods of Part 6. The fundamental screening-out criteria are as follows (quoting from 6-2.3):

"A hazard can be screened out if

(1) it meets the criteria in the U.S. Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) [9-3] or a later revision; or

(2) it can be shown, by using a demonstrably conservative analysis, that the mean value of the design-basis hazard event used in the plant design is less than $\sim10^{-5}$/yr and that the conditional core damage probability is $<10^{-1}$, given the occurrence of the design-basis-hazard event; or

(3) it can be shown, by using a demonstrably conservative analysis, that the CDF is $<10^{-6}$/yr"

It is recognized that for some hazards, although it may be difficult or impossible to demonstrate that any of these criteria are met by using screening or demonstrably conservative analysis, nevertheless the risk posed by the entire hazard category is quite small, as measured by the hazard's contribution to CDF and LERF. Given this possibility, although the detailed analysis contemplated in this Part is intended to be a realistic analysis, it is quite acceptable to introduce conservatisms in any given step, provided that, at the end, the overall contributions to CDF and LERF are demonstrably small. However, if the contributions to either CDF or LERF turn out to be "important" — presumably, important compared to other CDF and/or LERF contributions from other initiators — then the PRA analyst team is obliged to revisit the analysis here to make it as realistic as feasible.

Although the usual format for these supporting technical requirements is to organize them into three "Capability Categories," there are no entries herein for Capability Category I. Before applying the requirements in this Part, the analyst has presumably subjected the hazard group to a screening analysis by using the requirements in Part 6, but it was not possible to screen out some hazards in this group. Therefore, it is necessary to perform a more detailed analysis by using the requirements in this Part. It is assumed for many SRs that if a more detailed analysis of this hazard group is needed, then the analyst will desire that the PRA have capability at least corresponding to Capability Category II and perhaps even to Category III for some issues. In these cases, the Capability Category I requirements are not defined. Some SRs call for the use or adaptation of the internal-events PRA. In these cases, is recognized that portions of the PRA adopted for the plant response model used will correspond to the associated Capability Category from Part 2.

(b) Rationale and Structure of the Requirements. There are three technical elements in the PRA of any hazard. They are described briefly as follows:

(1) Other Hazard Analysis (XHA). This element involves the evaluation of the frequency of occurrence of different intensities of the hazard based on a site-specific probabilistic evaluation reflecting recent available data and site-specific information.

(2) Other Hazard Fragility Evaluation (XFR). This element evaluates the fragilities of the SSCs as a function of the intensity of the hazard by using plant-specific, SSC-specific information and an accepted engineering method for evaluating the postulated failure.

(3) Other Hazard Plant Response Model (XPR). This element develops a plant response model that addresses the initiating events and other failures resulting from the effects of the hazard that can lead to core damage or large early release. The model is based on the internal-events, at-power PRA model to incorporate those aspects that are different, due to the hazard's effects, from the corresponding aspects of the internal-events, at-power model. The conditional CDF and LERF obtained from this model are combined with the frequency of the plant damage states obtained by convoluting the hazard and fragility curves to estimate the unconditional CDF and LERF.

(c) Aircraft-Impact PRA. For the PRA of aircraft impact, the requirements herein apply. However, another acceptable method for meeting aspects of this Part is to follow the methodology in the U.S. Department of Energy (DOE) standard "Accident Analysis for Aircraft Crash Into Hazardous Facilities" [9-4], which is a methodology standard for aircraft-impact PRA developed by the DOE for analyzing impacts on various DOE facilities. This DOE methodology may be used as an alternative way to satisfy in full the intent of the hazard analysis and fragility analysis technical elements (Requirements HLR-XHA-A and HLR-XHA-B and

of their supporting requirements). It would still be necessary to meet the requirements under HLR-XHA-C ("Systems Analysis and Quantification"). Please note that the aircraft-impact issue addressed herein covers accidental aircraft crashes only.

9-2.1 OTHER HAZARD ANALYSIS (XHA)

The objective of the hazard analysis is to assess the frequency of occurrence of the external hazard as a function of intensity on a site-specific basis.

Table 9-2.1-1 High Level Requirements for Other Hazard Analysis (XHA)

Designator	Requirement
HLR-XHA-A	The analysis of the hazard (the frequency of occurrence of different intensities of the hazard) shall be based on a site-specific probabilistic evaluation that uses recent available data and site-specific information. The analysis can be based on either historical data or a phenomenological model, or a mixture of the two.
HLR-XHA-B	Documentation of the hazard analysis shall be consistent with the applicable supporting requirements

Table 9-2.1-2 Supporting Requirements for HLR-XHA-A

The analysis of the hazard (the frequency of occurrence of different intensities of the hazard) shall be based on a site-specific probabilistic evaluation that uses recent available data and site-specific information. The analysis can be based on either historical data or a phenomenological model, or a mixture of the two (HLR-XHA-A).

Index No. XHA-A	Requirement
XHA-A1	ENSURE that the hazard analysis is site specific and plant specific to the extent necessary for the analysis.
Commentary: Although a site-specific and plant-specific hazard analysis is always desirable, it is often acceptable to develop a hazard on some other basis (for example, a regional or even generic basis), provided that the uncertainties introduced are acceptable for the applications contemplated. The phrase "to the extent necessary" in the requirement is intended to allow approximations provided that the error or uncertainty introduced is not dominant in the analysis. Model uncertainties are especially difficult to quantify in some cases.	
XHA-A2	In the hazard analysis for the hazard, USE up-to-date databases. PROPAGATE uncertainties in the models and parameter values to obtain a family of hazard curves from which a mean hazard curve can be derived.
Commentary: In general, the hazard posed by any hazard can only be described by a multitude of variables related to the "size" of the event. Often, some of these variables are probabilistically dependent on other variables. However, for simplicity, the hazard function is generally described, albeit imperfectly, in terms of a limited number of variables — typically, one. For example, although a proper characterization of the hazard from a potential chemical explosion from a nearby railroad train carrying chemicals should include blast distance, duration, instantaneous pressure duration, shape of the pressure pulse as a function of frequency, chemical form of the explosive, and so on, the hazard would likely be characterized by only one or two of these parameters in any actual analysis. The other variables that would be needed for a "complete" description of the hazard would typically be considered in the response analysis and fragility evaluation, or may represent an irreducible variability in the hazard, or some of each. The output of the hazard analysis is a so-called "hazard curve" — actually, a family of hazard curves accounting for uncertainties — of exceedance frequency versus hazard intensity. The PRA Procedures Guide [9-1] has a useful discussion of the general considerations involved in hazard analysis.	

Table 9-2.1-2 Supporting Requirements for HLR-XHA-A (Cont'd)

The analysis of the hazard (the frequency of occurrence of different intensities of the hazard) shall be based on a site-specific probabilistic evaluation that uses recent available data and site-specific information. The analysis can be based on either historical data or a phenomenological model, or a mixture of the two (HLR-XHA-A).

Index No. XHA-A	Requirement
XHA-A3	To develop the PRA model, SPECIFY the hazard curve in terms of the parameter that best represents a measure of the intensity of the hazard.
Commentary: None	
XHA-A4	If expert elicitation or another use-of-experts process is used in developing the hazard information, PERFORM it in accordance with established guidelines.
Commentary: The discussion in Section 5-2, which introduces the hazard requirements for seismic PRA, and the corresponding supporting requirements and commentary in 5-2.1 in Requirements SHA-A4, SHA-C2, and SHA-D2 contain useful guidance on this subject. Also, Part 2 contains requirements on the use of experts. Adapting these requirements to the situation of the "other" hazard analyzed herein is acceptable.	

Table 9-2.1-3 Supporting Requirements for HLR-XHA-B

The hazard analysis shall be documented consistent with the applicable supporting requirements (HLR-XHA-B).

Index No. XHA-B	Requirement
XHA-B1	DOCUMENT the hazard analysis manner that facilitates PRA applications, upgrades, and peer review.
Commentary: None	
XHA-B2	DOCUMENT the processes used to define and quantify the hazard. For example, this documentation typically includes a description of the specific methods used for determining the hazard curves, including the technical interpretations that are the basis for the inputs and results.
Commentary: None	
XHA-B3	DOCUMENT the sources of model uncertainty and related assumptions associated with the hazard analysis.
Commentary: None	

9-2.2 OTHER HAZARD FRAGILITY ANALYSIS (XFR)

The objective of the fragility analysis is to identify those SSCs that are susceptible to the effects of the hazard and to determine their plant-specific failure probabilities as a function of the intensity of the hazard. [Note that in this context, the plant operators are included as components of the system, since some hazards (e.g., toxic gas) may affect operators rather than equipment.]

Table 9-2.2-1 High Level Requirements for Other Hazard Fragility Analysis (XFR)

Designator	Requirement
HLR-XFR-A	The fragility of an SSC shall be evaluated by using plant-specific, SSC-specific information and an accepted engineering method for evaluating the postulated failure.
HLR-XFR-B	Documentation of the hazard fragility analysis shall be consistent with the applicable supporting requirements

Table 9-2.2-2 Supporting Requirements for HLR-XFR-A

The fragility or vulnerability of an SSC shall be evaluated by using plant-specific, SSC-specific information and an accepted engineering method for evaluating the postulated failure (HLR-XFR-A).

Index No. XFR-A	Requirement
XFR-A1	ENSURE that the fragility estimates are site specific and plant specific to the extent necessary for the purposes of the analysis.
Commentary: Although a site-specific and plant-specific analysis of the fragilities of SSCs is always desirable, it is often acceptable to develop fragility estimates on some other basis (e.g., generic information), provided that the uncertainties introduced are acceptable for the applications contemplated. The phrase "to the extent necessary" in the requirement is intended to allow approximations provided that the error or uncertainty introduced is not dominant in the analysis. Model uncertainties are especially difficult to quantify in some cases.	
XFR-A2	EVALUATE the fragilities of SSCs by using plant-specific data to the extent necessary for the analysis. INCLUDE the findings of a plant walkdown in this evaluation.
Commentary: The fragility of an SSC is estimated from the actual capacity of the SSC for a *given failure mode*. Thus, a failure-mode identification is a crucial aspect of this work. Another crucial aspect is an engineering evaluation of how the effect of the hazard is transmitted to the SSC — what force or effect leads to the specified failure mode. The PRA Procedures Guide [9-1] has a useful discussion of the general considerations involved in fragility evaluation. The phrase "to the extent necessary" in the requirement is intended to allow approximations provided that the error or uncertainty introduced is not dominant in the analysis. Model uncertainties are especially difficult to quantify in some cases.	
XFR-A3	SPECIFY the fragility curve for each failure mode as a function of the same parameter used to represent the intensity of the hazard.
Commentary: To make the PRA analysis tractable, the fragility should be expressed as a function of the same variable — related to the "size" of the hazard — of which the hazard curves are functions. This allows the convolution of the hazard curves and fragility curves during the quantification step to be done in a mathematically straightforward way.	
XFR-A4	In the fragility analysis, USE the uncertainties in the underlying information and the models used.
Commentary: The analysis of the fragility or vulnerability of an SSC must account for the various uncertainties in both underlying data and models. The requirements and commentary on this subject given in 5-2.2 on seismic PRA fragility analysis contain useful guidance. Adapting these requirements to the situation of the "other" hazard analyzed here is acceptable. Attention to model uncertainty is important and is implied in the requirement.	

Table 9-2.2-3 Supporting Requirements for HLR-XFR-B

Documentation of the hazard fragility analysis shall be consistent with the applicable supporting requirements (HLR-XFR-B).

Index No. XFR-B	Requirement
XFR-B1	DOCUMENT the hazard fragility analysis in a manner that facilitates PRA applications, upgrades, and peer review.
Commentary: None	
XFR-B2	DOCUMENT the processes used to define and quantify the hazard fragilities. For example, this documentation typically includes *(a)* a description of the specific methods used for determining the hazard curves, including the technical interpretations that are the basis for the inputs and results *(b)* the methodologies used to quantify the fragilities of SSCs, together with key assumptions *(c)* the basis for the screening out of any generic high-capacity SSCs *(d)* a detailed list of SSC fragility values that includes the method of analysis, the dominant failure mode(s), the sources of information, and the location of each SSC
Commentary: None	
XFR-B3	DOCUMENT the sources of model uncertainty and related assumptions associated with the hazard fragility analysis.
Commentary: None	

9-2.3 OTHER HAZARD PLANT RESPONSE MODEL (XPR)

The objectives of this element are to

(a) develop a plant response model by modifying the internal-events, at-power PRA model to include the effects of the hazard in terms of initiating events and failures caused

(b) quantify this model by calculating the conditional core damage probability (CCDP) and conditional large early release probability (CLERP) for each defined hazard plant damage state

(c) evaluate the unconditional CDF and LERF by integrating the CCDP/CLERP with the frequencies of the plant damage states obtained by combining the hazard analysis and fragility analysis

Table 9-2.3-1 High Level Requirements for Other Hazard Plant Response Model and Quantification (XPR)

Designator	Requirement
HLR-XPR-A	The hazard PRA plant model shall include hazard-induced initiating events and other failures that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate hazard-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model.
HLR-XPR-B	The analysis to calculate core damage and large early release frequencies shall appropriately integrate the hazard, the fragilities, and the plant response aspects.
HLR-XPR-C	Documentation of the external hazard plant response analysis and quantification shall be consistent with the applicable supporting requirements.

Table 9-2.3-2 Supporting Requirements for HLR-XPR-A

The hazard PRA plant model shall include hazard-induced initiating events and other failures that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate hazard-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model (HLR-XPR-A).

Index No. XPR-A	Capability Category I	Capability Category II	Capability Category III
XPR-A1	IDENTIFY those SSCs required to maintain the plant in operation or that are required to respond to an initiating event to prevent core damage that are vulnerable to the hazard, and determine their failure modes.		
Commentary: It is very important that site-specific failure events, e.g., wind-caused structural, mechanical, and electrical failures, be thoroughly investigated. Also, multiple-unit impacts and dependencies should be considered, as appropriate, including, for example, recovery resources that could be affected by high wind. Attention to both the core damage frequency (CDF) endpoint and the large early release frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement. This means that initiating events and SSC failures that could lead to LERF-type consequences need to be included in the systems model even if the CDF frequency is quite low.			
XPR-A2	USE the event trees and fault trees from the internal-event, at-power PRA model as the basis for the hazard accident sequence analysis.		
Commentary: None			
XPR-A3	ENSURE that the PRA systems models include hazard-induced failures as well as other unavailabilities and human errors that give rise to significant accident sequences or significant accident progression sequences.		
Commentary: The analysis may group hazard-induced failures if the leading failure in the group is modeled. The event trees and fault trees from the internal-events, at-power PRA model are generally used as the basis for the hazard-initiated accident sequences/event trees. This captures the thinking that has gone into their development and assists in allowing comparisons between the internal-events PRA and the hazard event PRA to be made on a common basis. In some circumstances, it is acceptable to develop an ad hoc systems model tailored especially to the hazard PRA situation being modeled, instead of adapting the internal-events model. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause–effect relationships of the failures. Attention to both the core damage frequency (CDF) endpoint and the large early release frequency (LERF) endpoint of the PRA analysis is necessary to meet this requirement. This means that intiating events and SSC failures that could lead to LERF-type consequences need to be included in the systems model even if the CDF frequency is quite low.			

Table 9-2.3-2 Supporting Requirements for HLR-XPR-A (Cont'd)

The hazard PRA plant model shall include hazard-induced initiating events and other failures that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate hazard-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model (HLR-XPR-A).

