AN AMERICAN NATIONAL STANDARD

ASME RA-Sa–2003

ADDENDA

to

ASME RA-S–2002 STANDARD FOR PROBABILISTIC RISK ASSESSMENT FOR NUCLEAR POWER PLANT APPLICATIONS

THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS
Three Park Avenue • New York, NY 10016

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Date of Issuance: December 5, 2003

ASME is the registered trademark of The American Society of Mechanical Engineers.

This code or standard was developed under procedures accredited as meeting the criteria for American National Standards. The Standards Committee that approved the code or standard was balanced to assure that individuals from competent and concerned interests have had an opportunity to participate. The proposed code or standard was made available for public review and comment that provides an opportunity for additional public input from industry, academia, regulatory agencies, and the public-at-large.

ASME does not "approve," "rate," or "endorse" any item, construction, proprietary device, or activity.

ASME does not take any position with respect to the validity of any patent rights asserted in connection with any items mentioned in this document, and does not undertake to insure anyone utilizing a standard against liability for infringement of any applicable letters patent, nor assume any such liability. Users of a code or standard are expressly advised that determination of the validity of any such patent rights, and the risk of infringement of such rights, is entirely their own responsibility.

Participation by federal agency representative(s) or person(s) affiliated with industry is not to be interpreted as government or industry endorsement of this code or standard.

ASME accepts responsibility for only those interpretations of this document issued in accordance with the established ASME procedures and policies, which precludes the issuance of interpretations by individuals.

No part of this document may be reproduced in any form,
in an electronic retrieval system or otherwise,
without the prior written permission of the publisher.

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

Copyright © 2003 by
THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS
All Rights Reserved
Printed in U.S.A.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

ASME RA-Sa–2003

Following approval by the ASME Board on Nuclear Codes and Standards and after public review, ASME RA-Sa–2003 was approved by the American National Standards Institute on October 22, 2003.

Addenda to ASME RA-S–2002 are issued in the form of replacement pages. Revisions, additions, and deletions are incorporated directly into affected pages. It is advisable, however, that this page, the Addenda title and copyright pages, and all replaced pages be retained for reference.

SUMMARY OF CHANGES

This is the first Addenda to be published to ASME RA-S–2002.

Replace or insert pages listed. Changes given below are identified on the pages by a margin note **(a03)** next to the affected area. The pages not listed are the reverse sides of the listed pages and contain no changes.

Page	*Location*	*Change*
2	1.1	Revised
	1.3	Revised
3	Table 1.3-1	Revised
6, 7, 9, 10	2.2	(1) *accident sequence* revised (2) *best estimate* deleted (3) *key safety functions* revised (4) *large early release* revised (5) *screening analysis* revised (6) *skill of the craft* added (7) *unavailability* revised
15	3.3.1	Example revised
16	3.4	Revised
17	Fig. 3.1-1	Section C, Part 9 revised
19	Section 4 Contents	Revised
21	4.3.6(b)	Revised
22–24	Table 4.5.1-2(a)	(1) IE-A2 entry revised (2) IE-A4 entry revised (3) IE-A5 entry revised (4) IE-A6 entry revised
25–30.2	Table 4.5.1-2(b)	(1) IE-B1 entry revised (2) IE-B4 entry revised

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Page	*Location*	*Change*
	Table 4.5.1-2(c)	(1) IE-C1 entry revised (2) IE-C5 entry revised (3) IE-C9 entry revised (4) IE-C10 entry revised (5) IE-C12 entry revised
	Table 4.5.1-2(d)	(1) IE-D1 entry revised (2) IE-D4 entry deleted
	4.5.2.1(a)	Revised
31, 32	Table 4.5.2-1	HLR-AS-B entry revised
	Table 4.5.2-2(a)	(1) AS-A7 entry revised (2) AS-A10 entry revised
33	Table 4.5.2-2(b)	AS-B6 entry revised
35	Table 4.5.3-1	HLR-SC-B entry revised
37, 38	Table 4.5.3-2(b)	(1) SC-B1 entry revised (2) SC-B6 entry deleted
	Table 4.5.3-2(c)	(1) SC-C1 entry revised (2) SC-C3 entry revised (3) SC-C4 entry revised
39, 40, 42–45	Table 4.5.4-2(a)	(1) SY-A8 entry revised (2) SY-A14 entry revised (3) SY-A22 entry deleted; former SY-A23 renumbered SY-A22
	Table 4.5.4-2(b)	(1) SY-B1 entry revised (2) SY-B5 entry revised (3) SY-B12 entry revised
50	Table 4.5.5-2(d)	HR-D2 entry revised
51	Table 4.5.5-2(e)	HR-E2 entry revised
52–54	Table 4.5.5-2(g)	(1) HR-G1 entry revised (2) HR-G4 entry revised (3) HR-G8 entry revised
	Table 4.5.5-2(h)	(1) HR-H1 entry revised (2) HR-H2 entry revised
55	Table 4.5.5-2(i)	HE-I1 entry revised
	Table 4.5.6-1	HLR-DA-D entry revised
56.1	Table 4.5.6-2(b)	DA-B2 entry revised
59	Table 4.5.6-2(c)	(1) DA-C12 entry revised (2) DA-C14 entry revised

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Page	*Location*	*Change*
60–62	Table 4.5.6-2(d)	(1) DA-D1 entry revised (2) DA-D3 entry revised (3) DA-D5 entry revised (4) DA-D6 entry revised (5) DA-D7 entry revised
65	Table 4.5.7-2(b)	IF-B4 entry revised
	Table 4.5.7-2(c)	(1) IF-C2 entry revised (2) IF-C5 entry revised
66, 67	Table 4.5.7-2(d)	IF-D4 entry revised
68, 69	Table 4.5.7-2(e)	IF-E5 entry revised
70	4.5.8.1	Revised
	Table 4.5.8-1	(1) HLR-QU-D entry revised (2) HLR-QU-E entry revised
71–75	Table 4.5.8-2(a)	QU-A2 entry revised
	Table 4.5.8-2(b)	(1) QU-B2 entry revised (2) QU-B3 entry revised (3) QU-B5 entry revised (4) QU-B6 entry revised
	Table 4.5.8-2(c)	QU-C3 entry revised
	Table 4.5.8-2(d)	(1) QU-D1 entry revised (2) QU-D3 entry revised (3) QU-D4 entry revised (4) QU-D5 entry revised
	Table 4.5.8-2(e)	(1) Introductory paragraph revised (2) QU-E3 entry revised
	Table 4.5.8-2(f)	(1) QU-F1 entry revised (2) QU-F2 entry revised (3) QU-F3 entry revised
76	4.5.9.1	Revised
	Table 4.5.9-1	Revised
78	Table 4.5.9-2(b)	Revised in its entirety
79–81	Table 4.5.9-2(c)	Revised in its entirety
82–85	Table 4.5.9-2(d)	Revised in its entirety
85	Table 4.5.9-2(e)	(1) LE-E1 entry revised (2) LE-E2 entry revised
86, 86.1	Table 4.5.9-2(f)	Revised in its entirety

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Page	*Location*	*Change*
	Table 4.5.9-2(g)	(1) LE-G2 entry revised (2) LE-G3 entry revised (3) LE-G5 entry revised (4) LE-G6 entry revised (5) LE-G7 entry revised
86.2	Table 4.5.9-3	Revised
88	5.4	Revised
	5.5	Revised
91	6.1	Revised
	6.2.2	Revised
92–94	6.2.3	Revised
	6.3	Revised in its entirety
	6.5	Revised
	6.6.1	Revised

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 1

1.1 Scope 2
1.2 Applicability 2
1.3 PRA Capability Categories 2
1.4 Requirements for PRA Elements 2
1.5 Risk Assessment Application Process 4
1.6 PRA Configuration Control 4
1.7 Peer Review Requirements 4

Table

1.3-1 Bases for PRA Capability Categories 3

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 1
INTRODUCTION

(a03) 1.1 Scope

This Standard sets forth 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. (Additional or revised requirements may be needed for other reactor designs.)

1.2 Applicability

This Standard applies to PRAs used to support applications of risk-informed decision making related to design, licensing, procurement, construction, operation, and maintenance. This Standard establishes requirements for a Level 1 analysis of internal events while at power. In addition, this Standard establishes requirements for a limited Level 2 analysis sufficient to evaluate the large early release frequency (LERF) for internal events while at power.

(a03) 1.3 PRA Capability Categories

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 of the factors that determine the risk significance of the proposed changes. To determine the needed capabilities, the application is evaluated by considering the above attributes, as amplified in para. 3.2.2. For example, a proposed change to the maintenance practice for a given component may be evaluated in a particular application. The capabilities of the PRA for those accident sequences involving the failure or unavailability of the component are relevant to such an application, whereas the capabilities of the PRA for other sequences are not as relevant since there are no changes that need to be resolved for these parts.

The required category of PRA capabilities may vary over different elements of the PRA, within a given element, across different accident sequences or classes of accident sequences, initiating events, basic events, and end states, depending on the application. Although the range of capabilities required for each part of the PRA to support an application falls on a continuum, three Capability Categories are defined in this Standard so that requirements can be developed and presented in a manageable way. They are designated as PRA Capability Categories I, II, and III. Table 1.3-1 describes for three principal attributes of a PRA the bases for defining the capability categories. These were used to develop Section 4 requirements for each of the Capability Categories.

This Standard includes High Level Requirements (HLRs) for each element that are the same for all applications, and Supporting Requirements (SRs) that are specified for each element that are differentiated by Capability Category. This differentiation facilitates the determination of the appropriate requirements for each part of a PRA that is necessary to support a given application.

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 parts 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 parts of the PRA within the elements that fail to meet the SRs for any of these Capability Categories.) While all parts of the PRA need not have the same capability, the PRA model should be coherent. The SRs have been written so that, within a Category, the interfaces between parts 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 part of the PRA, and hence which SRs apply to the application. (See Section 3.)

1.4 Requirements for PRA Elements

1.4.1 PRA Elements. The requirements of this Standard are organized by nine PRA Elements that comprise an internal-events, at-power, Level-1 and

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

(a03)

Table 1.3-1 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 including associated human actions.	Resolution and specificity sufficient to identify the relative importance of the significant contributors at the component level including associated human actions, as necessary [Note (1)].	Resolution and specificity sufficient to identify the relative importance of the contributors at the component level including associated human actions, as necessary [Note (1)].
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 (2)].	Departures from realism will have small impact on the conclusions and risk insights as supported by good practices [Note (2)].	Departures from realism will have negligible impact on the conclusions and risk insights as supported by good practices [Note (2)].

NOTES:

(1) 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.

(2) 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.

Not for Resale

Level-2/LERF PRA. They and their abbreviations are as follows:

(a) Initiating Events 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) Internal Flooding (IF)
(h) Quantification (QU)
(i) LERF Analysis (LE)

1.4.2 High Level Requirements. A set of Objectives and HLRs are provided for each PRA Element in Section 4. All PRAs using this Standard shall satisfy each of these HLRs as explained in para. 1.3. The HLRs set forth the minimum requirements for meeting this Standard in general terms and present the top level logic for the derivation of more detailed SRs for each of the PRA Capability Categories. 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.

1.4.3 Supporting Requirements. The SRs for each of the nine PRA Elements are presented in Section 4 as action statements, using the three Capability Categories described in para. 1.3. 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 statement extends to more than one category, it applies equally to each Capability Category, but the scope of applicability will be commensurate with the Capability Category criteria in Table 1.3-1 and the scope and level of detail required by other associated SRs. It is intended that, by meeting all the SRs under a given HLR, a PRA will meet that HLR. Section 4 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 to the requirements of this 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 6.

1.5 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 3 describes 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 para. 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 Section 4. Requirements for supplementary analysis are outside the scope of this Standard.

1.6 PRA Configuration Control

Section 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.

1.7 Peer Review Requirements

Section 6 provides the requirements for a peer review to determine if the PRA methodology and its implementation meet the requirements of Section 4 of this Standard.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 2

2.1 Acronyms 6
2.2 Definitions 6

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 2
ACRONYMS AND DEFINITIONS

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

2.1 Acronyms

ADS: Automatic Depressurization System

AOT: Allowed Outage Time

ARI: Alternate Rod Insertion

ASEP: Accident Sequence Evaluation Program

ATWS: Anticipated Transient Without Scram

BWR: Boiling Water Reactor

CCW: Component Cooling Water

ECCS: Emergency Core Cooling System

EDG: Emergency Diesel Generator

EOPs/AOPs: Emergency Operating Procedures/Abnormal Operating Procedures

HFE: Human Failure Event

HLR: High Level Requirement

HPCI: High Pressure Coolant Injection

HVAC: Heating, Ventilation, and Air Conditioning

ISLOCA: Interfacing Systems Loss of Coolant Accident

LOCA: Loss of Coolant Accident

LOOP: Loss of Offsite Power

MOV: Motor Operated Valve

NPSH: Net Positive Suction Head

NRC: Nuclear Regulatory Commission

NSSS: Nuclear Steam Supply System

P&IDs: Piping and Instrumentation Drawings (or Diagrams)

PDS: Plant Damage State

PWR: Pressurized Water Reactor

RCIC: Reactor Core Isolation Cooling

RCP: Reactor Coolant Pump

RCS: Reactor Coolant System

RPT: Reactor Pump Trip

RPV: Reactor Pressure Vessel

RWST: Refueling Water Storage Tank

SAR: Safety Analysis Report

SBO: Station Blackout

SGTR: Steam Generator Tube Rupture

SLCS: Standby Liquid Control System

SORV: Stuck Open Relief Valve

SR: Supporting Requirements

SSCs: Structures, Systems, and Components

SW: Service Water

THERP: Technique for Human Error Rate Prediction (see NUREG/CR-1278[1])

TS: Technical Specifications

2.2 Definitions (a03)

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 (i.e., system, function, or operator performance) that can lead to undesired consequences, with a specified end state (e.g., core damage or large early release).

[1] NUREG/CR-1278, Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications; A. D. Swain and H. E. Guttmann; August 1983

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

accident sequence, dominant: an accident sequence that is usually represented by the top 10 or 20 events or groups of events modeled in a PRA and accounts for a large fraction of the core damage or large early release frequency.

accident sequence, modeled: an accident sequence contained in the PRA above the model truncation level.

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.

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.

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.

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

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

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 is anticipated and involving enough of the core to cause a significant release.

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

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.

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

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.

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

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

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

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

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

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

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).

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

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).

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).

front-line system: a system (safety or non-safety) 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 non-minimal 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.

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

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.

initiating event: any event either internal or external to the plant that perturbs the steady state operation of the plant, if operating, thereby initiating an abnormal event such as transient or LOCA within the plant. Initiating events trigger sequences of events that challenge plant control and safety systems whose failure could potentially lead to core damage or large early release.

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.

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 over-pressurization of a low-pressure system when subjected to RCS pressure and can result in containment bypass.

internal event: an event originating within a nuclear power plant that, in combination with safety system failures and/or operator errors, can affect the operability of plant systems and may lead to core damage or large early release. By convention, loss of offsite power is considered to be an internal event, and internal fire is considered to be an external event.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

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.

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.

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

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.

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

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

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

post-initiator human failure events: human failure events that represent the impact of human errors committed during actions performed in response to an accident initiator.

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. 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).

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

recovery: a general term describing restoration and repair acts required to change the initial or current state of a system or component into a position or condition needed to accomplish a desired function for a given plant state.

recovery action: a human action performed to regain equipment or system operability from a specific failure

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

or human error to mitigate or reduce the consequences of the failure.

reliability: the complement of unreliability.

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 importance to a PRA.

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

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

risk: probability and consequences of an event, as expressed by the "risk triplet" that is the answer to the following three questions: (1) What can go wrong? (2) How likely is it? and (3) 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.

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

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

screening analysis: an analysis 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.

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.

shall: used to state a mandatory requirement.

should: used to state a recommendation.

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

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.

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

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.

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: termination of the ability of a system to perform any one of its critical design functions.

NOTE: Failure of a line/train within a system may occur in such a way that the system retains its ability to perform all its required functions; in this case, the system has not failed.

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.

truncation limit: the numerical cutoff value of probability or frequency below which results are not retained in the quantitative PRA model or used in subsequent calculations (such limits can apply to accident sequences/cut sets, system level cut sets, and sequence/cut set 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,

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Category I. Hence, the initiating events, accident sequences, data parameters, system models, human actions, and quantification process for those sequences and cutsets impacted by the AOT changes are in PRA Capability Category II, and the remaining parts 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 a baseline core damage frequency that was greater than $1x10^{-4}$ or baseline LERF greater than $1x10^{-5}$, 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 parts 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.

3.3 Assessment of PRA for Necessary Scope, Results, and Models (Stage B)

(a03) **3.3.1 Necessary Scope and Results.** Determine if the PRA provides the results needed to assess the plant or operational change (Box 5 of Fig. 3.1-1). If some aspects of the PRA are insufficient to assess the change, then upgrade them in accordance with the SRs of Section 4 for its corresponding Capability Category (Box 6a of Fig. 3.1-1), or generate supplementary analyses. (See para. 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 5.

Example: The proposed change in the SW AOT has been determined to affect the SW unavailability. 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, the scope of the initiating event analysis element of the PRA must include: (1) LOCA initiators, since the change in SW unavailability will affect ECCS pump cooling in the recirculation phase, (2) loss of offsite power initiators, since the SW change will affect the diesel generators, and (3) 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. 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.

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. 3.1-1). If the affected SSCs or plant activities are not modeled, then either upgrade the PRA to include the SSCs in accordance with the SRs of Section 4 for their corresponding Capability Category (Box 6a of Fig. 3.1-1), or generate supplementary analyses. (See para. 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 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 that contribute to the change in CDF (i.e., ECCS, DGs, feedwater, and CCW) need to be modeled in the PRA. For example, if, as is likely, the loss of feedwater initiator is modeled as one global initiator,

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

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 the standard.

3.3.3 Peer Review. The parts of a PRA that are needed for an application shall have been reviewed pursuant to the requirements of Section 6.

(a03) 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 Section 4, for the corresponding Capability Categories determined in para. 3.2.2, are sufficient to assess the application under consideration (Box 8 of Fig. 3.1-1).

If it is determined that the standard lacks specific requirements, their relevance to the application shall be assessed (Box 9 of Fig. 3.1-1). 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. 3.1-1).

3.5 Comparison of PRA Model to Standard (Stage D)

Determine if each part of the PRA satisfies the SRs at the appropriate Capability Category needed to support the application (Box 10 of Fig. 3.1-1). The results of the Peer Review (Section 6) 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. 3.1-1). The bases for this determination shall be documented.

If the PRA does not satisfy a SR for the appropriate Capability Category, then determine if the difference is significant (Box 12 of Fig. 3.1-1). Acceptable requirements for determining the significance of this difference include the following:

(a) The difference is not applicable or does not affect quantification relative to the impact of the proposed application, or

(b) Modeled accident sequences accounting for at least 90% of CDF/LERF, 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 Section 4 as applied to the application.

Determination of significance will depend on the particular application being considered and may involve determinations made by an expert panel.

If the difference is not significant, then the PRA is acceptable for the application. If the difference is significant, then either upgrade the PRA to address the corresponding SRs stated in Section 4 (Box 6b of Fig. 3.1-1), or generate supplementary analyses. (See para. 3.6.) Any upgrade of the PRA shall be done and documented in accordance with Section 5. The examples provided under para. 3.3 are applicable.

3.6 Use of Supplementary Analyses/ Requirements (Stage E)

If the scope of either the PRA or the standard is insufficient, supplementary analyses or requirements may be used (Box 7 of Fig. 3.1-1). 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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

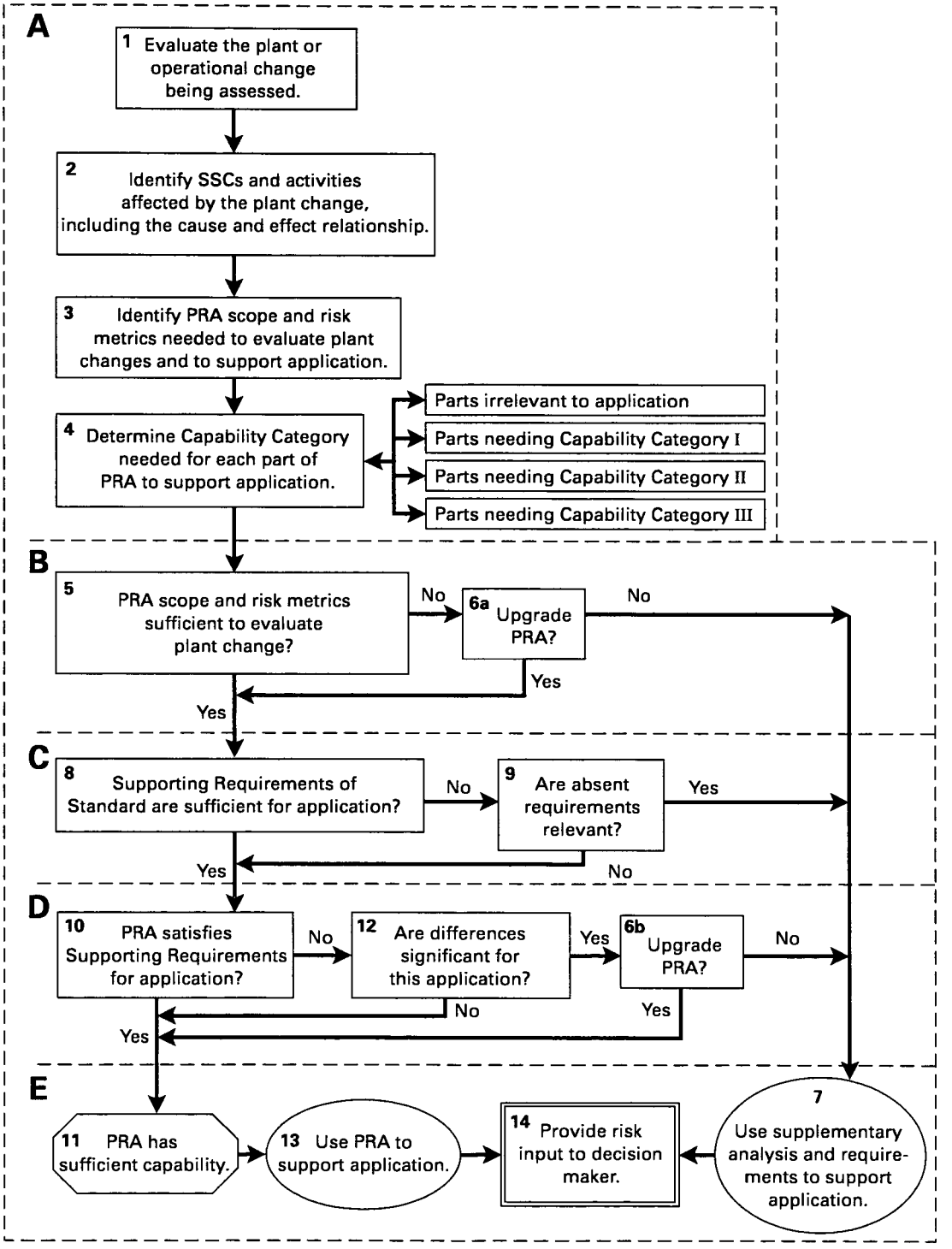


Fig. 3.1-1 Application Process Flowchart **(a03)**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

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.

Example of supplementary requirements: 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.

Second example of supplementary requirements: It is desired to rank the snubbers in a plant according to their risk significance for the purpose of 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,[3] 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.

Third example of supplementary requirements: 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. The issue of seismic performance of these valves is beyond the scope of this Standard. To address it, supplementary requirements, found in *Nuclear Engineering and Design*,[4] 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.

If it has been determined that the PRA has sufficient capability, its results can be used to support the application (Box 13 of Fig. 3.1-1). 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. 3.1-1). Such supplementary analyses/requirements are outside the scope of this Standard.

---

[3] ASME OM Code for Operation and Maintenance of Nuclear Power Plants, Code Case OMN-10: Requirements for Safety Significance Categorization of Snubbers Using Risk Insights and Testing Strategies for Inservice Testing of LWR Power Plants

[4] Kennedy, R. P. and M. K. Ravindra, "Seismic Fragilities for Nuclear Power Plant Risk Studies," *Nuclear Engineering and Design*, Vol. 79, No. 1, pp. 47–68 (1984)

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

SECTION 4 (a03)

4.1 Purpose 20
4.2 Process Check 20
4.3 Use of Expert Judgment 20
4.4 Derivation of PRA Requirements 21
4.5 PRA Requirements 21

Objectives and Tables

4.5.1 Initiating Events Analysis (IE) 22
4.5.2 Accident Sequence Analysis (AS) 31
4.5.3 Success Criteria (SC) 35
4.5.4 Systems Analysis (SY) 39
4.5.5 Human Reliability Analysis (HR) 47
4.5.6 Data Analysis (DA) 55
4.5.7 Internal Flooding (IF) 63
4.5.8 Quantification (QU) 70
4.5.9 LERF Analysis (LE) 76

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

SECTION 4
RISK ASSESSMENT TECHNICAL REQUIREMENTS

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.

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.

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.

NUREG/CR-6372[1] and NUREG/CR-1563[2] 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 include Pacific Gas and Electric's Diablo Canyon seismic study[3] and the Yucca Mountain project's study of volcanic hazards.[4] 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-1563.

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).

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).

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

[1] 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

[2] 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

[3] NRC 50-275, 50-323, Final Report of the Diablo Canyon Long-Term Seismic Program, Pacific Gas and Electric Company

[4] BA000-1717-2200-00082, Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada, U.S. Department of Energy Yucca Mountain Project, Geomatrix Consultants, Inc., 1996

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

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

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.

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

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; for example, an individual who evaluates data and develops a particular hypothesis to explain the data

(a03) *(b)* technical experts with knowledge of a particular technical area of relevance to the issue

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.

4.4 Derivation of PRA Requirements

Objectives were established for each of the nine elements used to characterize a PRA. The objectives reflect substantial experience accumulated with PRA development and usage, and are consistent with the PRA Procedures Guide[5] and the NEI-00-02 Peer Review Process Guidance.[6] These objectives form the basis for development of the high level requirements (HLRs) for each element that were used, in turn, to define the supporting requirements (SRs).

In setting the HLRs for each element, the goal was to derive, based on the objectives, an irreducible set of firm requirements, applicable to PRAs that support all levels of application, to guide the development of SRs. This goal reflects the diversity of approaches that have been used to develop existing PRAs and the need to allow for technological innovations in the future. An additional goal was to derive a reasonably small set of HLRs that capture all the important technical issues that were identified in the efforts to develop this Standard and to implement the NEI-00-02 PRA Peer Review process guidance.

The HLRs generally address attributes of the PRA element such as

(a) scope and level of detail

(b) model fidelity and realism

(c) output or quantitative results (if applicable)

(d) documentation

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

4.5 PRA Requirements

Tables of HLRs and SRs for the nine PRA elements are provided in paras. 4.5.1 through 4.5.9. 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 statement extends to more than one category, it applies equally to each Capability Category, but the scope of applicability will be commensurate with the Capability Category criteria in Table 1.3-1 and the scope and level of detail required by other associated SRs. It is intended that, by meeting all the SRs under a given HLR, a PRA will meet that HLR.

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

[5] NUREG/CR-2300, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, 1983

[6] NEI-00-02, Probabilistic Risk Assessment Peer Review Process Guidance, 2000

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

4.5.1 Initiating Event Analysis (IE)

4.5.1.1 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 4.5.1-1 High Level Requirements for 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	The initiating event analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed to select, group, and screen the initiating event list and to model and quantify the initiating event frequencies, with assumptions and bases stated.

(a03)

Table 4.5.1-2(a) Supporting Requirements for HLR-IE-A

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

Index No. IE-A	Capability Category I	Capability Category II	Capability Category III
IE-A1	USE a structured, systematic process for identifying initiating events. 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.1-2(a) 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-A2	IDENTIFY those initiating events and event categories that challenge normal plant operation and that require successful mitigation to prevent core damage. In the identification, ACCOUNT FOR the potential for plant-specific features to influence initiating events. INCLUDE in the spectrum of internal-event challenges at least the following general categories, and within each general category. INCORPORATE each initiating event category in the model quantitatively in terms of its relative frequency. In the categorization, SEPARATE into different categories based on whether events have different impacts on plant performance, safety functions, and possibilities for recovery. The following list is not intended to be all-inclusive: *(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. DIFFERENTIATE the LOCA initiators into at least the following categories, using a defined rationale for the differentiation: *(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: pipe breaks outside containment *(c)* ISLOCAs: INCLUDE postulated events representing active components (i.e., components that will need to change state) 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 [e.g., interfacing systems LOCAs (ISLOCAs)]. *(d)* Special initiators (e.g., support systems failures, instrument line breaks) [Note (1)]. *(e)* Internal flooding initiators (see IF-D1 and D2) [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. REVIEW experience and analyses at similar plants to assess whether the list of challenges included in the model accounts for industry experience.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.1-2(a) 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-A4	PERFORM a systematic evaluation of each system to assess the possibility of an initiating event occurring due to a failure of the system.	PERFORM a systematic evaluation of each system to assess the possibility of an initiating event occurring due to a failure of the system.	PERFORM a systematic evaluation of each system 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 potential system initiating events.	**USE a structured approach (such as a system-by-system review of initiating event potential, or an FMEA [failure modes and effects analysis] or fault tree) to assess and document the possibility of an initiating event resulting from individual systems or train failures.**	**DEVELOP a detailed model of system interfaces including fault tree development. PERFORM an FMEA (failure modes and effects analysis) to assess and document the possibility of an initiating event resulting from individual systems or train failures.**
	INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause.	INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause.	INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause.
IE-A5	In the identification of the initiating events, INCORPORATE *(a)* events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions, unless it is determined that an event is not applicable to at-power operation). *(b)* events resulting in a controlled shutdown that include a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to at-power operation.		
IE-A6	No requirement for interviews.	INTERVIEW plant operations, maintenance, engineering, and safety analysis personnel to determine if potential initiating events have been overlooked. **Information from interviews conducted at similar plants may be used in lieu of plant-specific interviews.**	INTERVIEW plant operations, maintenance, engineering, and safety analysis personnel to determine if potential initiating events have been overlooked.
IE-A7	No requirement for precursor review.	In searching for initiating events, ACCOUNT FOR initiating event precursors, to help identify initiating events and to provide a partial basis for quantifying their frequencies.	
IE-A8	No requirement to account for the system alignments cited.	In searching for initiating events, ACCOUNT FOR each system and supporting system alignment, such as temporary alignments during maintenance, that could either influence the likelihood that failures cause an initiating event, or could increase the severity of the effect on plant safety functions that would result from such an event.	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.1-2(a) 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 to include support system failures in a realistic fashion, nor to treat explicitly the cited support systems.	INCLUDE support system failures as initiating events quantitatively in the PRA in a realistic fashion. TREAT EXPLICITLY the individual support systems (or trains) that can cause a plant trip.	
IE-A10	INCLUDE those multi-unit site initiators such as dual unit LOOP events or total loss of service water that may impact the model at multi-unit sites with shared systems.		