Index No. XPR-A	Capability Category I	Capability Category II	Capability Category III
XPR-A4	In each of the following aspects of the hazard PRA systems-analysis work, SATISFY the corresponding requirements in Part 2, except where they are not applicable or where this Part includes additional requirements. SPECIFY a defined basis to support the claimed nonapplicability of any exceptions. The aspects governed by this requirement are *(a)* initiating-event analysis *(b)* accident-sequence analysis *(c)* success-criteria analysis *(d)* systems analysis *(e)* data analysis *(f)* human-reliability analysis *(g)* use of expert judgment When the Part 2 requirements are used, USE the Capability Category designations in Part 2, and for consistency, USE the same Capability Category in this analysis.		
Commentary: These Sections of Part 2 are effectively incorporated here by reference. A few aspects, however, do not apply in detail. Whenever an exception is taken, the PRA analyst team needs to be cognizant of the underlying rationale for the specific Part 2 requirement so as to ensure that this rationale is considered when the exception is taken.			
XPR-A5	In the human reliability analysis (HRA) aspect, EVALUATE additional stresses that can increase the likelihood of human errors or inattention, compared to the likelihood assigned in the internal events HRA when the same activities are undertaken in non–internal-events accident sequences. Whether or not increases in error probabilities are used, JUSTIFY the basis for this decision about what error rates to use.		
Commentary: The human-error probabilities may be increased for some hazard actions, compared to the probabilities assigned in analogous internal-events–initiated sequences. The corresponding technical requirements in Part 2 should be consulted in performing the HRA aspect of the hazard PRA.			
XPR-A6	If any screening is performed, PERFORM it using defined criteria that are documented in the PRA.		
XPR-A7	PERFORM an analysis of hazard-induced dependencies and correlations in a way so that any screening out of SSCs appropriately includes those dependencies.		
Commentary: It is vital that the analysis capture the important dependencies among hazard-induced failures (e.g., spatial or environmental dependencies). Common-cause failure analysis is important in all PRAs, but particularly for other hazards.			

Table 9-2.3-2 Supporting Requirements for HLR-XPR-A (Cont'd)

The hazard PRA plant model shall include hazard-induced initiating events and other failures that can lead to core damage or large early release. The model shall be adapted from the internal-events, at-power PRA systems model to incorporate hazard-analysis aspects that are different from the corresponding aspects in the internal-events, at-power PRA systems model (HLR-XPR-A).

Index No. XPR-A	Capability Category I	Capability Category II	Capability Category III
XPR-A8	ENSURE that any screening of human-error basic events and non–hazard-induced-failure basic events does not eliminate any significant accident sequences or significant accident progression sequences.		
Commentary: To make the systems-analysis models more manageable, some of the non–hazard-induced failures and human errors may be screened out of the model if their contribution to the results is demonstrably very small.			
XPR-A9	In the systems-analysis models, for each basic event that represents a hazard-induced failure, INCLUDE the complementary "success" state where applicable to a particular SSC in cases where the hazard-induced failure probability is high.		
Commentary: For some hazards, some SSCs whose hazard-induced failure is important to safety at higher levels will not fail or will fail with only modest probability. The modeling of the nonfailure (that is, the "success") of such SSCs is an important aspect of the systems model, and excluding these "success" states can lead to erroneous PRA results.			
XPR-A10	EVALUATE the possibility that the hazard can cause damage or plant conditions that preclude personnel access to safety equipment or controls, thereby inhibiting operator actions that might otherwise be credited.		
Commentary: This information is most effectively gathered during the walkdown, which must be structured to search for access issues. Coordination with the human-reliability–analysis aspect of the PRA is important. In making these evaluations, it MAY be assumed that portable lighting is available and that breathing devices are available, if in fact the plant configuration includes them.			
XPR-A11	Not Defined	EVALUATE the likelihood that system recoveries modeled in the internal-events PRA may be more complex or even not possible after a hazard, and ADJUST the recovery models accordingly.	
Commentary: The restoration of safety functions can be inhibited by any of several types of causes, including damage or failure, access problems, confusion, loss of supporting personnel to other post–hazard-recovery functions, and so on. Careful consideration of these causes must be given before recoveries are credited in the initial period after the hazard. This is especially true for externally caused loss of off-site power (LOOP), given that the damage could be to switchyard components or to the off-site grid towers, which are generally difficult to fix quickly.			

GENERAL NOTE: While the most common procedure for developing the hazard PRA systems model is to adapt the internal-events systems by adding and trimming, in some circumstances it is acceptable to develop an ad hoc hazard PRA plant response tailored especially to the situation being modeled. If this approach is used, it is especially important that the resulting model be consistent with the internal-events systems model regarding plant response and the cause–effect relationships of the failures. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects.

Table 9-2.3-3 Supporting Requirements for HLR-XPR-B

The analysis to quantify core damage and large early release frequencies shall appropriately integrate the hazard, the fragilities, and the plant response aspects (HLF-XPR-B).

Index No. XPR-B	Requirement
XPR-B1	CALCULATE the CCDP taking into account the initiating events caused by the hazard, and the systems or functions rendered unavailable. Adapting the internal-events PRA model, as appropriate, by using conservative assessments of the impact of the hazard (fragility analysis) is an acceptable approach.
Commentary: None	
XPR-B2	EVALUATE the accident sequences initiated by the hazard to estimate core damage frequency (CDF) and large early release frequency (LERF) contributions. In the analysis, USE as appropriate the applicable hazard curves and the fragilities of structures and equipment.
Commentary: The PRA systems-analysis model for any external hazard is almost always based on the internal-events, at-power PRA systems model, to which are added basic failure events derived from the information developed in the specific hazard's fragility analysis. Considerable screening out of parts of the internal-events systems model is also common, where appropriate. The analysis consists of developing event trees and fault trees in which the initiating event can be either a transient or loss-of-coolant accident induced by the event, or another initiating event specific to the hazard being analyzed. Various accident sequences that lead to core damage or large early release are identified, and their conditional probabilities of occurrence are calculated. The frequency of core damage or large early release is obtained by a convolution of these conditional probabilities over the relevant range of hazard intensities. The procedure for determining the accident sequences is similar to that used in seismic-PRA systems analysis, and following the requirements therein represents one acceptable approach, after they are adapted to apply to the PRA situation represented by the specific hazard. (See the requirements and commentary in Section 5-2 and the discussion about seismic PRA methods in Nonmandatory Appendix 5-A.) Other factors to be considered include non–hazard-related unavailabilities or failures of equipment; operator errors; unique aspects of common causes, correlations, and dependencies; any warning time available to take mitigating steps; the possibility of recovery actions by operators and replacement by substitutes to accomplish the needed function; and the likelihood of common-cause failures.	
XPR-B3	In the integration-quantification, INCLUDE all important dependencies and correlations and for the uncertainties in each of the inputs.
Commentary: The usefulness of the "final results" of the PRA for the hazard are dependent on performing sufficient assessment to understand the dependencies, correlations, and uncertainties and to account for them quantitatively if they are important. Considerable judgment is needed on the part of the analyst. This integration-quantification aspect should be a focus of the peer review.	

GENERAL NOTE: In special circumstances, it is acceptable to develop an ad hoc systems model tailored especially to the hazard being analyzed instead of adapting the internal-events systems model. If this approach is used, it is especially important that a peer review be undertaken that concentrates on this aspect.

Table 9-2.3-4 Supporting Requirements for HLR-XPR-C

Documentation of the hazard plant response analysis and quantification shall be consistent with the applicable supporting requirements (HLF-XPR-C).

Index No. XPR-C	Capability Category I	Capability Category II	Capability Category III
XPR-C1	DOCUMENT the hazard plant response analysis and quantification in a manner that facilitates PRA applications, upgrades, and peer review.		
Commentary: None			
XPR-C2	DOCUMENT the process used in the hazard fragility analysis. For example, this documentation typically includes a description of *(a)* the specific adaptations made to the internal-events PRA model to produce the hazard-PRA model, and their motivation *(b)* the final results of the PRA analysis in terms of core damage frequency and large early release frequency, as well as selected intermediate results		
Commentary: None			
XPR-C3	DOCUMENT the sources of model undertainty and related assumptions associated with the hazard plant response model development.		
Commentary: None			

Section 9-3
Peer Review for Other Hazards At-Power PRA

9-3.1 PURPOSE

This Section provides requirements for peer review of an Other Hazards Level 1 and LERF at-power PRA.

9-3.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATION

The peer review team shall have knowledge and collective experience in the subjects of systems engineering, evaluation of the relevant hazard(s), and evaluation of how the hazard(s) could damage the nuclear plant's SSCs, as applicable to the scope of the review. Section 1-6 provides general requirements for peer review. Subsection 1-6.2 specifies requirements for peer review team knowledge and collective experience. Paragraph 1-6.1.1 specifies requirements regarding peer review scope.

9-3.3 REVIEW OF OTHER HAZARDS PRAs ELEMENTS TO CONFIRM THE METHODOLOGY

9-3.3.1 Hazard Selection

The peer review team shall evaluate whether the hazard used in the PRA is appropriately specific to the site and has met the relevant requirements of this Standard.

9-3.3.2 Hazard-Caused Initiating Events

The peer review team shall evaluate whether the initiating events postulated to be caused by the hazard are properly identified, the SSCs are properly modeled, and the accident sequences are properly quantified.

9-3.3.3 "Fragility" Analysis Methods and Data

The peer review team shall evaluate whether the methods and data used in the "fragility" analysis of SSCs are adequate for the purpose and meet the relevant requirements of this Standard. The review team should perform independent fragility calculations of a selected sample of SSCs covering different categories and contributions to core damage frequency and large early release frequency.

9-3.3.4 Plant Walkdown

The peer review team shall review the walkdown of the plant to ensure the validity of the findings of the analysis in terms of screening, any spatial interactions, and the identification of critical failure modes.

9-3.3.5 Quantification Method

The peer review team shall evaluate whether the quantification method used in the PRA is appropriate and provides all of the results and insights needed for risk-informed decisions. If the analysis contains screening assumptions, or assumptions that the analysis team claims to be demonstrably conservative, the peer review team shall review the validity of these assumptions. The review shall focus on the core damage frequency and large early release frequency estimates and uncertainty bounds and on the significant risk contributors.

Section 9-4
References

[9-1] J. Hickman et al., "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Report NUREG/CR-2300, American Nuclear Society, Institute of Electrical and Electronic Engineers, and U.S. Nuclear Regulatory Commission (1983)

[9-2] M. McCann, J. Reed, C. Ruger, K. Shiu, T. Teichmann, A. Unione, and R. Youngblood, "Probabilistic Safety Analysis Procedures Guide," Report NUREG/CR-2815, Vol. 2, Brookhaven National Laboratory and U.S. Nuclear Regulatory Commission (1985)

[9-3] "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," Report NUREG-75/087, U.S. Nuclear Regulatory Commission (1975)

[9-4] Standard DOE-STD-3014-96: U.S. Department of Energy, "Accident Analysis for Aircraft Crash Into Hazardous Facilities" (1996)

PART 10
(a) # SEISMIC MARGIN
(b) # ASSESSMENT REQUIREMENTS AT-POWER

Section 10-1
Overview of Requirements for Seismic Margins At-Power

10-1.1 SCOPE

This Part establishes the technical requirements for a seismic margin assessment while at-power.

10-1.2 COORDINATION WITH OTHER PARTS OF THIS STANDARD

This Part is intended to be used together with Part 1 of this Standard. Some notes and requirements in Part 5 may also provide useful reference information.

10-1.3 SEISMIC MARGINS SCOPE

The scope of this Part includes the widely used SMA methodology. SMA methods employ many of the same tools as a seismic PRA, although they are not full-scope seismic PRAs (i.e., they do not provide CDF or LERF estimates). SMA methods can be used, as appropriate, for risk-informed applications.

The scope of a seismic margin assessment (SMA) covered by this Part is limited to analyzing nuclear power plant seismic capacities according to the Electric Power Research Institute (EPRI) method ("EPRI method") [10-1], the U.S. Nuclear Regulatory Commission (NRC) method ("NRC method") [10-2], or the "PRA-based seismic margin evaluation" method [10-16].

10-1.4 FIDELITY: PLANT VERSUS SEISMIC MARGIN ASSESSMENT

It is important that the SMA reasonably reflect the actual as-built, as-operated nuclear power plant being analyzed. Several mechanisms are used to achieve this fidelity between plant and analysis. One key mechanism is called "plant familiarization." During this phase, plant information is collected and examined. This involves

(a) information sources, including design information, operational information, maintenance information, and engineering information

(b) plant walkdowns, both inside and outside the plant

Later, if the plant or the PRA is modified, it remains important to ensure that fidelity is preserved, and hence, further plant-familiarization work is necessary.

Throughout this Standard, requirements can be found whose objective is to ensure fidelity between plant and analysis. Because SMAs depend critically on plant walkdowns, both inside and outside the plant, to ascertain the physical configurations of important SSCs and the environments to which they are exposed, this Part places special emphasis on *walkdowns*, through requirements in the relevant Parts dealing with SSC fragilities due to earthquakes (see 5-2.2 and Tables 10-2-3, 10-2-4, and 10-2-5), and Section 10-3, which addresses peer review.

Section 10-2
Technical Requirements for Seismic Margin At-Power

In the mid-1980s, two different methodologies for the seismic margin assessment (SMA) of nuclear power plants were developed. These are the "NRC method" [10-2, 10-3] and the "EPRI method" [10-1]. Recently, an SMA method known as the "PRA-based seismic margin evaluation" method has evolved and come into use, especially for certain design-certification applications to the NRC for new LWR designs [10-16]. *The Requirements herein are explicitly directed toward an analysis using the "EPRI method,"* which employs success-path–type systems-analysis methods.

Using the "NRC method" SMA: If an SMA uses the "NRC method," then only some of the requirements of this Section are applicable. Specifically, an NRC-type SMA uses fault-space systems-analysis logic but limits the scope of SSCs to what the NRC guidance documents call the "Group A" safety functions, namely, reactivity control, normal cooldown, and inventory control during early times after the earthquake. These are not all of the important safety functions — for example, no consideration is given in an NRC-type SMA to maintaining extended inventory control or to mitigation-type safety functions such as the performance of containment or containment systems (fans, sprays, pressure suppression, etc). Hence, the scope of the systems-analysis part of an NRC-type SMA is less than the scope of a full seismic PRA.

To meet this Part with an SMA that uses the "NRC method," all of the high-level requirements (and supporting requirements) apply except Requirements HLR-SM-B and HLR-SM-G. Instead, the requirements in Part 5, covering the systems-analysis part of a seismic PRA but limited to the "Group A" safety functions, must be used. *An "NRC method" SMA meets this Part by meeting the above combination of requirements.*

Using the "PRA-based seismic margin evaluation" method: If an SMA uses this method, then only some of the requirements of this Part are applicable. Specifically, this method uses fault-space systems-analysis logic similar to that in a seismic PRA.

To meet this Part with an SMA that uses this new method, all of the high-level requirements (and supporting requirements) apply except Requirements HLR-SM-B and HLR-SM-G. Instead, the requirements in Part 5, covering the systems-analysis part of a seismic PRA, must be used. *A "PRA-based seismic margin evaluation" SMA meets this Part by meeting the above combination of requirements.*

Both the "NRC method" and the "PRA-based seismic margin evaluation" method employ a few special features different from the "EPRI method" besides the systems-analysis difference cited just above, but these have not been considered important enough to merit special requirements herein because the likelihood of a misapplication is judged to be small, assuming that the systems-analysis aspects meet the relevant seismic-PRA requirements in Part 5. One important difference between the three SMA methods is that the "NRC method" and the "PRA-based seismic margin evaluation" method explicitly treat nonseismic failures and human errors, along with seismic-caused failures, in an integrated systems analysis that uses fault-space methods instead of the two success paths used in the "EPRI method." Thus, both these methods are capable of certain insights that an EPRI-type SMA cannot identify without enhancements of the type discussed in Nonmandatory Appendix 10-B.

The technical requirements for SMA have been developed based on the SMA methodology guidance developed by both EPRI [10-1] and NRC [10-2, 10-3], plus the experience gained in performing several dozen SMAs for nuclear power plants. Other useful references include references [10-4, 10-5], and [10-6] through [10-9].

A primer about the SMA methodology can be found in Nonmandatory Appendix 10-A, and an extended discussion of the SMA methodology and applications using it can be found in Nonmandatory Appendix 10-B. Nonmandatory Appendix 10-B discusses applications for which a well-executed SMA that meets this Part is suited, applications for which it could be suited if certain enhancements are accomplished, and the limitations of the methodology.

This Part permits the use of issue-focused specific PRA evaluations or SMA enhancements (see Nonmandatory Appendix 10-B) to augment an SMA. The analyst needs to document the technical basis for the adequacy of the methodology, and a peer review needs to focus on it. The EPRI SMA guidance document [10-1] gives just this guidance on this issue in the following quote: "Still another approach would be to perform a limited-scope Seismic PRA, which focuses on the particular function that is questionable, and be able to demonstrate an acceptable risk. This approach would have merit if limiting systems were in alternate parallel paths but have limited benefit if the limiting system(s) was required to support all paths. If this approach were taken, most of the systems work done during the [SMA] review would be directly applicable. Seismic hazard curves and equipment fragilities for the specific equipment would have to be developed, but much of the plant modeling work would have been done."

As discussed in detail in Nonmandatory Appendix 10-B, an SMA can be used to support a variety of risk applications. These can be categorized roughly as follows, while noting that various enhancements (discussed in Nonmandatory Appendix 10-B) can provide stronger support if needed for any of these types of applications, and also noting that whether a specific application can be supported will depend on the following details:

(a) determination that the plant risk profile is acceptably low
(b) evaluation of component significance in a risk-ranking application
(c) implications of risk profile for components within the safe shutdown path
(d) assessment of component significance for those components not included in a safe shutdown path

All of these types of applications involve an assessment of the safety significance of a particular activity or characteristic of the plant. This can sometimes be determined qualitatively by evaluating the nature of the component, system, or activity and its relationship to the way overall safety is ensured.

Table 10-2-1 High Level Requirements for Seismic Margin Assessment: Technical Requirements (SM)

Designator	Requirement
HLR-SM-A	A review level earthquake characterized by a ground motion spectrum shall be selected to facilitate screening of SSCs and performance of seismic margin calculations.
HLR-SM-B	A minimum of two diverse success paths shall be developed consisting of structures and equipment that can be used to bring the plant to a safe stable state and maintain this condition for at least 72 hr following an earthquake equal to or larger than the review level earthquake.
HLR-SM-C	Seismic responses calculated for the review level earthquake shall be median centered, shall be based on current state-of-the-art methods of structural modeling, and shall include the effects of soil-structure interaction where applicable.
HLR-SM-D	The screening of components and subsequent seismic margin calculations shall incorporate the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support, and potential spatial interactions.
HRL-SM-E	Seismic margin calculations shall be performed for critical failure modes of SSCs such as structural failure modes and functional failure modes identified through the review of plant design documents, including analysis and test reports, and the results of a plant walkdown supplemented by earthquake experience data, fragility test data, and generic qualification test data.
HRL-SM-F	The calculation of seismic margins [or so-called high confidence of low probability of failure (HCLPF) capacities] shall be based on plant-specific data supplemented by earthquake experience data, fragility test data, and generic qualification test data. Use of such generic data shall be justified.
HRL-SM-G	The plant seismic margin shall be reported based on the margins calculated for the success paths.
HRL-SM-H	Documentation of the seismic margin assessment shall be consistent with the applicable supporting requirements.

Table 10-2-2 Supporting Requirements for HLR-SM-A

A review level earthquake characterized by a ground motion spectrum shall be selected to facilitate screening of SSCs and performance of seismic margin calculations (HLR-SM-A).

Index No. SM-A	Requirement
SM-A1	SELECT a review level earthquake as an earthquake larger than the safe shutdown earthquake for the plant.
Commentary: The seismic margin methodology is designed to demonstrate sufficient margin over the safe shutdown earthquake (SSE) to ensure plant safety and to find any weak links that might limit the plant's capability to safely withstand a seismic event larger than the SSE. The review level earthquake (RLE) is used to screen components based on generic seismic capacity. Screening is done in a seismic margin assessment (SMA) to optimize the resources needed and to focus attention on more critical and potentially seismically weak components. Reference [10-1] contains useful guidance on the selection of the RLE. The seismic margin method typically uses two review or screening levels geared to peak ground accelerations (PGAs) of 0.3*g* and 0.5*g*. Based on the guidance given in NUREG-1407 [10-4], most plants in the central and eastern U.S. have selected 0.3*g* PGA as the RLE for their SMAs. For some sites where the seismic hazard is judged to be low (i.e., $<10^{-4}$/year at SSE), a reduced-scope margin assessment relying mainly on a walkdown has been considered acceptable. NUREG-1407 further states that an RLE of 0.5*g* should be used for sites in the western U.S. except for the California coastal sites, for which the seismic margin methodology is not acceptable.	
SM-A2	CHARACTERIZE the review level earthquake by a ground motion spectrum appropriate for the site conditions.
Commentary: Based on the guidance in NUREG-1407 [10-4], seismic margin assessments have been done using the 5% damped NUREG/CR-0098 [10-10] median rock or soil spectrum anchored at 0.3*g* or 0.5*g* [depending on the review level earthquake (RLE) for the site]. Alternative approaches for selecting the RLE spectrum are described in reference [10-1]. The shape of the RLE ground motion spectrum is needed to develop seismic responses of structures and equipment for the calculation of seismic margins.	

Table 10-2-3 Supporting Requirements for HLR-SM-B

A minimum of two diverse success paths shall be developed consisting of structures and equipment that can be used to bring the plant to a safe stable state and maintain this condition for at least 72 hr following an earthquake larger than the review level earthquake (HLR-SM-B).

Index No. SM-B	Requirement
SM-B1	SELECT a primary success path and an alternative success path, one of which is capable of mitigating a small loss-of-coolant accident. In the success paths, INCLUDE systems whose function is to prevent severe core damage and their support systems.

Commentary: A set of 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 is known as a "success path." Based on the selected success paths, a safe shutdown equipment list (SSEL) is then developed for subsequent screening, walkdown, and margin evaluation.

It is advisable to compare the SSEL for reasonableness with comparable SSEL lists compiled for seismic margin assessments at other similar nuclear power plants.

Index No. SM-B	Requirement
SM-B2	ENSURE that the success paths have the following properties: they are those for which there is a high likelihood of an adequate seismic margin, they are compatible with plant operating procedures, and they have acceptable operational reliability.

Commentary: It is desirable that to the maximum extent possible, the alternative path involves operational sequences, systems, distribution systems (i.e., piping, raceways, duct, and tubing), and components different from those used in the primary path. Reference [10-1] contains useful guidance on the selection of success paths, on the use of success path logic diagrams in their selection, and on how "acceptable operational reliability" is defined for the seismic margin assessment review. Generally, the approach is to choose one success path that can mitigate sequences that start with a loss of off-site power transient and the other success path that can mitigate a small LOCA. Also, the SSCs are generally to be selected to enhance diversity and to avoid those with low reliability. See NUREG-1407 [10-4] for further guidance.

Index No. SM-B	Requirement
SM-B3	ASSUME that off-site power has failed and is not recoverable during the 72-hr period of interest following the review level earthquake.

Commentary: Earthquake experience has shown that off-site power is almost always lost after any earthquake larger than the safe shutdown earthquake. Because of the potential damage to the electric grid and the region surrounding the plant, it is judged that the off-site power may not be recovered for up to 72 hr. Therefore, the selected success paths should be able to provide core cooling and decay heat removal for at least 72 hr following the earthquake, without recourse to off-site power. Although no credit for off-site power is taken in the seismic margin assessment (SMA), one also must be aware of possible adverse effects if off-site power remains available or is restored. In the internal-event PRA, the analyst assumes that there would be a successful scram given the loss of off-site power. The probability of mechanical binding of control rods is deemed low; hence, there is no need to examine if the reactor protection system will function.

However, in the case of SMA, the analyst should verify if the reactor protection system works and if the control rod could drop given the potential for seismically induced deformation of the reactor internals and failure of the control rod drive mechanism. Further, the power conversion system (e.g., the main condenser) should be assumed as not available for heat sink function, and any equipment powered by nonvital alternating-current is also considered unavailable.

Table 10-2-3 Supporting Requirements for HLR-SM-B (Cont'd)

A minimum of two diverse success paths shall be developed consisting of structures and equipment that can be used to bring the plant to a safe stable state and maintain this condition for at least 72 hr following an earthquake larger than the review level earthquake (HLR-SM-B).

Index No. SM-B	Requirement
SM-B4	In the seismic margin assessment, ANALYZE at least seismically initiated transient events and small seismically induced primary coolant leakage events (referred to as "small LOCA").
Commentary: A detailed walkdown within the containment, to verify that all small instrumentation or impulse lines can withstand the review level earthquake (RLE) and that there are no potential spatial interactions resulting in their failure to add up to an area of 25-mm diameter, would lead to excessive radiation exposure of the walkdown team. Therefore, it is considered prudent and expedient to concede that a small LOCA will occur after an RLE and to include the required mitigation systems in the success path (see Requirement SPR-B8 in Part 5).	
SM-B5	If one element in the success path logic diagram represents a multitrain system, DETERMINE safety function success at the system level, not at the train level.
Commentary: If one train of a system is judged to be seismically rugged (exclusive of a train-specific spatial interaction failure), then all trains of that system are considered rugged if the equipment items are identical. Reference [10-1] states further that this assumption is valid if the trainwise layout is similar, although train-specific systems interaction problems may invalidate this assumption.	
SM-B6	ENSURE that nonseismic failure modes and human actions identified on the success paths have low enough probabilities so as not to affect the seismic margin evaluation. USE a documented method for ensuring this.
Commentary: Non–seismic-caused component system unavailabilities are not explicitly addressed in a seismic margin assessment (SMA) by quantifying them, but they are identified and avoided on the success paths where necessary. This issue is covered implicitly in the Electric Power Research Institute SMA guidance [10-1] by the requirement therein to avoid unreliable equipment. This should be reasonable for systems that have multiple and redundant trains but should be treated with caution for a single train with recognized high unavailability. The screening criteria cited in the NRC's IPEEE guidance, NUREG/CR-5679 [10-11], addressing both single-train and multitrain systems, MAY be used as guidance.	
SM-B7	EVALUATE the potential effects of seismically induced relay and contactor chatter as well as the operator actions that may be required to recover from any such effects.
Commentary: Guidance on evaluation of relay chatter effects is given in references [10-1], [10-4], and [10-12].	
SM-B8	As part of the seismic margin assessment, EVALUATE SSCs needed to prevent early containment failure following core damage.
Commentary: NUREG-1407 [10-4] identifies these functions. These functions are containment integrity, containment isolation, prevention of bypass, and some specific systems depending on the containment design (for example, igniters or ice baskets). The purpose of this examination is to evaluate whether these SSCs have enough seismic margin to function at earthquake levels above the design basis.	

Table 10-2-4 Supporting Requirements for HLR-SM-C

Seismic responses calculated for the review level earthquake shall be median centered, shall be based on current state-of-the-art methods of structural modeling, and shall include the effects of soil-structure interaction where applicable (HLR-SM-C).

Index No. SM-C	Requirement
SM-C1	ENSURE that seismic responses calculated for the review level earthquake are median centered, are based on current state-of-the-art methods of structural modeling, and include the effects of soil-structure interactions where applicable.
Commentary: The median-centered responses are calculated using EPRI-NP-6041-SL, Rev. 1 [10-1]. Here, "median centered" means that medians are being used to establish distributions, and not that medians represent best estimates for single-value calculations.	
SM-C2	DERIVE realistic seismic responses.
Commentary: Depending on the site conditions and response analysis methods used in the plant design, realistic seismic responses could be obtained using a combination of scaling, new analysis, and new structural models.	
SM-C3	For soil sites or when the design response analysis models are judged not to be realistic and state of the art, or when the design input ground motion is significantly different from the site-specific input motion, PERFORM new analysis to obtain realistic structural loads and floor response spectra.
Commentary: Further details about the basis for this requirement can be found in reference [10-13].	
SM-C4	ENSURE that soil-structure interaction (SSI) analysis is median centered using median properties at soil strain levels corresponding to the review level earthquake input ground motion. CONDUCT at least three SSI analyses to investigate the effects on response due to uncertainty in soil properties. ENSURE that one analysis is at the median low strain soil shear modulus and additional analyses at the median value times $(1 + C_v)$ and the median value divided by $(1 + C_v)$, where C_v is a factor that accounts for uncertainties in the SSI analysis and soil properties. If adequate soil investigation data are available, SPECIFY the mean and standard deviation of the low strain shear modulus for every soil layer. SPECIFY the value of C_v so that it will cover the mean plus or minus one standard deviation for every layer. For the minimum value of C_v, USE 0.5. When insufficient data are available to address uncertainty in soil properties, USE C_v at a value not less than 1.0.
Commentary: Further details about the basis for this requirement can be found in reference [10-13].	

Table 10-2-5 Supporting Requirements for HLR-SM-D

The screening of components and subsequent seismic margin calculations shall incorporate the findings of a detailed walkdown of the plant focusing on the anchorage, lateral seismic support, and potential spatial interactions (HLR-SM-D).

Index No. SM-D	Requirement
SM-D1	If SSCs on the safe shutdown equipment list are screened out on the basis of their generic high seismic capacity exceeding the review level earthquake, CONFIRM the basis for such screening through a walkdown. (See Requirement SM-H2.)
Commentary: None	
SM-D2	CONDUCT a detailed walkdown of the plant, focusing on equipment anchorage, lateral seismic support, and potential systems spatial interactions. The purposes of such a walkdown are to find as-designed, as-built, and as-operated seismic weaknesses in the plant and to ensure that the seismic margins are realistic and plant specific.
Commentary: None	
SM-D3	CONDUCT the walkdown consistent with the guidance given in reference [10-1].
Commentary: None	
SM-D4	If components are screened out during or following the walkdown, EVALUATE anchorages to justify such screening out.
Commentary: Normally, an anchorage calculation is required to support the screening. In some cases, the analyst MAY use judgment in deciding the adequacy of anchorage. Such judgments should be documented. For details and scope of anchorage evaluation, the reader is referred to references [10-1] and [10-24].	
SM-D5	During the walkdown, IDENTIFY the potential for seismically induced fire and flooding following the guidance given in NUREG-1407 [10-4].
Commentary: Normally, if the walkdown team identifies a potential seismically induced fire issue or a seismically induced flooding issue, it should be reviewed by the plant personnel and is either dismissed on a defined basis or remedied if necessary. Only rarely is the seismic margin assessment (SMA) analysis team faced with the task of quantifying a seismic margin for seismically induced fire or flooding issues. However, if this is needed, the assessment must quantify the relevant high confidence of low probability of failure (HCLPF) capacities and integrate these with the systems-analysis aspect as in any other aspect of SMA.	
SM-D6	In the walkdown, EXAMINE potential sources of spatial interaction (e.g., II/I issues, impact between cabinets, flooding, and spray) and consequences of such interactions on SSCs contained in the safe shutdown equipment list, and INCLUDE them in the analysis as appropriate.
Commentary: A “II/I issue” refers to any object (whether seismically qualified to the plant design basis or not) that can fall on and damage any item on the SSEL. The HCLPF capacity of the falling object may control the HCLPF capacity of the success path and potentially the plant HCLPF capacity if it is less than the HCLPF capacity of the weakest item on the SSEL.	