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

Table 4.5.1-2(b) Supporting Requirements for HLR-IE-B (a03)

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	COMBINE initiating events into groups to facilitate definition of accident sequences in the Accident Sequence Analysis element (para. 4.5.2) and to facilitate quantification in the Quantification element (para. 4.5.8). When initiating events are not grouped, PROVIDE a separate accident-sequence evaluation for each selected initiating event.		
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).		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.1-2(b) Supporting Requirements for HLR-IE-B (Cont'd)

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-B3	GROUP initiating events only when the following can be assured: *(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 subsumed into a group and bounded by the worst case impacts within the "new" group.		GROUP initiating events only when the following can be assured: *(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 subsumed into a group and bounded by the worst case impacts within the "new" group. **To avoid conservatism, DO NOT ADD initiating events to a group and DO NOT SUBSUME events into a group unless the impacts are comparable to or less than those of the remaining events in that group, or it is demonstrated that such grouping does not appreciably impact CDF or LERF.**
IE-B4	GROUP separately from other initiating event categories those categories with different plant response (i.e., those with different success rate 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.		

(a03)

Table 4.5.1-2(c) 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).

Index No. IE-C	Capability Category I	Capability Category II	Capability Category III
IE-C1	CALCULATE the initiating event frequency from plant-specific data, if sufficient data are available. Otherwise, USE generic data (see IE-C2). 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). CREDIT recovery actions as appropriate; JUSTIFY each such credit (as evidenced such as through procedures or training). Data from the initial year of commercial operation may be excluded; if excluded, JUSTIFY.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.1-2(c) 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-C2	If necessary because insufficient plant-specific data are available (see IE-C1), USE generic industry data in the quantification of initiating event frequencies.	If necessary because insufficient plant-specific data are available (see IE-C1), USE **a Bayesian update process of** generic industry data in the quantification of initiating event frequencies.	
IE-C3	CALCULATE initiating event frequencies on a reactor-year basis. Specifically, for sequences initiated at power, ACCOUNT in the initiating event analysis for the plant availability, such that the frequencies are weighted by the fraction of time the plant is at-power. ACCOUNT FOR differences between historical plant availability over the period of event occurrences in the plant database and present or expected future plant availability that could be different from historical values.		
IE-C4	USE as screening criteria no higher than the following characteristics (or more stringent characteristics as devised by the analyst) to eliminate initiating events from further evaluation: *(a)* the frequency of the event is less than 1E-7 per reactor-year (/ry) 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/ry 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, then CONFIRM that the value specified in the criterion meets the requirements in the Data Analysis section (para. 4.5.6) and the Level 1 Quantification section (para. 4.5.8).		
IE-C5	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 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). One acceptable methodology for time-trend analysis is found in NUREG/CR-5750 [Note (1)].
IE-C6	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-specific design features. When the fault-tree approach is used, USE the appropriate systems-analysis requirements for fault-tree modeling found in the Systems Analysis section (para. 4.5.4).		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.1-2(c) 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-C7	When using fault tree models for initiating events, QUANTIFY 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 the Systems Analysis section, para. 4.5.4.). Thus, 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. APPLY the relevant requirements in the Data Analysis section, para. 4.5.6, for the data used in the fault-tree quantification.		
IE-C8	If fault-tree modeling is used, CAPTURE 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.		
IE-C9	No requirement to use plant-specific information in the fault-tree modeling.	If fault-tree modeling is used, USE plant-specific information in the assessment and quantification of recovery actions where available; consistent with Human Reliability Analysis (see para. 4.5.5 for further guidance).	
IE-C10	COMPARE and RESOLVE the results of the initiating event analysis with generic data sources to provide a reasonableness check of the results.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.1-2(c) 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	For rare initiating events, USE industry generic data and **ACCOUNT for plant-specific functions.** For extremely rare initiating events, engineering judgment may be used; if used, AUGMENT with applicable generic data sources. Refer to para. 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 para. 4.3, Use of Expert Judgment, as appropriate.
	For purposes of this Requirement, a "rare event" might be expected to occur one or a few times throughout the world nuclear industry over many years. An "extremely rare event" would not be expected to occur even once throughout the industry over many years.		For purposes of this Requirement, a "rare event" might be expected to occur one or a few times throughout the world nuclear industry over many years. An "extremely rare event" would not be expected to occur even once throughout the industry over many years. **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**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.1-2(c) 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-C12	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, existence and positioning of relief valves *(b)* provision of protective interlocks *(c)* relevant surveillance test procedures		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,** ***(d)* EVALUATE surveillance procedure steps** ***(e)* INCLUDE surveillance test intervals explicitly** ***(f)* ASSESS on-line surveillance testing quantitatively** ***(g)* QUANTIFY pipe rupture probability** ***(h)* ADDRESS explicitly valve design (e.g., air operated testable check valves)** ***(i)* INCLUDE quantitatively the valve isolation capability given the high-to-low-pressure differential.**

NOTE:

(1) NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants, Idaho National Engineering and Environmental Laboratory, Idaho Falls, February 1999

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.1-2(d) Supporting Requirements for HLR-IE-D (a03)

The initiating event analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed to select, group, and screen the initiating event list and to model and quantify the initiating event frequencies, with assumptions and bases stated (HLR-IE-D).

Index No. IE-D	Capability Category I	Capability Category II	Capability Category III
IE-D1	Selection of initiating events: *(a)* LIST and JUSTIFY (by plant-specific or applicable generic analyses) functional categories considered, in accordance with the safety functions considered in the accident sequence model. *(b)* For each functional category, DOCUMENT the specific initiating events considered. *(c)* DOCUMENT the systematic search for plant-unique and plant-specific support system initiators, along with the resulting support system initiators disposition. *(d)* DOCUMENT the systematic search for RCS pressure boundary failures and interfacing system LOCAs. *(e)* DOCUMENT the approach for assessing completeness and consistency of initiating events with plant-specific experience, industry experience, other comparable PRAs and FSAR initiating events. *(f)* DOCUMENT the assumptions.		
IE-D2	Grouping and screening of initiating events: *(a)* DOCUMENT the basis for screening out initiators as risk insignificant. *(b)* DOCUMENT the basis for grouping and subsuming initiating events. This may interface with the required success criteria from the Systems Analysis section (para. 4.5.4) and Success Criteria section (para. 4.5.3) of this Standard. *(c)* DOCUMENT the assumptions. *(d)* DOCUMENT the dismissal of any observed initiating events, including any credit for recovery.		
IE-D3	Quantification of initiating event frequencies: *(a)* DOCUMENT the derivation of the initiating event frequencies and the recoveries used in conjunction with the initiating event. *(b)* DOCUMENT the approach to quantification of each initiating event frequency as data analysis or model approach. *(c)* EXPLAIN any large deviations (such as an order-of-magnitude difference) from comparable generic data. When fault tree models are used to estimate initiating event frequencies, APPLY appropriate aspects of system analysis documentation requirements including any modeling assumptions. *(d)* When fault trees are used to develop initiating event frequencies, DOCUMENT how the applicable system failure modes are taken into account for each fault tree. *(e)* DOCUMENT the methodology and approach when using data analysis methods to estimate initiating event frequencies; also, IDENTIFY the data used. *(f)* DOCUMENT the justification for exclusion of any data. *(g)* DOCUMENT the basis for the availability factor used to convert initiating event frequencies to events per reactor year. *(h)* IDENTIFY potential time dependent aspects of the initiating event frequencies, and DOCUMENT assumptions made to obtain average frequencies. *(i)* DOCUMENT the process for computing initiating event frequencies. *(j)* DOCUMENT the key assumptions made in the analysis that affect the results.		

4.5.2 Accident Sequence Analysis (AS)

4.5.2.1 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 and LERF in such a way that

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

(a03) *(a)* 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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.2-1 High Level Requirements for Accident Sequence Analysis (AS) (a03)

Designator	Requirement
HLR-AS-A	The accident sequence analysis shall describe the plant-specific scenarios that can lead to core damage following each initiating event or initiating event category. 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 shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases.

Table 4.5.2-2(a) Supporting Requirements for HLR-AS-A (a03)

The accident sequence analysis shall describe the plant-specific scenarios that can lead to core damage following each initiating event or initiating event category. 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	CHOOSE a method for accident sequence analysis that explicitly MODELS the appropriate combinations of system responses and operator actions that affect the key safety functions for each modeled initiating event and provides a framework to support sequence quantification. CHOOSE a method that includes an "event tree structure" or equivalent such that the accident sequence logic (progression) is graphically represented.		
AS-A2	For each initiating event group, DEFINE in the model the key safety functions that are necessary to reach a safe, stable state and prevent core damage.		
AS-A3	For each initiating event group, using the defined success criteria for each key safety function, IDENTIFY the systems that can be used to mitigate the initiator.		
AS-A4	For each initiating event group, using the defined success criteria for each key safety function, IDENTIFY the procedurally directed operator actions.		
AS-A5	DEFINE functions and structure of the accident sequence model in a manner that is consistent with the plant-specific EOPs, abnormal procedures, training simulator exercises, and existing plant transient analysis.		
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 initiating event group, **unless the sequences can be shown to be a non-contribution using qualitative arguments.**		DELINEATE the possible accident sequences for each initiating event group.
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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.2-2(a) 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 initiating event or initiating event category. 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-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.	**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.
AS-A10	For each initiating group, **INCLUDE bounding critical safety function status that captures timing and operator actions as the individual events in the accident sequence.**	For each initiating event group, **DEVELOP the accident sequence model to sufficient detail that significant differences in requirements on systems and operator responses are captured. For example, diverse systems providing a similar function need not be modeled separately if choosing one over another does not substantially impact the sequence development. If, however, choosing one over another changes the requirements for operator intervention or the need for other systems, they should be modeled separately.**	For each initiating event group, **explicitly INCLUDE each system and operator action required for each critical 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.2-2(b) Supporting Requirements for HLR-AS-B (a03)

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 the initiating event, IDENTIFY the mitigating systems impacted by the occurrence of the initiator and the extent of the impact. INCLUDE the impact of initiating events on accident progression.		
AS-B2	For each critical safety function, IDENTIFY its dependence on the success or failure of preceding functions. INCLUDE the impact on accident progression. 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 If functional event trees are used, INCLUDE a model that accounts for interactions and dependencies between events.		
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]. INCLUDE the impact of the accident progression phenomena.		
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, PLACE Event A to the left of Event B in the ordering of event tops.		
AS-B5	For the event trees with conditional split fraction method, DEVELOP the event trees to a level of detail sufficient to identify intersystem dependencies and train level interfaces. For the fault tree linking method, DEVELOP fault trees and apply flag settings and mutually exclusive files or comparable method to resolve these same dependencies. 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. PROVIDE one event sequence model or set of event trees that accounts for each initiating event or initiating event category defined in the Initiating Event Analysis element so that initiating event dependencies can be properly modeled.		
AS-B6	INCLUDE events for which time-phased dependencies might exist. Examples are: *(a)* For SBO/LOOP sequences, INCLUDE 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), INCLUDE 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		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.2-2(c) Supporting Requirements for HLR-AS-C

Documentation shall be performed in a manner that facilitates peer review, as well as future updates and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-AS-C).

Index No. AS-C	Capability Category I	Capability Category II	Capability Category III
AS-C1	DOCUMENT the results of the accident sequence analysis consistent with the process that was used for its development. PROVIDE the basis for the accident sequence process.		
AS-C2	DOCUMENT the treatment of each initiator and event tree to support reviews and applications.		
AS-C3	DOCUMENT the following interfaces between accident sequence analysis and other PRA tasks: *(a)* a link between the definition of initiating event category in the Initiating Event Analysis section and the event sequence model *(b)* the definition of core damage and associated success criteria that is consistent with that documented in the Success Criteria Task (para. 4.5.3) *(c)* key definitions of operator actions, sequence specific timing, and dependencies reflected in the event trees that are traceable to the HRA for these actions *(d)* a description of the interface of the accident sequence models with plant damage states *(e)* a framework for an integrated treatment of dependencies in the Initiating Events Analysis, Systems Analysis, Data Analysis, Human Reliability Analysis, and Level 1 Quantification		
AS-C4	DOCUMENT the following: *(a)* the success criteria established for each initiating event category 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) *(b)* the models used (including all sequences) for each initiating event category *(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)* any assumptions that were made in developing the accident sequences, and the bases for the assumptions and their impact on the final results *(e)* existing analyses or plant-specific calculations performed to arrive at success criteria and expected sequence phenomena including necessary timing considerations *(f)* sufficient system operation information to support the modeled dependencies *(g)* calculations or other bases used to justify equipment operability beyond its "normal" design parameters and for which credit has been taken *(h)* how all requirements for accident sequence analysis have been satisfied when sequences are modeled using a single top event linked fault tree		

4.5.3 Success Criteria (SC)

4.5.3.1 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 and large early release).

(b) success criteria are defined for critical safety functions, supporting systems, structures, components 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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.3-1 High Level Requirements for Success Criteria (SC) (a03)

Designator	Requirement
HLR-SC-A	The overall success criteria for the PRA and the system, structure, component, 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 and LERF, 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 shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA, by describing the processes that were used, and providing details of the assumptions made and their bases.

Table 4.5.3-2(a) Supporting Requirements for HLR-SC-A

The overall success criteria for the PRA and the system, structure, component, 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 2 of this Standard. If core damage has been defined differently than in Section 2: *(a)* IDENTIFY any substantial differences from the Section 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. SELECT these parameters such that the determination of core damage is as realistic as practical, consistent with current best practice. DEFINE computer code-predicted acceptance criteria with sufficient margin between actual limits and code-calculated values to allow for limitations of the codes, sophistication of the models, and uncertainties in the results. Examples of measures for core damage that have been used in PRAs include: *(a)* Collapsed liquid level less than $\frac{1}{3}$ 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 core peak node temperature >2,200°F using a code with detailed core modeling, or code-predicted core peak node 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)		
SC-A3	SPECIFY the minimum set of mitigative functions to prevent core damage or radioactivity release in the accident sequences, for each initiating event group.		
SC-A4	SPECIFY success criteria for each of the mitigating functions for each initiating event group. IDENTIFY systems capable of meeting the specified mitigating function success criteria. IDENTIFY mitigating systems that are shared between units, and the manner in which the sharing is performed should both units experience a common intitiating event (e.g., LOOP).		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.3-2(a) Supporting Requirements for HLR-SC-A (Cont'd)

The overall success criteria for the PRA and the system, structure, component, 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 sequences in which stable plant conditions would not be achieved by 24 hr using the modeled plant equipment and human actions, USE a longer mission time if needed to achieve stable plant conditions. PERFORM additional evaluation or modeling for sequences in which a safe, stable state has not been achieved by the end of the mission time defined for the PRA 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 the Systems Analysis and Human Reliability sections of this Standard, to demonstrate that a successful outcome is achieved.		
SC-A6	CONFIRM that the models and inputs for thermal/hydraulic, structural, and other supporting engineering bases are consistent with the features, procedures, and operating philosophy of the plant.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.3-2(b) Supporting Requirements for HLR-SC-B (a03)

The thermal/hydraulic, structural, and other supporting engineering bases shall be capable of providing success criteria and event timing sufficient for quantification of CDF and LERF, determination of the relative impact of success criteria on SSC and human action importance, 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 (e.g., thermal-hydraulic codes such as RELAP, MAAP, SAFER/GESTER, RETRAN, or equivalent) for thermal/ hydraulic, structural, and other supporting engineering bases in support of success criteria requiring detailed computer modeling. 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 (e.g., thermal-hydraulic codes such as RELAP, MAAP, SAFER/ GESTER, RETRAN, or equivalent) for thermal/ hydraulic, structural, and other supporting engineering bases in support of success criteria requiring detailed computer modeling. DO NOT USE assumptions that could yield conservative or optimistic success criteria.**
SC-B2	No restrictions regarding the use of expert judgment, but requirements in 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 para. 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 (HLR-IE-B) and accident sequence modeling (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/LERF, 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., Owner's Group generic studies) may be used. USE computer codes and models only within known limits of applicability.		
SC-B5	CHECK 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		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

(a03)

Table 4.5.3-2(c) Supporting Requirements for HLR-SC-C

Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA, by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-SC-C).

Index No. SC-C	Capability Category I	Capability Category II	Capability Category III
SC-C1	**DOCUMENT bases, references, and assumptions for success criteria. IDENTIFY which of the key assumptions are conservative or optimistic, and IDENTIFY their general impacts on the results.**	**DOCUMENT each of the success criteria and the supporting engineering bases, references, and key assumptions for success criteria and the supporting engineering calculations performed in support of the PRA.** *(a)* **IDENTIFY conservative, optimistic, or simplifying assumptions or conditions.** *(b)* **PROVIDE specific justification, based on results of evaluation or quantification, as appropriate to the application Category, for use of conservative, optimistic, or simplifying assumptions or conditions.** *(c)* **PROVIDE the basis for the success criteria development process and the supporting engineering calculations.**	
SC-C2	DOCUMENT uses of and rationale for expert judgment.		
SC-C3	DOCUMENT the rationale used in the application of success criteria for situations in the PRA for which there is more than one technical approach, none of which is universally accepted as correct, and the approach represents a key source of uncertainty.		
SC-C4	DOCUMENT the following, to the extent they are not included in the documentation under AS-C4 and SY-C1: *(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)* identification of key assumptions used in establishing success criteria *(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		

4.5.4 Systems Analysis (SY)

4.5.4.1 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 the HRA element.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

(c) different initial system alignments are evaluated to the extent needed for CDF and LERF 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 4.5.4-1 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	The systems analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed to select, to model, and to quantify the system unavailability. Assumptions and bases shall be stated.

Table 4.5.4-2(a) Supporting Requirements for HLR-SY-A (a03)

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 system. 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 final or updated SAR, 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 abnormal conditions *(h)* system configuration during normal and abnormal conditions		
SY-A4	No requirement to perform plant walkdowns or interviews.	PERFORM plant walkdowns and interviews with system engineers and plant operators to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	
SY-A5	In the system model, INCLUDE those conditions that prevent the system from meeting the desired system function. INCLUDE the effects of both normal and alternate system alignments, to the extent needed for CDF and LERF determination.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.4-2(a) 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-A6	In defining the system model boundary [see SY-A3(a)], INCLUDE within the boundary the components required for system operation, support systems interface required for actuation and operation of the system components, and other components whose failures would degrade or fail the system.		
SY-A7	DEVELOP detailed systems models, unless sufficient system-level data are available to quantify the system failure probability, or 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. A single data value may be used for systems with no equipment or human-action dependencies, 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. A system model may be developed in which a single failure of a super component is used to represent the collective impact of failures of several components. In such a "reduced" model, RETAIN the major contributors to system unavailability, and INCLUDE components or support systems shared with other modeled systems. 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.		
SY-A8	IDENTIFY 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 in the system model if the pump failure data used in quantifying the system model include control circuit failures. MODEL separately portions of a component boundary that are shared by another component or affect another component, in order to account for the dependent failure mechanism.		
SY-A9	If a detailed system model is developed, MODEL separately all trains of a multi-train system in the fault tree models.		
SY-A10	If super components or modules are used to simplify system fault trees, 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 that are dependent on the scenario. Examples of such events include *(a)* hardware failures that are not recoverable versus actuation signals, which are recoverable *(b)* HE events 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 used by other systems		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.4-2(a) 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-A11	INCORPORATE the effect of variable success criteria into the system modeling. Example causes of variable system success criteria are *(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 accident initiating event category). *(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 non-critical 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 when both units are challenged by the same initiating event (e.g., LOOP).		
SY-A12	INCLUDE in the system model the equipment and components whose failure would affect system operability (as identified in the system success criteria), except when excluded using the criteria in SY-A14. 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. 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 failure modes for components contained in the model, consistent with available data and model level of detail, except where excluded using the criteria in SY-A14. 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 *(p)* other failures of a component to perform its required function		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.4-2(a) 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-A14	In meeting SY-A12 and SY-A13, 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, taking into account the same effect on system operation. *(c)* The screened contributors are position faults for components (such as those that occur during or following test and maintenance activities) for which the component receives an automatic signal to place it in its required state and no other position faults exists (e.g., pulled breakers) that would preclude the component from receiving the signal.		
SY-A15	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, para. 4.5.5.)		In the systems analysis, INCLUDE HFEs that cause the system or component to be unavailable when demanded. These events are referred to as pre-initiator human events. **To avoid double counting, CHECK that the data within the equipment-failure data base that are used for the equipment failure rates do not include events that are captured in the pre-initiator-HEP calculation.** (See also Human Reliability Analysis, para. 4.5.5.)
SY-A16	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 in the Accident Sequence Analysis. These HFEs are referred to as post-initiator human actions. (See also Human Reliability Analysis, para. 4.5.5, and Accident Sequence Analysis, para. 4.5.2.)		
SY-A17	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 SHOW that their exclusion does not to 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		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.4-2(a) 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-A18	In the systems model, INCLUDE out-of-service unavailability for components in the system model, unless screened, 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: *(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-A19	MODEL explicitly system conditions that cause a loss of desired system function by using realistic functional requirements that are supported with engineering analysis. For example: excessive heat loads, excessive electrical loads, etc. to support continued operation of the system for the required mission time that are based on plant-specific or acceptable generic analyses. **If engineering analyses are not available, ASSUME that the equipment/system fails with a probability of 1.0 or JUSTIFY the assumed failure probability.**		MODEL explicitly system conditions that cause a loss of desired system function by using realistic functional requirements that are supported with engineering analysis. For example: excessive heat loads, excessive electrical loads, etc. to support continued operation of the system for the required mission time that are based on plant-specific or acceptable generic analyses.
SY-A20	DO NOT TAKE CREDIT for system or component operability beyond rated or design capabilities unless justified, based on an appropriate combination of *(a)* test or operational data *(b)* calculations *(c)* vendor input *(d)* expert judgment JUSTIFY the basis for any credit taken.		
SY-A21	DEVELOP 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-A22	DO NOT MODEL the repair of hardware faults, unless the probability of repair is justified through an adequate recovery analysis or examination of data. (See DA-C14.)		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

(a03)

Table 4.5.4-2(b) 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, which is consistent with DA-D5), or SHOW 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 [Note (1)]. 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-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 SHOW that they do not impact the results.
SY-B3	ESTABLISH 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.		
SY-B4	INCORPORATE common cause failures into the system model consistent with the common cause model used for data analysis. (See DA-D6.)		
SY-B5	ACCOUNT explicitly for the modeled system's dependency on support systems in the modeling process. This may be accomplished by *(a)* fault tree linking *(b)* dependency matrices that are translated into event tree structure, event tree logic rules, or conditional split fractions		
SY-B6	PERFORM engineering analyses of 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. Bounding or generic engineering analyses (i.e., tests, operational experience, or calculations) may be used when it is not expected that these analyses would interfere with the realistic quantification of CDF or LERF.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.4-2(b) 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-B7	BASE support system modeling on **the use of conservative success criteria and timing.**	BASE support system modeling on **realistic success criteria and timing, unless a conservative approach can be justified. i.e., if their use does not impact risk significant contributors.**	BASE support system modeling on **realistic plant-specific success criteria and timing.**
SY-B8	IDENTIFY spatial and environmental hazards that may impact system operation, and ACCOUNT for them in the system fault tree or the accident sequence evaluation. Example: Use results of plant walkdowns as a source of information and resolution of issues in the evaluation of their impacts.		
SY-B9	INCLUDE explicit treatment of containment vent effects (BWRs) and containment failure effects on system operation in the consideration of possible hazards.		
SY-B10	When modeling a system, INCLUDE the support systems required for successful operation of the system for a required mission time. Examples: *(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-B11	**IDENTIFY** those systems that are required for initiation and actuation of a system. MODEL them unless a justification is provided. 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.	
SY-B12	MODEL the ability of the available inventories of air, power, and cooling to support the mission time. TREAT these inventories in the model unless a justification is provided.		
SY-B13	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.		
SY-B14	Some systems use components and equipment that are required for operation of other systems. INCLUDE components that may otherwise be screened from a system model, if their failure affects more than one system (e.g., a common suction pipe feeding two separate systems).		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.4-2(b) 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-B15	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)* 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		
SY-B16	INCLUDE operator interface dependencies across systems or trains, where applicable.		

NOTE:

(1) NUREG/CR-5485, Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment, November 20, 1998

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.5-2(b) 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, ESTABLISH rules for screening **classes of** activities from further consideration. Example: Screen maintenance and test activities 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, ESTABLISH rules for screening **individual** activities from further consideration. Example: Screen maintenance and test activities from further consideration only if ***(a)* equipment is automatically re-aligned on system demand, or** ***(b)* following maintenance activities, a post-maintenance functional test is performed that reveals misalignment, or** ***(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 activities that could simultaneously have an impact on multiple trains of a redundant system or diverse systems.		

Table 4.5.5-2(c) 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	DEFINE HFEs at a level of detail consistent with that of the system and accident sequence models.		
HR-C2	For each unscreened activity, 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	For each unscreened activity, 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 **ADD 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

(a03)

Table 4.5.5-2(d) 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 [Note (1)] and ASEP [Note (2)].		
HR-D2	USE **screening estimates** in the quantification of the pre-initiator HEPs.	USE **detailed assessments** in the quantification of pre-initiator HEPs **for significant human failure basic events. Screening values may be used in the quantification of the pre-initiator HEPs for nonsignificant human failure basic events.**	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 of written procedures (for performing tasks) and administrative controls (for independent review) *(b)* the quality of the human-machine interface, including both the equipment configuration, and instrumentation and control layout	
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 consistent with selected methodology. If recovery of pre-initiator errors is credited *(a)* ESTABLISH 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: *(1)* post-maintenance or post-calibration tests required and performed by procedure *(2)* independent verification, using a written check-off list, which verify component status following maintenance/testing *(3)* original performer, using a written check-off list, makes a separate check of component status at a later time *(4)* work shift or daily checks of component status, using a written check-off list		
HR-D5	ASSESS the joint probability of those HFEs identified as having some degree of dependency (i.e., having some common elements in their causes, such as performed by the same crew in the same time-frame).		
HR-D6	PROVIDE an assessment of the uncertainty in the HEPs. USE mean values when providing point estimates of HEPs.		
HR-D7	CHECK the reasonableness of the HEPs in light of the plant's history, procedures, operational practices, and experience. Operating experience may be used to support quantification of impact that test, maintenance, and calibration activities have on overall system unavailability.		

NOTES:

(1) NUREG/CR-1278, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications; A.D. Swain and H.E. Guttmann; August 1983 (THERP)

(2) NUREG/CR-4772, Accident Sequence Evaluation Program Human Reliability Analysis Procedure; A.D. Swain; February 1987 (ASEP)

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.5-2(e) Supporting Requirements for HLR-HR-E (a03)

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 *(a)* REVIEW the plant-specific emergency operating procedures, and other relevant procedures (e.g., AOPs, annunciator response procedures) in the context of the accident scenarios *(b)* REVIEW system operation such that an understanding of how the system(s) functions and the human interfaces with the system is obtained		
HR-E2	INCLUDE *(a)* those actions 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)* those actions performed by the control room staff either in response to procedural direction or as skill-of-the-craft to recover a failed function, system or component that is used in the performance of a response action as identified in 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.**	**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 is consistent with plant observations and training procedures.**	
HR-E4	No requirement for using simulator observations or talk-throughs with operators to confirm response models.	USE simulator observations or talk-throughs with operators to confirm the response models for scenarios modeled.	

Table 4.5.5-2(f) 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 a set of Human Failure Events (HFEs) as unavailabilities of functions, systems or components as appropriate to the level of detail in the accident sequence and system models. 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.5-2(f) Supporting Requirements for HLR-HR-F (Cont'd)

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-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., component level) required to achieve the goal of the response**

(a03)

Table 4.5.5-2(g) 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 human failure basic events. 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.5-2(g) 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-G3	**USE an approach that takes the following into account:** *(a)* the complexity of the response *(b)* the time available and time required to complete the response *(c)* some measure of scenario-induced stress The ASEP Approach 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 the cues/indications** ***(e)* human-machine interface** *(f)* time available and time required to complete the response *(g)* complexity of 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	BASE the time available to complete actions on **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.	BASE the time available to complete actions on **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.	BASE the time available to complete actions **on plant-specific thermal/hydraulic analysis, or simulations.** SPECIFY the point in time at which operators are expected to receive relevant indications.
HR-G5	**ESTIMATE** the time required to complete actions. **The approach described in ASEP is an acceptable approach.**	**BASE** the required time to complete actions **for significant human failure basic events on actual time measurements in either walkthroughs or talk-throughs of the procedures or simulator observations.**	**BASE** the required time to complete actions **on actual time measurements in either walkthroughs or talk-throughs of the procedures or simulator observations.**
HR-G6	CHECK the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to check 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, ASSESS the degree of dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including *(a)* the 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.)		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.5-2(g) 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-G8	DEFINE and JUSTIFY (accounting for the dependencies identified in supporting requirement HR-G7) the minimum probability to be used for the joint probability of multiple human errors occurring in a given cutset.		
HR-G9	Characterize the uncertainty in the estimates of the HEPs, and PROVIDE mean values for use in the quantification of the PRA results.		

(a03)

Table 4.5.5-2(h) 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).

Index No. HR-H	Capability Category I	Capability Category II	Capability Category III
HR-H1	INCLUDE human recovery actions that can restore the functions, systems, or components **on an as-needed basis to eliminate unnecessarily conservative contributions to accident sequences.**		INCLUDE human recovery actions that can restore the functions, systems, or components.
HR-H2	CREDIT operator recovery actions only if, on a plant-specific basis: *(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 HR-G3 *(d)* there is sufficient manpower to perform the action		
HR-H3	ACCOUNT for any dependency between the HFE for recovery and any other HFEs in the sequence, scenario, or cutset to which the recovery is applied.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.5-2(i) Supporting Requirements for HLR-HR-I (a03)

The HRA shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-HR-I).