Table 10-2-6 Supporting Requirements for HLR-SM-E

Seismic margin calculations shall be performed for critical failure modes of SSCs such as structural failure modes and functional failure modes identified through the review of plant design documents including analysis and test reports and the results of a plant walkdown supplemented by earthquake experience data, fragility test data, and generic qualification test data (HLR-SM-E).

Index No. SM-E	Requirement
SM-E1	IDENTIFY realistic failure modes of screened-in structures, distribution systems, and components that interfere with the operability of equipment during or after the earthquake through review of plant design documents and the walkdown.
Commentary: None	
SM-E2	EVALUATE all relevant failure modes of structures (e.g., sliding, overturning, yielding, and excessive drift), equipment (e.g., anchorage failure, impact with adjacent equipment or structures, bracing failure, and functional failure), and soil (i.e., liquefaction, slope instability, and excessive differential settlement), and CALCULATE the high confidence of low probability of failure (HCLPF) capacities for the critical failure modes.
Commentary: The concept of high confidence of low probability of failure (HCLPF) capacity as an indicator of seismic margin was introduced in reference [10-2]. Examples of calculations of HCLPF capacities for a selected set of SSCs can be found in reference [10-6]. Detailed and more prescriptive guidance on methods for calculating HCLPF capacities of SSCs under different critical failure modes can be found in references [10-1] and [10-7]. Past seismic margin assessment reviews and seismic PRAs MAY also be used as guidance.	

Table 10-2-7 Supporting Requirements for HLR-SM-F

The calculation of seismic margins [or so-called high confidence of low probability of failure (HCLPF) capacities] shall be based on plant-specific data supplemented by earthquake experience data, fragility test data, and generic qualification test data. Use of such generic data shall be justified (HLR-SM-F).

Index No. SM-F	Requirement
SM-F1	CALCULATE the high confidence of low probability of failure (HCLPF) capacities for all components and structures that are screened in based on plant-specific information, such as site-specific seismic input, anchoring and installation of the component or structure, spatial interaction, and plant-specific material test data.

Commentary: The component high confidence of low probability of failure (HCLPF) capacities can be calculated using the conservative deterministic failure margin method proposed in reference [10-1]. Note that the HCLPF capacity is calculated assuming that only normal operating loads are present at the time of the seismic margin earthquake. In the case of boiling water reactors, the safety relief valve response is combined with the seismic response.

Index No. SM-F	Requirement
SM-F2	CALCULATE seismic high confidence of low probability of failure (HCLPF) capacities for SSCs that are identified in the internal-events-PRA systems model as playing a role in the large early release frequency part of the PRA analysis (see Requirement HLR-SPR-A in Part 5).

Commentary: Generally, the concern is the seismically induced early failure of containment functions. NUREG-1407 [10-4] describes these functions as containment integrity, containment isolation, prevention of bypass functions, and some specific systems, depending on the containment design (e.g., igniters, suppression pools, and ice baskets).

Table 10-2-8 Supporting Requirements for HLR-SM-G

The plant seismic margin shall be reported based on the margins calculated for the success paths (HLR-SM-G).

Index No. SM-G	Requirement
SM-G1	REPORT plant seismic margin based on the margins calculated for the SSCs on the success paths.

Commentary: The various individual high confidence of low probability of failure (HCLPF) capacities are combined by using the so-called "min-max" method, described in reference [10-3].

Table 10-2-9 Supporting Requirements for HLR-SM-H

Documentation of the seismic margin assessment shall be consistent with the applicable supporting requirements (HLR-SM-H).

Index No. SM-H	Requirement
SM-H1	DOCUMENT the seismic margin assessment in a manner that facilitates PRA applications, upgrades, and peer review.
Commentary: None	
SM-H2	DOCUMENT the process used in the seismic margin assessment. For example, this documentation typically includes a description of *(a)* the methodologies used to quantify the seismic margins or high confidence of low probability of failure (HCLPF) capacities of SSCs, together with key assumptions *(b)* a detailed list of SSC margin values that includes the method of seismic qualification, the dominant failure mode(s), the source of information, and the location of each SSC *(c)* for each analyzed SSC, the parameter values defining the seismic margin [i.e., the high confidence of low probability of failure (HCLPF) capacity and any other parameter values such as the median acceleration capacity and the beta values] and the technical bases for them *(d)* the basis for screening of any SSCs on the safe shutdown equipment list based on their generic high seismic capacity exceeding the review level earthquake (see Requirement SM-D1) *(e)* the key aspects of the seismic margin assessment, such as *(1)* the selection of the review level earthquake *(2)* the development of success paths and the safe shutdown equipment list *(3)* the seismic response analysis; the screening *(4)* the walkdown; the review of design documents *(5)* the identification of critical failure modes for each SSC *(6)* the calculation of high confidence of low probability of failure (HCLPF) capacities for each screened-in SSC
Commentary: The documentation requirements given in references [10-1] and [10-4] may be used as guidance.	
SM-H3	DOCUMENT the sources of model uncertainty and related assumptions associated with the seismic margin assessment.
Commentary: None	

Section 10-3
Peer Review for Seismic Margins At-Power

10-3.1 PURPOSE

This Section provides requirements for peer review of a seismic margin assessment at-power.

10-3.2 PEER-REVIEW COMPOSITION AND PERSONNEL QUALIFICATIONS

The peer-review team shall have knowledge and collective experience in the areas of systems engineering, seismic hazard, seismic capability engineering, and seismic PRAs or seismic margin methodologies, as applicable to the scope of the review. Section 1-6 provides general requirements for peer review. Subsection 1-6.2 specifies requirements for peer-review team knowledge and collective experience. Paragraph 1-6.1.1 specifies requirements regarding peer-review scope. The reviewer(s) focusing on the seismic-fragility work shall have demonstrated experience in seismic walkdowns of nuclear power plants.

10-3.3 REVIEW OF SEISMIC MARGIN ELEMENTS TO CONFIRM THE METHODOLOGY

10-3.3.1 Review Level Earthquake Selection

The peer-review team shall evaluate whether the selection of the review level earthquake used in the seismic margin assessment is appropriately specific to the site and has met the relevant requirements of this Standard.

10-3.3.2 Success Path Selection or Other Systems Analysis

The peer-review team shall evaluate whether the success paths are chosen properly and reflect the systems and operating procedures in the plant and that the preferred and alternative paths are reasonably redundant. (If either the "NRC method" or the "PRA-based seismic margin evaluation" method is used, the peer-review team shall evaluate whether the systems analysis has been accomplished properly.) The review team shall ensure that the safe shutdown equipment list is reasonable for the plant considering the reactor type, design vintage, and specific design.

10-3.3.3 Seismic Response Analysis

The peer-review team shall evaluate whether the seismic response analysis used in the development of seismic margins meets the relevant requirements of this Standard. Specifically, the review should focus on the input ground motion (i.e., spectrum or time history), structural modeling including soil-structure-interaction effects, parameters of structural response (e.g., structural damping and soil damping), and the reasonableness of the calculated seismic response for the review level earthquake input.

10-3.3.4 Seismic Walkdown

The peer-review team shall review the seismic walkdown of the plant to ensure the validity of the findings of the seismic review team on screening, seismic spatial interactions, and identification of critical failure modes.

10-3.3.5 Component Methods and Data

The peer-review team shall evaluate whether the methods and data used in the seismic margin analysis of components are adequate for the purpose. The review team should perform independent high confidence of low probability of failure (HCLPF) calculations of a selected sample of components covering different categories and contributions to plant margin.

10-3.3.6 Seismic Margin Assessment Methodology

The peer-review team shall evaluate whether the seismic margin assessment method used is appropriate and provides all the results and insights needed for risk-informed decisions. The review should focus on the high confidence of low probability of failure (HCLPF) capacities of components that contribute most to the seismic margins.

Section 10-4
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NONMANDATORY APPENDIX 10-A SEISMIC MARGIN ASSESSMENT METHODOLOGY: PRIMER

10-A.1 INTRODUCTION

The objective of a seismic margin review of a plant is to determine if the plant can safely withstand an earthquake larger than the design-basis earthquake (DBE), the safe shutdown earthquake (SSE). In the literature two seismic margin assessment (SMA) methods are described; one methodology was developed for the Electric Power Research Institute (EPRI) [10-A-1],[1] and another was developed for the U.S. Nuclear Regulatory Commission (NRC) [10-A-2, 10-A-3].

Most of the SMA reviews of nuclear power plants were performed to fulfill the requirements of the NRC's Individual Plant Examination of External Events (IPEEE) program [10-A-4], and most of them used the EPRI methodology. The NRC's IPEEE guidance asked that the SMA methodology be enhanced to include certain additional features (see below), and the enhanced SMA methodology that includes these additional features provides the major basis for the SMA requirements in this Part. It is also the basis for this appendix.

The margin methodology (either NRC or EPRI) utilizes a so-called review level earthquake (RLE). There is explicit guidance for two RLEs, one at 0.30*g* and the other at 0.50*g* [peak ground acceleration (PGA)]. The RLE for each plant in the U.S. was assigned by the NRC [10-A-3] based on the Lawrence Livermore National Laboratory and EPRI seismic hazard estimates [10-A-6, 10-A-7], sensitivity studies, seismological and engineering judgment, and plant design considerations. The type of SMA review has been further divided into three scopes by the NRC: a reduced-scope margin methodology that emphasizes plant walkdowns, a focused-scope methodology, and a full-scope methodology. The level of effort in the analysis of relay chatter is the major difference between the focused-scope and full-scope methodologies. The discussion presented in the following is primarily applicable to the focused- and full-scope seismic margin studies using the EPRI SMA methodology.

The EPRI methodology is based on a "success path" approach. Two success paths must be identified (see below). Each success path consists of a selected group of safety functions capable of bringing the nuclear plant to a safe state (hot or cold shutdown) after an earthquake larger than the DBE, and of maintaining it there for 72 hr. The individual structures, systems, or components, or a combination thereof (SSCs) needed to accomplish each of the success paths are then identified and become the basis for the rest of the SMA analysis. The SMA defines and evaluates the seismic capacity of each of the SSCs on the two success paths. Of course, for any nuclear plant several paths may exist. The NRC's IPEEE guidance [10-A-4, 10-A-5] required that the two success paths be selected so that they involve to the maximum extent possible systems, piping runs, and components that differ between the primary and the alternate success path. The NRC SMA Methodology is based on the fault tree approach whose systems-analysis elements are very similar to those in a seismic PRA. In the following, the discussion is limited to the EPRI SMA.

The "bottom-line results" of a well-executed SMA consist of estimates of the seismic capacities of each of the SSCs analyzed, from which are derived estimates of the seismic capacities of the needed safety functions, and then of the two success paths, leading ultimately to an estimate of the seismic capacity of the plant as a whole. In actual practice, a typical SMA is usually structured so that the estimated seismic capacities of many of the SSCs under consideration are lower bounds for the capacities rather than realistic estimates. The SMA capacity estimates are worked out in terms of the so-called high confidence of low probability of failure (HCLPF) capacity, which is expressed in terms of the earthquake "size" (say, 0.22*g* PGA or 0.29*g* spectral acceleration at 5 Hz) for which the analyst has a high confidence that the particular SSC will continue to perform its safety function.

When such an SMA has been completed, the principal results and insights are reported by findings such as "SSC number 4 has an HCLPF capacity of 0.22*g*," or ". . . has an HCLPF capacity of at least 0.30*g*." Using combinatorial rules that are intended to be conservative, the individual SSC capacities can then be combined to provide results such as "The service-water system has an HCLPF capacity of 0.22*g*," or "The residual heat removal safety function has an HCLPF capacity of 0.22*g*," or ultimately that "The plant as a whole has an HCLPF capacity of 0.22*g*," or of course perhaps ". . . has an HCLPF capacity of at least 0.30*g*."

10-A.2 THE SEVEN ELEMENTS OR STEPS

The seven elements of an SMA, as set down in the EPRI guidance [10-A-1], are summarized as follows (a

[1] The numeric citations in this Nonmandatory Appendix can be found in Section 1-7 of the main text.

more detailed discussion of each will be presented in the next section):

(a) Selection of the Seismic Margin Earthquake. This involves the specification of the earthquake for which the SMA is to be conducted. The NRC has defined this level for all plants as the RLE in NUREG-1407 [10-A-4].

(b) Selection of Assessment Team. The assessment team, called the seismic review team (SRT), is made up of senior systems engineers and seismic capability engineers. In accordance with the NRC guidance for the IPEEE, the SRT should incorporate utility personnel, to the maximum extent possible, so that results and insights obtained during the SMA can be utilized in plant operation, seismic upgrading, and accident management.

(c) Preparatory Work Prior to Walkdowns. The preparatory work prior to walkdowns consists of gathering and reviewing information about the plant design and operation. During this step, the systems engineers define the candidate success paths and the associated frontline and support systems and components. Preliminary or final estimates of realistic floor response spectra to the RLE are also developed in this step. The potential for soil liquefaction and slope instability is assessed considering the seismic sources in the site region and soil conditions. The objective is to assess if soil failures are likely at the RLE and to estimate the potential consequences on buildings, buried piping, and ground-mounted tanks.

(d) Systems and Elements Selection ("Success Paths") Walkdown. The primary objective of this step is a preliminary assessment of the relative seismic ruggedness of the major equipment in the candidate success paths and the selection of a preferred success path and an alternate success path.

(e) Seismic Capability Walkdown. This step involves the identification of any potential weak links in the SSCs required for the selected success paths. SSCs in the systems are screened in this step from further evaluation based on the EPRI screening criteria. Weak links to be considered include the potential for seismic spatial systems interactions, equipment anchorage, etc. Systems include all fluid, electrical power, and instrumentation systems in the success paths, as well as the frontline safety systems.

(f) Seismic Margin Assessment. This step carries out the SMA to demonstrate structural capacity or operability of those structures and equipment that are not screened out in steps 4 and 5. Seismic HCLPF capacity calculations are done to verify if sufficient margin over the RLE exists in the components selected in the success paths.

(g) Documentation. The documentation of the SMA, including information gathered in walkdowns, is completed in this step. Requirements on contents of the reports are given in EPRI NP-6041-SL, Rev. 1 [10-A-1] and NUREG-1407, Appendix C [10-A-4].

10-A.3 ENHANCEMENTS

In addition to the requirements outlined above in terms of the seven elements or steps, the following four *enhancements* to the EPRI SMA methodology are required to satisfy this Standard.

(a) Selection of Alternative Success Paths. The EPRI SMA as currently developed calls for evaluation of one preferred path and one alternative path. Following NUREG-1407, this Part recommends that a fuller set of potential success paths be set down initially. From this set, the number of paths is narrowed to one primary path and an alternate path.

(b) Treatment of Nonseismic Failures and Human Actions. This step involves the identification of nonseismic failures and human actions in the success paths. The success paths are chosen based on a screening criterion applied to nonseismic failures and needed human actions. It is important that the nonseismic failures and human actions identified have low enough failure probabilities so as not to affect the seismic capabilities of the success paths.

(c) Evaluation of Containment and Containment Systems. This step is intended to identify vulnerabilities that involve early failure of containment functions including containment integrity, containment isolation, prevention of bypass functions, and some specific systems that are included in the success paths.

(d) Relay Chatter Review. This step is intended to identify any vulnerabilities that might result from the seismic-caused chatter of relays and contactors.

Section 10-A.4 describes in more detail the seven elements or steps (1 through 7) and the four enhancements (A through D) required to carry out an SMA that meets this Part.

10-A.4 THE SEVEN ELEMENTS OR STEPS: DETAILED DISCUSSION

10-A.4.1 Step 1: Seismic Margin Earthquake and Its Level

The EPRI SMA methodology is based upon the selection of a seismic margin earthquake (SME). The SME as defined in reference [10-A-1] is equivalent to the RLE specified in reference [10-A-4], so here the terms are used interchangeably. The SME is defined as the earthquake level for which survivability is to be demonstrated for those systems and components that are required to bring the plant to, and maintain, a safe shutdown condition following the postulated earthquake. The SME is not a new design earthquake. It is a stylized earthquake used to evaluate whether the existing nuclear plant can perform safely during and after the earthquake is postulated to strike.

The RLE defines the screening level at which components and structures considered in the success paths are to be examined. For those SSCs that are not screened out during the walkdown phase, additional analyses are necessary to determine their HCLPF seismic capacities. It is likely that some SSCs will have HCLPF capacities that are below the RLE as determined from the detailed analysis. Thus, it is possible that the plant-level HCLPF capacity is found to be less than the RLE.

EPRI NP-6041-SL, Rev. 1 [10-A-1] discusses the selection of SME levels and the response spectrum shape for performing SMAs. Four different alternatives for specifying the SME and its level are discussed, using one of the following:

(a) horizontal PGA

(b) uniform hazard spectra

(c) earthquake magnitude range

(d) standard trial SME spectrum

For the seismic IPEEE using the seismic margin method, NUREG-1407 [10-A-4] specifies using a $0.3g$ RLE for most of the plant sites in the central and eastern U.S. and the NUREG/CR-0098 median rock or soil spectrum [10-A-8] anchored at the assigned PGA.

10-A.4.2 Step 2: Selection of Assessment Team

An SRT is formed consisting of senior seismic capability engineers who are responsible for the seismic capability walkdowns and for screening out components from further SMA. They also define any required SMA scope of work for those components not screened out. The SRT is assisted by other seismic capability engineers in collecting data and conducting HCLPF calculations.

The SRT consists of three to five members who possess the following qualifications:

(a) knowledge of the failure modes and performance of structures, tanks, piping, process and control equipment, active electrical components, etc., during strong earthquakes

(b) knowledge of nuclear design standards, seismic design practices, and equipment qualification practices for nuclear power plants

(c) ability to perform fragility/margins-type capability evaluations including structural/mechanical analyses of essential elements of nuclear power plants

(d) some general understanding of seismic PRA systems analysis and conclusions

(e) some general knowledge of the plant systems and functions

It is not necessary that each member of the team individually have strong capability in all of these areas or strong seismic experience for all of the elements identified in the success paths being considered. However, in the composite, the SRT should be strong in all of these areas. A good composite makeup of the SRT would include systems engineers, plant operations personnel, and seismic capability engineers.

Systems engineers must identify all reasonable alternate means to bring the plant to a stable condition. They also must identify all elements that comprise the frontline and support system components together with the associated electrical, fluid, and pneumatic systems for each of these success paths. The systems engineers have the principal responsibility for selecting the two success paths for which the seismic capability is to be assessed in detail.

Plant operations personnel on the SRT should be intimately knowledgeable about normal and emergency operating procedures and operator responses to abnormal situations. These experts should be aware of instrumentation and actuation systems required to support those operator actions that may be required to accomplish the safe shutdown objectives associated with the preferred and alternative success paths selected.

10-A.4.3 Step 3: Preparatory Work Prior to Walkdowns

10-A.4.3.1 Collection and Review of Plant Design Information. Considerable preparatory work in both the systems area and the seismic capability area is necessary prior to the walkdown. The systems engineers should initially review the plant design documents and familiarize themselves with the plant design features. Information is contained in the final safety analysis report (FSAR), piping and instrumentation drawings, electrical one-line drawings, plant arrangement drawings, topical reports, and plant specifications. Representative lists of safety functions, frontline systems that perform the functions, support systems and components, and dependency matrices between frontline and support systems should be reviewed. These lists should be made more plant specific prior to review by the systems personnel. The plant operations personnel familiar with the systems are the logical choice to perform a prescreening of any representative lists. These engineers should be able to

(a) identify the important plant functions

(b) identify the frontline and supporting systems required to perform necessary functions for plant shutdown

(c) identify alternate sequences to shut down the plant (success path logic diagrams)

(d) identify the elements of each system in each of the success paths

10-A.4.3.2 Preparation for the Systems and Element Selection. At this point, the systems engineers will be ready for the systems and element selection walkdown. At the same time, the plant seismic design documents should be reviewed by all or part of the SRT or by a seismic capability engineer under the direction of the SRT. The purpose of the review is to determine conformance of the individual elements of the plant design with screening guidelines. This review includes the seismic sections of the FSAR, sample equipment qualification reports, sample equipment specifications, seismic analyses conducted for defining floor spectra, floor spectra provided as required response spectra (RRSs) to equipment vendors, relay chatter documentation, representative equipment seismic anchorage analyses and designs, seismic qualification review team (SQRT) forms if available [10-A-1], and any topical reports associated with seismic issues.

Prior to the SRT walkdown, a summary of all the review items should be provided to the SRT. The SRT should be familiar with the plant design basis prior to the walkdown. A thorough understanding of the seismic design basis and approaches used for equipment qualification and anchorage is necessary for a credible screening of elements for the RLE. The SRT must have preliminary estimates of realistic floor spectra resulting from the RLE. Judgments can only be made on the adequacy of seismic ruggedness with an understanding of the seismic demand at the RLE level, and some measure of equipment anchorage capacity.

10-A.4.3.3 Development of Realistic Floor Spectra. Realistic median-centered response to the RLE of the structures and equipment that comprise the success paths is estimated in this task, to facilitate

(a) screening of structures and equipment

(b) evaluation of seismic HCLPF capacities of screened-in SSCs

Median in-structure responses could be obtained either by scaling of the SSE design analysis responses or by new analysis. EPRI NP-6041-SL, Rev. 1 [10-A-1] describes the conditions under which each method is appropriate.

10-A.4.4 Step 4: Systems and Elements Selection ("Success Paths") Walkdown

The systems and elements selection walkdown is an initial walkdown carried out by the systems engineers, one or more plant operations experts, and preferably at least one seismic capability engineer.

10-A.4.4.1 Purpose. The purposes of the walkdown are to

(a) review the previously developed plant system models (candidate success paths) for obvious RLE evaluation problems, such as missing anchorage or seismic spatial system interaction issues.

(b) select a primary success path and an alternate success path for the SMA, eliminating those elements or paths that cannot be evaluated for seismic adequacy economically. Ensure that one of these two paths is capable of mitigating a small loss-of-coolant accident. It is important that this initial screening be closely monitored by members of the SRT and thoroughly documented.

The primary success path should be that path for which it is judged easiest to demonstrate a high seismic margin and one that the plant operators would employ after a large earthquake based upon procedures and training. The primary success path should be a logical success path consistent with plant operational procedures.

Remote success paths unlikely to be used may have higher seismic margins exceeding RLE; however, their selection is inadvisable. The alternate path should involve operational sequences, systems piping runs, and components different from those in the preferred path.

The alternate path should contain levels of redundancy on the same order as that of the primary success path. In accordance with NRC guidelines in NUREG-1407 [10-A-4], a reasonably complete set of potential success paths should be initially identified. From this set, the number of paths is narrowed to the primary and alternative success paths following procedures established in EPRI NP-6359-D [10-A-7].

10-A.4.4.2 Communication Between Systems Engineers and Seismic Capability Engineers. The following information should be provided by the systems engineers to the seismic capability engineers prior to the seismic capability walkdown:

(a) a list of the primary and alternate success paths that are to be evaluated in the SMA, together with all important elements in these paths

(b) the components in each success path, clearly marked on plant arrangement drawings

(c) instrumentation required for safe shutdown

(d) a list of relays and contactors for which seismic-induced chatter must be precluded

10-A.4.5 Step 5: Seismic Capability Walkdown

The seismic capability walkdown is the responsibility of the SRT, assisted by seismic capability engineers. A systems engineer who was engaged in the system and element selection walkdown and a person knowledgeable in plant operations should also accompany the SRT. The seismic capability walkdown should concentrate on rooms that contain elements of the success paths previously selected by the systems engineer. The SRT should also be aware of seismic spatial interaction effects and make note of any deficiencies as they are generally an indicator of a lack of seismic concern on the part of plant operations and design personnel. The purposes of the seismic capability walkdown are to

(a) screen from the margin review all elements for which they estimate HCLPFs to exceed the RLE level based upon their combined experience and judgment and use of earthquake experience data as appropriate

(b) define the failure modes for elements that are not screened and the types of review analysis that should be conducted

(c) add to the list any systems interaction items that are judged to be potentially serious problems

Each item is to be reviewed by at least two members of the SRT. Decisions to screen should be unanimous.

Otherwise, concerns should be documented on walkdown forms for further review. All decisions to screen are documented on walkdown forms. The seismic capacity screening criteria in Tables 2-3 and 2-4 of EPRI NP-6041-SL, Rev. 1 [10-A-1] for civil structures and equipment and subsystems along with applicable caveats could be used for the screening. It is to be noted that ground motion levels in terms of the 5% damped peak spectral acceleration are used in the screening criteria because the spectral acceleration is a better descriptor of the potential for earthquake damage than is the PGA.

The SRT should "walk by" all components that are reasonably accessible and in nonradioactive or low-radioactive environments. Components that are inaccessible could be evaluated by alternative means such as photographic inspection or reliance on seismic reanalysis. If several components are similar, and are similarly anchored, then a sample component from this group could be inspected for the purpose of qualifying the group. The "similarity basis" is developed during the seismic capability preparatory work by reference to drawings, calculations, or specifications.

The 100% "walk-by" is to look for outliers, lack of similarity, anchorage that is different from that shown on drawings or prescribed in criteria for that component, potential systems interaction problems, situations that are at odds with the team members' experience, and other areas of seismic concern. If concerns exist, then the limited sample size for thorough inspection should be increased accordingly.

A major part of the walkdown is devoted to the evaluation of equipment anchorage, which typically consists of expansion bolts installed in concrete, cast-in-place bolts embedded in concrete, and welds to embedded steel members and to the equipment itself. Generic anchorage calculations for typical anchorage configurations and equipment types should be made prior to the walkdown in order to assist the SRT with making screening decisions in the field. All anchorage for equipment should be analyzed by either generic bounding or by analysis for individual equipment items. Generic bounding evaluation of equipment is preferred since it can be used to screen out whole classes of equipment. This minor effort performed prior to walkdowns ultimately saves time by narrowing the scope of the SMA work. EPRI NP-5228 [10-A-9][2] could be used as a guideline in evaluating generic capacities for common anchorage configurations.

The walk-by of subsystems (distribution systems such as piping; cable trays; conduit; and heating, ventilating, and air-conditioning ducting) could be handled on a sampling basis. The sample size will depend upon the seismic design basis and upon the number of seismic concerns expressed by the SRT during the walk-by of the selected sample.

For each of the elements that are not screened by the SRT walkdown and for each spatial interaction issue raised by the SRT, it may be necessary to gather field data. The amount of data to be gathered is dependent upon the amount of documentation that exists prior to the walkdown. The level of existing documentation is established during the seismic capability preparatory phase. Particularly, the SRT will determine during this walkdown whether the documentation accurately describes element anchorage details and seismic support details. If discrepancies are found, they are noted for further evaluation.

10-A.4.6 Step 6: Seismic Margin Assessment

At the completion of the walkdowns, a relatively small list of elements will remain for which a detailed review is required. For these elements, the SRT should have documented exactly what needs to be reviewed (anchorage, support details, seismic qualification test data, etc.).

Experience has shown that most of the SMA work will be concerned with support and anchorage details.

For those components requiring review, realistic median-centered input motion (demand) associated with the RLE will be available from the results of the work in step 3. This seismic demand will be specified in terms of in-structure (floor) response spectra at the base of the component. Once this demand is established, the next step is to compare it to the demand used in the seismic qualification of the component [i.e., SSE required response spectrum (RRS)]. When the RLE demand, throughout the frequency range of interest, is less than or approximately equal to the design demand for which the component has been previously designed and qualified, no further work is necessary to demonstrate capability to withstand the RLE.

In those instances where the RLE demand significantly exceeds the design demand in an important frequency range, or where the component has not had previous seismic qualification, seismic HCLPF capacity evaluations for the component are necessary. Capacity evaluations can be performed analytically for items such as equipment anchorage and storage tank, or can be performed by comparison with generic equipment qualification or fragility test data for functional failure mode of electromechanical equipment. If an analysis is required to determine the seismic HCLPF capacity of a component, the conservative deterministic failure margin (CDFM) approach discussed in EPRI NP-6041-SL, Rev. 1 [10-A-1] is used.

HCLPF capacities are documented for all elements in the primary and alternate success paths that have capacities less than the specified RLE. The element with the lowest HCLPF capacity in a success path establishes the seismic HCLPF capacity for the path. The higher

[2] Citations appearing in this appendix separate from the main text and not appearing in the main text are designated with "B" and are listed in 10-B.10.

seismic HCLPF capacity of the primary and alternative success paths is the seismic HCLPF capacity of the plant as a whole if both paths can mitigate an SLOCA or only one path can mitigate an SLOCA but the SLOCA path has a higher HCLPF than the other path. However, in the case where only one success path can mitigate an SLOCA and the path also has a lower HCLPF than the other path, then the plant HCLPF is governed by the SLOCA success path HCLPF.

10-A.4.7 Step 7: Documentation

Documentation requirements for the SMA are given in NUREG-1407, Appendix C [10-A-4]. Typical aspects that are documented include the selection of the RLE, the development of success paths and the safe shutdown equipment list, the seismic response analysis, the screening, the walkdown, the review of design documents, the identification of critical failure modes for each SSC, and the calculation of HCLPF capacities for each screened-in SSC.

10-A.5 THE FOUR ENHANCEMENTS: DETAILED DISCUSSION

As discussed in Section 10-A.1, the SMA, as documented in EPRI NP-6041-SL, Rev. 1 [10-A-1], is not sufficient to meet the requirements for the seismic IPEEE as specified in NUREG-1407 [10-A-4]. This subsection describes the methodology to be followed in meeting the additional requirements called for in NUREG-1407 for plants binned in the focused scope or full-scope review level.