Index No. HR-I	Capability Category I	Capability Category II	Capability Category III
HR-I1	DOCUMENT the HRA in enough detail to reproduce results and permit reviewers to understand limitations imposed by the models, assumptions, and data, including the following: *(a)* HRA methodology and process used to identify pre- and post-initiator HEPs *(b)* generic and plant-specific assumptions that were made in the HRA, including: *(1)* the bases for the assumptions *(2)* their impact on the CDF and LERF results *(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)* source(s) of data used to quantify human actions, including *(1)* screening values and their bases *(2)* best estimates with uncertainties and their bases *(3)* the method and treatment of dependencies for post-initiator actions *(4)* all pre- and post-initiator human actions evaluated by model, system, initiating event, and function *(5)* the HEPs for each post-initiator human action and the identified dependency effects. *(e)* justification of the minimum probability used for the joint probability of multiple human errors occurring in a given outset.		

4.5.6 Data Analysis (DA)

4.5.6.1 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.

Table 4.5.6-1 High Level Requirements for Data Analysis (DA) (a03)

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 plant-specific data shall be collected consistent with the parameter definitions of HLR A and the grouping rationale of HLR 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 shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.6-2(a) 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. ESTABLISH definitions of SSC boundaries, failure modes, and mission success criteria consistent with corresponding basic event definitions in Systems Analysis (SY-A4, SY-A7, and SY-A8) for failure rates and common cause failure parameters, and ESTABLISH boundaries of unavailability events consistent with corresponding definitions in Systems Analysis (SY-A19). Basic events typically 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	USE an appropriate probability model for each basic event. Examples include *(a)* binomial distributions for failure on demand *(b)* Poisson distributions for standby and operating failures and initiating events		
DA-A3	IDENTIFY the parameter to be estimated and the data required. Examples are as follows: *(a)* For failures on demand or unavailability due to test or maintenance, the parameter is the probability of failure or unavailability on demand, 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.6-2(b) Supporting Requirements for HLR-DA-B (a03)

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: ***(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:** ***(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.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.6-2(c) Supporting Requirements for HLR-DA-C (Cont'd)

Generic parameter estimates shall be chosen and plant-specific data shall be collected in accordance with the parameter definitions of HLR A and the grouping rationale of HLR B (HLR-DA-C).

Index No. DA-C	Capability Category I	Capability Category II	Capability Category III
DA-C12	EVALUATE 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. Special attention should be paid to the case of a multi-plant site with shared systems, 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, **provide conservative estimates.**	EVALUATE 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. Special attention should be paid to the case of a multi-plant site with shared systems, when the 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 plant maintenance and operations staff 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-C13	EXAMINE coincident outage times for redundant equipment (both intra- and inter-system) based on actual plant experience. CALCULATE outage unavailabilities that reflect actual plant experience.		
DA-C14	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-C15	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.		

NOTES:

(1) NUREG/CR-4639, Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR), Vols. 1–5, 1994
(2) NUREG/CR-4550, Auxiliary Feedwater System Risk-Based Inspection Guide for the Salem Nuclear Generating Station, October 1990
(3) NUREG/CR-5497, Common-Cause Failure Parameter Estimations
(4) NUREG/CR-6268, Common Cause Failure Database and Analysis System, Vols. 1–4, 1998
(5) NUREG/CR-5496, Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1986
(6) NUREG/CR-5032, Modeling Time to Recover and Initiate Even Frequency for Loss-of-Offsite Power Incidents at Nuclear Power Plants, March 1988

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

(a03)

Table 4.5.6-2(d) 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 DA-D2; USE generic information for the remaining events.**	**CALCULATE realistic parameter estimates for significant basic events; if sufficient plant-specific data is not available, use a Bayesian update process of generic industry data. CHOOSE prior distributions as either noninformative, or representative of variability in industry data. CALCULATE parameter estimates for the remaining events by using generic industry data.**	**CALCULATE realistic parameter estimates; if sufficient plant-specific data is not available, use a Bayesian update process of generic industry data. CHOOSE 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.		
DA-D3	PROVIDE a **characterization (e.g., qualitative discussion)** of the uncertainty intervals for the estimates of those parameters **used for estimating the probabilities of the basic events that contribute measurably to CDF and LERF.**	PROVIDE a **mean value of and a statistical representation** of the uncertainty intervals for the parameter estimates **of significant basic events. Acceptable systematic methods include Bayesian updating, frequentist method, or expert judgment.**	PROVIDE a **mean value of and a statistical representation** of the uncertainty intervals for the parameter estimates. **Acceptable systematic methods include Bayesian updating, frequentist method, 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, PERFORM the following tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application: *(a)* CONFIRM that the Bayesian updating does not produce a posterior distribution with a single bin histogram *(b)* EXAMINE the cause of any unusual posterior distribution shapes *(c)* EXAMINE inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate *(d)* CONFIRM that the Bayesian updating algorithm provides valid results over the range of values being considered *(e)* CONFIRM the reasonableness of the posterior distribution mean value	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.6-2(d) 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) or an equivalent for the estimation of 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** common cause failure probabilities.	USE **realistic** common cause failure 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 [Note (1)]. DETERMINE that the systems and data models are consistent.**	USE **realistic** common cause failure 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 [Note (1)]. DETERMINE that the systems and data models are consistent.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.6-2(d) 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-D7	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.

NOTE:

(1) NUREG/CR-5485, Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment, November 20, 1998

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.7-2(b) Supporting Requirements for HLR-IF-B (Cont'd) (a03)

The potential flood sources in the plant and their associated flooding mechanisms shall be identified (HLR-IF-B).

Index No. IF-B	Capability Category I	Capability Category II	Capability Category III
IF-B4	In each flood area, IDENTIFY any floor drains (i.e., any physical structure that can function as a drain) or sumps (i.e., any physical structure that allows for the accumulation and retention of water). DETERMINE the capacity of the drains and the amount of water retained by the sumps. If these are larger than a flood source in the area and the flood source cannot cause additional equipment damage or failure (see IF-C4 and IF-C5), then the flood source may be eliminated.		

Table 4.5.7-2(c) Supporting Requirements for HLR-IF-C (a03)

The potential flooding scenarios shall be developed for each flood source by identifying the propagation path(s) of the water and the affected SSCs (HLR-IF-C).

Index No. IF-C	Capability Category I	Capability Category II	Capability Category III
IF-C1	For each flood source, IDENTIFY the propagation path from the flood source area to its accumulation point; INCLUDE the normal flow path from one area to another via drain lines; and areas connected via back flow 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.		
IF-C2	IDENTIFY plant design features or operator actions that have the ability to terminate the flood propagation. INCLUDE the availability of flood alarms, flood dikes, curbs, drains, sump pumps, spray shields, water-tight doors, and operator actions. INCLUDE credit for non-flood proof doors or barriers for isolation of flooding only where the credit is justified by consideration of methods of detection, accessibility to the isolation device, and time available to perform the action with due consideration of procedures and skill of the craft.		
IF-C3	IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence, jet impingement, spray, pipe whip, humidity, condensation, temperature concerns, and any other identified failure modes in the identification process. JUSTIFY exclusion of any SSC's susceptibility to a flood-induced environment based on appropriate documented criteria such as test or experimental data, equipment qualification data, or other analyses. If susceptibility information cannot be ascertained, ASSUME the equipment will fail in the presence of the associated flood-induced environment.		
IF-C4	DEVELOP flood scenarios by examining potential propagation paths, giving credit for appropriate flood mitigation systems or operator actions, and identifying susceptible SSCs. VERIFY any information used from documents during plant walkdown.		
IF-C5	USE as criteria to screen out flood areas one or more of the following: *(a)* an area (including adjacent areas where flood sources can propagate) with no mitigating equipment modeled in the PRA; *(b)* a flood within the area does not cause an initiating event, including a manual scram; or *(c)* an area with no flood sources sufficient (e.g., through either spray or immersion) to cause failure of equipment; *(d)* an area with flooding mitigation systems (e.g., drains or sump pumps) capable of preventing unacceptable flood levels and the nature of the flood does not cause equipment failure (e.g., through spray) JUSTIFY any other qualitative screening criteria (provide evidence that the qualitative alternative used is unacceptable).		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.7-2(c) Supporting Requirements for HLR-IF-C (Cont'd)

The potential flooding scenarios shall be developed for each flood source by identifying the propagation path(s) of the water and the affected SSCs (HLR-IF-C).

Index No. IF-C	Capability Category I	Capability Category II	Capability Category III
IF-C6	**USE potential human mitigative actions as additional criteria for screening if all the following can be shown:** ***(a)*** **an area has small or modest flood sources that is not in a propagation path from small or modest sources** ***(b)*** **the time to the damage of safe shutdown equipment is greater than 2 hr for the worst flooding initiator** ***(c)*** **flood indication is available in the control room** ***(d)*** **the flood sources in the area can be isolated**	**USE potential human mitigative actions as additional criteria for screening if all the following can be shown:** ***(a)*** **an area has small or modest flood sources that is not in a propagation path from small or modest sources** ***(b)*** **the mitigative action can be performed with high reliability for the worst flooding initiator** ***(c)*** **flood indication is available in the control room** ***(d)*** **the flood sources in the area can be isolated**	**DO NOT SCREEN flood scenarios that rely on operator action to prevent challenges to normal plant operations.**

(a03)

Table 4.5.7-2(d) Supporting Requirements for HLR-IF-D

Flooding-induced initiating events shall be identified and their frequencies estimated (HLR-IF-D).

Index No. IF-D	Capability Category I	Capability Category II	Capability Category III
IF-D1	USE a structured, systematic process to identify those flood-induced initiating events that challenge normal plant operation and that require successful mitigation to prevent core damage. INCLUDE the potential for a flooding-induced transient or LOCA.		
IF-D2	No requirement to review the impact of plant-specific initiating event precursors and system alignments.	In searching for flood-induced initiating events, REVIEW the impact of plant-specific initiating event precursors and system alignments. INCLUDE alignments of supporting systems.	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.7-2(d) Supporting Requirements for HLR-IF-D (Cont'd)

Flooding-induced initiating events shall be identified and their frequencies estimated (HLR-IF-D).

Index No. IF-D	Capability Category I	Capability Category II	Capability Category III
IF-D3	GROUP flooding-induced initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, and timing; or *(b)* events can be subsumed into a group and bounded by the worst-case impacts within the new group.	GROUP flooding-induced initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, and timing, **and the effect on the operability and performance of operators and relevant mitigating systems;** or *(b)* events can be subsumed into a group and bounded by the worst case impacts within the new group. **To avoid excess conservatism, DO NOT ADD initiating events to a group and DO NOT SUBSUME events into a group unless the impacts are comparable to or less than those of the remaining events in that group.**	GROUP flooding-induced initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, and timing, **and the effect on the operability and performance of operators and relevant mitigating systems;** or *(b)* events can be subsumed into a group and bounded by the worst case impacts within the new group. **To avoid conservatism, DO NOT ADD initiating events to a group and DO NOT SUBSUME events into a group unless the impacts are comparable to or less than those of the remaining events in that group, or it is demonstrated that such grouping does not appreciably impact CDF or LERF.**
IF-D4	For multi-unit sites with shared systems or structures, **INCLUDE multi-unit internal flood initiators.**	For multi-unit sites with shared systems or structures, **TREAT quantitatively multi-unit internal flood initiators.**	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.7-2(d) Supporting Requirements for HLR-IF-D (Cont'd)

Flooding-induced initiating events shall be identified and their frequencies estimated (HLR-IF-D).

Index No. IF-D	Capability Category I	Capability Category II	Capability Category III
IF-D5	DETERMINE the flood initiating event frequency by using the applicable requirements in IE-C (specifically IE-C1 through IE-C5) and one or more of the following: *(a)* an assessment of **applicable** generic operating experience of internal flooding *(b)* an evaluation of pipe, component, and tank rupture failure rates from generic data sources *(c)* a probabilistic fracture mechanics evaluation of the probability of pipe leaks or ruptures **representative of** plant-specific conditions *(d)* a combination of operating experience, and generic pipe and component failure rates *(e)* a combination of one of the above approaches with expert judgment	DETERMINE the flood initiating event frequency by using the applicable requirements in IE-C (specifically IE-C1 through IE-C5) and one or more of the following: *(a)* an assessment of generic **and plant-specific** operating experience of internal flooding *(b)* an evaluation of pipe, component, and tank rupture failure rates from generic data sources **enhanced by any plant-specific information** *(c)* a probabilistic fracture mechanics evaluation of the probability of pipe leaks or ruptures **using** plant-specific information *(d)* a combination of **plant-specific** operating experience and generic pipe and component failure rates *(e)* a combination of one of the above approaches with expert judgment	

(a03) **Table 4.5.7-2(e) Supporting Requirements for HLR-IF-E**

Flood-induced accident sequences shall be quantified (HLR-IF-E).

Index No. IF-E	Capability Category I	Capability Category II	Capability Category III
IF-E1	REVIEW the accident sequence results obtained by following the applicable requirements described in para. 4.5.2 and MODIFY sequences as necessary to account for any flood-induced phenomena.		
IF-E2	PERFORM any necessary engineering calculations for flood rate, time to reach vulnerable equipment, and the structural capacity of SSCs according to the requirements described in SC-B1 through SC-B6.		
IF-E3	MODIFY the systems analysis results obtained by following the applicable requirements described in para 4.5.4 to include flood-induced failures identified by IF-C3.		
IF-E4	PERFORM any additional data analysis to the applicable requirements described in para. 4.5.6.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.7-2(e) Supporting Requirements for HLR-IF-E (Cont'd)

Flood-induced accident sequences shall be quantified (HLR-IF-E).

Index No. IF-E	Capability Category I	Capability Category II	Capability Category III
IF-E5	PERFORM any human reliability analysis to the applicable requirements described in para. 4.5.5 (HLR-HR-E through HLR-HR-G), and INCLUDE the following scenario specific PSFs for control room and ex-control room actions as appropriate: *(a)* additional workload and stress (above that for similar sequences not caused by internal floods) *(b)* uncertainties in event progression (e.g., cue availability and timing concerns caused by flood) *(c)* effect of flood on mitigation, required response, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) *(d)* flooding-specific job aids and training (e.g., procedures, training exercises) JUSTIFY the use of extraordinary recovery actions that are not proceduralized (i.e., provide evidence of appropriate training that would ensure knowledge, skill of the craft).		
IF-E6	PERFORM internal flood sequence quantification in accordance with the applicable requirements described in para. 4.5.8, including any quantitative screening. INCLUDE the combined effects of failures caused by flooding and those coincident with the flooding due to independent causes including equipment failures, unavailability due to maintenance, and other credible causes. INCLUDE 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.		
IF-E7	REVIEW the LERF analysis results obtained by following the applicable requirements described in para. 4.5.9 and MODIFY as necessary to account for any flood-induced phenomena.		

Table 4.5.7-2(f) Supporting Requirements for HLR-IF-F

The internal flooding analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed, with assumptions and bases stated (HLR-IF-F).

Index No. IF-F	Capability Category I	Capability Category II	Capability Category III
IF-F1	DOCUMENT the process used to identify flood sources, flood areas, flood pathways, flood scenarios, and their screening, and internal flood model development and quantification.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.7-2(f) Supporting Requirements for HLR-IF-F (Cont'd)

The internal flooding analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed, with assumptions and bases stated (HLR-IF-F).

Index No. IF-F	Capability Category I	Capability Category II	Capability Category III
IF-F2	DOCUMENT the following results *(a)* flood sources identified in the analysis, any rules used to screen out these sources, and the resulting list of sources to be further examined *(b)* flood areas used in the analysis and the reason for eliminating any of these areas from further analysis *(c)* propagation pathways between flood areas and any assumptions, calculations, or other bases for eliminating or justifying any of these propagation pathways *(d)* accident mitigating features and barriers credited in the analysis, the extent to which they were credited, and associated justification *(e)* component fragilities and any associated assumptions or calculations used in the determination of the impacts of submergence, spray, temperature, or other flood-induced effects on equipment operability *(f)* screening criteria used in the analysis *(g)* flooding scenarios considered, screened, and the remaining scenarios, as well as how the internal event analysis models were modified to model these remaining scenarios for the internal flooding analysis *(h)* flood frequencies, component unreliabilities/unavailabilities, and HEPs used in the analysis (i.e., the data values unique to the flooding analysis) *(i)* any calculations or other analyses used to support or refine the flooding evaluation *(j)* results of the internal flooding analysis including results from each accident sequence, results from the combined accident sequence model (i.e., the total plant model), results from sensitivity and uncertainty analyses, and results from importance measure calculations		

4.5.8 Quantification (QU)

(a03) **4.5.8.1 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 and appropriately quantified.

(a03) **Table 4.5.8-1 High Level Requirements for the Level 1 Accident Sequence Quantification (QU) and Results Interpretation**

Designator	Requirement
HLR-QU-A	The Level 1 quantification shall quantify core damage frequency.
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, 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. Key sources of model uncertainty and key assumptions shall be identified, and their potential impact on the results understood.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.8-1 High Level Requirements for the Level 1 Accident Sequence Quantification (QU) and Results Interpretation (Cont'd)

Designator	Requirement
HLR-QU-F	Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assuumptions made and their bases.

Table 4.5.8-2(a) Supporting Requirements for HLR-QU-A (a03)

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 sequence delineation, 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 point estimate** CDF from internal events. PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF to identify significant sequences and confirm the sequence logic is appropriately reflected. The estimates may be accomplished by using either fault tree linking or event trees with conditional split fractions.	**ESTIMATE the mean** CDF from internal events, **ensuring that the "state-of-knowledge" correlation between event probabilities is taken into account.** PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF to identify significant sequences and confirm the sequence logic is appropriately reflected. The estimates may be accomplished by using either fault tree linking or event trees with conditional split fractions.	CALCULATE the mean CDF from internal events **by propagating the uncertainty distributions, ensuring that the "state-of-knowledge" correlation between event probabilities is taken into account.** PROVIDE estimates of the individual sequences in a manner consistent with the estimation of total CDF to identify significant sequences and confirm the sequence logic is appropriately reflected. The estimates may be accomplished by using either fault tree linking or event trees with conditional split fractions.
QU-A3	SELECT a method that is capable of discriminating the contributors to the CDF commensurate with the level of detail in the model.		
QU-A4	INCLUDE recovery actions in the quantification process in applicable sequences and cutsets. Recovery actions credited in the evaluation may be either proceduralized or have reasonable likelihood of success assuming that trained and qualified personnel are performing the recovery action(s).		

Table 4.5.8-2(b) Supporting Requirements for HLR-QU-B (a03)

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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.8-2(b) Supporting Requirements for HLR-QU-B (Cont'd)

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-B2	TRUNCATE accident sequences and associated system models at a sufficiently low cutoff value that dependencies (e.g., human failure dependencies) associated with significant cutsets or accident sequences are not eliminated.		
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.		
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 [Note (1)]. When resolving circular logic, AVOID introducing unnecessary conservatisms or non-conservatisms.		
QU-B6	ACCOUNT for 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. CORRECT by either: *(a)* designing path-dependent logic to eliminate mutually exclusive situations, or *(b)* deleting cutsets containing mutually exclusive events.		
QU-B8	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-B9	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)		

NOTE:
(1) NUREG/CR-2728, Interim Reliability Evaluation Program Procedures Guide, March 3, 1983

(a03) **Table 4.5.8-2(c) 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 by requantifying the PRA model with HEP values set to values that are sufficiently high that the cutsets are not truncated. AVOID premature truncation of cutsets with multiple HFEs, by selecting appropriate HEP screening values in the initial quantification. The final quantification of these post-initiator HFEs may be done at the cutset or saved sequence level.		
QU-C2	ASSESS the degree of dependency between the HFEs in the cutset or sequence in accordance with HR-D5 and HR-G7. INCLUDE both the cognitive aspects, and the constraints imposed by available time or shared equipment. ESTIMATE the combined probability of the HFEs taking into account the dependency.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.8-2(c) 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-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 4.5.8-2(d) Supporting Requirements for HLR-QU-D (a03)

The quantification results shall be reviewed and significant contributors to CDF, such as initiating events, accident sequences, 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.

Index No. QU-D	Capability Category I	Capability Category II	Capability Category III
QU-D1	REVIEW a sample of the significant cutsets and sequences sufficient to determine that the logic of the cutset or sequence is reasonable and to identify that there are no anomalies in the results. REVIEW the results of the PRA for modeling consistency (e.g., event sequence models consistency with systems models and success criteria) and operational consistency (e.g., plant configuration, procedures, and plant-specific and industry experience). REVIEW results to determine that the flag event settings, mutually exclusive event rules, and recovery rules yield logical results.		
QU-D2	IDENTIFY the modeling assumptions that drive the results. QUESTION modeling assumptions, asking if conditions outside those modeled could occur and, if so, could success criteria or other assumptions change. QUESTION modeled human actions for consistency with plant procedures and the range of conditions that would be obtained in the associated PRA sequence.		
QU-D3	No requirement to compare results to those from similar plants	COMPARE results to those from similar plants and IDENTIFY causes for differences in significant contributors.	
QU-D4	REVIEW a sampling of nonsignificant accident cutsets or sequences to determine they are reasonable and have physical meaning.		
QU-D5	IDENTIFY significant contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors. An acceptable approach is the use of importance measures. REVIEW the importance values for components and basic events to determine they make logical sense.	IDENTIFY significant contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors. An acceptable approach is the use of importance measures. **EXAMINE the importance of SSCs that contribute to initiating event frequencies.** REVIEW the importance values for components and basic events to determine they make logical sense.	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

(a03) **Table 4.5.8-2(e) Supporting Requirements for HLR-QU-E**

Uncertainties in the PRA results shall be characterized. Key sources of model uncertainty and key 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	CHARACTERIZE parameter uncertainty consistent with DA-D3.		
QU-E2	IDENTIFY key sources of model uncertainty, and the assumptions made or models adopted in response to those uncertainties.		
QU-E3	**ESTIMATE the uncertainty interval of the overall CDF results. Provide a basis for the estimate.**	**ESTIMATE the uncertainty interval of the overall CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties taking into account the "state-of-knowledge" correlation.**	**PROPAGATE parameter uncertainties 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 probabilities is taken into account.**
QU-E4	**PROVIDE an assessment of the impact of the key model uncertainties on the results of the PRA.**	**EVALUATE the sensitivity of the results to uncertain model boundary conditions and other key assumptions using sensitivity analyses. EXAMINE key assumptions and parameters both individually and in logical combinations For example, a sensitivity analysis of logical combinations may evaluate the combined effects of modeling assumptions, HEP, CCF probabilities, and safety function success criteria.**	**EVALUATE the sensitivity of the results to uncertain model boundary conditions and other key assumptions using sensitivity analyses. EXAMINE key assumptions and parameters both individually and in logical combinations. For example, sensitivity analyses of logical combinations may evaluate the combined effects of modeling assumptions, HEPs, CCF probabilities, and safety function success criteria unless such sources of uncertainties have been adequately treated in the quantitative uncertainty analysis.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.8-2(f) Supporting Requirements for HLR-QU-F (a03)

Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-QU-F).

Index No. QU-F	Capability Category I	Capability Category II	Capability Category III
QU-F1	DOCUMENT the model integration process. INCLUDE any recovery analysis, and the results of the quantification including uncertainty and sensitivity analyses. 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 accidents to be non-dominant *(h)* the results of all sensitivity studies *(i)* the uncertainty distribution for the total CDF *(j)* importance measure results *(k)* a list of mutually exclusive events eliminated from the resulting cutsets and their bases for elimination *(l)* key assumptions used in the quantification and an assessment of their risk significance		
QU-F2	DESCRIBE the significant contributors (such as initiating events, accident sequences, basic events) to CDF in the PRA results summary.	DESCRIBE the significant contributors (such as initiating events, accident sequences, basic events) to CDF in the PRA results summary. **PROVIDE a detailed description of dominant accident sequences or functional failure groups.**	
QU-F3	DOCUMENT key assumptions and causes of uncertainty, such as: possible optimistic or conservative success criteria, suitability of the reliability data, possible modeling uncertainties (modeling limitations due to the method selected), degree of completeness in the selection of initiating events, possible spatial dependencies, etc.		
QU-F4	DOCUMENT asymmetries in quantitative modeling to provide application users the necessary understanding regarding why such asymmetries are present in the model.		
QU-F5	DOCUMENT the process used to illustrate the computer code(s) used to perform the quantification will yield correct results process.		
QU-F6	DOCUMENT limitations that would impact applications.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

4.5.9 LERF Analysis (LE)

(a03) **4.5.9.1 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 LERF or non-LERF.

(a03) **Table 4.5.9-1 High Level Requirements for LERF Analysis (LE)**

Description	Designator	Requirement
Plant Damage Analysis	HLR-LE-A	Core damage sequences shall be grouped into plant damage states based on their accident progression attributes.
Accident Progression Analysis	HLR-LE-B	The accident progression analyses shall include an evaluation of the credible 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 analysis shall include an evaluation of the containment structural capability for those containment challenges that would result in a large early release.
LERF Quantification	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 uncertainty shall be identified and their impact characterized.
Documentation	HLR-LE-G	The documentation of the LERF Analysis shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases.

Table 4.5.9-2(a) Supporting Requirements for HLR-LE-A

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

Index No. LE-A	Capability Category I	Capability Category II	Capability Category III
LE-A1	IDENTIFY those physical characteristics of Level 1 end states that can influence LERF. Examples include: *(a)* RCS pressure at core damage (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		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(a) Supporting Requirements for HLR-LE-A (Cont'd)

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

Index No. LE-A	Capability Category I	Capability Category II	Capability Category III
LE-A2	IDENTIFY the accident sequence characteristics that lead to the physical characteristics identified in 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 The references in Notes (1) and (2) provide lists of typical characteristics.		
LE-A3	SELECT a method for binning the accident sequences into plant damage states; this method may include extension of the accident sequences, development of a bridge tree, or some other approach.		
LE-A4	Either EXPAND the accident sequences, CONSTRUCT the bridge tree, or TRANSFER the Level 1 information to explicitly account for the LE-A1 and LE-A2 characteristics and ensure that dependencies are properly treated.		
LE-A5	DEFINE plant damage states based upon information in AS-A8, LE-A1, LE-A2, LE-A3, and LE-A4.		

NOTES:

(1) Nuclear Power Plant Response to Severe Accident, IDCOR Technical Summary Report, Atomic Industrial Forum, November 1984

(2) NUREG 1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, December 1990

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

(a03)

Table 4.5.9-2(b) Supporting Requirements for HLR-LE-B

The accident progression analysis shall include an evaluation of the credible 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 significant LERF contributors from the set identified in Table 4.5.9-3. An acceptable approach for identifying contributors that could influence LERF for the various containment types is contained in NUREG/CR-6595, January 1999.** ***(a)* INCLUDE as appropriate, unique plant issues as determined by expert judgment and/or past plant analyses.** ***(b)* EVALUATE qualitatively those contributors that are not quantified in NUREG/CR-6595 [Note (1)] (e.g., induced steam generator tube rupture).**	**IDENTIFY credible LERF contributors from the set identified in Table 4.5.9-3, sufficient to support development of realistic significant accident progression sequences. INCLUDE, as appropriate, unique plant issues as determined by expert judgment and/or past plant analyses.**	**INCLUDE the credible LERF contributors sufficient to support development of realistic accident progression sequences. Consider those contributors identified by IDCOR [Note (2)] and NUREG-1150 [Note (3)].**
LE-B2	DETERMINE the containment challenges (e.g., pressure loads, debris impingement) resulting from contributors identified in LE-B1 in a **conservative (generic or plant-specific analyses)** manner. **An acceptable alternative is the approach in NUREG/CR-6595, January 1999 [Note (1)].**	DETERMINE the containment challenges (e.g., **temperature,** pressure loads, debris impingement) resulting from contributors identified in LE-B1 in a **realistic** manner. **Conservative treatment or a combination of conservative and realistic treatment is used for nonsignificant phenomena. CONSIDER differential pressure loadings on the RCS, as RCS motions may impact containment integrity.**	DETERMINE the containment challenges (e.g., pressure loads, debris impingement) resulting from contributors identified in LE-B1 in a **realistic** manner. **CONSIDER differential pressure loadings on the RCS, as RCS motions may impact containment integrity.**

NOTES:

(1) NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events

(2) Nuclear Power Plant Response to Severe Accident, IDCOR Technical Summary Report, Atomic Industrial Forum, November 1984

(3) NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, December 1990

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(c) Supporting Requirements for HLR-LE-C (a03)

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 containment event tree(s) (CET) or equivalent structure(s) to propagate plant damage states in order to identify the accident progression sequences resulting in a large early release. The accident sequences are developed to a level of detail to account for the potential contributors identified in LE-B1 and analyzed in LE-B2. Containment event trees developed in NUREG/CR-6595 [Note (1)] (with plant-specific modifications, if needed) are acceptable.		
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 [Note (1)].** USE HFEs consistent with EOPs/Severe Accident Management Guidelines (SAMGs), proceduralized actions, or Technical Support Center guidance.	INCLUDE **realistic** treatment of feasible operator actions following the onset of core damage. **Repair of equipment may be considered if justified through an adequate recovery analysis (e.g., one that considers equipment availability, repair procedure availability, adequate time available, and environmental conditions appropriate to allow repair).** USE HFEs consistent with EOPs/SAMGs, proceduralized actions, or Technical Support Center guidance.	
LE-C3	INCLUDE those branch points necessary to provide **an accident progression sequences resulting in a large early release. Containment event trees developed in NUREG/CR-6595 [Note (1)] (with plant-specific modifications, if needed) are acceptable.**	INCLUDE those branch points necessary to provide **a realistic estimation of the significant accident progression sequences resulting in a large early release. INCLUDE significant mitigating actions by operating staff, 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) supporting the inclusion of any of these features.**	INCLUDE those branch points necessary to provide **a realistic estimation of the accident progression sequences resulting in a large early release. INCLUDE mitigating actions by operating staff, 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.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(c) 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	USE **conservative** system success criteria. **Realistic criteria may be used.**	USE **realistic** system success criteria for the significant accident progression sequences. USE conservative or a combination of conservative and realistic system success criteria for non-risk significant accident progression sequences.	USE **realistic** system success criteria.
LE-C5	DEVELOP system models that support the accident progression analysis consistent with the applicable requirements for para. 4.5.4, as appropriate for the level of detail of the analysis.		
LE-C6	DEFINE HFEs that support the accident progression analysis consistent with the applicable requirements of para. 4.5.5 as appropriate for the level of detail of the analysis.		
LE-C7	INCLUDE accident sequence dependencies in the accident progression sequences consistent with the applicable requirements of para. 4.5.2, as appropriate for the level of detail of the analysis.		
LE-C8	TREAT containment environmental impacts on continued operation of equipment and operator actions in a **conservative** manner. **An acceptable alternative is the approach in NUREG/CR-6595, January 1999 [Note (1)].**	TREAT containment environmental impacts on continued operation of equipment and operator actions in a **realistic** manner for **risk significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative and realistic treatment for nonsignificant accident progression sequences.**	TREAT containment environmental impacts on continued operation of equipment and operator actions in a **realistic** manner.
LE-C9	TREAT containment failure impacts on continued operation of equipment and operator actions in a **conservative** manner. **An acceptable alternative is the approach in NUREG/CR-6595, January 1999 [Note (1)].**	TREAT containment failure impacts on continued operation of equipment and operator actions in a **realistic** manner **for risk significant accident progression sequences resulting in a large early release. USE conservative or a combination of conservative and realistic treatment for nonsignificant accident progression sequences.**	TREAT containment failure impacts on continued operation of equipment and operator actions in a **realistic** manner.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(c) 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	TREAT containment bypass events in a **conservative** manner. **DO NOT TAKE CREDIT for reducing the class of the release by considering scrubbing. An acceptable alternative is the approach NUREG/CR-6595 [Note (1)].**	TREAT containment bypass in a **realistic** manner. **JUSTIFY any credit taken for reducing the class of the release by scrubbing (i.e., provide the source of the decontamination factor used).**	

NOTE:

(1) NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

(a03)

Table 4.5.9-2(d) 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	DETERMINE 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. EVALUATE impact of containment seals, penetrations, hatches, drywell heads (BWRs), and vent piping bellows and INCLUDE as potential containment challenges, as required. Quasi-static containment capability evaluations are acceptable unless hydrogen concentrations are expected to result in potential detonations. Such considerations need to be included for small volume containments, such as the ice-condenser type. An acceptable alternative is the approach in NUREG/CR-6595 [Note (1)].**	DETERMINE 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. EVALUATE impact of containment seals, penetrations, hatches, drywell heads (BWRs), vent pipe bellows and INCLUDE as potential containment challenges, as required. 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. Such considerations need to be included for small volume containments such as the ice-condenser type.**	DETERMINE 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. INCLUDE behavior of *(a)* containment seals *(b)* penetrations *(c)* hatches *(d)* drywell head (BWRs) *(e)* vent pipe bellows (BWRs) beyond the design basis temperature and pressure conditions. PROVIDE static and dynamic failure capabilities, as appropriate.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(d) 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-D2	When containment failure location [Note (2)] affects the classification of the accident progression as a large early release, DEFINE failure location based on a **conservative** plant-specific containment assessment. **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 [Note (1)].**	When containment failure location [Note (2)] affects the event classification of the accident progression as a large early release, DEFINE failure location based on a **realistic** plant-specific containment assessment.	
LE-D3	**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. **Evaluation of failure probability may include conservatisms. 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 exchanges 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 exchanges at applicable temperature and pressure conditions. PROVIDE static and dynamic failure capabilities, as appropriate.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.9-2(d) 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-D4	**USE a conservative evaluation of** secondary side isolation capability for significant accident progression sequences caused by SG 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 SG tube failure resulting in a large early release. **Evaluation of capability may include conservatisms. 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 relaistic comparison with similar isolation capability in similar containment designs.**	**PERFORM a realistic** secondary side isolation capability **analysis** for the accident progression sequences caused by SG tube failure resulting in a large early release. **INCLUDE behavior of relief and isolation valves at applicable temperatures and pressure conditions.**
LE-D5	TREAT induced SG tube rupture in a **conservative** manner. **A realistic treatment may be used.**	TREAT induced SG tube rupture in a **realistic** manner, **when practical. Conservative treatment may be used, when justified.**	TREAT induced SG tube rupture in a **realistic** manner.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.9-2(d) 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	TREAT containment isolation 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.	TREAT containment isolation 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.	TREAT containment isolation 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.

NOTES:
(1) NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events
(2) 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 4.5.9-2(e) Supporting Requirements for HLR-LE-E **(a03)**

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 used in accident progression analysis consistent with the applicable requirements of paras. 4.5.5 and 4.5.6 including consideration of the severe accident plant conditions for both equipment and operators, as appropriate for the level of detail of the analysis.		
LE-E2	USE **conservative** parameter estimates **for determination of CET branch points. A conservative data set for some key parameters is included in NUREG/CR-6595 [Note (1)].**	USE **realistic** parameter estimates **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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.9-2(e) 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-E3	Starting with plant damage states, QUANTIFY LERF consistent with the applicable requirements of para. 4.5.8, as appropriate for the level of detail of the analysis.		

NOTE:
(1) NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events

(a03)

Table 4.5.9-2(f) 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 impact characterized.

Index No. LE-F	Capability Category I	Capability Category II	Capability Category III
LE-F1	**IDENTIFY the significant contributors to large early release (e.g., plant damage states, containment failure modes, containment system unavailabilities). REVIEW for reasonableness.**	**PERFORM an importance analysis to identify the significant contributors to LERF.**	
LE-F2	**PROVIDE a qualitative assessment of the key sources of uncertainty.** **Examples:** ***(a)* Identify bounding assumptions.** ***(b)* Identify conservative treatment of phenomena.**	**PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies for the significant contributors to LERF.**	**PROVIDE uncertainty analysis that identifies the key sources of uncertainty and includes sensitivity studies.**

(a03)

Table 4.5.9-2(g) Supporting Requirements for HLR-LE-G

The documentation of the LERF analysis shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-LE-G).

Index No. LE-G	Capability Category I	Capability Category II	Capability Category III
LE-G1	DOCUMENT *(a)* the physical characteristics and accident sequence characteristics used to group the accident sequences into plant damage states *(b)* the method used to bin the accident sequences into plant damage states *(c)* the plant damage states and their attributes, as used in the analysis		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(g) Supporting Requirements for HLR-LE-G (Cont'd)

The documentation of the LERF analysis shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-LE-G).

Index No. LE-G	Capability Category I	Capability Category II	Capability Category III
LE-G2	DOCUMENT containment failure modes, phenomena, equipment failures, and human actions considered in the development of the accident progression sequences, where appropriate, including *(a)* containment isolation failure *(b)* containment bypass *(c)* energetic containment failures *(d)* hydrogen related phenomena *(e)* liner melt-through (Mark I containment) *(f)* containment failure at vessel breach *(g)* induced SGTRs (PWRs) *(h)* containment venting *(i)* suppression pool bypass mechanism (BWRs) *(j)* containment flooding (BWRs)		
LE-G3	DOCUMENT treatment of factors influencing containment challenges and containment capability, as appropriate for the level of detail of the analysis, including *(a)* design details (i.e., heat sink distribution, circulation paths, ignition sources, water availability, gravity drain paths, and cavity geometry) *(b)* equipment survivability credited beyond design basis envelope		
LE-G4	DOCUMENT, as appropriate for the level of detail of the analysis, *(a)* containment challenges treated *(b)* containment failure modes identified *(c)* containment event tree (CET) and basis for event tree nodes *(d)* containment capacity and its basis *(e)* containment failure locations and probabilities *(f)* basis for parameter estimates		
LE-G5	DOCUMENT the model integration process. INCLUDE the results of the quantification including uncertainty and sensitivity analyses, as appropriate for the level of detail of the analysis. Documentation typically includes *(a)* a general description of the quantification *(b)* key assumptions that affect the results *(c)* the total plant LERF and contributions from the different plant damage states and accident classes *(d)* equipment or human actions that are significant basic events *(e)* the results of all sensitivity studies, (as applicable)		
LE-G6	DESCRIBE the significant contributors to LERF.	PROVIDE a detailed description of the significant contributors (i.e., plant damage states, accident progression sequences, phenomena, containment challenges, and containment failure modes) to LERF.	
LE-G7	DOCUMENT sources of uncertainty consistent with QU-F3.		
LE-G8	IDENTIFY limitations that would impact applications.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

(a03)

Table 4.5.9-3 LERF Contributors To Be Considered

	Containment Design				
LERF Contributor	**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
(c) De-inerted operation	...	...	x	x	x
(d) Core debris impingement	[Note (2)]	x	x	x	
Steam explosion [Note (3)]	...	...	x	x	x
Shell melt-through	...	...	x	...	...
Suppression pool bypass	x	x	x	x	x
RPV and/or containment venting	...	...	x	x	x
Isolation condenser tube rupture	...	...	x (if applicable)	...	...
Vacuum breaker failure	...	...	x	x	x
Hydrodynamic loads under severe accident conditions	...	...	x	...	...
Containment flooding	...	...	x	...	...
In-vessel recovery	x	x	x	x	x
ATWS-induced failure	...	...	x	x	x

GENERAL NOTE: Combinations of phenomena should also be considered where appropriate.

NOTES:

(1) dry well (DW) isolation failure

(2) steel shell designs

(3) Steam explosion challenges are of low probability for PWRs.

Not for Resale

SECTION 5

5.1 Purpose 88
5.2 PRA Configuration Control Program 88
5.3 Monitoring PRA Inputs and Collecting New Information 88
5.4 PRA Maintenance and Upgrades 88
5.5 Pending Changes 88
5.6 Previous PRA Applications 88
5.7 Use of Computer Codes 88
5.8 Documentation 89

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 5
PRA CONFIGURATION CONTROL

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.

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 evaluates the impact of changes on previously implemented risk-informed decisions that have used the PRA

(e) a process that maintains configuration control of computer codes used to support PRA quantification

(f) documentation of the Program

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 sub-system 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.

(a03) 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 para. 5.3 shall be evaluated to determine whether such information warrants PRA maintenance or PRA upgrade. (See Section 2 for the distinction between PRA maintenance and PRA upgrade.) Changes that would impact risk-informed decisions should be prioritized to ensure that the most significant changes are 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 para. 3.5.

Changes to a PRA due to PRA maintenance and PRA upgrade shall meet the requirements of Section 4. Upgrades of a PRA shall receive a peer review in accordance with the requirements specified in Section 6, but limited to aspects of the PRA that have been upgraded.

5.5 Pending Changes (a03)

This Standard recognizes that immediately following a plant change, or upon identification of a subject for model improvement, a PRA may not represent the plant until the change is incorporated. Therefore, the PRA configuration control process shall consider the cumulative impact of pending changes on the application being performed. These changes shall be addressed in a fashion similar to the approach used in Section 3 to address elements that are determined to be inadequate.

5.6 Previous PRA Applications

A process shall exist to evaluate the impact of a PRA change on previously implemented risk-informed decisions that relied upon PRA information and that affect the safe operation of the plant.

5.7 Use of Computer Codes

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

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

SECTION 6
PEER REVIEW

(a03) 6.1 Purpose

This Section provides requirements for peer review of a PRA to be used in risk-informed decisions for commercial nuclear power plants. PRAs used for applications applying this Standard shall be peer reviewed. The peer review shall assess the PRA to the extent necessary to determine if the methodology and its implementation meet the requirements of this Standard. Another purpose of the peer review is to determine strengths and weaknesses in the PRA. The peer review need not assess all aspects of the PRA against all Section 4 requirements; however, enough aspects of the PRA shall be reviewed for the reviewers to achieve consensus on the adequacy of methodologies and their implementation for each PRA Element.

6.1.1 Frequency. Only a single complete peer review is necessary prior to using a PRA. In addition, Section 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.

6.1.2 Methodology. The review shall be performed using a written methodology that assesses the requirements of Section 4 and addresses the requirements of Section 6.

The peer review methodology shall consist of the following elements:

(a) 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 Section 4 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

NEI-00-02[1] provides an example of an acceptable review methodology; however, the differences between the supporting requirements of Section 4 of this Standard and the supporting requirements of Appendix B of NEI-00-02 shall be evaluated. This evaluation shall be documented.

6.2 Peer Review Team Composition and Personnel Qualifications

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 Section 4 and the interfaces between those elements

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

6.2.2 General (a03)

6.2.2.1

(a) The peer review team members individually shall

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

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

(b) The peer review team members shall

(1) not be allowed to review their own work or work for which they have contributed

(2) not be allowed to review a PRA for which they have a conflict of interest, such as a financial or career path incentive or disincentive that may influence the outcome of the peer review

6.2.2.2 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] NEI-00-02, Probabilistic Risk Assessment (PRA) Peer Review Process Guidance, 2002

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

(a03) **6.2.3 Specific.** 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.

One member of the peer review team (the technical integrator) shall be familiar with all the PRA Elements identified in this Standard and shall have demonstrated the capability to integrate these PRA Elements.

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.

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.

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.

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 para. 6.6 of this Standard.

(a03) 6.3 Review of PRA Elements to Confirm the Methodology

The peer review team shall use the requirements of this Subsection for the PRA Elements being reviewed to determine if the methodology and the implementation of the methodology for each PRA Element meet the requirements of this Standard. Some subparagraphs of para. 6.3 contain specific suggestions for the review team to consider during the review. Additional material for those Elements may be reviewed depending on the results obtained. These suggestions are not intended to be a minimum or comprehensive list of requirements. The judgment of the reviewer shall be used to determine the specific scope and depth of the review in each PRA Element.

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

The HLRs and the composite of the SRs of Section 4 shall be used by the peer review team to assess the completeness of a PRA Element.

6.3.1 Initiating Event Analysis (IE). The entire initiating event analysis shall be reviewed.

6.3.2 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/Station Blackout 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

6.3.3 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,

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

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

6.3.4 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 non-safety loads are isolated)

6.3.5 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

6.3.6 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

6.3.7 Internal Flooding (IF). A review shall be performed on the internal flooding analysis.

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

(a) internal flood initiating event frequencies

(b) internal flooding scenario involving each identified flood source

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

(d) internal flooding scenario that involves each of the flood-induced component failure mechanisms (i.e., one flood scenario for each mechanism)

(e) one internal flooding scenario involving each type of identified accident initiator (e.g., transient and LOCA)

6.3.8 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

6.3.9 LERF Analysis (LE). The LERF analysis and the Level 1/LERF interface process shall be reviewed.

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

(a) accident characteristics chosen for carryover to

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

LERF analysis (and for binning of PDSs if PDS methods were used)

(b) interface mechanism used

(c) CDF carryover

6.3.9.2 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 functional events necessary to achieve a safe stable containment endstate

(h) sensitivity analysis

(i) the containment response calculations, performed specifically for the PRA, for the dominant plant damage states

6.4 Expert Judgment

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

(a03) 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.

6.6 Documentation

6.6.1 Peer Review Team Documentation. The peer review team's documentation shall demonstrate that the review process appropriately implemented the review requirements. (a03)

Specifically, the peer review documentation shall include the following:

(a) identification of the version of the PRA reviewed

(b) the names of the peer review team members

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

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

(e) a discussion of the extent to which each PRA Element was reviewed

(f) results of the review identifying any differences between the requirements in Sections 4 and 5 of this Standard and the methodology implemented, defined to a sufficient level of detail that will allow the resolution of the differences

(g) identification and significance of exceptions and deficiencies with respect to SRs, including an assessment of PRA assumptions that the reviewers have determined to be relevant

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

(i) recommended alternatives for resolution of any differences

(j) identification of the strengths and weaknesses in the PRA

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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

STANDARD FOR PROBABILISTIC RISK ASSESSMENT FOR NUCLEAR POWER PLANT APPLICATIONS

ASME RA-S–2002

AN AMERICAN NATIONAL STANDARD



The American Society of Mechanical Engineers

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

The American Society of Mechanical Engineers

AN AMERICAN NATIONAL STANDARD

STANDARD FOR PROBABILISTIC RISK ASSESSMENT FOR NUCLEAR POWER PLANT APPLICATIONS

ASME RA-S–2002

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Date of Issuance: April 5, 2002

The 2002 edition of this Standard is being issued with an automatic addenda subscription service. The use of addenda allows revisions made in response to public review comments or committee actions to be published as necessary; revisions published in addenda will become effective 6 months after the Date of Issuance of the addenda. The next edition of this Standard is scheduled for publication in 2005.

ASME issues written replies to inquiries concerning interpretations of technical aspects of this Standard. The interpretations will be included with the above addenda service.

ASME is the registered trademark of The American Society of Mechanical Engineers.

This code or standard was developed under procedures accredited as meeting the criteria for American National Standards. The Standards Committee that approved the code or standard was balanced to assure that individuals from competent and concerned interests have had an opportunity to participate. The proposed code or standard was made available for public review and comment that provides an opportunity for additional public input from industry, academia, regulatory agencies, and the public-at-large.

ASME does not "approve," "rate," or "endorse" any item, construction, proprietary device, or activity.

ASME does not take any position with respect to the validity of any patent rights asserted in connection with any items mentioned in this document, and does not undertake to insure anyone utilizing a standard against liability for infringement of any applicable letters patent, nor assume any such liability. Users of a code or standard are expressly advised that determination of the validity of any such patent rights, and the risk of infringement of such rights, is entirely their own responsibility.

Participation by federal agency representative(s) or person(s) affiliated with industry is not to be interpreted as government or industry endorsement of this code or standard.

ASME accepts responsibility for only those interpretations of this document issued in accordance with the established ASME procedures and policies, which precludes the issuance of interpretations by individuals.

No part of this document may be reproduced in any form,
in an electronic retrieval system or otherwise,
without the prior written permission of the publisher.

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

Copyright © 2002 by
THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS
All Rights Reserved
Printed in U.S.A.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

CONTENTS

(A detailed Contents precedes each Section.)

Foreword iv
Committee Roster v
Preparation of Technical Inquiries to the Committee on Nuclear Risk Management vi
Preface vii

Sections

1 Introduction 1
2 Acronyms and Definitions 5
3 Risk Assessment Application Process 12
4 Risk Assessment Technical Requirements 19
5 PRA Configuration Control 87
6 Peer Review 90

References 95

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

FOREWORD

The ASME Board on Nuclear Codes and Standards (BNCS) began considering the development of a consensus Standard for the use of Probabilistic Risk Assessment (PRA) in risk-informed decision making in the summer of 1997. Newly published ASME Code Cases for risk-informed applications provided an impetus for a PRA Standard on the technical capability of a PRA necessary to support risk-informed changes to nuclear power plant design and operations.

The BNCS and the ASME Council on Codes and Standards evaluated this consideration with regard to ASME safety criteria and activities associated with risk-informed applications. Given the advancements in developing risk-informed Code Cases issued by the Boiler and Pressure Vessel Committee and the Operations and Maintenance Committee, it was determined that a need exists for a Standard to address the PRA capability necessary to support ASME applications of this emerging technology. After approval by the ASME Council on Codes and Standards, an ASME Project Team and a Standards Committee were formed in early 1998 to develop a PRA Standard that would provide a foundation for existing and future risk-informed applications for nuclear power plants. The Committee and Project Team charged with drafting the standard received strong support from the U.S. Nuclear Regulatory Commission (NRC) and industry, and maintains liaison with the American Nuclear Society (ANS), National Fire Protection Association (NFPA), and Institute of Electrical and Electronic Engineers (IEEE) nuclear standards developing groups.

The Project Team (i.e., the writing group) comprised key individuals with the direct knowledge and experience to produce a technically adequate document in a timely manner under the ASME Codes and Standards Redesign Process. A unique part of this process was the review of two drafts of the Standard by experts inside and outside the ASME Committee structure prior to the formal public review and approval process. Comments provided by these reviewers have been addressed by the Project Team and incorporated within the Standard where they were considered to be appropriate.

The U.S. nuclear industry has developed a Peer Review process for assessing the technical capability and adequacy of a PRA to support risk-informed regulatory licensing applications (NEI 00-02). Peer Reviews have been conducted on most U.S. nuclear power plants. The guidelines of NEI 00-02 have been considered in the development of this PRA Standard.

Upon completion of all reviews, the draft Standard was approved by the consensus technical standards committee, the Committee on Nuclear Risk Management (CNRM).

CNRM operates under procedures accredited by the American National Standards Institute (ANSI) as meeting the criteria of consensus procedures for American National Standards. This Standard was approved by the ASME Board on Nuclear Codes and Standards and subsequently approved by ANSI on January 31, 2002.

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

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

COMMITTEE ON NUCLEAR RISK MANAGEMENT

As of June 11, 2001

MAIN COMMITTEE

S. A. Bernsen, *Chair*
G. L. Zigler, *Vice Chair*
G. M. Eisenberg, *Secretary*
R. E. Bradley
H. D. Brewer
R. J. Budnitz
M. A. Cunningham
K. N. Fleming
H. A. Hackerott
R. E. Hall
R. A. Hill
T. G. Hook
S. H. Levinson
T. A. Meyer
W. J. Parkinson
L. Sage
R. L. Simard
F. A. Simonen
J. J. Zudans

PROJECT TEAM

R. L. Simard, *Chair*
G. M. Eisenberg, *Secretary*
H. D. Brewer
R. J. Budnitz
M. Drouin
K. N. Fleming
C. R. Grantom
H. A. Hackerott
R. A. Hill
G. A. Krueger
S. H. Levinson
B. W. Logan
B. B. Mrowca
W. J. Parkinson
F. J. Rahn
R. E. Schneider
B. D. Sloane
I. B. Wall
R. A. West

Other individuals and organizations providing substantial support and input to the standard include:

D. M. Bucheit
E. T. Burns
A. Camp
J. Chapman
M. C. Cheok
W. Hannaman
S. Kojima
J. L. LaChance
J. R. Lehner
G. W. Parry
W. T. Pratt
N. O. Siu
D. E. True
D. W. Whitehead
American Nuclear Society
B&W Owners' Group
BWR Owners' Group
CE Owners' Group
Electric Power Research Institute
Westinghouse Owners' Group

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

PREPARATION OF TECHNICAL INQUIRIES TO THE COMMITTEE ON NUCLEAR RISK MANAGEMENT

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, Three Park Avenue, New York, NY 10016-5990.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

PREFACE

ORGANIZATION OF THIS STANDARD

This Standard is organized into Sections as follows:

1 Introduction
2 Acronyms and Definitions
3 Risk Assessment Application Process
4 Risk Assessment Technical Requirements
5 PRA Configuration Control
6 Peer Review

Each Section is subdivided into paragraphs, subparagraphs, and breakdowns, as in the following example:

paragraph: 3.1

subaragraph: 3.1.2

When paragraphs are used to identify sequential requirements they will be identified by adding a lowercase letter such as (*a*), (*b*), (*c*), etc.

Tables and figures provided in this Standard are identified by the applicable paragraph or subparagraph number for which they apply, with either "Table" or "Fig." and labeled sequentially as follows: 3.2.1-1, 3.2.1-2, etc. Each table or figure is placed near the text that applies to its use.

References are identified sequentially within the text as footnotes. Within tables, references are identified sequentially as Note (1), Note (2), etc. A complete listing of the references, including publishers' information, is located in the References section.

When required by context in this Standard, the singular shall be interpreted as the plural, and vice versa. The feminine, masculine, or neutral gender shall be treated as such other gender as appropriate.

DESCRIPTION OF SECTIONS IN THIS STANDARD

The following descriptions of the individual Sections in this Standard are intended to provide the reader with general information on the scope of coverage and the rationale applied in their development.

1 INTRODUCTION

This Section summarizes the scope, applicability, and contents of the Standard.

2 ACRONYMS AND DEFINITIONS

This Section identifies and describes unique terms, abbreviations, and acronyms that are used in this Standard.

3 RISK ASSESSMENT APPLICATION PROCESS

This Section describes a process for determining the capability of a PRA to support specific risk-informed applications.

4 RISK ASSESSMENT TECHNICAL REQUIREMENTS

This Section contains objectives, high level requirements (HLRs), and supporting requirements (SRs) for a PRA to be used in support of risk-informed decision-making within the scope of this Standard.

5 PRA CONFIGURATION CONTROL

This Section provides requirements for maintaining and updating a PRA to be used in support of risk-informed decision-making within the scope of this Standard.

6 PEER REVIEW

This Section provides the requirements for a peer review of a PRA to be used in support of risk-informed decision making within the scope of this Standard.

PROPOSED SECTION EXPANSIONS

In addition to the criteria provided in this Standard, consideration will be given in the future to expanding the Standard to other risk assessment methodologies beyond a Level 1 analysis of internal events (excluding fires) while at power and the limited Level 2 (LERF) analysis provided.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

USER RESPONSIBILITY

Users of this Standard are cautioned that they are responsible for all technical assumptions inherent in the use of PRA models, computer programs, and analysis performed to meet the requirements of this Standard.

CORRESPONDENCE

Suggestions for improvements to this Standard or inclusion of additional topics shall be sent to the following address: Secretary, Committee on Nuclear Risk Management, The American Society of Mechanical Engineers, Three Park Avenue, New York, NY 10016-5990.

ADDENDA SERVICE

This edition of ASME RA-S–2002 includes an automatic addenda subscription service with no set schedule up to the publication of the next edition. The addenda subscription service includes approved new Sections, revisions to existing Sections, and issued interpretations. The interpretations included as part of the addenda service are not part of this Standard.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 1

1.1 Scope 2
1.2 Applicability 2
1.3 PRA Capability Categories 2
1.4 Requirements for PRA Elements 2
1.5 Risk Assessment Application Process 4
1.6 PRA Configuration Control 4
1.7 Peer Review Requirements 4

Table

1.3-1 Bases for PRA Capability Categories 3

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 1
INTRODUCTION

1.1 Scope

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

1.2 Applicability

This Standard applies to PRAs used to support applications of risk-informed decision making related to design, licensing, procurement, construction, operation, and maintenance. This Standard establishes requirements for a Level 1 analysis of internal events while at power. In addition, this Standard establishes requirements for a limited Level 2 analysis sufficient to evaluate the large early release frequency (LERF) for internal events while at power.

1.3 PRA Capability Categories

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 of the factors that determine the risk significance of the proposed changes. To determine the needed capabilities, the application is evaluated by considering the above attributes, as amplified in para. 3.2.2. For example, a proposed change to the maintenance practice for a given component may be evaluated in a particular application. The capabilities of the PRA for those accident sequences involving the failure or unavailability of the component are relevant to such an application, whereas the capabilities of the PRA for other sequences are not as relevant since there are no changes that need to be resolved for these parts.

The required category of PRA capabilities may vary over different elements of the PRA, within a given element, across different accident sequences or classes of accident sequences, initiating events, basic events, and end states, depending on the application. Although the range of capabilities required for each part of the PRA to support an application falls on a continuum, three Capability Categories are defined in this Standard so that requirements can be developed and presented in a manageable way. They are designated as PRA Capability Categories I, II, and III. Table 1.3-1 describes the attributes of a PRA that were used to develop Section 4 requirements for each of these Capability Categories.

This Standard includes High Level Requirements (HLRs) for each element that are the same for all applications, and Supporting Requirements (SRs) that are specified for each element that are differentiated by Capability Category. This differentiation facilitates the determination of the appropriate requirements for each part of a PRA that is necessary to support a given application.

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 parts 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 parts of the PRA within the elements that fail to meet the SRs for any of these Capability Categories.) While all parts of the PRA need not have the same capability, the PRA model should be coherent. The SRs have been written so that, within a Category, the interfaces between parts 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 part of the PRA, and hence which SRs apply to the application. (See Section 3.)

1.4 Requirements for PRA Elements

1.4.1 PRA Elements. The requirements of this Standard are organized by nine PRA Elements that comprise an internal-events, at-power, Level-1 and

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Table 1.3-1 Bases for PRA Capability Categories

Criteria	Capability Category I	Capability Category II	Capability Category 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 including associated human actions.	Resolution and specificity sufficient to identify the relative importance of the dominant contributors at the component level including associated human actions, as necessary [Note (1)].	Resolution and specificity sufficient to identify the relative importance of the contributors at the component level including associated human actions, as necessary [Note (1)].
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 dominant 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 (2)].	Departures from realism will have small impact on the conclusions and risk insights as supported by good practices [Note (2)].	Departures from realism will have negligible impact on the conclusions and risk insights as supported by good practices [Note (2)].

NOTES:

(1) 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.

(2) 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.

Not for Resale

Level-2/LERF PRA. They and their abbreviations are as follows:

(*a*) Initiating Events 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*) Internal Flooding (IF)
(*h*) Quantification (QU)
(*i*) LERF Analysis (LE)

1.4.2 High Level Requirements. A set of Objectives and HLRs are provided for each PRA Element in Section 4. All PRAs using this Standard shall satisfy each of these HLRs as explained in para. 1.3. The HLRs set forth the minimum requirements for meeting this Standard in general terms and present the top level logic for the derivation of more detailed SRs for each of the PRA Capability Categories. 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.

1.4.3 Supporting Requirements. The SRs for each of the nine PRA Elements are presented in Section 4 as action statements, using the three Capability Categories described in para. 1.3. 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 statement extends to more than one category, it applies equally to each Capability Category, but the scope of applicability will be commensurate with the Capability Category criteria in Table 1.3-1 and the scope and level of detail required by other associated SRs. It is intended that, by meeting all the SRs under a given HLR, a PRA will meet that HLR. Section 4 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 to the requirements of this 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 6.

1.5 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 3 describes 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 para. 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 Section 4. Requirements for supplementary analysis are outside the scope of this Standard.

1.6 PRA Configuration Control

Section 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.