10-A.5.1 Enhancement A: Selection of Alternative Success Paths

The incorporation of this enhancement in the seismic margin IPEEE was discussed above. The selection process of the final two success paths (primary and alternative) should be documented in accordance with NUREG-1407.

10-A.5.2 Enhancement B: Analysis of Nonseismic Failures and Human Actions

The analysis of nonseismic failures (i.e., random failures and maintenance unavailability) and human actions is of paramount importance. The success paths often rely upon certain human actions in order to bring the plant to safe shutdown conditions. Failure modes and the associated human actions should be identified, and it should be ensured that they have low enough failure probabilities so as not to affect the seismic margin evaluation. Those success paths that contain nonseismic failures and human actions with relatively high rates of failure are screened out. Redundancies along the primary and alternative success paths are analyzed and documented. This documentation should include those instances where a single component is isolated in performing a vital function along a success path.

10-A.5.3 Enhancement C: Evaluation of Containment and Containment Systems

Vulnerabilities that involve early failure of containment functions are identified and reviewed. The scope of the review is determined based upon the internal-events PRA. The evaluation of the containment performance follows the same methodology as described above. The walkdown of the containment systems would take place at the same time the seismic capability walkdown for SMA is being completed.

The integrity of the containment hatch, personnel air lock, and penetrations following the postulated event are addressed, as well as the capacities and anchorages on containment heat removal/pressure suppression systems. Seismic HCLPFs of containment components (e.g., containment fan coolers) are developed.

10-A.5.4 Enhancement D: Relay Chatter Evaluation

The relay chatter evaluation addresses the questions of

(a) whether the overall plant safety system could be adversely affected by relay malfunction in a seismic event

(b) whether the relays for which malfunction is unacceptable have an adequate seismic capacity

10-A.5.4.1 Procedure. The procedure for evaluating relays consists of the following three major steps:

(a) identification of the list of relays needing evaluation

(b) system consequence evaluation

(c) seismic HCLPF capacity evaluation

The first step consists of the identification of the set of relays associated with the systems and items of equipment that are considered in the success paths. The second step is a system-type screening process that evaluates the consequences of malfunction of the associated relays on system performance to determine if proper function of the relays is essential to safe shutdown. Credit is also taken for any existing procedures or operator actions that can rectify relay chatter-induced problems. Relays whose malfunction is acceptable are not required to be seismically rugged. This screening process is intended to reduce significantly the number of relays whose fragility must be evaluated in the third step. The seismic HCLPF capacities of the screened-in relays can be evaluated using the CDFM [10-A-1].

For a focused-scope margin review, only low seismic ruggedness relays (so-called "bad actor" relays) are examined [10-A-4, 10-A-5]. If important plant systems have such bad actor relays, electrical circuitry analysis is conducted to determine the impact of relay chatter. Relays whose chatter would have an adverse impact on the system performance are identified for replacement or further testing to verify seismic adequacy.

10-A.6 REFERENCES

[10-A-1] NTS Engineering, RPK Structural Mechanics Consulting, Pickard Lowe & Garrick, Woodward Clyde Consultants, and Duke Power Company, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Report EPRI NP-6041-SL, Rev. 1, Electric Power Research Institute (1991)

[10-A-2] R. J. Budnitz, P. J. Amico, C. A. Cornell, W. J. Hall, R. P. Kennedy, J. W. Reed, and M. Shinozuka, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," Report NUREG/CR-4334, Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission (1985)

[10-A-3] P. G. Prassinos, M. K. Ravindra, and J. B. Savy, "Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants," Report NUREG/CR-4482, Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission (1986)

[10-A-4] "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Report NUREG-1407, U.S. Nuclear Regulatory Commission (1991)

[10-A-5] "Individual Plant Examination for External Events (IPEEE) for Severe Accident Vulnerabilities-10 CFR 50.54(f)," Generic Letter No. 88-20, Supplement 4, U.S. Nuclear Regulatory Commission (1991)

[10-A-6] "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," Report NUREG-1488, U.S. Nuclear Regulatory Commission (1993)

[10-A-7] "Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue," Report EPRI NP-6395-D, Electric Power Research Institute (1989)

[10-A-8] N. W. Newmark and W. Hall, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants," Report NUREG/CR-0098, U.S. Nuclear Regulatory Commission (1978)

[10-A-9] "Seismic Verification of Nuclear Plant Equipment Anchorage," EPRI NP-5228, Electric Power Research Institute

NONMANDATORY APPENDIX 10-B SEISMIC MARGIN ASSESSMENT APPLICATIONS GUIDANCE, INCLUDING SEISMIC MARGIN ASSESSMENT WITH ENHANCEMENTS[1]

The objective of this Nonmandatory Appendix is to explore the extent to which a seismic margin assessment (SMA)[2] that meets this Part can be used to obtain various types of risk insights, either as is or after it has been enhanced in certain ways, some of which are relatively simple and straightforward.

Various seismic analysis methods may be used to obtain qualitative and/or quantitative risk insights to support risk-informed decision making. To describe the insights adequately, it is necessary to consider the different types of applications to which the insights might be applied.

10-B.1 DEFINITION OF A RISK INSIGHT

In its broadest sense, a risk insight is any statement that characterizes the risk of a facility or the role of components, procedures, systems, or structures in the risk profile. The risk insight can be either quantitative or qualitative. Further, the risk insight may be supported by detailed assessments or by simpler analyses sufficient to support the conclusion being stated. It may involve defining the relationship of the component or system to the suite of postulated initiators and the associated plant response.

It may be further described by doing numerical analysis, which adds additional information regarding the significance and importance of the component or system.

To summarize, insights often relate to the role of a system, procedure, structure, or component in responding to postulated events, as well as to the nature of the response or the significance of a failure to respond.

10-B.2 SPECIALIZED RISK INSIGHTS DERIVABLE FROM SEISMIC PRAS AND SEISMIC MARGIN ASSESSMENTS

There is a long list of risk insights derivable from probabilistic analyses of various kinds, be they internal-events probabilistic risk assessments (PRAs), external-events PRAs, SMAs, screening-type PRAs, or other specialized PRAs. This nonmandatory appendix will not dwell on all of them. However, there are a few types of insights that are *tailored to seismic-safety issues* and hence are specifically derivable from a seismic PRA or an SMA. This short subsection will discuss these to provide a context for the remainder of this Nonmandatory Appendix, which concentrates on applications using SMAs, including SMAs with various enhancements.

10-B.2.1 Types of Seismic-Related Insights

The specialized types of seismic-related insights can be broadly categorized as follows:

(a) What is the seismic risk (annual frequency of unacceptable seismic performance), usually cast in terms of core damage frequency (CDF) or large early release frequency (LERF), but also sometimes using other endpoints such as failure of a core damage success path or of a plant damage state?

(b) What is the seismic ground motion range that dominates the seismic risk?

(c) Which structures, or systems, or components, or a combination thereof (SSCs) are the significant contributors to the plant's seismic risk, measured by CDF, LERF, or another endpoint as in subpara. (a)?

(d) What is the median (or mean) seismic capacity of the plant as a whole as measured in terms of CDF or LERF, or of an individual SSC, or of a success path?

(e) What is the high confidence of low probability of failure (HCLPF) seismic capacity below which it is very unlikely that an individual SSC, or a success path, or the plant as a whole would suffer seismic damage?[3]

(f) Are there any "weaker" SSCs that reduce the HCLPF capacity of the plant as a whole below some predetermined earthquake review level?

10-B.2.2 Important Observations

A few important observations about the insights in 10-B.2.2 are as follows:

(a) A seismic PRA that meets this Part is capable of addressing all six types of insights in 10-B.2.2.

[1] In this Nonmandatory Appendix, as elsewhere, when the term "SMA" is used, the term is intended to refer to the Electric Power Research Institute (EPRI)–type seismic margin assessment methodology [10-B-4][2] unless explicitly stated otherwise.

[2] The numeric citations in this Nonmandatory Appendix can be found in Part 8 of the main text.

[3] See Nonmandatory Appendix 10-A for a definition of "HCLPF capacity."

(b) The seismic individual plant examinations of external events (IPEEEs) [10-B-1, 10-B-2, 10-B-3] had as their principal objective to address insight (f).

(c) Also, note that the SMA methodology, as originally conceived [10-B-4, 10-B-5], was directed at insights (e) and (f) but unless enhanced is not directly suited to addressing insights (a) through (d).

As discussed above, a principal objective of this appendix is to explore to what extent an SMA that meets this Standard can address insights of types listed in 10-B.2.2(a) through 10-B.2.2(d) if it is enhanced in certain ways, some of which are relatively simple and straightforward.

10-B.3 RISK-INFORMED APPLICATIONS

Risk-assessment studies have been found to contribute considerable valuable information, which can be communicated to plant operators, maintenance personnel, engineers, regulators, and the public. Both a general sense of the risk level and an appreciation of the risk contributors have value for these groups. These applications may require the blending of deterministic and risk information.

10-B.4 APPLICATIONS USING SEISMIC MARGIN ASSESSMENT METHODS

An SMA can be used to support a variety of risk applications. These can be categorized roughly as follows, while noting that various enhancements (discussed below) can provide stronger support if needed for any of these types of applications, and also noting that whether a specific application can be supported will depend on the details:

(a) determination that the plant risk profile is acceptably low

(b) evaluation of component significance in a risk-ranking application

(c) implications of risk profile for components within the safe shutdown path

(d) assessment of component significance for those components not included in a safe shutdown path

All of these types of applications involve an assessment of the safety significance of a particular activity or characteristic of the plant. This can sometimes be determined qualitatively by evaluating the nature of the component, system, or activity and its relationship to the way overall safety is assured.

10-B.5 QUALITATIVE INSIGHTS

Although the scope of an EPRI-type SMA is limited compared to that of a full seismic PRA, a wide variety of risk-informed applications can be supported by an SMA. (For our purposes here, the phrase "a well-executed SMA" translates into the phrase "an SMA that meets this Standard.") Furthermore, if an SMA is judged incapable of supporting an important class of risk-informed applications, several types of enhancements are available, ranging from modest extensions to the number of the SSCs considered to improving the approach in the systems analysis, to working out an approximate CDF, to developing a full-scope seismic PRA. The insights can be either qualitative (discussed in this subsection) or quantitative (discussed in 10-B.6).

A partial list of qualitative insights related to seismic issues that may support certain types of risk-informed decision making include the following:

(a) identification of SSCs not significantly impacted by seismic events

(b) identification of SSCs significantly impacted by seismic events

(c) potential modifications to SSCs that do not significantly impact their seismic capacity

(d) potential modifications to SSCs that significantly impact their seismic capacity

(e) identification of operator actions not significantly impacted by seismic events

(f) identification of operator actions potentially impacted by seismic events

In evaluating a given nuclear power plant, an SMA begins with the identification of two "success paths," each consisting of a selected group of safety functions capable of bringing the plant to a safe state after a large earthquake and of maintaining it there. The individual SSCs needed to accomplish each of these success paths are then identified and become the basis for the rest of the analysis.

Logically, it can be concluded that SSCs and operator actions within the SMA success path are important to postearthquake safe shutdown. Similarly, one may conclude that SSCs and operator actions outside the SMA seismic paths likely have less importance to seismic safety. However, this latter conclusion would have to take into account other factors (e.g., the need for support systems). Also, whether a particular operator action or a particular nonseismic failure of equipment is important for safety depends on detailed analysis (see below).

The "bottom-line results" of an SMA consist of *estimates of the seismic capacities* of each of the SSCs analyzed, from which are derived estimates of the seismic capacities of the needed safety functions, and then of the two success paths, leading ultimately to an *estimate of the seismic capacity of the plant as a whole*. In actual practice, a typical SMA is usually structured so that the estimated seismic capacities of many of the SSCs under consideration are lower bounds on the capacities rather than realistic estimates. The SMA capacity estimates are worked out in terms of the so-called HCLPF capacity, which is expressed in terms of the earthquake "size" [say, 0.22*g* peak ground acceleration (PGA), or 0.29*g* spectral acceleration at 5 Hz] for which the analyst has a high confidence that the particular SSC will continue to perform its safety function.

When such an SMA has been completed, the principal results and insights are reported by findings such as "SSC number 4 has an HCLPF capacity of 0.22*g*," or ". . . has an HCLPF capacity of at least 0.30*g*." Using combinational rules that are intended to be conservative [10-B-4], the individual SSC capacities can then be combined to provide results such as "The service-water system has an HCLPF capacity of 0.22*g*," or "The residual heat removal safety function has an HCLPF capacity of 0.22*g*," or ultimately that "The plant as a whole has an HCLPF capacity of 0.22*g*," or of course perhaps ". . . has an HCLPF capacity of at least 0.30*g*."

As it turns out, certain risk-informed applications may need no more information than statements like those above. Such applications can be supported fully by a well-executed SMA. (The examples in 10-B.8 and 10-B.9 illustrate some of the types of applications that can be supported.)

10-B.6 QUANTITATIVE INSIGHTS

However, some applications will require more quantitative information (see below), and to support them it would be necessary to enhance the SMA.[4] The simplest enhancement is to use the site-specific seismic hazard curves to calculate the mean annual frequency of the earthquake whose "size" corresponds to the HCLPF capacity of the SSC or function of interest. Given the knowledge of that frequency (call it "F"), the statement that "SSC number 4 has an HCLPF capacity of 0.22*g*" can be converted to a statement like "There is high confidence that an earthquake of mean annual frequency F, or any smaller earthquake, will not cause the failure of SSC number 4." (Here the mean annual frequency F corresponds to 0.22*g* according to the mean hazard curve.) Of course, if the HCLPF capacity for the plant as a whole is used, then the high confidence for the frequency F represents a high-confidence statement about the plant's seismic-caused CDF, although to make such a CDF statement in a robust way requires taking careful account of any nonseismic failures or human errors that could contribute.

While some care must be used in determining the frequency F, including attention to the uncertainty with which F is known, this type of insight can be very useful. Also, depending on whether the analysis uses the full family of hazard curves, or an approximation such as the mean curve, there will be a different level of confidence attached to the conclusions reached — and in any event, without further work it is difficult to ascertain exactly how much confidence (85% confidence? 99% confidence?) is embedded in the "high confidence" statement just mentioned.

[4] While this discussion speaks of enhancements to an EPRI-type SMA, it is of course feasible to develop an "enhanced SMA" from scratch.

Another type of enhancement is to develop a seismic-fragility curve, or a set of such curves, for each SSC of interest rather than working only with each SSC's HCLPF capacity. This enables the analyst to derive more accurate conclusions about the annual frequency of earthquake-induced undesired outcomes (SSC failure, system or function failure, etc.) than the high-confidence/bounding statement available using only the HCLPF seismic capacity. This is done by convolving the fragility curves with the hazard curves. Methods for accomplishing this type of seismic-fragility enhancement, either approximately or more rigorously, are well documented [10-B-6, 10-B-7] and are not difficult to execute.