1.7 Peer Review Requirements

Section 6 provides the requirements for a peer review to determine if the PRA methodology and its implementation meet the requirements of Section 4 of this Standard.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

SECTION 2

2.1 Acronyms 6
2.2 Definitions 6

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 2
ACRONYMS AND DEFINITIONS

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

2.1 Acronyms

ADS: Automatic Depressurization System

AOT: Allowed Outage Time

ARI: Alternate Rod Insertion

ASEP: Accident Sequence Evaluation Program

ATWS: Anticipated Transient Without Scram

BWR: Boiling Water Reactor

CCW: Component Cooling Water

ECCS: Emergency Core Cooling System

EDG: Emergency Diesel Generator

EOPs/AOPs: Emergency Operating Procedures/Abnormal Operating Procedures

HFE: Human Failure Event

HLR: High Level Requirement

HPCI: High Pressure Coolant Injection

HVAC: Heating, Ventilation, and Air Conditioning

ISLOCA: Interfacing Systems Loss of Coolant Accident

LOCA: Loss of Coolant Accident

LOOP: Loss of Offsite Power

MOV: Motor Operated Valve

NPSH: Net Positive Suction Head

NRC: Nuclear Regulatory Commission

NSSS: Nuclear Steam Supply System

P&IDs: Piping and Instrumentation Drawings (or Diagrams)

PDS: Plant Damage State

PWR: Pressurized Water Reactor

RCIC: Reactor Core Isolation Cooling

RCP: Reactor Coolant Pump

RCS: Reactor Coolant System

RPT: Reactor Pump Trip

RPV: Reactor Pressure Vessel

RWST: Refueling Water Storage Tank

SAR: Safety Analysis Report

SBO: Station Blackout

SGTR: Steam Generator Tube Rupture

SLCS: Standby Liquid Control System

SORV: Stuck Open Relief Valve

SR: Supporting Requirements

SSCs: Structures, Systems, and Components

SW: Service Water

THERP: Technique for Human Error Rate Prediction (see NUREG/CR-1278[1])

TS: Technical Specifications

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 combination of system, function, and operator failures or successes, of an accident that can lead to undesired consequences, with a specified end state (e.g., core damage or large early

[1] NUREG/CR-1278, Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications; A. D. Swain and H. E. Guttmann; August 1983

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

release). An accident sequence may contain many unique variations of events (minimal cut sets) that are similar.

accident sequence, dominant: an accident sequence that is usually represented by the top 10 or 20 events or groups of events modeled in a PRA and accounts for a large fraction of the core damage or large early release frequency.

accident sequence, modeled: an accident sequence contained in the PRA above the model truncation level.

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.

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.

best estimate: the point estimate of a parameter that is not biased by conservatism or optimism. Generally, the best estimate of a parameter is represented as a mean value.

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.

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

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

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 is anticipated and involving enough of the core to cause a significant release.

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

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.

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

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.

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

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

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

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

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

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

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

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).

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).

front-line system: a system (safety or non-safety) 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 non-minimal 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.

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

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.

initiating event: any event either internal or external to the plant that perturbs the steady state operation of the plant, if operating, thereby initiating an abnormal event such as transient or LOCA within the plant. Initiating events trigger sequences of events that challenge plant control and safety systems whose failure could potentially lead to core damage or large early release.

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.

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 over-pressurization of a low-pressure system when subjected to RCS pressure and can result in containment bypass.

internal event: an event originating within a nuclear power plant that, in combination with safety system failures and/or operator errors, can affect the operability

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

of plant systems and may lead to core damage or large early release. By convention, loss of offsite power is considered to be an internal event, and internal fire is considered to be an external event.

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, core heat removal, reactor coolant inventory control, reactor coolant 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.

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.

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.

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

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.

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

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

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

post-initiator human failure events: human failure events that represent the impact of human errors committed during actions performed in response to an accident initiator.

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. 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).

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

recovery: a general term describing restoration and repair acts required to change the initial or current state of a system or component into a position or condition needed to accomplish a desired function for a given plant state.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

recovery action: a human action performed to regain equipment or system operability from a specific failure or human error to mitigate or reduce the consequences of the failure.

reliability: the complement of unreliability.

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 importance to a PRA.

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

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

risk: probability and consequences of an event, as expressed by the "risk triplet" that is the answer to the following three questions: (1) What can go wrong? (2) How likely is it? and (3) 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.

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

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

screening analysis: an analysis that eliminates items from further consideration based on their negligible contribution to the probability of a significant 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.

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.

shall: used to state a mandatory requirement.

should: used to state a recommendation.

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.

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

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.

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: termination of the ability of a system to perform any one of its critical design functions.

NOTE: Failure of a line/train within a system may occur in such a way that the system retains its ability to perform all its required functions; in this case, the system has not failed.

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.

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

unavailability: the fraction of time 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,

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

and evaluating their impact on the PRA results and developing a quantitative measure to the extent practical.

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

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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 3

3.1 Purpose 13
3.2 Identification of Application and Determination of Capability Categories (Stage A) 13
3.3 Assessment of PRA for Necessary Scope, Results, and Models (Stage B) 15
3.4 Determination of the Standard's Scope and Level of Detail (Stage C) 16
3.5 Comparison of PRA Model to Standard (Stage D) 16
3.6 Use of Supplementary Analyses/Requirements (Stage E) 16

Figure

3.1-1 Application Process Flowchart 17

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

SECTION 3
RISK ASSESSMENT APPLICATION PROCESS

3.1 Purpose

This Section describes required activities to determine the capability of a PRA needed to support a particular risk-informed application. For a specific application, PRA capabilities are evaluated to determine the appropriate Supporting Requirements (SRs) rather than by specifying a single Capability Category for the whole PRA. Depending on the application, the required category of PRA capabilities may vary over different elements of the PRA, within a given element, across different accident sequences or classes of accident sequences, initiating events, basic events, and end states. The process is intended to be used with PRAs that have had a peer review that meets the requirements of Section 6 of this Standard.

Figure 3.1-1 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 parts of a PRA affected by the plant change are determined, and the PRA scope 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 parts of a PRA model that are particularly sensitive to the proposed change, the Capability Categories for each part of the PRA necessary to support the application are determined. Different parts 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 parts of a PRA may be irrelevant to the application.

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

(c) Stage C. An evaluation is performed to determine whether the SRs from the Standard for each part of the PRA and its identified Capability Category 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 part 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 PRA has adequate capability, needs upgrading to meet the appropriate set of SRs, or needs supplementary analyses as described in Stage E.

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

It is noted that the scope of the activities in Fig. 3.1-1 determines how to evaluate the role of the PRA in the application and how to determine which Capability Categories are needed for each part 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 parts of the PRA used in the application meet the HLRs 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.

3.2 Identification of Application and Determination of Capability Categories (Stage A)

3.2.1 Identification of Application. Define the application by

(a) evaluating the plant design or operational change being assessed (Box 1 of Fig. 3.1-1)

(b) identifying the SSCs and plant activities affected by the change including the cause and effect relationship between the plant design or operational change and the PRA model (Box 2 of Fig. 3.1-1)

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

(c) identifying the PRA scope and PRA risk metrics that are needed to assess the change (Box 3 of Fig. 3.1-1)

EPRI TR-105396[1] and NUREG-0800[2] provide guidance for the above activities.

Example: 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 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.

3.2.2 Determination of Capability Categories. Section 4 of this Standard sets forth SRs for three PRA Capability Categories whose attributes are described in para. 1.3. For the application, determine the Capability Category for each part of the PRA needed to support the application (Box 4 of Fig. 3.1-1). This determination dictates which SRs are used to evaluate the capabilities of each part 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. 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) required scope/level of detail of the PRA models for each part of the PRA and the PRA results relative to the needs of the specified application

(d) degree of accuracy and evaluation of uncertainties and sensitivities required of the PRA results

(e) degree of confidence in the results that is required to support the decision

(f) 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 a baseline 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 parts 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 parts of the PRA needed to determine CDF are determined to require only PRA Capability

[1] EPRI TR-105396, PSA Applications Guide; D. True, et al.; August 1995

[2] NUREG-0800, Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decision-Making: General Guidance, *NRC Standard Review Plan,* Chapter 19, 1998

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Category I. Hence, the initiating events, accident sequences, data parameters, system models, human actions, and quantification process for those sequences and cutsets impacted by the AOT changes are in PRA Capability Category II, and the remaining parts 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 a baseline core damage frequency that was greater than $1x10^{-4}$ or baseline LERF greater than $1x10^{-5}$, 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 parts 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.

3.3 Assessment of PRA for Necessary Scope, Results, and Models (Stage B)

3.3.1 Necessary Scope and Results. Determine if the PRA provides the results needed to assess the plant or operational change (Box 5 of Fig. 3.1-1). If some aspects of the PRA are insufficient to assess the change, then upgrade them in accordance with the SRs of Section 4 for its corresponding Capability Category (Box 6a of Fig. 3.1-1), or generate supplementary analyses. (See para. 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 5.

Example: The proposed change in the SW AOT has been determined to affect the SW unavailability. 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, the scope of the initiating event analysis element of the PRA must include: (1) LOCA initiators, since the change in SW unavailability will affect ECCS pump cooling in the recirculation phase, (2) loss of offsite power initiators, since the SW change will affect the diesel generators, and (3) 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. It is determined that the changes in maintenance unavailability are too small to consider significant 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.

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. 3.1-1). If the affected SSCs or plant activities are not modeled, then either upgrade the PRA to include the SSCs in accordance with the SRs of Section 4 for their corresponding Capability Category (Box 6a of Fig. 3.1-1), or generate supplementary analyses. (See para. 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 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 that contribute to the change in CDF (i.e., ECCS, DGs, feedwater, and CCW) need to be modeled in the PRA. For example, if, as is likely, the loss of feedwater

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

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 the standard.

3.3.3 Peer Review. The parts of a PRA that are needed for an application shall have been reviewed pursuant to the requirements of Section 6.

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 Section 4, for the corresponding Capability Categories determined in para. 3.2.2, are sufficient to assess the application under consideration (Box 8 of Fig. 3.1-1).

If it is determined that the standard lacks specific requirements, their significance to the application shall be assessed (Box 9 of Fig. 3.1-1). If the absent requirements are not significant, 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 significant, supplementary requirements may be used (Box 7 of Fig. 3.1-1).

3.5 Comparison of PRA Model to Standard (Stage D)

Determine if each part of the PRA satisfies the SRs at the appropriate Capability Category needed to support the application (Box 10 of Fig. 3.1-1). The results of the Peer Review (Section 6) 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. 3.1-1). The bases for this determination shall be documented.

If the PRA does not satisfy a SR for the appropriate Capability Category, then determine if the difference is significant (Box 12 of Fig. 3.1-1). Acceptable requirements for determining the significance of this difference include the following:

(a) The difference is not applicable or does not affect quantification relative to the impact of the proposed application, or

(b) Modeled accident sequences accounting for at least 90% of CDF/LERF, 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 Section 4 as applied to the application.

Determination of significance will depend on the particular application being considered and may involve determinations made by an expert panel.

If the difference is not significant, then the PRA is acceptable for the application. If the difference is significant, then either upgrade the PRA to address the corresponding SRs stated in Section 4 (Box 6b of Fig. 3.1-1), or generate supplementary analyses. (See para. 3.6.) Any upgrade of the PRA shall be done and documented in accordance with Section 5. The examples provided under para. 3.3 are applicable.

3.6 Use of Supplementary Analyses/ Requirements (Stage E)

If the scope of either the PRA or the standard is insufficient, supplementary analyses or requirements may be used (Box 7 of Fig. 3.1-1). 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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

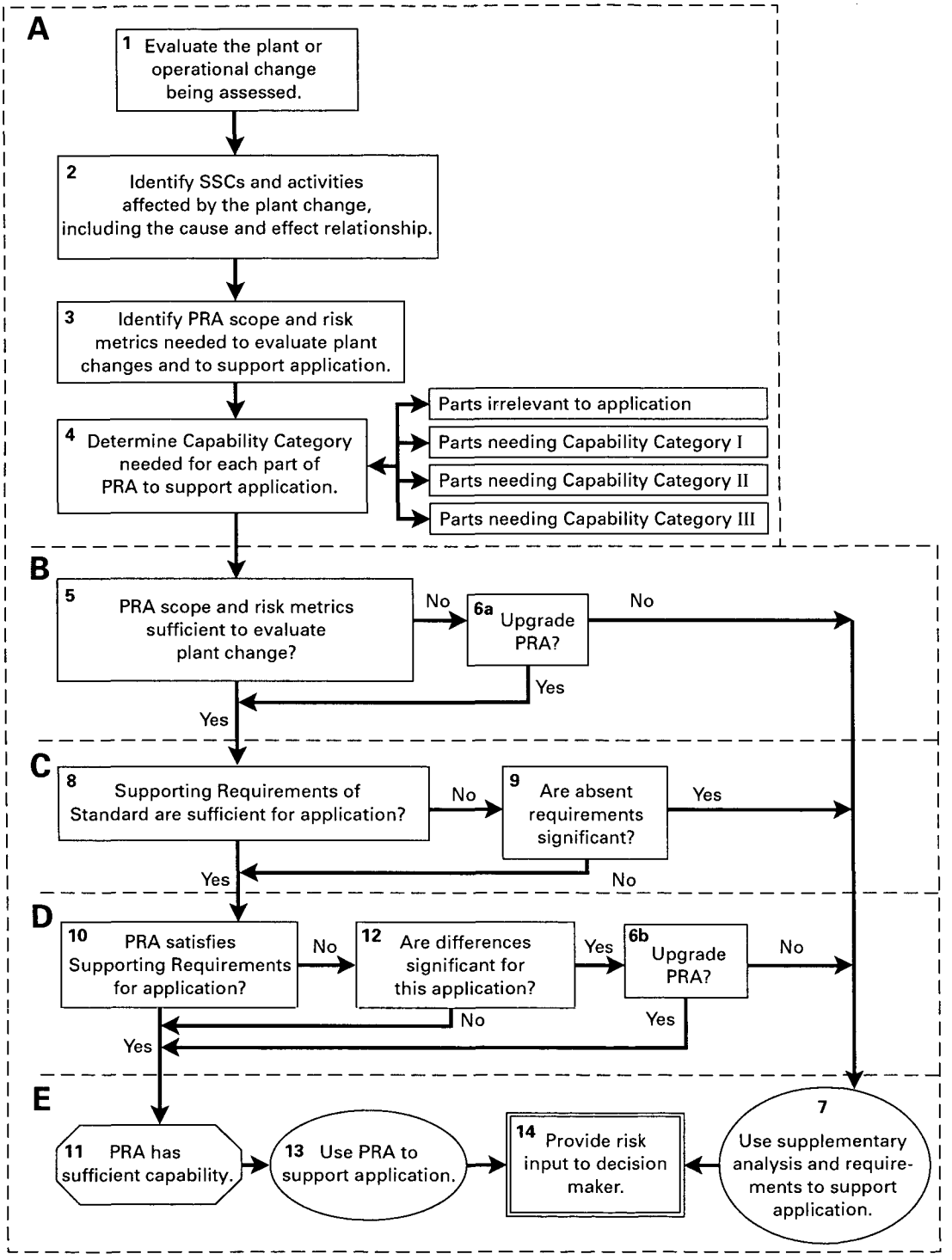


Fig. 3.1-1 Application Process Flowchart

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

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.

Example of supplementary requirements: 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.

Second example of supplementary requirements: It is desired to rank the snubbers in a plant according to their risk significance for the purpose of 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,[3] 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.

Third example of supplementary requirements: 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. The issue of seismic performance of these valves is beyond the scope of this Standard. To address it, supplementary requirements, found in *Nuclear Engineering and Design*,[4] 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.

If it has been determined that the PRA has sufficient capability, its results can be used to support the application (Box 13 of Fig. 3.1-1). 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. 3.1-1). Such supplementary analyses/requirements are outside the scope of this Standard.

---

[3] ASME OM Code for Operation and Maintenance of Nuclear Power Plants, Code Case OMN-10: Requirements for Safety Significance Categorization of Snubbers Using Risk Insights and Testing Strategies for Inservice Testing of LWR Power Plants

[4] Kennedy, R. P. and M. K. Ravindra, "Seismic Fragilities for Nuclear Power Plant Risk Studies," *Nuclear Engineering and Design*, Vol. 79, No. 1, pp. 47–68 (1984)

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

SECTION 4

4.1 Purpose 20
4.2 Process Check 20
4.3 Use of Expert Judgment 20
4.4 Derivation of PRA Requirements 21
4.5 PRA Requirements 21

Objectives and Tables

4.5.1 Initiating Events Analysis (IE) 22
4.5.2 Accident Sequence Analysis (AS) 30
4.5.3 Success Criteria (SC) 34
4.5.4 Systems Analysis (SY) 39
4.5.5 Human Reliability Analysis (HR) 47
4.5.6 Data Analysis (DA) 55
4.5.7 Internal Flooding (IF) 63
4.5.8 Quantification (QU) 70
4.5.9 LERF Analysis (LE) 76

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 4
RISK ASSESSMENT TECHNICAL REQUIREMENTS

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.

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.

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.

NUREG/CR-6372[1] and NUREG/CR-1563[2] 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 include Pacific Gas and Electric's Diablo Canyon seismic study[3] and the Yucca Mountain project's study of volcanic hazards.[4] 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-1563.

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).

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).

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

[1] 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

[2] 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

[3] NRC 50-275, 50-323, Final Report of the Diablo Canyon Long-Term Seismic Program, Pacific Gas and Electric Company

[4] BA000-1717-2200-00082, Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada, U.S. Department of Energy Yucca Mountain Project, Geomatrix Consultants, Inc., 1996

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

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

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.

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

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; for example, 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 importance to the issue

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.

4.4 Derivation of PRA Requirements

Objectives were established for each of the nine elements used to characterize a PRA. The objectives reflect substantial experience accumulated with PRA development and usage, and are consistent with the PRA Procedures Guide[5] and the NEI-00-02 Peer Review Process Guidance.[6] These objectives form the basis for development of the high level requirements (HLRs) for each element that were used, in turn, to define the supporting requirements (SRs).

In setting the HLRs for each element, the goal was to derive, based on the objectives, an irreducible set of firm requirements, applicable to PRAs that support all levels of application, to guide the development of SRs. This goal reflects the diversity of approaches that have been used to develop existing PRAs and the need to allow for technological innovations in the future. An additional goal was to derive a reasonably small set of HLRs that capture all the important technical issues that were identified in the efforts to develop this Standard and to implement the NEI-00-02 PRA Peer Review process guidance.

The HLRs generally address attributes of the PRA element such as

(a) scope and level of detail

(b) model fidelity and realism

(c) output or quantitative results (if applicable)

(d) documentation

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

4.5 PRA Requirements

Tables of HLRs and SRs for the nine PRA elements are provided in paras. 4.5.1 through 4.5.9. 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 statement extends to more than one category, it applies equally to each Capability Category, but the scope of applicability will be commensurate with the Capability Category criteria in Table 1.3-1 and the scope and level of detail required by other associated SRs. It is intended that, by meeting all the SRs under a given HLR, a PRA will meet that HLR.

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

[5] NUREG/CR-2300, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, 1983

[6] NEI-00-02, Probabilistic Risk Assessment Peer Review Process Guidance, 2000

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

4.5.1 Initiating Event Analysis (IE)

4.5.1.1 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 4.5.1-1 High Level Requirements for 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	The initiating event analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed to select, group, and screen the initiating event list and to model and quantify the initiating event frequencies, with assumptions and bases stated.

Table 4.5.1-2(a) Supporting Requirements for HLR-IE-A

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

Index No. IE-A	Capability Category I	Capability Category II	Capability Category III
IE-A1	USE a structured, systematic process for identifying initiating events. 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.1-2(a) 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-A2	IDENTIFY those initiating events and event categories that challenge normal plant operation and that require successful mitigation to prevent core damage. In the identification, ACCOUNT FOR the potential for plant-specific features to influence initiating events. INCLUDE in the spectrum of internal-event challenges at least the following general categories, and within each general category. INCORPORATE each initiating event category in the model quantitatively in terms of its relative frequency. In the categorization, SEPARATE into different categories based on whether events have different impacts on plant performance, safety functions, and possibilities for recovery. The following list is not intended to be all-inclusive: *(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. DIFFERENTIATE the LOCA initiators into at least the following categories, using a defined rationale for the differentiation: *(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: pipe breaks outside containment *(c)* ISLOCAs: INCLUDE postulated events representing active components 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 [e.g., interfacing systems LOCAs (ISLOCAs)]. *(d)* Special initiators (e.g., support systems failures, instrument line breaks) [Note (1)]. *(e)* Internal flooding initiators (see IF-D1 and D2) [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. REVIEW experience and analyses at similar plants to assess whether the list of challenges included in the model accounts for industry experience.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.1-2(a) 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-A4	PERFORM a systematic evaluation of each system to assess the possibility of an initiating event occurring due to a failure of the system.	PERFORM a systematic evaluation of each system to assess the possibility of an initiating event occurring due to a failure of the system.	PERFORM a systematic evaluation of each system 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 risk-significant system initiating events.	**USE a structured approach (such as a system-by-system review of initiating event potential, or an FMEA [failure modes and effects analysis] or fault tree) to assess and document the possibility of an initiating event resulting from system failures.**	**DEVELOP a detailed model of system interfaces including fault tree development. PERFORM an FMEA (failure modes and effects analysis) to assess and document the possibility of an initiating event resulting from individual systems or train failures.**
	INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause.	INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause.	INCLUDE initiating events resulting from multiple failures, if the equipment failures result from a common cause.
IE-A5	In the identification of the initiating events, INCORPORATE *(a)* events that have occurred at conditions other than at-power operation (i.e., during low-power or shutdown conditions) *(b)* events resulting in a controlled shutdown that include a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to at-power operation.		
IE-A6	No requirement for interviews.	INTERVIEW plant operations, maintenance, engineering, and safety analysis personnel to determine if potential initiating events have been overlooked. **Information from interviews conducted at similar plants may be used.**	INTERVIEW plant operations, maintenance, engineering, and safety analysis personnel to determine if potential initiating events have been overlooked.
IE-A7	No requirement for precursor review.	In searching for initiating events, ACCOUNT FOR initiating event precursors, to help identify initiating events and to provide a partial basis for quantifying their frequencies.	
IE-A8	No requirement to account for the system alignments cited.	In searching for initiating events, ACCOUNT FOR each system and supporting system alignment, such as temporary alignments during maintenance, that could either influence the likelihood that failures cause an initiating event, or could increase the severity of the effect on plant safety functions that would result from such an event.	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.1-2(a) 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 to include support system failures in a realistic fashion, nor to treat explicitly the cited support systems.	INCLUDE support system failures as initiating events quantitatively in the PRA in a realistic fashion. TREAT EXPLICITLY the individual support systems (or trains) that can cause a plant trip.	
IE-A10	INCLUDE those multi-unit site initiators such as dual unit LOOP events or total loss of service water that may impact the model at multi-unit sites with shared systems.		

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

Table 4.5.1-2(b) 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	COMBINE initiating events into groups to facilitate definition of accident sequences in the Accident Sequence Analysis element (para. 4.5.2) and to facilitate quantification in the Quantification element (para. 4.5.8). Functional initiating event categories refer to initiating events grouped for the purpose of accident sequence definition, while quantification initiating event categories refer to those grouped for separate quantification of the accident sequences. When initiating events are not grouped for either of these purposes, PROVIDE a separate accident-sequence evaluation for each selected initiating event.		
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).		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.1-2(b) Supporting Requirements for HLR-IE-B (Cont'd)

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-B3	GROUP initiating events only when the following can be assured: *(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 subsumed into a group and bounded by the worst case impacts within the "new" group.		GROUP initiating events only when the following can be assured: *(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 subsumed into a group and bounded by the worst case impacts within the "new" group. **To avoid conservatism, DO NOT ADD initiating events to a group and DO NOT SUBSUME events into a group unless the impacts are comparable to or less than those of the remaining events in that group, or it is demonstrated that such grouping does not appreciably impact CDF or LERF.**
IE-B4	GROUP separately from other initiating event categories those categories with significantly different plant response 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.		

Table 4.5.1-2(c) Suppporting 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).

Index No. IE-C	Capability Category I	Capability Category II	Capability Category III
IE-C1	CALCULATE the initiating event frequency from plant-specific data, if sufficient data are available. Otherwise, USE generic data (see IE-C2). USE the most recent applicable data to quantify the initiating event frequencies. CREDIT recovery actions as appropriate; JUSTIFY each such credit. Data from the initial year of commercial operation may be excluded; if excluded, JUSTIFY.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.1-2(c) Suppporting 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-C2	If necessary because insufficient plant-specific data are available (see IE-C1), USE generic industry data in the quantification of initiating event frequencies.	If necessary because insufficient plant-specific data are available (see IE-C1), USE **a Bayesian update process of** generic industry data in the quantification of initiating event frequencies.	
IE-C3	CALCULATE initiating event frequencies on a reactor-year basis. Specifically, for sequences initiated at power, ACCOUNT in the initiating event analysis for the plant availability, such that the frequencies are weighted by the fraction of time the plant is at-power. ACCOUNT FOR differences between historical plant availability over the period of event occurrences in the plant database and present or expected future plant availability that could be different from historical values.		
IE-C4	USE as screening criteria no higher than the following characteristics (or more stringent characteristics as devised by the analyst) to eliminate initiating events from further evaluation: *(a)* the frequency of the event is less than 1E-7 per reactor-year (/ry) 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/ry 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, then CONFIRM that the value specified in the criterion meets the requirements in the Data Analysis section (para. 4.5.6) and the Level 1 Quantification section (para. 4.5.8).		
IE-C5	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 exclusion of earlier years that are not representative of current data. One acceptable methodology for time-trend analysis is found in NUREG/CR-5750 [Note (1)].
IE-C6	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-specific design features. When the fault-tree approach is used, USE the appropriate systems-analysis requirements for fault-tree modeling found in the Systems Analysis section (para. 4.5.4).		
IE-C7	When using fault tree models for initiating events, QUANTIFY 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 the Systems Analysis section, para. 4.5.4.). Thus, 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. APPLY the relevant requirements in the Data Analysis section, para. 4.5.6, for the data used in the fault-tree quantification.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.1-2(c) Suppporting 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-C8	If fault-tree modeling is used, CAPTURE 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.		
IE-C9	No requirement to use plant-specific information in the fault-tree modeling.	If fault-tree modeling is used, USE plant-specific information in the assessment and quantification of recovery actions where available. See Human Reliability Analysis (para. 4.5.5) for further guidance.	
IE-C10	COMPARE and RESOLVE the results of the initiating event analysis with generic data sources to provide a reasonableness check of the quantitative and qualitative results.		
IE-C11	For rare initiating events, USE industry generic data and **ACCOUNT for plant-specific functions.** For extremely rare initiating events, engineering judgment may be used; if used, AUGMENT with applicable generic data sources. Refer to para. 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 para. 4.3, Use of Expert Judgment, as appropriate.
	For purposes of this Requirement, a "rare event" might be expected to occur one or a few times throughout the world nuclear industry over many years. An "extremely rare event" would not be expected to occur even once throughout the industry over many years.		For purposes of this Requirement, a "rare event" might be expected to occur one or a few times throughout the world nuclear industry over many years. An "extremely rare event" would not be expected to occur even once throughout the industry over many years. **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**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.1-2(c) Suppporting 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-C12	In the ISLOCA frequency analysis, INCLUDE features of plant and procedures that could significantly influence the ISLOCA frequency.		In the ISLOCA frequency analysis, INCLUDE features of the plant and procedures that could significantly influence the ISLOCA frequency: *(a)* **EVALUATE surveillance procedure steps** *(b)* **INCLUDE surveillance test intervals explicitly** *(c)* **ASSESS on-line surveillance testing quantitatively** *(d)* **QUANTIFY pipe rupture probability** *(e)* **ADDRESS explicitly valve design (e.g., air operated testable check valves)** *(f)* **INCLUDE quantitatively the valve isolation capability given the high-to-low-pressure differential.**

NOTE:
(1) NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants, Idaho National Engineering and Environmental Laboratory, Idaho Falls, February 1999

Table 4.5.1-2(d) Supporting Requirements for HLR-IE-D

The initiating event analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed to select, group, and screen the initiating event list and to model and quantify the initiating event frequencies, with assumptions and bases stated (HLR-IE-D).

Index No. IE-D	Capability Category I	Capability Category II	Capability Category III
IE-D1	Selection of initiating events: *(a)* LIST and JUSTIFY functional categories considered, in accordance with the safety functions considered in the accident sequence model. *(b)* For each functional category, DOCUMENT the specific initiating events considered. *(c)* DOCUMENT the systematic search for plant-unique and plant-specific support system initiators, along with the resulting support system initiators disposition. *(d)* DOCUMENT the systematic search for RCS pressure boundary failures and interfacing system LOCAs. *(e)* DOCUMENT the approach for assessing completeness and consistency of initiating events with plant-specific experience, industry experience, other comparable PRAs and FSAR initiating events. *(f)* DOCUMENT the assumptions.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.1-2(d) Supporting Requirements for HLR-IE-D (Cont'd)

The initiating event analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed to select, group, and screen the initiating event list and to model and quantify the initiating event frequencies, with assumptions and bases stated (HLR-IE-D).

Index No. IE-D	Capability Category I	Capability Category II	Capability Category III
IE-D2	Grouping and screening of initiating events: *(a)* DOCUMENT the basis for screening out initiators as risk insignificant. *(b)* DOCUMENT the basis for grouping and subsuming initiating events. This may interface with the required success criteria from the Systems Analysis section (para. 4.5.4) and Success Criteria section (para. 4.5.3) of this Standard. *(c)* DOCUMENT the assumptions. *(d)* DOCUMENT the dismissal of any observed initiating events, including any credit for recovery.		
IE-D3	Quantification of initiating event frequencies: *(a)* DOCUMENT the derivation of the initiating event frequencies and the recoveries used in conjunction with the initiating event. *(b)* DOCUMENT the approach to quantification of each initiating event frequency as data analysis or model approach. *(c)* EXPLAIN any large deviations (such as an order-of-magnitude difference) from comparable generic data. When fault tree models are used to estimate initiating event frequencies, APPLY appropriate aspects of system analysis documentation requirements including any modeling assumptions. *(d)* When fault trees are used to develop initiating event frequencies, DOCUMENT how the applicable system failure modes are taken into account for each fault tree. *(e)* DOCUMENT the methodology and approach when using data analysis methods to estimate initiating event frequencies; also, IDENTIFY the data used. *(f)* DOCUMENT the justification for exclusion of any data. *(g)* DOCUMENT the basis for the availability factor used to convert initiating event frequencies to events per reactor year. *(h)* IDENTIFY potential time dependent aspects of the initiating event frequencies, and DOCUMENT assumptions made to obtain average frequencies. *(i)* DOCUMENT the process for computing initiating event frequencies. *(j)* DOCUMENT the important assumptions made in the analysis that affect the results.		
IE-D4	Interfaces with other PRA tasks: DOCUMENT specific interfaces with other PRA tasks for traceability, and to facilitate configuration control when interfacing tasks are updated.		