A more extensive enhancement would be to supplement the two-success-path systems-analysis approach by a partial or perhaps even a full fault-space systems analysis similar to that employed in a seismic PRA. A truncated systems-analysis approach along these lines is what characterizes a U.S. Nuclear Regulatory Commission (NRC)–type SMA [10-B-5, 10-B-8] and is what differentiates it from the more commonly applied EPRI-type SMA [10-B-4], so performing this enhancement would be equivalent to developing an NRC-type SMA. Specifically, an NRC-type SMA uses fault-space systems-analysis logic (event trees and fault trees) but limits the scope of SSCs to what the NRC guidance documents call the "Group A" safety functions, namely, reactivity control, normal cooldown, and inventory control during early times after the earthquake. These are not all of the important safety functions — for example, no consideration is given in an NRC-type SMA to maintaining extended inventory control or to mitigation-type safety functions such as the performance of containment or containment systems (fans, sprays, pressure suppression, etc.). Hence, the scope of the systems-analysis part of an NRC-type SMA is less than the scope of a full seismic PRA. The "results" of such an SMA, like the "results" of an EPRI-type SMA, are limited (unless enhanced using approaches described herein) to statements about the plant-level seismic HCLPF capacity and corresponding subsidiary HCLPF capacities such as the HCLPF capacities of key accident sequences and SSCs. One important advantage of using fault-space systems-analysis logic is that nonseismic failures and human errors are incorporated fully and naturally into the analysis, which is not the case for the success-path-type systems-analysis logic of an EPRI SMA.

Another and more extensive enhancement, along the same lines, would be to expand the systems-analysis scope to include all of the SSCs normally included in a seismic PRA. Unless enhanced, this so-called "PRA-based SMA" still produces results that are limited to HCLPF capacities, but the approach can provide a full evaluation of all relevant SSCs, including all safety functions. (Of course, various enhancements to obtain approximate CDFs like those discussed elsewhere in this appendix are as fully applicable to this "PRA-based SMA" as they are to an EPRI-type SMA.) One example of how this type of PRA-based SMA has been used in the past is in analyzing the seismic margin of an advanced design, such as an analysis to support NRC's design-certification review. Because an advanced design is not linked to a specific site when it is being evaluated for certification, no site-specific seismic hazard curve is available. However, using a full PRA-type systems analysis coupled with an SMA-based HCLPF-capacity evaluation can provide very useful insights into the overall seismic capacity of the advanced design; it can also illuminate how balanced the risk contributors are across different types of SSCs and systems.

Finally, of course, the most extensive enhancement would be to use a well-executed SMA as the springboard for developing a full-scope seismic PRA. Much of the SMA's fragilities work can be used directly, as can important parts of the systems-analysis work.

None of these enhancements are technically difficult in the hands of skilled practitioners, although of course more resources are needed and more technical challenges ensue for the more complex enhancements. Importantly, each allows the analyst to support a range of risk-informed applications beyond those that the original (unenhanced) SMA can support. (Section 10-2 in the main text of this Part, which refers to and relies on Part 2, provides the requirements and guidance for using this Standard for risk-informed applications.)

10-B.7 UNCERTAINTY IN QUANTITATIVE SEISMIC RISK ESTIMATES

To utilize a risk study, it is important for the analyst to assure that the quality of the PRA is commensurate with what is needed in any given application. In this context, quality must be related directly to the application and involve consideration of the detail required to support the application as well as the role that the PRA result might play in the decision making. With respect to seismic risk, an obvious PRA-quality issue is the ability to make statements about the inherent uncertainties in the seismic risk information.

A risk profile by its very definition is intended to be a realistic estimate, about which uncertainty exists. For many applications, the ability to characterize the uncertainty distribution is every bit as important as the mean value or median value that might be quoted. Only by understanding the distribution, which represents the analyst's entire state of knowledge, is it possible to understand the risk itself.

The uncertainty associated with seismic risk is typically dominated by the uncertainty in the initiating-event frequency, local building response, and component seismic capacity. Sometimes, one or more elements are conservatively rather than realistically treated (for example, the local response is sometimes conservatively treated). Significant assumptions such as this can in some cases make it difficult to use the seismic risk profile, which is why realistic analysis is to be preferred.

10-B.8 QUALITATIVE EXAMPLES

It is useful to show, through a few illustrative examples, how a well-executed SMA that meets this Part, either as is or with certain enhancements, can be used to support various risk-informed decisions, and what the limitations are. We assume that the SMA has identified two success paths, determined the HCLPF seismic capacities of the important SSCs in each path, and from these determined the HCLPF seismic capacities of each success path and hence of the plant as a whole.

The examples below are hypothetical but realistic enough that they might apply to any plant that possesses a well-executed SMA. The list of examples below largely tracks the short list of qualitative-type insights that are presented in 10-B.5.

10-B.8.1 Example A: Identification of an SSC That Is Not Significantly Impacted by Earthquakes

Suppose that a particular SSC is found, using the SMA, to possess an HCLPF seismic capacity well in excess of 1*g* PGA. In general, except for sites with very high seismicity such as in coastal California, one can state with high confidence that such an SSC will not contribute significantly to seismic risk due to seismic-caused failures. A well-executed SMA can make such identifications.

Indeed, depending on how one defines "significantly," such a statement could be made for an SSC with an HCLPF capacity above, say, 0.30*g* PGA: recall that in the IPEEE reviews for most eastern-U.S. plants, 0.30*g* was used as the SMA review level earthquake (RLE) [10-B-1], and an SSC with HCLPF = 0.30*g* PGA was judged not to represent a "vulnerability" using the IPEEE program's definition [10-B-3].

10-B.8.2 Example B: Identification of an SSC Significantly Impacted by Earthquakes

Suppose that a particular SSC is found, using the SMA, to possess an HCLPF seismic capacity in the range of 0.05*g*. (Such a capacity is very weak, at the low end of capacities for most equipment even if not specifically designed for earthquakes.) If that SSC plays an important role in plant safety after an earthquake, for example, by being an essential part of one of the success paths, then one can conclude that the SSC is surely "significantly impacted" seismically. A well-executed SMA can make such identifications.

10-B.8.3 Example C: Potential Modification to an SSC That Does Not Significantly Impact Its Seismic Capacity

An important category of risk-informed decisions involves a proposal to modify an SSC in a way that does not significantly impact its seismic capacity. For example, suppose that the seismic capacity of a particular motor-operated valve is high and is controlled by its very strong anchorage and mounting. Suppose that a proposal is made to test the valve for operability only every 3 mo instead of monthly. A well-executed SMA can support the conclusion that the proposed testing-schedule change will not impact the valve's seismic capacity.

10-B.8.4 Example D: The Reverse of Example C

Suppose that a proposed modification clearly has some impact on the seismic capacity of a given SSC, which requires evaluation. An example would be a modification to the support of a pipe-supported valve by attaching it instead to a wall in order to alleviate a certain load on the associated pipe. A well-executed SMA can evaluate whether (or not) the support modification would change the seismic capacity of that valve, and if so by how much, and if so whether the change is "significant." In this case, "significant" would need to be defined in the context of the particular safety issue under study. (However, understanding the full contribution of the valve to risk is beyond the capability of an SMA unless it is enhanced; see 10-B.9 for discussions of some such enhancements.)

10-B.8.5 Example E: Identification of Operator Actions Significantly Impacted by a Large Earthquake

Suppose that a risk-informed decision depends on the safety significance of a specific operator action.

An example would be the action of switching over from injection mode to recirculation mode after an earthquake-caused small loss-of-coolant accident (LOCA) in the piping of a pressurized water reactor. If in fact this operator action is very likely to be needed after an important and challenging earthquake, a well-executed SMA should be able to ascertain this by identifying and evaluating the specific seismic small-LOCA vulnerability and the success path used to respond, which presumably would be a success path that requires the switchover action. (However, understanding the full contribution of the switchover action to risk is beyond the capability of an SMA unless it is enhanced; see 10-B.9 for discussions of some such enhancements.)

In each of the examples above, the safety-relevant risk insight can be derived from an SMA without necessarily enhancing it to obtain an approximate CDF. In that sense, this type of insight is "qualitative," although of course any SMA used to support such an insight must involve enough quantitative analysis to sort out what is and what is not important.

10-B.9 QUANTITATIVE EXAMPLES

It is useful to show, through some illustrative *quantitative examples*, how this all might work out in practice for a hypothetical plant that has completed an EPRI SMA that has been peer reviewed. We assume that the SMA has identified two success paths, determined the HCLPF capacities of the important SSCs in each path, and from these determined the HCLPF capacities of each complete success path.

In our hypothetical example, suppose that the SMA analysis determines that the SSC in Success Path 1 with the lowest HCLPF capacity is "Valve A," one particular valve in the safety-injection line, with HCLPF = 0.18*g* PGA and a failure mode of "failed closed." In this plant, if "Valve A" fails closed, the success path cannot be used. Suppose that the only other important SSC in this success path is found to be the refueling water storage tank used for safety injection, with HCLPF = 0.28*g* PGA. All other SSCs have significantly higher HCLPF capacities. For Success Path 2, every SSC has an HCLPF capacity of at least 0.30*g* PGA.

Given the above, the SMA determines that the plant as a whole has an HCLPF capacity of *at least 0.30g* because the "stronger" success path determines the plant's HCLPF capacity. This is equivalent to the statement, "There is high confidence that an earthquake whose "size" corresponds to 0.30*g* PGA will not cause a core damage accident."

10-B.9.1 Example 1: Determining a Bounding CDF

With the above information, a very simple and approximate earthquake-initiated CDF upper bound can easily be obtained. The approach is to calculate the mean annual frequency of the earthquake whose "size" corresponds to 0.30*g* PGA. Let us assume that using the site seismic hazard curves, the mean frequency of earthquakes at 0.30*g* is found to be 3×10^{-5}/yr. With this information, one can reach the following conclusion: "There is high confidence that an earthquake of annual frequency 3×10^{-5}, or any smaller earthquake, will not cause a core damage accident." This is equivalent to "There is high confidence that the plant's seismic-caused CDF is smaller than 3×10^{-5}/yr," although to make such a CDF statement in a robust way requires taking careful account of any nonseismic failures or human errors to assure that they are not important. Of course, as mentioned in 10-B.7, without further work it is difficult to ascertain exactly how much confidence (85% confidence? 99% confidence?) is embedded in the "high confidence" statement just mentioned. Also, this very simple and approximate CDF estimate can be improved upon substantially without much extra effort (see the further examples below).

10-B.9.2 Example 2: A Bounding CDF for a Slightly Different Case

Let us assume, as a variant case, that Success Path 2 is very weak seismically and that Success Path 1 is thus the only means of shutting down the plant after a major earthquake. Then, Success Path 1's HCLPF capacity represents the seismic capacity of the plant as a whole. In this case, the SMA finds that "Valve A," with HCLPF = 0.18*g* PGA, dominates the plant's seismic CDF. Again, as in Example 1, we can use the site seismic hazard curves to calculate the mean annual frequency of exceedance of the earthquake whose "size" corresponds to 0.18*g* PGA. Suppose that this mean frequency is found to be 8×10^{-5}/yr. With this information, one can reach the following conclusion: "There is high confidence that an earthquake of annual frequency 8×10^{-5}, or any smaller earthquake, will not cause a core damage accident." This is equivalent to the following: "There is high confidence that the plant seismic-caused mean CDF is smaller than 8×10^{-5}/yr." (Again, to make such a CDF statement in a robust way requires taking careful account of any nonseismic failures or human errors to assure that they are not important.) As with Example 1, without further work it is difficult to ascertain exactly how much confidence (85% confidence? 99% confidence?) is embedded in the "high confidence" statement just mentioned. Furthermore, if two SSCs on the same success path have approximately equal HCLPF seismic capacities that are both "low" and hence "significant," the actual HCLPF capacity of that success path will depend on how these are combined. The SMA guidance on this, using the min-max approach [10-4], has limitations under some circumstances that the analyst should be aware of and would need to overcome if a more accurate result were needed. Also, again as with Example 1, this very simple and approximate CDF estimate can be improved upon substantially without much extra effort (see the further examples below).

10-B.9.3 Example 3: A Better Estimate of CDF

We continue for this example with the variant of Example 2, in which Success Path 2 is very weak seismically, so that "Valve A" in Success Path 1 represents the weakest component. If a better estimate of CDF is sought, one approach is to develop a seismic-fragility curve for Valve A, using, for example, the guidance in reference [10-B-6] or reference [10-B-7]. By convolving this fragility curve with the site-specific seismic hazard curves, a better estimate can be obtained for the CDF. In fact, working simply with the two mean values gives a rough, albeit somewhat nonconservative, estimate. If, for example, the mean seismic capacity of Valve A (from the fragility curve) equals 0.45*g* PGA, and if the mean hazard curve at 0.45*g* PGA has a frequency of, say, 1×10^{-5}/yr, one can conclude that "the mean frequency with which Valve A will fail in earthquakes is ~1×10^{-5}/yr." If Valve A completely dominates the seismic capacity of the plant, then one can conclude that "the CDF is ~1×10^{-5}/yr." One can do better still, as shown in reference [10-B-7], by using the ground motion corresponding to the 10% confidence point on the seismic-fragility curve; the seismic CDF turns out to be approximately 0.5 times the frequency from the mean seismic hazard curve corresponding to that ground motion, with the caveat that careful account must be taken of any nonseismic failures or human errors that could contribute. The uncertainties surrounding this CDF estimate can also be estimated by using the full family of fragility curves and the full family of seismic hazard curves, as discussed below under 10-B.9.7 (Example 6).

10-B.9.4 Example 4: A Better Upper Bound on CDF

Let us return to the case in Example 1 in which both success paths exist and Success Path 2 is stronger and hence controls the seismic risk profile. Recall that every SSC in Success Path 2 was found in the SMA to have an HCLPF capacity in excess of 0.30*g* PGA. In Example 1, we determined a simple bounding CDF by assuming that it is equal to the mean annual frequency of a site earthquake motion exceeding 0.30*g* PGA, assuming as always that one has taken careful account of any nonseismic failures or human errors to assure that they are not important. To obtain a better upper bound, one can develop a set of full approximate fragility curves for a surrogate component with HCLPF capacity = 0.30*g*. The analyst could use generic values for the "beta" parameters in this work, as described in references [10-B-6] and [10-B-7]. By convolving the set of fragility curves with the full set of site hazard curves, a better value for the CDF upper bound can be obtained. This upper-bound-type conclusion is correct because the actual SSCs whose capacities govern the seismic capacity of the plant (and hence the seismic CDF) are known to have HCLPF capacities above 0.30*g*. However, we do not know how far above 0.30*g* they lie and hence how much lower the actual plant seismic-caused CDF might be. (It is possible, for example, that a single SSC with HCLPF at, say, 0.35*g* governs the seismic capacity, which would produce a plant seismic-caused CDF not very much lower than the upper-bound CDF we ascertained using the surrogate fragility curve as above.) For this case as for the case in Example 3, approaches described in reference [10-B-7] can be used to obtain approximate numerical results that may be sufficiently accurate for the analyst's purpose at hand.

10-B.9.5 Notes About Examples 1 Through 4

In all of the four examples above, a warning has been written that it is necessary to take careful account of any nonseismic failures or human errors that might contribute. Taking these into account, if they matter, is something that is not easily accomplished with an SMA whose systems-analysis aspect is based on evaluating two success paths. This is an intrinsic limitation, and to overcome it, one needs a systems analysis based on fault-space methods. These methods are discussed in the next two examples.

10-B.9.6 Example 5: An Improved Estimate of the Plant-as-a-Whole HCLPF Capacity

To arrive at a better estimate of the HCLPF capacity for the plant as a whole, one could use the seismic-capacity information in the SMA but could supplement it by developing a fault-space systems analysis so that, in effect, an NRC-type SMA has been developed. (The NRC-type SMA uses the same HCLPF-based seismic-capacity analysis as for an EPRI-type SMA, but instead of a two-success-path systems analysis, it uses a PRA-type fault-space systems analysis, albeit truncated compared to the fault-space systems analysis in a full seismic PRA.) Following the guidance in the NRC SMA methodology reports [10-B-5, 10-B-8], the analyst would need to develop a PRA-type seismic event tree supported by fault trees, using techniques that are well established. That is, the analyst would either start with the internal-events-PRA event-tree structure and prune away the branches that are not relevant or would develop a special event tree tailored specifically to earthquake initiators.