4.5.2 Accident Sequence Analysis (AS)

4.5.2.1 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 and LERF 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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.2-1 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 initiating event or initiating event category. 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 due to initiating events, human interface, functional dependencies, environmental and spatial impacts, and common cause failures shall be addressed.
HLR-AS-C	Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases.

Table 4.5.2-2(a) Supporting Requirements for HLR-AS-A

The accident sequence analysis shall describe the plant-specific scenarios that can lead to core damage following each initiating event or initiating event category. 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	CHOOSE a method for accident sequence analysis that explicitly MODELS the appropriate combinations of system responses and operator actions that affect the key safety functions for each modeled initiating event and provides a framework to support sequence quantification. CHOOSE a method that includes an "event tree structure" or equivalent such that the accident sequence logic (progression) is graphically represented.		
AS-A2	For each initiating event group, DEFINE in the model the key safety functions that are necessary to reach a safe, stable state and prevent core damage.		
AS-A3	For each initiating event group, using the defined success criteria for each key safety function, IDENTIFY the systems that can be used to mitigate the initiator.		
AS-A4	For each initiating event group, using the defined success criteria for each key safety function, IDENTIFY the procedurally directed operator actions.		
AS-A5	DEFINE functions and structure of the accident sequence model in a manner that is consistent with the plant-specific EOPs, abnormal procedures, training simulator exercises, and existing plant transient analysis.		
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.		
AS-A7	DELINEATE the possible accident sequences for each initiating event group, **unless the sequences can be shown to be a non-contribution using qualitative arguments.**		DELINEATE the possible accident sequences for each initiating event group.
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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.2-2(a) 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 initiating event or initiating event category. 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-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.	**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.
AS-A10	For each initiating group, **INCLUDE bounding critical safety function status that captures timing and operator actions as the individual events in the accident sequence.**	For each initiating event group, **DEVELOP the accident sequence model to sufficient detail that significant differences in requirements on systems and operator responses are captured. For example, diverse systems providing a similar function need not be modeled separately if choosing one over another does not substantially impact the sequence development. If, however, choosing one over another significantly changes the requirements for operator intervention or the need for other systems, they should be modeled separately.**	For each initiating event group, **explicitly INCLUDE each system and operator action required for each critical 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.2-2(b) 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 the initiating event, IDENTIFY the mitigating systems impacted by the occurrence of the initiator and the extent of the impact. INCLUDE the impact of initiating events on accident progression.		
AS-B2	For each critical safety function, IDENTIFY its dependence on the success or failure of preceding functions. INCLUDE the impact on accident progression. 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 If functional event trees are used, INCLUDE a model that accounts for interactions and dependencies between events.		
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]. INCLUDE the impact of the accident progression phenomena.		
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, PLACE Event A to the left of Event B in the ordering of event tops.		
AS-B5	For the event trees with conditional split fraction method, DEVELOP the event trees to a level of detail sufficient to identify intersystem dependencies and train level interfaces. For the fault tree linking method, DEVELOP fault trees and apply flag settings and mutually exclusive files or comparable method to resolve these same dependencies. 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. PROVIDE one event sequence model or set of event trees that accounts for each initiating event or initiating event category defined in the Initiating Event Analysis element so that initiating event dependencies can be properly modeled.		
AS-B6	INCLUDE events for which time-phased dependencies might exist. *(a)* For SBO/LOOP sequences, INCLUDE 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), INCLUDE 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		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.2-2(c) Supporting Requirements for HLR-AS-C

Documentation shall be performed in a manner that facilitates peer review, as well as future updates and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-AS-C).

Index No. AS-C	Capability Category I	Capability Category II	Capability Category III
AS-C1	DOCUMENT the results of the accident sequence analysis consistent with the process that was used for its development. PROVIDE the basis for the accident sequence process.		
AS-C2	DOCUMENT the treatment of each initiator and event tree to support reviews and applications.		
AS-C3	DOCUMENT the following interfaces between accident sequence analysis and other PRA tasks: *(a)* a link between the definition of initiating event category in the Initiating Event Analysis section and the event sequence model *(b)* the definition of core damage and associated success criteria that is consistent with that documented in the Success Criteria Task (para. 4.5.3) *(c)* key definitions of operator actions, sequence specific timing, and dependencies reflected in the event trees that are traceable to the HRA for these actions *(d)* a description of the interface of the accident sequence models with plant damage states *(e)* a framework for an integrated treatment of dependencies in the Initiating Events Analysis, Systems Analysis, Data Analysis, Human Reliability Analysis, and Level 1 Quantification		
AS-C4	DOCUMENT the following: *(a)* the success criteria established for each initiating event category 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) *(b)* the models used (including all sequences) for each initiating event category *(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)* any assumptions that were made in developing the accident sequences, and the bases for the assumptions and their impact on the final results *(e)* existing analyses or plant-specific calculations performed to arrive at success criteria and expected sequence phenomena including necessary timing considerations *(f)* sufficient system operation information to support the modeled dependencies *(g)* calculations or other bases used to justify equipment operability beyond its "normal" design parameters and for which credit has been taken *(h)* how all requirements for accident sequence analysis have been satisfied when sequences are modeled using a single top event linked fault tree		

4.5.3 Success Criteria (SC)

4.5.3.1 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 and large early release).

(b) success criteria are defined for critical safety functions, supporting systems, structures, components 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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.3-1 High Level Requirements for Success Criteria (SC)

Designator	Requirement
HLR-SC-A	The overall success criteria for the PRA and the system, structure, component, 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 and LERF, determination of the relative impact of success criteria on SSC and human action importance, and the impact of uncertainty on this determination.
HLR-SC-C	Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA, by describing the processes that were used, and providing details of the assumptions made and their bases.

Table 4.5.3-2(a) Supporting Requirements for HLR-SC-A

The overall success criteria for the PRA and the system, structure, component, 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 2 of this Standard. If core damage has been defined differently than in Section 2: *(a)* IDENTIFY any substantial differences from the Section 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. SELECT these parameters such that the determination of core damage is as realistic as practical, consistent with current best practice. DEFINE computer code-predicted acceptance criteria with sufficient margin between actual limits and code-calculated values to allow for limitations of the codes, sophistication of the models, and uncertainties in the results. Examples of measures for core damage that have been used in PRAs include: *(a)* Collapsed liquid level less than $\frac{1}{3}$ 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 core peak node temperature >2,200°F using a code with detailed core modeling, or code-predicted core peak node 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)		
SC-A3	SPECIFY the minimum set of mitigative functions to prevent core damage or radioactivity release in the accident sequences, for each initiating event group.		
SC-A4	SPECIFY success criteria for each of the mitigating functions for each initiating event group. IDENTIFY systems capable of meeting the specified mitigating function success criteria. IDENTIFY mitigating systems that are shared between units, and the manner in which the sharing is performed should both units experience a common intitiating event (e.g., LOOP).		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.3-2(a) Supporting Requirements for HLR-SC-A (Cont'd)

The overall success criteria for the PRA and the system, structure, component, 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 sequences in which stable plant conditions would not be achieved by 24 hr using the modeled plant equipment and human actions, USE a longer mission time if needed to achieve stable plant conditions. PERFORM additional evaluation or modeling for sequences in which a safe, stable state has not been achieved by the end of the mission time defined for the PRA 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 the Systems Analysis and Human Reliability sections of this Standard, to demonstrate that a successful outcome is achieved.		
SC-A6	CONFIRM that the models and inputs for thermal/hydraulic, structural, and other supporting engineering bases are consistent with the features, procedures, and operating philosophy of the plant.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.3-2(b) 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 and LERF, determination of the relative impact of success criteria on SSC and human action importance, 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 best-estimate generic analyses/ evaluations (e.g., thermal-hydraulic codes such as RELAP, MAAP, SAFER/ GESTER, RETRAN, or equivalent) for thermal/ hydraulic, structural, and other supporting engineering bases in support of success criteria requiring detailed computer modeling. Best-estimate 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 risk significant CDF/LERF sequences.**	**USE best-estimate, plant-specific models (e.g., thermal-hydraulic codes such as RELAP, MAAP, SAFER/ GESTER, RETRAN, or equivalent) for thermal/ hydraulic, structural, and other supporting engineering bases in support of success criteria requiring detailed computer modeling.**
SC-B2	No restrictions regarding the use of expert judgment, but requirements in 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 para. 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 (HLR-IE-B) and accident sequence modeling (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/LERF, 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., Owner's Group generic studies) may be used. USE computer codes and models only within known limits of applicability.		
SC-B5	CHECK 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		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.3-2(b) Supporting Requirements for HLR-SC-B (Cont'd)

The thermal/hydraulic, structural, and other supporting engineering bases shall be capable of providing success criteria and event timing sufficient for quantification of CDF and LERF, determination of the relative impact of success criteria on SSC and human action importance, 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-B6	**If significantly conservative or optimistic assumptions have been made in performing success criteria analyses, EVALUATE their impacts on CDF/LERF.**	**If significantly conservative or optimistic assumptions have been made in performing success criteria analyses, QUANTIFY their impacts on CDF/LERF.**	**DO NOT USE significantly conservative or optimistic assumptions in performing success criteria analyses.**

Table 4.5.3-2(c) Supporting Requirements for HLR-SC-C

Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA, by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-SC-C).

Index No. SC-C	Capability Category I	Capability Category II	Capability Category III
SC-C1	**DOCUMENT important bases, references, and assumptions for success criteria. IDENTIFY significantly conservative or optimistic assumptions and their general impacts on the results.**	**DOCUMENT each of the success criteria and the supporting engineering bases, references, and important assumptions for success criteria and the supporting engineering calculations performed in support of the PRA.** ***(a)* IDENTIFY conservative, optimistic, or simplifying assumptions or conditions.** ***(b)* PROVIDE specific justification, based on results of evaluation or quantification, as appropriate to the application Category, for use of conservative, optimistic, or simplifying assumptions or conditions.** ***(c)* PROVIDE the basis for the success criteria development process and the supporting engineering calculations.**	
SC-C2	DOCUMENT uses of and rationale for expert judgment.		
SC-C3	DOCUMENT the rationale used in the application of success criteria for situations in the PRA for which there is more than one technical approach, none of which is universally accepted as correct, and each approach results in significantly different PRA results or insights.		
SC-C4	DOCUMENT the following, to the extent they are not included in the documentation under AS-C4 and SY-C1: *(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)* identification of important assumptions used in establishing success criteria *(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		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

4.5.4 Systems Analysis (SY)

4.5.4.1 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 the HRA element.

(c) different initial system alignments are evaluated to the extent needed for CDF and LERF 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 4.5.4-1 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	The systems analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed to select, to model, and to quantify the system unavailability. Assumptions and bases shall be stated.

Table 4.5.4-2(a) 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 system. 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 final or updated SAR, 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 abnormal conditions *(h)* system configuration during normal and abnormal conditions		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.4-2(a) 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-A4	No requirement to perform plant walkdowns or interviews.	PERFORM plant walkdowns and interviews with system engineers and plant operators to confirm that the systems analysis correctly reflects the as-built, as-operated plant.	
SY-A5	In the system model, INCLUDE those conditions that prevent the system from meeting the desired system function. INCLUDE the effects of both normal and alternate system alignments, to the extent needed for CDF and LERF determination.		
SY-A6	In defining the system model boundary [see SY-A3(a)], INCLUDE within the boundary the components required for system operation, support systems interface required for actuation and operation of the system components, and other components whose failures would degrade or fail the system.		
SY-A7	DEVELOP detailed systems models, unless sufficient system-level data are available to quantify the system failure probability, or 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. A single data value may be used for systems with no equipment or human-action dependencies, 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. A system model may be developed in which a single failure of a super component is used to represent the collective impact of failures of several components. In such a "reduced" model, RETAIN the major contributors to system unavailability, and INCLUDE components or support systems shared with other modeled systems. 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.		
SY-A8	IDENTIFY the boundaries of the components required for system operation. MATCH the definitions used to establish the component failure data, or JUSTIFY an alternative assumption. For example, a control circuit for a pump does not need to be included in the system model if the pump failure data used in quantifying the system model include control circuit failures. MODEL separately portions of a component boundary that are shared by another component or affect another component, in order to account for the dependent failure mechanism.		
SY-A9	If a detailed system model is developed, MODEL separately all trains of a multi-train system in the fault tree models.		
SY-A10	If super components or modules are used to simplify system fault trees, 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 that are dependent on the scenario. Examples of such events include *(a)* hardware failures that are not recoverable versus actuation signals, which are recoverable *(b)* HE events 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 used by other systems		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.4-2(a) 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-A11	INCORPORATE the effect of variable success criteria into the system modeling. Example causes of variable system success criteria are *(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 accident initiating event category). *(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 non-critical 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 when both units are challenged by the same initiating event (e.g., LOOP).		
SY-A12	INCLUDE in the system model the equipment and components whose failure would affect system operability (as identified in the system success criteria), except when excluded using the criteria in SY-A14. 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. 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 failure modes for components contained in the model, consistent with available data and model level of detail, except where excluded using the criteria in SY-A14. 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 *(p)* other failures of a component to perform its required function		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.4-2(a) 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-A14	In meeting SY-A12 and SY-A13, 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, taking into account the same effect on system operation. *(c)* The screened contributors are position faults for components (such as those that occur during or following test and maintenance activities) for which the component receives an automatic signal to place it in its required state and no other position faults exists (e.g., pulled breakers) that would preclude the component from receiving the signal. *(d)* It is shown that the omission of the contributor does not have a significant impact on the results. DO NOT SCREEN components or failure modes using criteria (a), (b), or (c) if they could fail multiple systems or multiple trains of a system.		
SY-A15	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, para. 4.5.5.)		In the systems analysis, INCLUDE HFEs that cause the system or component to be unavailable when demanded. These events are referred to as pre-initiator human events. **To avoid double counting, CHECK that the data within the equipment-failure data base that are used for the equipment failure rates do not include events that are captured in the pre-initiator-HEP calculation.** (See also Human Reliability Analysis, para. 4.5.5.)
SY-A16	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 in the Accident Sequence Analysis. These HFEs are referred to as post-initiator human actions. (See also Human Reliability Analysis, para. 4.5.5, and Accident Sequence Analysis, para. 4.5.2.)		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.4-2(a) 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-A17	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 SHOW that their exclusion does not to 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		
SY-A18	In the systems model, INCLUDE out-of-service unavailability for components in the system model, unless screened, 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: *(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-A19	MODEL explicitly system conditions that cause a loss of desired system function by using realistic functional requirements that are supported with engineering analysis. For example: excessive heat loads, excessive electrical loads, etc. to support continued operation of the system for the required mission time that are based on plant-specific or acceptable generic analyses. **If engineering analyses are not available, ASSUME that the equipment/system fails with a probability of 1.0 or JUSTIFY the assumed failure probability.**		MODEL explicitly system conditions that cause a loss of desired system function by using realistic functional requirements that are supported with engineering analysis. For example: excessive heat loads, excessive electrical loads, etc. to support continued operation of the system for the required mission time that are based on plant-specific or acceptable generic analyses.
SY-A20	DO NOT TAKE CREDIT for system or component operability beyond rated or design capabilities unless justified, based on an appropriate combination of *(a)* test or operational data *(b)* calculations *(c)* vendor input *(d)* expert judgment JUSTIFY the basis for any credit taken.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.4-2(a) 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-A21	DEVELOP 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-A22	In the support-state approach, ASSIGN support states to account properly for system dependencies on other systems.		
SY-A23	DO NOT MODEL the repair of hardware faults, unless the probability of repair is justified through an adequate recovery analysis or examination of data. (See DA-C14.)		

Table 4.5.4-2(b) 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, **or SHOW 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 [Note (1)].** **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-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 SHOW that they do not impact the results.
SY-B3	ESTABLISH 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.4-2(b) 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-B4	INCORPORATE common cause failures into the system model consistent with the common cause model used for data analysis. (See DA-D6.)		
SY-B5	ACCOUNT explicitly for the modeled system's dependency on support systems in the modeling process. This may be accomplished by *(a)* fault tree linking *(b)* dependency matrices that are translated into event tree structure, event tree logic rules, or conditional split fractions *(c)* an evaluation that demonstrates that excluding the dependency does not significantly affect the system model		
SY-B6	PERFORM engineering analyses of 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. Bounding or generic engineering analyses (i.e., tests, operational experience, or calculations) may be used when it is not expected that these analyses would interfere with the realistic quantification of CDF or LERF.		
SY-B7	BASE support system modeling on **the use of conservative success criteria and timing.**	BASE support system modeling on **realistic success criteria and timing, unless a conservative approach can be justified. i.e., if their use does not impact risk significant contributors.**	BASE support system modeling on **realistic plant-specific success criteria and timing.**
SY-B8	IDENTIFY spatial and environmental hazards that may impact system operation, and ACCOUNT for them in the system fault tree or the accident sequence evaluation. Example: Use results of plant walkdowns as a source of information and resolution of issues in the evaluation of their impacts.		
SY-B9	INCLUDE explicit treatment of containment vent effects (BWRs) and containment failure effects on system operation in the consideration of possible hazards.		
SY-B10	When modeling a system, INCLUDE the support systems required for successful operation of the system for a required mission time. Examples: *(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		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.4-2(b) 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	**IDENTIFY** those systems that are required for initiation and actuation of a system. MODEL them unless a justification is provided. 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.	
SY-B12	COMPARE the available inventories of air, power, and cooling with those required to support the mission time. TREAT these inventories in the model unless a justification is provided.		
SY-B13	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.		
SY-B14	Some systems use components and equipment that are required for operation of other systems. INCLUDE components that may otherwise be screened from a system model, if their failure affects more than one system (e.g., a common suction pipe feeding two separate systems).		
SY-B15	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)* 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		
SY-B16	INCLUDE operator interface dependencies across systems or trains, where applicable.		

NOTE:

(1) NUREG/CR-5485, Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment, November 20, 1998

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.4-2(c) Supporting Requirements for HLR-SY-C

The systems analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer view by describing the processes that were followed to select, to model, and to quantify the system unavailability. Assumptions and bases shall be stated (HLR-SY-C).

Index No. SY-C	Capability Category I	Capability Category II	Capability Category III
SY-C1	DOCUMENT the system model used in the PRA 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. 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)* a list of all components and failure modes included in the model, along with 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)		
SY-C2	DOCUMENT basic events in the system fault trees so that they are traceable to modules and to cutsets.		
SY-C3	DOCUMENT the nomenclature used in the system models.		

4.5.5 Human Reliability Analysis (HR)[7]

4.5.5.1 Objectives. The objectives of the human reliability element of the PRA are 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.

[7] The following reference provides useful background information for human reliability analysis: EPRI TP-101711, Sharp 1 — Revised Systematic Human Action Reliability Procedure; D. T. Wakefield, G. W. Parry, G. W. Hannaman, A. J. Spurgin; 1992

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.5-1 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 which, 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, 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, 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 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	The HRA shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were used, and providing details of the assumptions made and their bases.

Table 4.5.5-2(a) 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, through a review of procedures and practices, those test and maintenance activities that require realignment of equipment outside its normal operational or standby status.		
HR-A2	IDENTIFY, through a review of procedures and practices, those calibration activities that if performed incorrectly can have an adverse impact on the automatic initiation of standby safety equipment.		
HR-A3	IDENTIFY those work practices that could introduce a mechanism which 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)].		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.5-2(b) 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, ESTABLISH rules for screening **classes of** activities from further consideration. Example: Screen maintenance and test activities 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, ESTABLISH rules for screening **individual** activities from further consideration. Example: Screen maintenance and test activities from further consideration only if ***(a)* equipment is automatically re-aligned on system demand, or** ***(b)* following maintenance activities, a post-maintenance functional test is performed that reveals misalignment, or** ***(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 activities that could simultaneously have an impact on multiple trains of a redundant system or diverse systems.		

Table 4.5.5-2(c) 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	DEFINE HFEs at a level of detail consistent with that of the system and accident sequence models.		
HR-C2	For each unscreened activity, 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	For each unscreened activity, 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 **ADD 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.5-2(d) 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 [Note (1)] and ASEP [Note (2)].		
HR-D2	USE **screening estimates** in the quantification of the pre-initiator HEPs.	USE **detailed assessments** in the quantification of pre-initiator HEPs **for dominant system contributors. Screening values may be used in the quantification of the pre-initiator HEPs for systems that do not appear in the *dominant sequences.***	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 of written procedures (for performing tasks) and administrative controls (for independent review) *(b)* the quality of the human-machine interface, including both the equipment configuration, and instrumentation and control layout	
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 consistent with selected methodology. If recovery of pre-initiator errors is credited *(a)* ESTABLISH 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: *(1)* post-maintenance or post-calibration tests required and performed by procedure *(2)* independent verification, using a written check-off list, which verify component status following maintenance/testing *(3)* original performer, using a written check-off list, makes a separate check of component status at a later time *(4)* work shift or daily checks of component status, using a written check-off list		
HR-D5	ASSESS the joint probability of those HFEs identified as having some degree of dependency (i.e., having some common elements in their causes, such as performed by the same crew in the same time-frame).		
HR-D6	PROVIDE an assessment of the uncertainty in the HEPs. USE mean values when providing point estimates of HEPs.		
HR-D7	CHECK the reasonableness of the HEPs in light of the plant's history, procedures, operational practices, and experience. Operating experience may be used to support quantification of impact that test, maintenance, and calibration activities have on overall system unavailability.		

NOTES:

(1) NUREG/CR-1278, Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications; A.D. Swain and H.E. Guttmann; August 1983 (THERP)

(2) NUREG/CR-4772, Accident Sequence Evaluation Program Human Reliability Analysis Procedure; A.D. Swain; February 1987 (ASEP)

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.5-2(e) 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 *(a)* REVIEW the plant-specific emergency operating procedures, and other relevant procedures (e.g., AOPs, annunciator response procedures) in the context of the accident scenarios *(b)* REVIEW system operation such that an understanding of how the system(s) functions and the human interfaces with the system is obtained		
HR-E2	INCLUDE those actions required to initiate (for those systems that are not automatically initiated), operate, control, isolate, or terminate systems and components used in preventing or mitigating core damage as defined by the success criteria [e.g., operator initiates RHR **as required by the EOPs to maintain suppression pool temperature below the defined limit (BWR)**].	INCLUDE *(a)* those actions 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)* those actions performed by the control room staff either in response to procedural direction or as skill-of-the-craft to recover a failed function, system or component that is used in the performance of a response action in dominant sequences (e.g., manual start of a standby pump following failure of auto-start)**	
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.**	**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 is consistent with plant observations and training procedures.**	
HR-E4	No requirement for using simulator observations or talk-throughs with operators to confirm response models.	USE simulator observations or talk-throughs with operators to confirm the response models for scenarios modeled.	

Table 4.5.5-2(f) 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 a set of Human Failure Events (HFEs) as unavailabilities of functions, systems or components as appropriate to the level of detail in the accident sequence and system models. 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.5-2(f) Supporting Requirements for HLR-HR-F (Cont'd)

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-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., component level) required to achieve the goal of the response**

Table 4.5.5-2(g) 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** or PERFORM detailed analyses for the estimation of the HEPs of the HFEs in dominant accident sequences that survive initial quantification. Screening values may be used for **the probabilities of HFEs** in nondominant sequences.	PERFORM detailed analyses for the estimation of those post-accident HEPs of the HFEs that survive initial quantification. Screening values may be used for **HEPs that only appear** in non-dominant sequences.	
HR-G2	USE an approach to estimation of HEPs that addresses failure in cognition as well as failure to execute.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.5-2(g) 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-G3	**USE an approach that takes the following into account:** *(a)* the complexity of the response *(b)* the time available and time required to complete the response *(c)* some measure of scenario-induced stress The ASEP Approach 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 the cues/indications** ***(e)* human-machine interface** *(f)* time available and time required to complete the response *(g)* complexity of 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	BASE the time available to complete actions on **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.	BASE the time available to complete actions **on plant-specific thermal/hydraulic analysis, or simulations.** SPECIFY the point in time at which operators are expected to receive relevant indications.	
HR-G5	**ESTIMATE** the time required to complete actions. **The approach described in ASEP is an acceptable approach.**	**BASE** the required time to complete actions **in dominant scenarios on actual time measurements in either walkthroughs or talk-throughs of the procedures or simulator observations.**	**BASE** the required time to complete actions **on actual time measurements in either walkthroughs or talk-throughs of the procedures or simulator observations.**
HR-G6	CHECK the consistency of the post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to check 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 cut set, ASSESS the degree of dependence. ACCOUNT for the influence of success or failure in preceding human actions and system performance on the human event under consideration including *(a)* the 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.)		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.5-2(g) 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-G8	DEFINE and JUSTIFY the minimum probability to be used for the joint probability of multiple human errors occurring in a given cutset.		
HR-G9	Characterize the uncertainty in the estimates of the HEPs, and PROVIDE mean values for use in the quantification of the PRA results.		

Table 4.5.5-2(h) 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).

Index No. HR-H	Capability Category I	Capability Category II	Capability Category III
HR-H1	INCLUDE human recovery actions that can restore the functions, systems, or components **contributing to the dominant sequences.**		INCLUDE human recovery actions that can restore the functions, systems, or components.
HR-H2	CREDIT operator recovery actions only if, on a plant-specific basis: *(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		
HR-H3	ACCOUNT for any dependency between the HFE for recovery and any other HFEs in the sequence, scenario, or cutset to which the recovery is applied.		

Table 4.5.5-2(i) Supporting Requirements for HLR-HR-I

The HRA shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-HR-I).

Index No. HR-I	Capability Category I	Capability Category II	Capability Category III
HR-I1	DOCUMENT the HRA in enough detail to reproduce results and permit reviewers to understand limitations imposed by the models, assumptions, and data, including the following: *(a)* HRA methodology and process used to identify pre- and post-initiator HEPs *(b)* generic and plant-specific assumptions that were made in the HRA, including: *(1)* the bases for the assumptions *(2)* their impact on the CDF and LERF results *(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)* source(s) of data used to quantify human actions, including *(1)* screening values and their bases *(2)* best estimates with uncertainties and their bases *(3)* the method and treatment of dependencies for post-initiator actions *(4)* all pre- and post-initiator human actions evaluated by model, system, initiating event, and function *(5)* all HEPs for each post-initiator human action and significant dependency effects		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

4.5.6 Data Analysis (DA)

4.5.6.1 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.

Table 4.5.6-1 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 plant-specific data shall be collected consistent with the parameter definitions of HLR A and the grouping rationale of HLR 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. Parameter estimates for the important parameters shall be accompanied by a characterization of the uncertainty.
HLR-DA-E	Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases.

Table 4.5.6-2(a) 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. ESTABLISH definitions of SSC boundaries, failure modes, and mission success criteria consistent with corresponding basic event definitions in Systems Analysis (SY-A4, SY-A7, and SY-A8) for failure rates and common cause failure parameters, and ESTABLISH boundaries of unavailability events consistent with corresponding definitions in Systems Analysis (SY-A19). Basic events typically 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	USE an appropriate probability model for each basic event. Examples include *(a)* binomial distributions for failure on demand *(b)* Poisson distributions for standby and operating failures and initiating events		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.6-2(a) Supporting Requirements for HLR-DA-A (Cont'd)

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-A3	IDENTIFY the parameter to be estimated and the data required. Examples are as follows: *(a)* For failures on demand or unavailability due to test or maintenance, the parameter is the probability of failure or unavailability on demand, 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.		

Table 4.5.6-2(b) 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: ***(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:** ***(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 obvious 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 obvious 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.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.6-2(c) Supporting Requirements for HLR-DA-C

Generic parameter estimates shall be chosen and plant-specific data shall be collected in accordance with the parameter definitions of HLR A and the grouping rationale of HLR B (HLR-DA-C).

Index No. DA-C	Capability Category I	Capability Category II	Capability Category III
DA-C1	COLLECT generic parameter estimates from recognized sources. IDENTIFY the derivation process and source of the generic parameter estimates, DETERMINE that the parameter definitions and boundary conditions are consistent with those determined in response to DA-A13. USE generic data for unavailability due to test, maintenance, and repair with caution since different plants can have different test and maintenance philosophies. Examples of parameter estimates and associated sources include *(a)* component failure rates and probabilities: NUREG/CR-4639 [Note (1)], NUREG/CR-4550 [Note (2)] *(b)* common cause failures: NUREG/CR-5497 [Note (3)], NUREG/CR-6268 [Note (4)] *(c)* AC off-site power recovery: NUREG/CR-5496 [Note (5)], NUREG/CR-5032 [Note (6)] *(d)* component recovery		
DA-C2	COLLECT plant-specific data for the basic event/parameter grouping corresponding to that defined by requirement DA-A1, DA-A2, DA-A3, DA-B1, and DA-B2.		
DA-C3	COLLECT plant-specific data from as broad a time period as possible, consistent with uniformity in design, operational practices, and experience. JUSTIFY the rationale for screening or disregarding plant-specific data (e.g., plant design modifications, changes in operating practices).		
DA-C4	When evaluating maintenance or other relevant records to extract plant-specific component failure event data, DEVELOP a clear basis for the identification of events as failures: *(a)* DISTINGUISH between those degraded states for which failure, as modeled in the PRA, would have occurred on demand (e.g., an operator discovers that a pump has no oil in its lubrication reservoir), and those that would not (e.g., slow pick-up to rated speed). *(b)* INCLUDE as failures all events that would have resulted in a failure to perform the mission as defined in the PRA.		
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	DETERMINE 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** number of surveillance tests and planned maintenance activities on **plant requirements.**	**BASE** number of surveillance tests on **plant surveillance requirements and actual practice. BASE** number of planned maintenance activities **on plant maintenance plans and actual practice. BASE 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.	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.6-2(c) Supporting Requirements for HLR-DA-C (Cont'd)

Generic parameter estimates shall be chosen and plant-specific data shall be collected in accordance with the parameter definitions of HLR A and the grouping rationale of HLR B (HLR-DA-C).