Once this systems-analysis work has been accomplished, the analyst can determine the plant-as-a-whole HCLPF capacity, and it will be more accurate than the corresponding capacity determined using the success-path approach. This is because the detail in the fault-space systems analysis, even though it is truncated if the NRC seismic-margins-methodology guidance is used, permits the analyst to ascertain whether any other cut sets make lesser but still nonnegligible contributions to the plant-level HCLPF capacity, and to include properly the contributions of any nonseismic failures or human errors. Since this HCLPF capacity has fewer approximations than that derived from an EPRI-type SMA, when it is convolved with the site hazard curve [as in 10-B.9.3 (Example 3) and 10-B.9.4 (Example 4) above], the bounding-CDF-type results also have stronger validity (insofar as these approximations are less important).

However, because the fragility aspect of the analysis uses an RLE-type screening level such as 0.30*g* or 0.50*g*, the issue remains of how to deal with the actual capacities of SSCs about which all that is known is that the HCLPF capacity exceeds the screening level. Without revisiting each such SSC to work out its actual HCLPF capacity or fragility curve, this approximation will remain a limitation.

10-B.9.7 Example 6: A Seismic-Caused CDF Derived From a Full Seismic PRA

The ultimate "enhancement" of an SMA is to convert it to a full seismic PRA, using as much of the SMA's analytical work as is feasible. The most important SMA results are the seismic HCLPF capacities of a large number of SSCs, and the engineering evaluations and walkdown information used to develop these can be utilized directly, although for each important SSC the SMA's HCLPF-capacity analysis must be enhanced to produce a full family of seismic-fragility curves. A full seismic-PRA systems analysis is also needed, along with a family of seismic hazard curves. (Note that for most U.S. nuclear power plant sites, both the Lawrence Livermore National Laboratory [LLNL] and the EPRI regional hazard studies [10-B-9, 10-B-10] can be used to develop site-specific seismic hazard curves.)

The advantage of a full seismic PRA is that a rigorous seismic-caused CDF can be developed, including non-seismic failures and human errors, and accounting for the dependencies that cannot be studied any other way. This CDF would be a much more accurate estimate than in Example 3.

Furthermore, with a seismic PRA a much better uncertainty analysis can be performed to provide insights into the state of knowledge of CDF. To do a complete uncertainty analysis, one would need a full family of fragility curves, plus a full family of hazard curves, which are not always readily available (for example, the LLNL and EPRI hazard studies typically contain only a mean hazard curve and curves representing 15%, 50%, and 85% confidence level curves). However, most of the important insights to be gained from uncertainty analysis can be developed even if full families of fragility curves and hazard curves are not used, provided the analyst uses a reasonable set and is aware of the approximations made.

10-B.9.8 Example 7: Estimating Figures of Merit Related to LERF

Neither an EPRI-type SMA nor an NRC-type SMA can evaluate LERF-type issues because neither evaluates any of the key safety functions that are required to understand LERF. The SMA's systems-analysis scope stops short of examining the functions and SSCs that must be understood to evaluate LERF, such as containment-isolation capability.

The simplest type of enhancement that can provide insights in this area would be extending the scope of the SSCs to be evaluated, so that the list includes those involved in LERF-type issues. (Note that these SSCs may not have been evaluated previously and therefore may require a walkdown.) For example, determining that every such SSC has a very strong seismic capacity would be an important insight, as would be the insight that a particular containment-isolation function possesses a relatively weak seismic capacity. To go further, the analyst would need to use one of the enhanced approaches above [see 10-B.9.6 (Example 5) and 10-B.9.7 (Example 6)] that lead to an estimate of (or a bound on) CDF. The analyst can then attempt to determine whether any of the sequences contributing to the CDF might lead to a seismic-initiated LERF sequence, for example, because the needed SSCs do not have enough seismic capacity to keep the consequences of the CDF sequence small enough, so that it would evolve into an LERF sequence.

10-B.10 REFERENCES

[10-B-1] "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Report NUREG-1407, U.S. Nuclear Regulatory Commission (1991).

[10-B-2] "Individual Plant Examination for External Events (IPEEE) for Severe Accident Vulnerabilities-10 CFR 50.54(f)," Generic Letter No. 88-20, Supplement 4, U.S. Nuclear Regulatory Commission (1991).

[10-B-3] "Perspectives Gained From the Individual Examination of External Events (IPEEE) Program," Report NUREG-1742, in two volumes, U.S. Nuclear Regulatory Commission (2001).

[10-B-4] NTS Engineering, RPK Structural Mechanics Consulting, Pickard Lowe & Garrick, Woodward Clyde Consultants, and Duke Power Company, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Report EPRI NP-6041-SL, Rev. 1, Electric Power Research Institute (1991).

[10-B-5] R. J. Budnitz, P. J. Amico, C. A. Cornell, W. J. Hall, R. P. Kennedy, J. W. Reed, and M. Shinozuka, "An Approach to the Quantification of Seismic Margins in Nuclear Power Plants," Report NUREG/CR-4334, Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission (1985).

[10-B-6] J. W. Reed and R. P. Kennedy, "Methodology for Developing Seismic Fragilities," Report TR-103959, Electric Power Research Institute (1994).

[10-B-7] R. P. Kennedy, "Overview of Methods for Seismic PRA and Margins Methods Including Recent Innovations," Proceedings of the Organization for the Economic Cooperation and Development/Nuclear Energy Agency Workshop on Seismic Risk, August 10–12, 1999, Tokyo, Japan.

[10-B-8] P. G. Prassinos, M. K. Ravindra, and J. B. Savy, Recommendations to the Nuclear Regulatory Commission on Trial Guidelines for Seismic Margin Reviews of Nuclear Power Plants," Report NUREG/CR-4482, Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission (1986).

[10-B-9] "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," Report NUREG-1488, U.S. Nuclear Regulatory Commission (1993).

[10-B-10] "Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Earthquake Issue," Report EPRI NP-6395-D, Electric Power Research Institute (1989).

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ASME/ANS RA-S INTERPRETATIONS VOLUME 3

Replies to Technical Inquiries July 2008 Through June 2013

FOREWORD

Each interpretation has been reviewed for applicability to the edition and supplements listed for that inquiry. In some instances, a review of the interpretation revealed a need for corrections of a technical nature. In these cases, a revised interpretation is presented bearing the original interpretation number with the suffix R and the original file number with an asterisk.

ASME procedures provide for reconsideration of these interpretations when or if additional information is available which might affect any interpretation. Further, persons aggrieved by any interpretation may appeal to the cognizant ASME committee or subcommittee. ASME does not "approve," "certify," "rate," or "endorse" any item, construction, proprietary device, or activity.

For detailed instructions on the preparation of technical inquiries, refer to Preparation of Technical Inquiries to the Committee on Nuclear Risk Management (p. v of ASME/ANS RA-S–2008).

Interpretation: 1-1R

Subject: ASME RA-Sb–2005, Table 4.5.6-2(c); ASME/ANS RA-Sb–2013, Part 2, Table 2-2.6-4; Supporting Requirements for HLR-DA-C, Index number DA-C6

Date Issued: June 6, 2013

File: 05-1605*

Question: Should the second action verb in Supporting Requirement DA-C6 of RA-S–2002, Addendum a (and unchanged in Addendum b [RA-Sb–2005] and RA-Sb–2013) be interpreted as follows: those (additional) demands that might have been performed during troubleshooting to determine the cause of the fault should not be included, since they are part of the repair process? A single demand related to full functional testing of the component after maintenance, but prior to declaring it operable, may or may not be included, depending on the relationship between the maintenance and the functional test.

Reply: Yes.

Interpretation: 1-2R

Subject: ASME RA-Sa–2003, Section 4, Risk Assessment Technical Requirements;
ASME/ANS RA-Sb–2013, Part 2

Date Issued: June 6, 2013

File: 06-609*

Question: Is it a requirement of Table 4.5.4-2(c) [Table 2-2.4-4 in RA-Sb–2013], Index number SY-C1; Table 4.5.8-2(f) [Table 2-2.7-7 in RA-Sb–2013], Index number QU-F1; and Table 4.5.9-2(g) [Table 2-2.8-8 in RA-Sb–2013], Index number LE-G5 that the lists prefaced by "documentation typically includes" are provided as minimum requirements for documentation?

Reply: No, the lists in SY-C1, QU-F1, and LE-G5 are provided as examples of documentation forms or types that may be used to meet the documentation requirements of the PRA Element. They should not be interpreted as specific requirements for the documentation. This is clarified by the language used in Addendum (b); for specific locations, see Note (1) below.

NOTES:

(1) When the inquiry was posed, the supporting requirements designator correctly referred to "documentation" lists. With the release of Addendum (b), these designators have changed, and there are "documentation" lists in other tables of Section 4. These are as follows:

Table 4.5.1-2(d) [Table 2-2.1-5 in RA-Sb–2013]	IE-D2
Table 4.5.2-2(c) [Table 2-2.2-4 in RA-Sb–2013]	AS-C2
Table 4.5.3-2(c) [Table 2-2.3-4 in RA-Sb–2013]	SC-C2
Table 4.5.4-2(c) [Table 2-2.4-4 in RA-Sb–2013]	SY-C2
Table 4.5.5-2(i) [Table 2-2.5-10 in RA-Sb–2013]	HR-I2
Table 4.5.6-2(e) [Table 2-2.6-6 in RA-Sb–2013]	DA-E2
Table 4.5.7-2(f) [See Note (2) below regarding RA-Sb–2013.]	IF-F2
Table 4.5.8-2(f) [Table 2-2.7-7 in RA-Sb–2013]	QU-F2 (An error in the Standard identifies this as QE-F2.)
Table 4.5.9-2(g) [Table 2-2.8-8 in RA-Sb–2013]	LE-G2

(2) With regard to ASME/ANS RA-Sb–2013, Part 2, the first two sentences of the Reply remain applicable. The affected supporting requirements are as listed in Note (1) above (although the tables have been renumbered), with the exception that IF-F2 is now a Part 3 (Internal Flood) requirement.

Interpretation: 1-3R

Subject: ASME RA-Sa–2003, Section 4, Risk Assessment Technical Requirements, Table 4.5.5-2(g), Index number HR-G3; ASME/ANS RA-Sb–2013, Part 2, Table 2-2.5-8

Date Issued: June 6, 2013

File: 06-610*

Question: Is it the intent of Table 4.5.5-2(g) [Table 2-2.5-8 in RA-Sb-2013], Index number HR-G3, Capability Categories II and III that an explicit evaluation of the impact for each of the listed performance shaping factors (PSF) is not required if the selected human response analysis methodology addresses these PSFs implicitly and provides a means for establishing reasonable confidence that the results implicitly include these considerations?

Reply: Yes.

Interpretation: 1-5R

Subject: ASME RA-Sb–2005, Section 4, Risk Assessment Technical Requirements; ASME/ANS RA-Sb–2013, Part 2, Table 2-2.1-2

Date Issued: June 6, 2013

File: 06-1060*

Question: Is it a requirement to include "non-forced" manual trips that are part of the normal shutdown procedure when counting initiating events?

Reply: No, a normal controlled shutdown would not present the same challenges as a trip from full power. This event is more appropriate for a transition model and outside of the scope of this Standard. If the manual trip was prompted by conditions other than the normal shutdown procedure that could occur at full power, it should be counted. This guidance is consistent with IE-A5(a) [IE-A7(a) in RA-Sb–2013] and IE-C4 [IE-C6 in RA-Sb–2013].

Interpretation: 1-6R

Subject: ASME RA-Sb–2005, Section 4, Risk Assessment Technical Requirements; ASME/ANS RA-Sb–2013, Part 2, Table 2-2.1-2

Date Issued: June 6, 2013

File: 07-213*

Question: Is it a requirement to include "forced" (e.g., technical specification 3.03 actions) or "non-forced" (e.g., manual shutdowns for refueling) when the resulting shutdown follows normal plant procedures with no off-normal conditions requiring a reactor scram?

Reply: No, the risk needs to be captured in a transition risk or low power risk model, which is outside the scope of RA-Sb–2005 and RA-Sb–2013.

Interpretation: 3-1

Subject: ASME RA-Sc–2007, Section 4, Supporting Requirement (SR) AS-A9;
ASME/ANS RA-Sb–2013, Part 2, SR AS-A9

Date Issued: February 9, 2009

File: 08-493

Question: Do the requirements in Supporting Requirement AS-A9 mean that plant-specific thermal-hydraulic calculations are not required to achieve Capability Category II?

Reply: Yes.

Interpretation: 3-2

Subject: ASME RA-Sc–2007, Section 4.3, Expert Judgment; ASME/ANS RA-Sb–2013, Part 1, Subsection 1-4.3

Date Issued: February 9, 2009

File: 08-501

Question (1): Do the requirements in Section 4.3 of the Standard [subsection 1-4.3 in RA-Sb–2013] mean that it is necessary to apply and document the expert judgment process described in Section 4.3 [subsection 1-4.3 in RA-Sb–2013] to a PRA Level 2/LERF model solely on the basis that the model was developed by an entity (e.g., consultant, consulting company, etc.) outside of the PRA owner?

Reply (1): No.

Question (2): Do the requirements in Section 4.3 of the Standard [subsection 1-4.3 in RA-Sb–2013] mean that it is necessary to apply and document the expert judgment process described in Section 4.3 [subsection 1-4.3 in RA-Sb–2013] to usage of reports that involve expert judgment (e.g., NUREG-1829, NUREG/CR-6936) in support of the PRA simply on the basis that expert judgment was used in preparation of those reports?

Reply (2): No.

Interpretation: 3-3

Subject: ASME RA-Sc–2007 up to and including ASME/ANS RA-Sb–2013, Supporting Requirement IF-C2c [IFSN-A5 in RA-Sb–2013]

Date Issued: September 10, 2009

File: 08-503

Question: Is it the case that SR IF-C2c [IFSN-A5] can only be met if individual components located in the flood area are documented?

Reply: No. However, if individual components are not identified, adequate justification to support the level at which SSCs are modeled should be documented.

Interpretation: 3-4

Subject: ASME RA-Sc-2007 up to and including ASME/ANS RA-Sb–2013, Supporting Requirements IF-E3 and IF-E4 [IFQU-A2 and IFQU-A4, respectively, in RA-Sb–2013]

Date Issued: September 10, 2009

File: 08-505

Question: Is it the case that SR IF-E3 [IFQU-A2] and IF-E4 [IFQU-A4] can only be met if individual components located in the flood area are modeled as failed?

Reply: No. The level of detail should be consistent with IF-C3 [IFSN-A6]. However, if individual components are not identified, adequate justification to support the level at which SSCs are modeled should be documented.

Interpretation: 3-5

Subject: ASME RA-Sa–2009 and ASME/ANS RA-Sb–2013, Supporting Requirement AS-A9

Date Issued: April 29, 2013

File: 13-53

Question: Does the phrase "operability of the mitigating systems" in AS-A9 mean the ability of the mitigating systems to support the key safety functions (as stated in HLR-AS-A)?

Reply: Yes.

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