Index No. DA-C	Capability Category I	Capability Category II	Capability Category III
DA-C9	**ESTIMATE** operational time from surveillance test **practices** for standby components, and from actual operational data.		**DETERMINE** operational time from surveillance test **records** for standby components, and from actual operational data.
DA-C10	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operation.	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operation. **If the component failure mode is decomposed into sub-elements (or causes) that are fully tested, then USE tests that exercise specific sub-elements in their evaluation. Thus, one sub-element sometimes has many more successes than another.**	When using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. COUNT only completed tests or unplanned operational demands as success for component operation. **DECOMPOSE the component failure mode into sub-elements (or causes) that are fully tested, and USE tests that exercise specific sub-elements in their evaluation. Thus, one sub-element 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.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.6-2(c) Supporting Requirements for HLR-DA-C (Cont'd)

Generic parameter estimates shall be chosen and plant-specific data shall be collected in accordance with the parameter definitions of HLR A and the grouping rationale of HLR B (HLR-DA-C).

Index No. DA-C	Capability Category I	Capability Category II	Capability Category III
DA-C12	EVALUATE 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. Special attention should be paid to the case of a multi-plant site with shared systems, 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, **provide conservative estimates.**	EVALUATE 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. Special attention should be paid to the case of a multi-plant site with shared systems, when the 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 plant maintenance and operations staff to generate estimates of ranges in the unavailable time per maintenance act for components, trains, or systems in dominant accident scenarios. DO NOT manipulate time periods to avoid specific maintenance events.**	
DA-C13	EXAMINE coincident outage times for redundant equipment (both intra- and inter-system) based on actual plant experience. CALCULATE outage unavailabilities that reflect actual plant experience.		
DA-C14	IDENTIFY instances of plant-specific component repair 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-C15	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.		

NOTES:

(1) NUREG/CR-4639, Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR), Vols. 1–5, 1994
(2) NUREG/CR-4550, Auxiliary Feedwater System Risk-Based Inspection Guide for the Salem Nuclear Generating Station, October 1990
(3) NUREG/CR-5497, Common-Cause Failure Parameter Estimations
(4) NUREG/CR-6268, Common Cause Failure Database and Analysis System, Vols. 1–4, 1998
(5) NUREG/CR-5496, Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1986
(6) NUREG/CR-5032, Modeling Time to Recover and Initiate Even Frequency for Loss-of-Offsite Power Incidents at Nuclear Power Plants, March 1988

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.6-2(d) 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 the unique design or operational features with generic information for remaining events.**	**CALCULATE realistic parameter estimates for dominant contributors using Bayesian updates. CHOOSE prior distributions as either non-informative, or representative of variability in industry data. CALCULATE plant-specific parameter estimates for the remaining events by using plant-specific 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.		
DA-D3	PROVIDE a **characterization (e.g., qualitative discussion)** of the uncertainty intervals for the estimates of those parameters **used for estimating the probabilities of the basic events that contribute measurably to CDF and LERF.**	PROVIDE a **mean value of, and a statistical representation** of, the uncertainty intervals for the parameter estimates **that contribute measurably to CDF and LERF. Acceptable systematic methods include Bayesian updating, frequentist method, or expert judgment.**	PROVIDE a **mean value of, and a statistical representation** of, the uncertainty intervals for the parameter estimates. **Acceptable systematic methods include: Bayesian updating, frequentist method, 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, PERFORM the following tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application: *(a)* CONFIRM that the Bayesian updating does not produce a posterior distribution with a single bin histogram *(b)* EXAMINE the cause of any unusual posterior distribution shapes *(c)* EXAMINE inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate *(d)* CONFIRM that the Bayesian updating algorithm provides valid results over the range of values being considered *(e)* CONFIRM the reasonableness of the posterior distribution mean value	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.6-2(d) 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 or an equivalent for the estimation of CCF parameters.**	USE **one of the following models for estimating CCF parameters for dominant CCF contributors:** ***(a)* Alpha Factor Model** ***(b)* Basic Parameter Model** ***(c)* Multiple Greek Letter Model** ***(d)* Binomial Failure Rate Model** **JUSTIFY the use of alternative methods.**	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.**
DA-D6	USE **generic** common cause failure probabilities.	USE **realistic** common cause failure probabilities **consistent with available plant-specific data, supported by plant-specific screening and mapping of industry-wide data, for dominant common-cause events. An example approach is provided in NUREG/CR-5485 [Note (1)]. DETERMINE that the systems and data models are consistent.**	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.6-2(d) 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-D7	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 significant 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 significant generic parameter estimates are available, USE the generic parameter estimates updated with plant-specific data as it becomes available **for dominant contributors,** 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 significant 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.

NOTE:

(1) NUREG/CR-5485, *Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment*, November 20, 1998

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.6-2(e) Supporting Requirements for HLR-DA-E

Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-DA-E).

Index No. DA-E	Capability Category I	Capability Category II	Capability Category III
DA-E1	DOCUMENT the following: *(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)* key assumptions made in the interpretation of data and the reasoning (based on engineering, systems modeling, operations, and statistical knowledge) supporting its use in parameter estimation *(g)* justification for exclusion of any data *(h)* the basis for the estimates of common cause failure probabilities, including justification for screening or mapping of generic and plant-specific data *(i)* the rationale for any distributions used as priors for Bayesian updates, where applicable *(j)* parameter estimate including the characterization of uncertainty as appropriate		

4.5.7 Internal Flooding (IF)

4.5.7.1 Objectives. The objectives of the internal flooding element are to ensure that the impact of internal flooding as the cause of either an accident or a system failure is evaluated in such a way that

(a) the water sources within the plant that could flood plant locations or create adverse conditions that could damage mitigative plant equipment are identified.

(b) the flood scenarios/sequences that contribute to the core damage frequency and large early release frequency are identified and quantified.

Table 4.5.7-1 High Level Requirements for Internal Flooding (IF)

Designator	Requirement
HLR-IF-A	Different flood areas of the plant and the SSCs located within the areas shall be identified.
HLR-IF-B	The potential flood sources in the plant and their associated flooding mechanisms shall be identified.
HLR-IF-C	The potential flooding scenarios shall be developed for each flood source by identifying the propagation path(s) of the water and the affected SSCs.
HLR-IF-D	Flooding-induced initiating events shall be identified and their frequencies estimated.
HLR-IF-E	Flood-induced accident sequences shall be quantified.
HLR-IF-F	The internal flooding analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed, with assumptions and bases stated.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.7-2(a) Supporting Requirements for HLR-IF-A

Different flood areas of the plant and the SSCs located within the areas shall be identified (HLR-IF-A).

Index No. IF-A	Capability Category I	Capability Category II	Capability Category III
IF-A1	DEFINE flood areas by dividing the plant into physically separate areas where a flood area is generally viewed as independent of other areas in terms of flooding effects and flood propagation. DEFINE flood areas by using *(a)* the presence of physical barriers (e.g., walls, floors, dikes) *(b)* mitigation features (e.g., sumps, drains) *(c)* propagation pathways (e.g., open hatches or doors)		
IF-A2	IDENTIFY the SSCs located in each flood area, including their spatial location in the area and any flooding mitigative features (e.g., shielding). INCLUDE SSCs modeled in the PRA as part of the success criteria and SSCs that can challenge normal plant operation requiring successful mitigation to prevent core damage. If a flood area can be screened out using the requirements in element IF-C5, then there is no requirement to identify SSCs within this flood area.		
IF-A3	USE plant information sources to support development of flood areas and to identify the SSCs located within each flood area.		
IF-A4	CONDUCT a plant walkdown 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)* SSCs located within each flood area *(c)* potential flood sources within each flood area		

Table 4.5.7-2(b) Supporting Requirements for HLR-IF-B

The potential flood sources in the plant and their associated flooding mechanisms shall be identified (HLR-IF-B).

Index No. IF-B	Capability Category I	Capability Category II	Capability Category III
IF-B1	For each flood area, IDENTIFY the potential sources of flooding water, which include *(a)* equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, component cooling water system, feedwater system, and reactor coolant system) *(b)* plant internal sources of water (e.g., tanks or pools) located in the area *(c)* plant external sources of water (e.g., reservoirs or rivers) that are connected to the area through some system or structure		
IF-B2	For each potential source of flooding water, IDENTIFY the flooding mechanisms that would result in the release of water. INCLUDE *(a)* failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc. *(b)* human-induced mechanisms that could lead to overfilling tanks, diversion of flow through openings created to perform maintenance; inadvertent actuation of fire suppression system *(c)* other events releasing water into the area		
IF-B3	For each source and its identified failure mechanism, IDENTIFY the characteristic of water release and the capacity of the source; INCLUDE *(a)* a characterization of the breach, including type (e.g., leak, rupture, spray) and form (e.g., a five-foot cone-shaped spray discharging to the northeast) *(b)* flow rate of water *(c)* capacity (e.g., gallons of water source)		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.7-2(b) Supporting Requirements for HLR-IF-B (Cont'd)

The potential flood sources in the plant and their associated flooding mechanisms shall be identified (HLR-IF-B).

Index No. IF-B	Capability Category I	Capability Category II	Capability Category III
IF-B4	In each flood area, IDENTIFY any floor drains (i.e., any physical structure that can function as a drain) or sumps (i.e., any physical structure that allows for the accumulation and retention of water). DETERMINE the capacity of the drains and the amount of water retained by the sumps. If these are larger than a flood source in the area and the flood source cannot cause additional equipment damage or failure (see IF-C4), then the flood source may be eliminated.		

Table 4.5.7-2(c) Supporting Requirements for HLR-IF-C

The potential flooding scenarios shall be developed for each flood source by identifying the propagation path(s) of the water and the affected SSCs (HLR-IF-C).

Index No. IF-C	Capability Category I	Capability Category II	Capability Category III
IF-C1	For each flood source, IDENTIFY the propagation path from the flood source area to its accumulation point; INCLUDE the normal flow path from one area to another via drain lines; and areas connected via back flow 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.		
IF-C2	IDENTIFY plant design features or operator actions that have the ability to terminate the flood propagation. INCLUDE the availability of flood alarms, flood dikes, curbs, drains, sump pumps, spray shields, water-tight doors, and operator actions. JUSTIFY any credit given, particularly any credit given for non-flood proof doors or barriers, and credit for isolation of a flood source including the method of detection, accessibility to the isolation device, and time available to perform actions.		
IF-C3	IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence, jet impingement, spray, pipe whip, humidity, condensation, temperature concerns, and any other identified failure modes in the identification process. JUSTIFY exclusion of any SSC's susceptibility to a flood-induced environment based on appropriate documented criteria such as test or experimental data, equipment qualification data, or other analyses. If susceptibility information cannot be ascertained, ASSUME the equipment will fail in the presence of the associated flood-induced environment.		
IF-C4	DEVELOP flood scenarios by examining potential propagation paths, giving credit for appropriate flood mitigation systems or operator actions, and identifying susceptible SSCs. VERIFY any information used from documents during plant walkdown.		
IF-C5	USE as criteria to screen out flood areas one or more of the following: *(a)* an area (including adjacent areas where flood sources can propagate) with no mitigating equipment modeled in the PRA; *(b)* a flood within the area does not cause an initiating event, including a manual scram; or *(c)* an area with no significant flood sources and the nature of the flood (e.g., spraying of equipment) does not cause equipment failure (i.e., areas where there are no flood sources or where the volumes of the flood sources are insufficient, through either spray or immersion to cause failure of equipment); *(d)* an area with mitigation systems (e.g., drains or sump pumps) capable of preventing unacceptable flood levels and other flooding effects are expected to be insignificant. JUSTIFY any other qualitative screening criteria.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.7-2(c) Supporting Requirements for HLR-IF-C (Cont'd)

The potential flooding scenarios shall be developed for each flood source by identifying the propagation path(s) of the water and the affected SSCs (HLR-IF-C).

Index No. IF-C	Capability Category I	Capability Category II	Capability Category III
IF-C6	**USE potential human mitigative actions as additional criteria for screening if all the following can be shown:** ***(a)*** **an area has small or modest flood sources that is not in a propagation path from small or modest sources** ***(b)*** **the time to the damage of safe shutdown equipment is greater than 2 hr for the worst flooding initiator** ***(c)*** **flood indication is available in the control room** ***(d)*** **the flood sources in the area can be isolated**	**USE potential human mitigative actions as additional criteria for screening if all the following can be shown:** ***(a)*** **an area has small or modest flood sources that is not in a propagation path from small or modest sources** ***(b)*** **the mitigative action can be performed with high reliability for the worst flooding initiator** ***(c)*** **flood indication is available in the control room** ***(d)*** **the flood sources in the area can be isolated**	**DO NOT SCREEN flood scenarios that rely on operator action to prevent challenges to normal plant operations.**

Table 4.5.7-2(d) Supporting Requirements for HLR-IF-D

Flooding-induced initiating events shall be identified and their frequencies estimated (HLR-IF-D).

Index No. IF-D	Capability Category I	Capability Category II	Capability Category III
IF-D1	USE a structured, systematic process to identify those flood-induced initiating events that challenge normal plant operation and that require successful mitigation to prevent core damage. INCLUDE the potential for a flooding-induced transient or LOCA.		
IF-D2	No requirement to review the impact of plant-specific initiating event precursors and system alignments.	In searching for flood-induced initiating events, REVIEW the impact of plant-specific initiating event precursors and system alignments. INCLUDE alignments of supporting systems.	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.7-2(d) Supporting Requirements for HLR-IF-D (Cont'd)

Flooding-induced initiating events shall be identified and their frequencies estimated (HLR-IF-D).

Index No. IF-D	Capability Category I	Capability Category II	Capability Category III
IF-D3	GROUP flooding-induced initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, and timing; or *(b)* events can be subsumed into a group and bounded by the worst-case impacts within the new group.	GROUP flooding-induced initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, and timing, **and the effect on the operability and performance of operators and relevant mitigating systems;** or *(b)* events can be subsumed into a group and bounded by the worst case impacts within the new group. **To avoid excess conservatism, DO NOT ADD initiating events to a group and DO NOT SUBSUME events into a group unless the impacts are comparable to or less than those of the remaining events in that group.**	GROUP flooding-induced initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, and timing, **and the effect on the operability and performance of operators and relevant mitigating systems;** or *(b)* events can be subsumed into a group and bounded by the worst case impacts within the new group. **To avoid conservatism, DO NOT ADD initiating events to a group and DO NOT SUBSUME events into a group unless the impacts are comparable to or less than those of the remaining events in that group, or it is demonstrated that such grouping does not appreciably impact CDF or LERF.**
IF-D4	For multi-unit sites with shared systems or structures, **PERFORM a qualitative evaluation to determine that the relative risk significance of modeled SSCs is not distorted if dual unit internal flood initiators are excluded from the analysis. If the qualitative evaluation cannot show that the relative risk significance of modeled SSCs is not distorted, then INCLUDE dual unit initiators.**	For multi-unit sites with shared systems or structures, **TREAT quantitatively dual unit internal flood initiators.**	

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.7-2(d) Supporting Requirements for HLR-IF-D (Cont'd)

Flooding-induced initiating events shall be identified and their frequencies estimated (HLR-IF-D).

Index No. IF-D	Capability Category I	Capability Category II	Capability Category III
IF-D5	DETERMINE the flood initiating event frequency by using the applicable requirements in IE-C (specifically IE-C1 through IE-C5) and one or more of the following: *(a)* an assessment of **applicable** generic operating experience of internal flooding *(b)* an evaluation of pipe, component, and tank rupture failure rates from generic data sources *(c)* a probabilistic fracture mechanics evaluation of the probability of pipe leaks or ruptures **representative of** plant-specific conditions *(d)* a combination of operating experience, and generic pipe and component failure rates *(e)* a combination of one of the above approaches with expert judgment	DETERMINE the flood initiating event frequency by using the applicable requirements in IE-C (specifically IE-C1 through IE-C5) and one or more of the following: *(a)* an assessment of generic **and plant-specific** operating experience of internal flooding *(b)* an evaluation of pipe, component, and tank rupture failure rates from generic data sources **enhanced by any plant-specific information** *(c)* a probabilistic fracture mechanics evaluation of the probability of pipe leaks or ruptures **using** plant-specific information *(d)* a combination of **plant-specific** operating experience and generic pipe and component failure rates *(e)* a combination of one of the above approaches with expert judgment	

Table 4.5.7-2(e) Supporting Requirements for HLR-IF-E

Flood-induced accident sequences shall be quantified (HLR-IF-E).

Index No. IF-E	Capability Category I	Capability Category II	Capability Category III
IF-E1	REVIEW the accident sequence results obtained by following the applicable requirements described in para. 4.5.2 and MODIFY sequences as necessary to account for any flood-induced phenomena.		
IF-E2	PERFORM any necessary engineering calculations for flood rate, time to reach vulnerable equipment, and the structural capacity of SSCs according to the requirements described in SC-B1 through SC-B6.		
IF-E3	MODIFY the systems analysis results obtained by following the applicable requirements described in para 4.5.4 to include flood-induced failures identified by IF-C3.		
IF-E4	PERFORM any additional data analysis to the applicable requirements described in para. 4.5.6.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.7-2(e) Supporting Requirements for HLR-IF-E (Cont'd)

Flood-induced accident sequences shall be quantified (HLR-IF-E).

Index No. IF-E	Capability Category I	Capability Category II	Capability Category III
IF-E5	PERFORM any human reliability analysis to the applicable requirements described in para. 4.5.5 (HLR-HR-E through HLR-HR-G), and INCLUDE the following scenario specific PSFs for control room and ex-control room actions as appropriate: *(a)* additional workload and stress (above that for similar sequences not caused by internal floods) *(b)* uncertainties in event progression (e.g., cue availability and timing concerns caused by flood) *(c)* effect of flood on mitigation, required response, and recovery activities (e.g., accessibility restrictions, possibility of physical harm) *(d)* flooding-specific job aids and training (e.g., procedures, training exercises) JUSTIFY the use of extraordinary recovery actions that are not proceduralized.		
IF-E6	PERFORM internal flood sequence quantification in accordance with the applicable requirements described in para. 4.5.8, including any quantitative screening. INCLUDE the combined effects of failures caused by flooding and those coincident with the flooding due to independent causes including equipment failures, unavailability due to maintenance, and other credible causes. INCLUDE 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.		
IF-E7	REVIEW the LERF analysis results obtained by following the applicable requirements described in para. 4.5.9 and MODIFY as necessary to account for any flood-induced phenomena.		

Table 4.5.7-2(f) Supporting Requirements for HLR-IF-F

The internal flooding analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed, with assumptions and bases stated (HLR-IF-F).

Index No. IF-F	Capability Category I	Capability Category II	Capability Category III
IF-F1	DOCUMENT the process used to identify flood sources, flood areas, flood pathways, flood scenarios, and their screening, and internal flood model development and quantification.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.7-2(f) Supporting Requirements for HLR-IF-F (Cont'd)

The internal flooding analysis shall be documented in a manner that facilitates PRA applications, upgrades, and peer review by describing the processes that were followed, with assumptions and bases stated (HLR-IF-F).

Index No. IF-F	Capability Category I	Capability Category II	Capability Category III
IF-F2	DOCUMENT the following results *(a)* flood sources identified in the analysis, any rules used to screen out these sources, and the resulting list of sources to be further examined *(b)* flood areas used in the analysis and the reason for eliminating any of these areas from further analysis *(c)* propagation pathways between flood areas and any assumptions, calculations, or other bases for eliminating or justifying any of these propagation pathways *(d)* accident mitigating features and barriers credited in the analysis, the extent to which they were credited, and associated justification *(e)* component fragilities and any associated assumptions or calculations used in the determination of the impacts of submergence, spray, temperature, or other flood-induced effects on equipment operability *(f)* screening criteria used in the analysis *(g)* flooding scenarios considered, screened, and the remaining scenarios, as well as how the internal event analysis models were modified to model these remaining scenarios for the internal flooding analysis *(h)* flood frequencies, component unreliabilities/unavailabilities, and HEPs used in the analysis (i.e., the data values unique to the flooding analysis) *(i)* any calculations or other analyses used to support or refine the flooding evaluation *(j)* results of the internal flooding analysis including results from each accident sequence, results from the combined accident sequence model (i.e., the total plant model), results from sensitivity and uncertainty analyses, and results from importance measure calculations		

4.5.8 Quantification (QU)

4.5.8.1 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) important contributors to CDF are identified in terms of initiating events, accident sequences, equipment failures, and operator errors.

(c) significant dependencies are accounted for.

(d) uncertainties are understood and appropriately quantified.

Table 4.5.8-1 High Level Requirements for the Level 1 Accident Sequence Quantification (QU) and Results Interpretation

Designator	Requirement
HLR-QU-A	The Level 1 quantification shall quantify core damage frequency.
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 important contributors to CDF, such as initiating events, accident sequences, equipment failures and operator errors, 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 key assumptions shall be identified, and their potential impact on the results understood.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.8-1 High Level Requirements for the Level 1 Accident Sequence Quantification (QU) and Results Interpretation (Cont'd)

Designator	Requirement
HLR-QU-F	Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assuumptions made and their bases.

Table 4.5.8-2(a) 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 sequence delineation, 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 overall CDF from internal events. QUANTIFY individual sequences to identify dominant sequences and confirm the sequence logic is appropriately reflected. The estimates may be accomplished by using either fault tree linking or event trees with conditional split fractions.		
QU-A3	SELECT a method that is capable of discriminating the contributors to the CDF commensurate with the level of detail in the model.		
QU-A4	INCLUDE recovery actions in the quantification process in applicable sequences and cut sets. Recovery actions credited in the evaluation may be either proceduralized or have reasonable likelihood of success assuming that trained and qualified personnel are performing the recovery action(s).		

Table 4.5.8-2(b) 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 significant dependencies are not eliminated.		
QU-B3	SELECT initial truncation limits (e.g., system level, sequence level) such that the quantified result converges toward a stable value. EVALUATE truncation effects both before and after recovery actions are applied to avoid discarding important cutsets and sequences. ESTABLISH final truncation limits by an iterative process of demonstrating that the overall model results are not significantly changed and that no important accident sequences are inadvertently eliminated.		
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 [Note (1)]. When resolving circular logic, AVOID introducing significant conservatisms or non-conservatisms.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.8-2(b) Supporting Requirements for HLR-QU-B (Cont'd)

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-B6	ACCOUNT for system successes in addition to system failures in the evaluation of accident sequences to the extent needed for realistic estimation of CDF and LERF. 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. CORRECT by either: *(a)* designing path-dependent logic to eliminate mutually exclusive situations, or *(b)* deleting cutsets containing mutually exclusive events.		
QU-B8	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-B9	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)		

NOTE:
(1) NUREG/CR-2728, Interim Reliability Evaluation Program Procedures Guide, March 3, 1983

Table 4.5.8-2(c) 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. Avoid premature truncation of cutsets with multiple HFEs, by selecting appropriate HEP screening values in the initial quantification. The final quantification of these post-initiator HFEs may be done at the cutset or saved sequence level.		
QU-C2	ASSESS the degree of dependency between the HFEs in the cutset or sequence in accordance with HR-D5 and HR-G7. INCLUDE both the cognitive aspects, and the constraints imposed by available time or shared equipment. ESTIMATE the combined probability of the HFEs taking into account the dependency.		
QU-C3	TRANSFER the important sequence characteristics between event trees, not just the sequence frequency. For example, sequence characteristics can be transferred to another event tree by using the appropriate cutsets.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.8-2(d) Supporting Requirements for HLR-QU-D

The quantification results shall be reviewed and important contributors to CDF, such as initiating events, accident sequences, equipment failures, and operator errors, 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 the dominant cutsets or sequences to determine that the logic of the cutset or sequence is reasonable and to identify that there are no anomalies in the results. REVIEW the results of the PRA for modeling consistency (e.g., event sequence models consistency with systems models and success criteria) and operational consistency (e.g., plant configuration, procedures, and plant-specific and industry experience). REVIEW results to determine that the flag event settings, mutually exclusive event rules, and recovery rules yield logical results.		
QU-D2	IDENTIFY the modeling assumptions that drive the results. QUESTION modeling assumptions, asking if conditions outside those modeled could occur and, if so, could success criteria or other assumptions change. QUESTION modeled human actions for consistency with plant procedures and the range of conditions that would be obtained in the associated PRA sequence.		
QU-D3	No requirement to compare results to those from similar plants	COMPARE results to those from similar plants and **CONFIRM validity of unique outliers.**	COMPARE results to those from similar plants and **REVIEW significant differences.**
QU-D4	REVIEW a sampling of non-dominant accident cutsets or sequences to determine they are reasonable and have physical meaning.		
QU-D5	IDENTIFY important contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors. An acceptable approach is the use of importance measures. REVIEW the importance values for components and basic events to determine they make logical sense.	IDENTIFY important contributors to CDF, such as initiating events, accident sequences, equipment failures, common cause failures, and operator errors. An acceptable approach is the use of importance measures. **EXAMINE the importance of SSCs that contribute to initiating event frequencies.** REVIEW the importance values for components and basic events to determine they make logical sense.	

Table 4.5.8-2(e) Supporting Requirements for HLR-QU-E

Uncertainties in the PRA results shall be characterized. Sources of model uncertainty and key 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	CHARACTERIZE parameter uncertainty consistent with DA-D3.		
QU-E2	IDENTIFY key sources of model uncertainty, and the assumptions made or models adopted in response to those uncertainties.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.8-2(e) Supporting Requirements for HLR-QU-E (Cont'd)

Uncertainties in the PRA results shall be characterized. Sources of model uncertainty and key 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-E3	**ESTIMATE the uncertainty interval of the overall CDF results. Provide a basis for the estimate.**	**ESTIMATE the uncertainty interval of the overall CDF results. ESTIMATE the uncertainty intervals associated with parameter uncertainties and characterize the uncertainty associated with key model uncertainties.**	**PROPAGATE parameter uncertainties 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 probabilities is taken into account.**
QU-E4	**PROVIDE an assessment of the impact of the key model uncertainties on the results of the PRA.**	**EVALUATE the sensitivity of the results to uncertain model boundary conditions and other key assumptions using sensitivity analyses. EXAMINE key assumptions and parameters both individually and in logical combinations For example, a sensitivity analysis of logical combinations may evaluate the combined effects of modeling assumptions, HEP, CCF probabilities, and safety function success criteria.**	**EVALUATE the sensitivity of the results to uncertain model boundary conditions and other key assumptions using sensitivity analyses. EXAMINE key assumptions and parameters both individually and in logical combinations. For example, sensitivity analyses of logical combinations may evaluate the combined effects of modeling assumptions, HEPs, CCF probabilities, and safety function success criteria unless such sources of uncertainties have been adequately treated in the quantitative uncertainty analysis.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.8-2(f) Supporting Requirements for HLR-QU-F

Documentation shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-QU-F).

Index No. QU-F	Capability Category I	Capability Category II	Capability Category III
QU-F1	DOCUMENT the model integration process. INCLUDE any recovery analysis, and the results of the quantification including uncertainty and sensitivity analyses. 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 accidents to be non-dominant *(h)* the results of all sensitivity studies *(i)* the uncertainty distribution for the total CDF *(j)* importance measure results *(k)* a list of mutually exclusive events eliminated from the resulting cutsets and their bases for elimination *(l)* assumptions used in the quantification and an assessment of the significance of important assumptions		
QU-F2	DESCRIBE the key contributors to CDF in the PRA results summary.	DESCRIBE the key contributors to CDF in the PRA results summary. **PROVIDE a detailed description of dominant accident sequences or functional failure groups.**	
QU-F3	No requirement to document important assumptions and causes of uncertainlty.		DOCUMENT important assumptions and causes of uncertainty, such as: possible optimistic or conservative success criteria, suitability of the reliability data, possible modeling uncertainties (modeling limitations due to the method selected), degree of completeness in the selection of initiating events, possible spatial dependencies, etc.
QU-F4	DOCUMENT asymmetries in quantitative modeling to provide application users the necessary understanding regarding why such asymmetries are present in the model.		
QU-F5	DOCUMENT the process used to illustrate the computer code(s) used to perform the quantification will yield correct results process.		
QU-F6	DOCUMENT limitations that would impact applications.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

4.5.9 LERF Analysis (LE)

4.5.9.1 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) significant 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 LERF or non-LERF.

Table 4.5.9-1 High Level Requirements for LERF Analysis (LE)

Description	Designator	Requirement
Plant Damage Analysis	HLR-LE-A	Core damage sequences shall be grouped into plant damage states based on their accident progression attributes.
Accident Progression Analysis	HLR-LE-B	LERF evaluations shall include an analysis of the credible severe accident phenomena.
	HLR-LE-C	LERF evaluations shall include an analysis of containment system performance.
	HLR-LE-D	LERF evaluations shall include an analysis of containment structural capability.
LERF Quantification	HLR-LE-E	The frequency of different containment failure modes leading to a large early release shall be quantified and aggregated.
	HLR-LE-F	LERF shall be quantified in a manner that captures factors important to risk and supports an understanding of the sources of uncertainty.
Documentation	HLR-LE-G	The documentation of the LERF Analysis shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases.

Table 4.5.9-2(a) Supporting Requirements for HLR-LE-A

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

Index No. LE-A	Capability Category I	Capability Category II	Capability Category III
LE-A1	IDENTIFY those physical characteristics of Level 1 end states that can influence LERF. Examples include: *(a)* RCS pressure at core damage (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		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.9-2(a) Supporting Requirements for HLR-LE-A (Cont'd)

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

Index No. LE-A	Capability Category I	Capability Category II	Capability Category III
LE-A2	IDENTIFY the accident sequence characteristics that lead to the physical characteristics identified in 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 The references in Notes (1) and (2) provide lists of typical characteristics.		
LE-A3	SELECT a method for binning the accident sequences into plant damage states; this method may include extension of the accident sequences, development of a bridge tree, or some other approach.		
LE-A4	Either EXPAND the accident sequences, CONSTRUCT the bridge tree, or TRANSFER the Level 1 information to explicitly account for the LE-A1 and LE-A2 characteristics and ensure that dependencies are properly treated.		
LE-A5	DEFINE plant damage states based upon information in AS-A8, LE-A1, LE-A2, LE-A3, and LE-A4.		

NOTES:

(1) Nuclear Power Plant Response to Severe Accident, IDCOR Technical Summary Report, Atomic Industrial Forum, November 1984

(2) NUREG 1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, December 1990

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.9-2(b) Supporting Requirements for HLR-LE-B

LERF evaluations shall include an analysis of the credible severe accident phenomena (HLR-LE-B).

Index No. LE-B	Capability Category I	Capability Category II	Capability Category III
LE-B1	INCLUDE **potential** severe accident phenomena **that are important LERF contributors from the set identified in Table 4.5.9-2(a).** **An acceptable approach for identifying severe accident phenomena that could influence failure modes of various containment types is outlined in the LERF event trees contained in NUREG/CR-6595 [Note (1)].** *(a)* **INCLUDE as appropriate, unique plant issues as determined by expert judgment and/or past plant analyses.** *(b)* **EVALUATE qualitatively those severe accident phenomena that are not quantified in NUREG/CR-6595 [Note (1)] (e.g., induced steam generator tube rupture).**	INCLUDE **potential** severe accident phenomena that **could impact LERF from the set identified in Table 4.5.9-2(a). INCLUDE as appropriate, unique plant issues as determined by expert judgment and/or past plant analyses.**	INCLUDE **all** severe accident phenomena **sufficient to support development of a realistic containment event tree.** **INCLUDE all applicable postulated failure modes. Consider those identified by IDCOR [Note (2)] or in NUREG-1150 [Note (3)]. Known plant-specific failure modes, not included in the preceding evaluations, should also be included.**
LE-B2	USE containment loads (e.g., temperature, pressure) that are **conservative** for significant challenges to containment. **An acceptable alternative is the approach in NUREG/CR-6595 [Note (1)]. Realistic loads may be used when available.**	USE containment loads (e.g., temperature, pressure) that **are realistic when possible** for significant challenges to containment. **Conservative treatment may be used for non-dominant LERF contributors.**	USE containment loads (e.g., temperature, pressure) that **are realistic when possible** for significant challenges to containment.
LE-B3	USE **available containment analyses from generic or plant-specific sources. An acceptable alternative is the approach in NUREG/CR-6595 [Note (1)].**	USE **plant-specific containment thermal hydraulic analyses to model containment and RPV/RCS response under severe accident progression.**	

NOTES:

(1) NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events

(2) Nuclear Power Plant Response to Severe Accident, IDCOR Technical Summary Report, Atomic Industrial Forum, November 1984

(3) NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, December 1990

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

Table 4.5.9-2(c) Supporting Requirements for HLR-LE-C

LERF evaluations shall include an analysis of containment system performance (HLR-LE-C).

Index No. LE-C	Capability Category I	Capability Category II	Capability Category III
LE-C1	DEVELOP containment event tree(s) (CET) or equivalent structure(s) to propagate plant damage states in order to identify LERF scenarios in a manner consistent with the containment challenges and failure modes and intended level of detail. **Containment event trees developed in NUREG/CR-6595 [Note (1)] (with plant-specific modifications, if needed) are acceptable.**	DEVELOP containment event tree(s) (CET) or equivalent structure(s) to propagate plant damage states in order to identify LERF scenarios in a manner consistent with the containment challenges and failure modes and intended level of detail.	
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 [Note (1)].** USE HFEs consistent with EOPs/Severe Accident Management Guidelines (SAMGs), proceduralized actions, or Technical Support Center guidance.	INCLUDE **realistic** treatment of feasible operator actions following the onset of core damage. **Repair of equipment may be considered if appropriately justified.** USE HFEs consistent with EOPs/SAMGs, proceduralized actions, or Technical Support Center guidance.	
LE-C3	INCLUDE those branch points necessary to provide a **conservative** LERF estimation. **Containment event trees developed in NUREG/CR-6595 [Note (1)] (with plant-specific modifications, if needed) are acceptable.**	INCLUDE those branch points necessary to provide a **realistic** LERF estimation. **It is acceptable to selectively include mitigating actions by operating staff, effect of fission product scrubbing on radionuclide release, and expected beneficial failures. PROVIDE technical justification supporting the inclusion of any of these features.**	INCLUDE those branch points necessary to provide a **realistic** LERF calculation. **INCLUDE risk significant mitigating actions by operating staff, effect of fission product scrubbing on radionuclide release, and expected beneficial failures. PROVIDE technical justification for the inclusion of any of these features.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Table 4.5.9-2(c) Supporting Requirements for HLR-LE-C (Cont'd)

LERF evaluations shall include an analysis of containment system performance (HLR-LE-C).

Index No. LE-C	Capability Category I	Capability Category II	Capability Category III
LE-C4	USE **conservative** system success criteria. **Realistic criteria may be used.**	USE **realistic** system success criteria. **Conservative system success criteria may be used for non-dominant LERF contribution.**	USE **realistic** system success criteria.
LE-C5	DEVELOP system models that support LERF consistent with the applicable requirements of para. 4.5.4, as appropriate for the level of detail of the analysis.		
LE-C6	DEFINE HFEs that support LERF consistent with the applicable requirements of para. 4.5.5 as appropriate for the level of detail of the analysis.		
LE-C7	INCLUDE accident sequence dependencies in LERF event trees consistent with the applicable requirements of para. 4.5.2, as appropriate for the level of detail of the analysis.		
LE-C8	TREAT containment environmental impacts on continued operation of equipment and operator actions in a **conservative** manner. **An acceptable alternative is the approach in NUREG/CR-6595 [Note (1)]. A realistic treatment may be used.**	TREAT containment environmental impacts on continued operation of equipment and operator actions in a **realistic** manner **when possible. Conservative treatment may be used for non-dominant LERF contributors.**	TREAT containment environmental impacts on continued operation of equipment and operator actions in a realistic manner **when possible.**
LE-C9	TREAT containment failure impacts on continued operation of equipment and operator actions in a **conservative** manner. **An acceptable alternative is the approach in NUREG/CR-6595 [Note (1)]. A realistic treatment may be used.**	TREAT containment failure impacts on continued operation of equipment and operator actions in a **realistic** manner when possible. **Conservative treatment may be used for non-dominant LERF contributors.**	TREAT containment failure impacts on continued operation of equipment and operator actions in a **realistic** manner **when possible.**
LE-C10	TREAT containment bypass events in a **conservative** manner. **DO NOT TAKE CREDIT for reducing the class of the release by considering scrubbing. An acceptable treatment of containment bypass is in NUREG/CR-6595 [Note (1)]. A realistic treatment may be used.**	TREAT containment bypass in a **realistic** manner. **JUSTIFY any credit taken for reducing the class of the release by scrubbing.**	

NOTE:

(1) NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(d) Supporting Requirements for HLR-LE-D

LERF evaluations shall include an analysis of containment structural capability (HLR-LE-D).

Index No. LE-D	Capability Category I	Capability Category II	Capability Category III
LE-D1	DETERMINE the containment ultimate capacity for the dominant challenges that result in LERF. **USE a conservative evaluation of containment capacity for dominant containment failure modes. A realistic evaluation may be used.** **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.** **EVALUATE impact** of containment seals, penetrations, hatches, drywell heads (BWRs) and vent pipe bellows. **INCLUDE in failure mode, as required.** **Quasi-static containment capability evaluations are acceptable unless hydrogen concentrations are expected to result in potential detonations. Such considerations may need to be included for small volume containments, such as the ice-condenser type.** **An acceptable alternative is the approach in NUREG/CR-6595 [Note (1)].**	DETERMINE the containment ultimate capacity for the dominant challenges that result in LERF. **PERFORM a realistic containment capacity analysis for dominant containment failure modes. The analysis may include some conservative parameters.** **EVALUATE impact** of containment seals, penetrations, hatches, drywell heads (BWRs), and vent pipe bellows. **INCLUDE in failure mode, as required.** **If generic calculations are used in support of the assessment, JUSTIFY applicability to the plant being evaluated.** **Quasi-static containment capability evaluations are acceptable unless hydrogen concentrations are expected to result in potential detonations. Such considerations may need to be considered for small volume containments, such as the ice-condenser type.**	DETERMINE the containment ultimate capacity for the dominant challenges that result in LERF. PERFORM a realistic containment capacity analysis for dominant containment failure modes by using plant-specific input. **INCLUDE behavior** of *(a)* containment seals *(b)* penetrations *(c)* hatches *(d)* drywell head (BWRs) *(e)* vent pipe bellows (BWRs) **beyond the design basis temperature and pressure conditions.** **PROVIDE static and dynamic failure capabilities, as appropriate.**

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(d) Supporting Requirements for HLR-LE-D (Cont'd)

LERF evaluations shall include an analysis of containment structural capability (HLR-LE-D).

Index No. LE-D	Capability Category I	Capability Category II	Capability Category III
LE-D2	When failure location [Note (2)] affects the event classification as a LERF, DEFINE failure location based upon a **conservative** plant-specific containment assessment. **JUSTIFY applicability of generic and other analyses. An acceptable alternative is the approach in NUREG/CR-6595 [Note (1)].**	When failure location [Note (2)] affects the event classification as a LERF, DEFINE failure location based upon **realistic** plant-specific containment assessment.	
LE-D3	**USE a conservative evaluation of interfacing system failure probability for dominant failure modes. A realistic evaluation is acceptable. If generic analyses generated for similar plants are used, JUSTIFY applicability to the plant being evaluated.**	**PERFORM an interfacing system failure probability analysis for dominant failure modes. Evaluation of failure probability may include conservatisms. INCLUDE behavior of piping, relief valves, pump seals, and heat exchanges at applicable temperature and pressure conditions. If generic analyses generated for similar plants are used, JUSTIFY applicability to the plant being evaluated.**	**PERFORM a realistic interfacing system failure probability analysis for dominant failure modes. USE plant-specific input. INCLUDE behavior of piping, relief valves, pump seals, and heat exchanges at applicable temperature and pressure conditions. PROVIDE static and dynamic failure capabilities, as appropriate.**
LE-D4	**USE a conservative evaluation** of secondary side isolation capability for dominant SG tube failure modes. **A realistic evaluation may be used. If generic analyses generated for similar plants are used, JUSTIFY applicability to the plant being evaluated.**	**PERFORM** a secondary side isolation capability **analysis** for dominant SG tube failure modes. **Evaluation of capability may include conservatisms. If generic analyses generated for similar plants are used, JUSTIFY applicability to the plant being evaluated.**	**PERFORM a realistic** secondary side isolation capability **analysis** for dominant SG tube failure modes. **INCLUDE behavior of relief and isolation valves at applicable temperature and pressure conditions.**
LE-D5	TREAT induced SG tube rupture in a **conservative** manner. **A realistic treatment may be used.**	TREAT induced SG tube rupture in a **realistic** manner, **when practical. Conservative treatment may be used, when justified.**	TREAT induced SG tube rupture in a **realistic** manner.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(d) Supporting Requirements for HLR-LE-D (Cont'd)

LERF evaluations shall include an analysis of containment structural capability (HLR-LE-D).

Index No. LE-D	Capability Category I	Capability Category II	Capability Category III
LE-D6	TREAT containment isolation in a **conservative** manner. **A realistic treatment may be used.** 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.	TREAT containment isolation in a **realistic** manner. **Conservative treatment may be used for non-dominant contributors.** INCLUDE consideration of both the failure of containment isolation systems to perform properly and the status of safety systems which do not have automatic isolation provisions.	TREAT containment isolation in a **realistic** manner. INCLUDE consideration of both the failure of containment isolation systems to perform properly and the status of safety systems which do not have automatic isolation provisions.

NOTES:
(1) NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events
(2) Containment failures below ground level may not be LERFs even if the timing is early. Such failures may arise as a result of failures in the basemat region.

Table 4.5.9-2(e) 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 used in LERF analysis consistent with the applicable requirements of paras. 4.5.5 and 4.5.6 including consideration of the severe accident plant conditions for both equipment and operators, as appropriate for the level of detail of the analysis.		
LE-E2	USE **conservative** parameter estimates for **determination of CET branch points. A conservative data set for some key parameters is included in NUREG/CR-6595 [Note (1)].**	USE **realistic** parameter estimates **when possible for dominant LERF sequences**.	USE **realistic** parameter estimates **when possible.**
LE-E3	Starting with plant damage states, QUANTIFY LERF consistent with the applicable requirements of para. 4.5.8, as appropriate for the level of detail of the analysis.		

NOTE:
(1) NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(f) Supporting Requirements for HLR-LE-F

LERF shall be quantified in a manner that captures factors important to risk and supports an understanding of sources of uncertainty (HLR-LE-F).

Index No. LE-F	Capability Category I	Capability Category II	Capability Category III
LE-F1	**LIST** the dominant contributors to LERF (e.g., HPME, steam explosions, ISLOCA). **REVIEW for reasonableness.**		**PERFORM an importance analysis to identify** the dominant contributors to LERF.
LE-F2	**PROVIDE a qualitative assessment of the key sources of uncertainty.** **Examples:** ***(a)* Identify bounding assumptions.** ***(b)* Identify conservative treatment of phenomena.**	**PROVIDE uncertainty analysis which identifies the key sources of uncertainty and includes sensitivity studies for dominant contributors to LERF.**	

Table 4.5.9-2(g) Supporting Requirements for HLR-LE-G

The documentation of the LERF analysis shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-LE-G).

Index No. LE-G	Capability Category I	Capability Category II	Capability Category III
LE-G1	DOCUMENT *(a)* the physical characteristics and accident sequence characteristics used to group the accident sequences into plant damage states *(b)* the method used to bin the accident sequences into plant damage states *(c)* the plant damage states and their attrbutes, as used in the analysis		
LE-G2	DOCUMENT the potential LERF contributors considered, where appropriate, including *(a)* containment isolation failure *(b)* containment bypass *(c)* energetic containment failures *(d)* hydrogen related phenomena *(e)* liner melt-through (Mark I containment) *(f)* containment failure at vessel breach *(g)* induced SGTRs (PWRs) *(h)* containment venting *(i)* suppression pool bypass mechanism (BWRs) *(j)* containment flooding (BWRs)		
LE-G3	DOCUMENT treatment of key factors influencing containment capability, as appropriate for the level of detail of the analysis, including *(a)* design details (i.e., heat sink distribution, circulation paths, ignition sources, water availability, gravity drain paths, and cavity geometry) *(b)* equipment survivability credited beyond design basis envelope		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

Table 4.5.9-2(g) Supporting Requirements for HLR-LE-G (Cont'd)

The documentation of the LERF analysis shall be performed in a manner that facilitates peer review, as well as future upgrades and applications of the PRA by describing the processes that were used, and providing details of the assumptions made and their bases (HLR-LE-G).

Index No. LE-G	Capability Category I	Capability Category II	Capability Category III
LE-G4	DOCUMENT, as appropriate for the level of detail of the analysis, *(a)* containment challenges treated *(b)* containment failure modes identified *(c)* containment event tree (CET) and basis for event tree nodes *(d)* containment capacity and its basis *(e)* containment failure locations and probabilities *(f)* basis for parameter estimates		
LE-G5	DOCUMENT the model integration process. INCLUDE the results of the quantification including uncertainty and sensitivity analyses, as appropriate for the level of detail of the analysis. Documentation typically includes *(a)* a general description of the quantification *(b)* important assumptions *(c)* the total plant LERF and contributions from the different plant damage states and accident classes *(d)* equipment or human actions that are the key factors *(e)* the results of all sensitivity studies, (as applicable)		
LE-G6	DESCRIBE the key contributors to LERF.	DESCRIBE the key contributors to LERF. **PROVIDE a detailed description of dominant plant damage states and accident progression sequences.**	
LE-G7	DOCUMENT sources of uncertainty.		
LE-G8	IDENTIFY limitations that would impact applications.		

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS Not for Resale

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Table 4.5.9-3 Potential LERF Contributors To Be Considered

	Containment Design				
Potential LERF Contributor	**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	...	...	...
(c) De-inerted operation	...	...	x	x	x
(d) Core debris impingement	[Note (2)]	x	x	x	
Steam explosion	...	...	x	x	x
Shell melt-through	...	...	x	...	...
Suppression pool bypass	...	x	x	x	...
RPV and/or containment venting	...	...	x	x	x
Isolation condenser tube rupture	...	x (if applicable)	...	...	...
Vacuum breaker failure	...	...	x	x	x
Hydrodynamic loads under severe accident conditions	...	...	x	...	...
Containment flooding	...	...	x	...	...
In-vessel recovery	x	x	x	x	x
ATWS-induced failure	...	...	x	x	x

GENERAL NOTE: Combinations of phenomena should also be considered where appropriate.
NOTES:
(1) dry well (DW) isolation failure
(2) steel shell designs

Not for Resale

SECTION 5

5.1 Purpose 88
5.2 PRA Configuration Control Program 88
5.3 Monitoring PRA Inputs and Collecting New Information 88
5.4 PRA Maintenance and Upgrades 88
5.5 Pending Changes 88
5.6 Previous PRA Applications 88
5.7 Use of Computer Codes 88
5.8 Documentation 89

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 5
PRA CONFIGURATION CONTROL

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.

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 evaluates the impact of changes on previously implemented risk-informed decisions that have used the PRA

(e) a process that maintains configuration control of computer codes used to support PRA quantification

(f) documentation of the Program

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 sub-system 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.

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 para. 5.3 shall be evaluated to determine whether such information warrants PRA maintenance or PRA upgrade. (See Section 2 for the distinction between PRA maintenance and PRA upgrade.) Changes that would impact risk-informed decisions should be prioritized to ensure that the most significant changes are 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 para. 3.5.

Changes to a PRA due to PRA maintenance shall meet the requirements of Section 4. Upgrades of a PRA shall satisfy the peer review requirements specified in Section 6, but limited to aspects of the PRA that have been upgraded.

5.5 Pending Changes

This Standard recognizes that immediately following a plant change, or upon identification of a subject for model improvement, a PRA may not represent the plant until the change is incorporated. Therefore, the PRA configuration control process shall consider the cumulative impact of pending changes on the application being performed. These changes should be addressed in a fashion similar to the approach used in Section 3 to address elements that are determined to be inadequate.

5.6 Previous PRA Applications

A process shall exist to evaluate the impact of a PRA change on previously implemented risk-informed decisions that relied upon PRA information and that affect the safe operation of the plant.

5.7 Use of Computer Codes

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

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

5.8 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

(f) record of the process and results used to address the cumulative impact of pending changes

(g) record of the process and results used to evaluate changes on previously implemented risk-informed decisions pursuant to para. 5.6

(h) a description of the process used to maintain software configuration control

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 6

6.1 Purpose 91
6.2 Peer Review Team Composition and Personnel Qualifications 91
6.3 Review of PRA Elements to Confirm the Methodology 92
6.4 Expert Judgment 94
6.5 PRA Configuration Control 94
6.6 Documentation 94

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

SECTION 6
PEER REVIEW

6.1 Purpose

This Section provides requirements for peer review of a PRA to be used in risk-informed decisions for commercial nuclear power plants. PRAs used for applications applying this Standard shall be peer reviewed. The peer review shall assess the PRA to the extent necessary to determine if the methodology and its implementation meet the requirements of this Standard. The peer review need not assess all aspects of the PRA against all Section 4 requirements; however, enough aspects of the PRA shall be reviewed for the reviewers to achieve consensus on the adequacy of methodologies and their implementation for each PRA Element.

6.1.1 Frequency. Only a single complete peer review is necessary prior to using a PRA. In addition, Section 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.

6.1.2 Methodology. The review shall be performed using a written methodology that assesses the requirements of Section 4 and addresses the requirements of Section 6.

The peer review methodology shall consist of the following elements:

(a) 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 Section 4 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

NEI-00-02[1] provides an example of an acceptable review methodology; however, the differences between the supporting requirements of Section 4 of this Standard and the supporting requirements of Appendix B of NEI-00-02 shall be evaluated. This evaluation shall be documented.

6.2 Peer Review Team Composition and Personnel Qualifications

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 Section 4 and the interfaces between those elements

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

6.2.2 General

6.2.2.1 The peer review team members individually shall

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

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

6.2.2.2 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.

The peer review team members shall

(a) not be allowed to review their own work or work for which they have contributed

(b) not be allowed to review a PRA for which they have a conflict of interest, such as a financial or career path incentive or disincentive that may influence the outcome of the peer review

[1] NEI-00-02, Probabilistic Risk Assessment (PRA) Peer Review Process Guidance, 2002

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

6.2.3 Specific. 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.

One member of the peer review team (the technical integrator) shall be familiar with all the PRA Elements identified in this Standard and shall have demonstrated the capability to integrate these PRA Elements.

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.

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.

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.

Exceptions to the requirements of this paragraph may be taken based on the availability of appropriate personnel to develop a team. All such exceptions shall be documented in accordance with para. 6.6 of this Standard.

6.3 Review of PRA Elements to Confirm the Methodology

The peer review team shall use the requirements of this Subsection for the PRA Elements being reviewed to determine if the methodology and the implementation of the methodology for each PRA Element meet the requirements of this Standard. Some subparagraphs of para. 6.3 contain specific suggestions for the review team to consider during the review. Additional material for those Elements may be reviewed depending on the results obtained. These suggestions are not intended to be a minimum or comprehensive list of requirements. The judgment of the reviewer shall be used to determine the specific scope and depth of the review in each PRA Element.

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

The HLRs and the composite of the SRs of Section 4 shall be used by the peer review team to assess the completeness of a PRA Element.

6.3.1 Initiating Event Analysis (IE). The entire initiating event analysis shall be reviewed.

6.3.2 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/Station Blackout 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

6.3.3 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 modeling of 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

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

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) the containment response calculations, performed specifically for the PRA, for the dominant plant damage states

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

6.3.4 Systems Analysis (SA). A review shall be performed on the systems analysis.

The portion of selected system models selected for review typically includes

(a) dominant systems contributing to the CDF or LERF calculated in the PRA

(b) different models reflecting different levels of detail

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

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

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

6.3.5 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) HEPs for dominant human actions contributing to the CDF or LERF calculated in the PRA

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

(c) post-accident HEPs

(d) pre-initiator HEPs for both instrumentation miscalibration and failure of equipment

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

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

(g) HEPs less than 1E-4

(h) HEPs involving remote actions in harsh environments

6.3.6 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 for component failure modes 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

6.3.7 Internal Flooding (IF). A review shall be performed on the internal flooding analysis.

The portion of the internal flooding analysis selected for review typically includes

(a) dominant internal flooding contributors to the CDF or LERF calculated in the PRA

(b) the screening of any flood areas

(c) internal flood initiating event frequencies

(d) internal flooding scenario involving each identified flood source

(e) internal flooding scenarios involving flood propagation to adjacent flood areas

(f) internal flooding scenario that involves each of the flood-induced component failure mechanisms (i.e., one flood scenario for each mechanism)

(g) one internal flooding scenario involving each type of identified accident initiator (e.g., transient and LOCA)

6.3.8 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)

6.3.9 LERF Analysis (LE). The LERF analysis and the Level 1/LERF interface process shall be reviewed.

6.3.9.1 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)

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

(b) interface mechanism used

(c) CDF carryover

6.3.9.2 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 functional events necessary to achieve a safe stable containment endstate

(h) sensitivity analysis

6.4 Expert Judgment

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

6.5 PRA Configuration Control

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

6.6 Documentation

6.6.1 Peer Review Team Documentation. The peer review team's documentation shall demonstrate that the review process appropriately implemented the review requirements.

Specifically, the peer review documentation shall include the following:

(a) identification of the version of the PRA reviewed

(b) the names of the peer review team members

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

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

(e) a discussion of the extent to which each PRA Element was reviewed

(f) results of the review identifying any differences between the requirements in Sections 4 and 5 of this Standard and the methodology implemented, defined to a sufficient level of detail that will allow the resolution of the differences

(g) identification and significance of exceptions and deficiencies with respect to SRs

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

(i) recommended alternatives for resolution of any differences

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.

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale

REFERENCES

ASME OM Code for Operation and Maintenance of Nuclear Power Plants, Code Case OMN-10, Requirements for Safety Significance Categorization of Snubbers Using Risk Insights and Testing Strategies for Inservice Testing of LWR Power Plants

Publisher: The American Society of Mechanical Engineers (ASME), Three Park Avenue, New York, NY 10016; Order Department: 22 Law Drive, Box 2300, Fairfield, NJ 07007

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

EPRI TR-101711, Sharp 1 — Revised Systematic Human Action Reliability Procedure; D. T. Wakefield, G. W. Parry, G. W. Hannaman, A. J. Spurgin; 1992

EPRI TR-105396, PSA Applications Guide; D. True, et al.; August 1995

Publisher: The Electric Power Research Institute (EPRI), 3412 Hillview Avenue, Palo Alto, CA 94304

Kennedy, R. P. and M. K. Ravindra, "Seismic Fragilities for Nuclear Power Plant Risk Studies," *Nuclear Engineering and Design,* Vol. 79, No. 1, pp. 47–68, 1984

Publisher: Elsevier Science, P. O. Box 945, New York, NY 10159

NEI-00-02, Probabilistic Risk Assessment Peer Review Process Guidance, 2000

Publisher: Nuclear Energy Institute (NEI), 1776 I Street NW, Suite 400, Washington, DC 20006

NRC 50-275, 50-323, Final Report of the Diablo Canyon Long-Term Seismic Program, Pacific Gas and Electric Company

NUREG-0800, Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decision-Making: General Guidance, *NRC Standard Review Plan,* Chapter 19, 1998

NUREG-1150, Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, December 1990

NUREG/CR-1278, Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications; A. D. Swain and H. E. Guttmann; August 1983 (THERP)

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

NUREG/CR-2728, Interim Reliability Evaluation Program Procedures Guide, March 3, 1983

NUREG/CR-2300, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, 1983

NUREG/CR-4550, Auxiliary Feedwater System Risk-Based Inspection Guide for the Salem Nuclear Generating Station, October 1990

NUREG/CR-4639, Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR), Vols. 1–5, 1994

NUREG/CR-4772, Accident Sequence Evaluation Program Human Reliability Analysis Procedure; A.D. Swain; February 1987 (ASEP)

NUREG/CR-5032, Modeling Time to Recover and Initiate Even Frequency for Loss-of-Offsite Power Incidents at Nuclear Power Plants, March 1988

NUREG/CR-5485, Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment; November 20, 1998

NUREG/CR-5496, Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1986

NUREG/CR-5497, Common-Cause Failure Parameter Estimations

NUREG/CR-5750, Rates of Initiating Events at U.S. Nuclear Power Plants, Idaho National Engineering and Environmental Laboratory, Idaho Falls, February, 1999

NUREG/CR-6268, Common Cause Failure Database and Analysis System, Vols. 1–4, 1998

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

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS
Not for Resale

NUREG/CR-6595, Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events

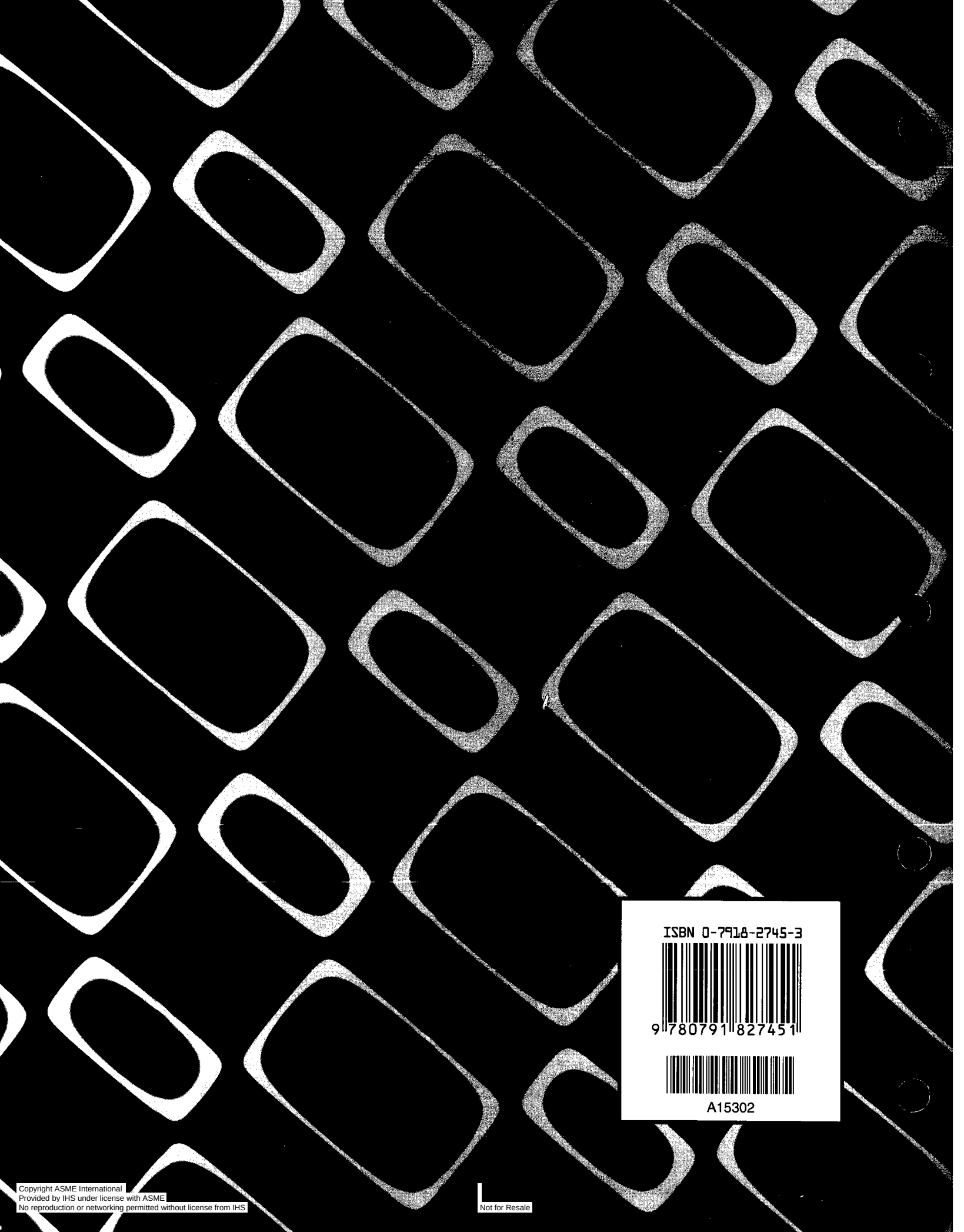
Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852

Nuclear Power Plant Response to Severe Accident, IDCOR Technical Summary Report, Atomic Industrial Forum, November 1984

Publisher: Atomic Industrial Forum, c/o Nuclear Energy Institute (NEI), 1776 I Street NW, Suite 400, Washington, DC 20006

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale



ISBN 0-7918-2745-3

A15302

Copyright ASME International
Provided by IHS under license with ASME
No reproduction or networking permitted without license from IHS

Not for Resale